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# Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada

## Volume 2: Repository Safety Before Permanent Closure

Office of Nuclear Material Safety and Safeguards

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# Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada

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**NOTE TO READER**: In June 2008, the U.S. Department of Energy (DOE) submitted a license application seeking authorization to construct a geologic repository at Yucca Mountain. After docketing the DOE license application, the U.S. Nuclear Regulatory Commission (NRC) staff began documenting its review in a Safety Evaluation Report (SER). In March 2010, DOE filed a motion to withdraw its application before the Atomic Safety and Licensing Board, which denied DOE's motion in June 2010. During this time period, Congress reduced funding for the NRC's review of the application, with no funds appropriated for Fiscal Year 2012. On September 30, 2010, DOE's Office of Civilian Radioactive Waste Management ceased operations and assigned the remaining Yucca Mountain-related responsibilities, such as site closure, to other offices within DOE. In October 2010, the NRC staff began orderly closure of its Yucca Mountain activities. In September 2011, the Commission announced it was evenly divided on whether to overturn or uphold the Atomic Safety and Licensing Board's decision denying DOE's motion to withdraw its application. The Commission directed the Board, in recognition of budgetary limitations, to complete all necessary and appropriate case management activities, and the Atomic Safety and Licensing Board suspended the proceeding on September 30, 2011.

In August 2013, the U.S. Court of Appeals for the District of Columbia Circuit issued a decision granting a writ of mandamus and directed NRC to resume the licensing process for DOE's license application. In November 2013, the Commission directed the NRC staff to complete and issue the SER associated with the license application. Because of the lapse in time and changes within DOE between license application submittal and the issuance of this SER volume, some information in the application does not reflect current circumstances. In addition, scientific information continues to be published in areas relevant to the topics considered in the license application. When these situations are relevant to the NRC staff's evaluation of the license application in this volume, the SER identifies and addresses them, as appropriate.

The SER details the NRC staff's review of DOE's license application and supporting information consistent with NRC regulations and the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by the Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63 (NRC, 2009ab).

This volume is one of five volumes that comprise the SER. Each volume was published as it was completed. The SER volume number and section number within a volume are based on the YMRP. Use of SER section numbers that correspond to the YMRP section numbers facilitated the NRC staff's writing of the SER and allows the reader to easily find the applicable review methods and acceptance criteria within the YMRP. The following table provides the topics and SER sections for each volume.

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**References**:

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: U.S. Nuclear Regulatory Commission.

# ABSTRACT

Volume 2, "*Repository Safety Before Permanent Closure*," of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the U.S. Department of Energy's (DOE) Safety Analysis Report (SAR), entitled, "*Repository Safety Before Permanent Closure,*" provided by DOE on June 3, 2008, as updated by DOE on February 19, 2009. In its application, DOE seeks authorization from the Commission to construct a repository for high-level radioactive waste at Yucca Mountain, Nevada. The NRC staff also reviewed information DOE provided in response to the NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, SER Volume 2 documents the results of the NRC staff's evaluation to determine whether the proposed repository design complies with the performance objectives and requirements that apply before the repository is permanently closed. Based on its review, and subject to the proposed conditions of Construction Authorization documented in Volume 2 of this SER, the NRC staff finds, with reasonable assurance, that DOE has demonstrated compliance with the NRC regulatory requirements for preclosure safety. This includes "Performance objectives for the geologic repository operations area through permanent closure" in 10 CFR 63.111, "Requirements for preclosure safety analysis of the geologic repository operations area" in 10 CFR 63.112, and "Preclosure Public Health and Environmental Standards" in 10 CFR Part 63, Subpart K.

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# EXECUTIVE SUMMARY

## 1.0 Background

Volume 2, *"Repository Safety Before Permanent Closure,"* of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Safety Analysis Report (SAR) the U.S. Department of Energy ("DOE" or the "applicant") provided in its June 3, 2008, license application (LA) submittal for construction authorization (DOE, 2008ab), as updated on February 19, 2009 (DOE, 2009av). The NRC staff also reviewed the information DOE provided in response to the NRC staff's requests for additional information (RAIs) and other information that DOE provided related to the SAR. In particular, this SER Volume 2 documents the results of the NRC staff's evaluation to determine whether the design of the proposed geologic repository operations area (GROA) for Yucca Mountain complies with the performance objectives and requirements that apply before the repository is permanently closed. These performance objectives and requirements can be found in NRC's regulations at 10 CFR Part 63, Subparts E and K. In conducting its review, the NRC staff was guided by the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa).

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. Pursuant to 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, a period of operations, and termination of the license. The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The period of operations includes (i) the time during which emplacement would occur, (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable, and (iii) permanent closure. The license application includes DOE's subsurface facility development plan (SAR Section 1.3.1) that explains operations in the subsurface facility will be preceded by a period of initial construction, during which three emplacement drifts will be built and commissioned to receive waste. According to DOE, the start of waste emplacement will mark the end of the period of initial construction and the beginning of repository operations in the subsurface facility. DOE stated its plans for the period of operation, also referred to as the preclosure period, is approximately 100 years.

**Preclosure Performance Objectives and Requirements**

In its review of DOE's application, the NRC staff used a risk-informed and performance-based review process and considered, among other things, whether the site and design comply with the performance objectives and requirements contained in 10 CFR Part 63, Subparts E and K. In accordance with 10 CFR 63.21, the applicant must include in its SAR a preclosure safety analysis (PCSA). As described in 10 CFR 63.102(f), the PCSA identifies and categorizes event sequences and identifies structures, systems, and components (SSCs) important to safety (ITS) and associated design bases and criteria. The PCSA is part of the risk-informed and performance-based review, which is described further in the following section. An event sequence, as defined in 10 CFR 63.2, is a series of actions and/or occurrences within the natural and engineered components of the facility that could potentially expose individuals to radiation. The applicant's PCSA must demonstrate that the repository, as proposed to be designed, constructed, and operated, will meet the specified radiological dose limits throughout the preclosure period. The applicant must also demonstrate that the GROA design will not preclude retrievability of the wastes, in whole or in part, from the underground facility where these wastes will be emplaced for permanent disposal (10 CFR 63.111).

**Risk-Informed, Performance-Based Review**

The PCSA quantifies GROA performance and is used to demonstrate compliance with the preclosure performance requirements in 10 CFR 63.111. The NRC staff evaluated DOE's PCSA using a risk-informed and performance-based review. A PCSA is a systematic analysis that answers three basic questions that are used to define risk: What can happen? How likely is it to happen? What are the resulting consequences? The applicant's PSCA includes a number of evaluations, such as identification of hazards and initiating event sequences; development and categorization of event sequences; failure mode and reliability assessments of structures, systems, and components (SSCs); and SSCs' fragility assessments. Because the PCSA encompasses a broad range of technical subjects, the NRC staff used risk information throughout the review process to ensure that the NRC staff's review focused on significant items that could affect preclosure performance. YMRP Section 2.1.1 provides guidance to the NRC staff on how to apply risk information throughout its review of the applicant's PCSA.

## 2.0 Sections of the Preclosure Review

DOE developed and implemented a PCSA to demonstrate that its proposed application for the GROA meets the preclosure performance objectives in 10 CFR Part 63. The NRC staff reviewed DOE's PCSA to determine whether the PCSA contains sufficient information to satisfy 10 CFR Part 63 preclosure requirements and whether the PCSA demonstrates that the repository meets the performance objectives of the GROA through permanent closure. Areas reviewed in SER Volume 2 are summarized in the following sections; these areas correspond with elements of DOE's PCSA.

## 2.1 Site Description as it Pertains to Preclosure Safety Analysis

SER Section 2.1.1.1 provides the NRC staff's review of DOE's Yucca Mountain site description as it pertains to the preclosure safety analysis (PCSA) and design of the GROA. The NRC staff focused its review on the adequacy of DOE's site characterization information to ensure that a sufficient level of detail is present to inform and permit the evaluation of both the PCSA and the design of the GROA.

The NRC staff has reviewed the SAR and other information submitted in support of the license application relevant to site characteristics of the Yucca Mountain site important to the preclosure safety of the facility and the GROA design, and finds, with reasonable assurance, that the relevant requirements of 10 CFR 63.21(c)(1)(i–iii), 10 CFR 63.21(c)(15), 10 CFR 63.112(b), and 10 CFR 63.112(c) are met, subject to a proposed condition of the construction authorization that DOE confirm that its site characterization information and related analyses in the SAR continue to be accurate with respect to (i) site boundaries, (ii) man-made features, (iii) previous land use, (iv) existing structures and facilities, and (v) potential exposure to residual radioactivity (SER Section 2.1.1.1.4).

## 2.2 Descriptions of Structures, Systems, Components, and Operational Activities as They Pertain to Preclosure Safety Analysis

SER Section 2.1.1.2 provides the NRC staff's review of DOE's description and design information of structures, systems, and components (SSCs); safety controls (SCs); equipment; and operational process activities, both important to safety (ITS) and not important to safety (non-ITS) in the surface and the subsurface facilities of the GROA for the application to receive a construction authorization under 10 CFR Part 63. The primary focus of Section 2.1.1.2 is for the NRC staff to assess the acceptability of the applicant's information related to description and design information of SSCs, SCs, equipment, radioactive wastes to be disposed, and operations of the GROA facility and PCSA. This SER section also provides the NRC staff's review of DOE's description and design of the non-ITS underground openings of the GROA.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, and with the proposed condition of construction authorization, finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), 10 CFR 63.21(c)(4), and 10 CFR 63.112(a) are satisfied in that DOE has provided an adequate description and design information for the structures, systems, components, equipment, and process activities of the geologic repository operations area. The NRC staff also finds, with reasonable assurance, that the requirements of 10 CFR 63.111(e), 10 CFR 63.112(a), 10 CFR 63.112(d), and 10 CFR 63.112(f) are satisfied in that an adequate description, discussion, and design information, which satisfactorily defines the relationship between design criteria and the performance objectives, and which identifies the relationship between the design bases and the design criteria, has been provided for non-ITS underground openings of the GROA. However, the NRC staff has found that DOE has not presented description of design and safety analyses for the multicanister overpacks (MCOs) for handling DOE spent nuclear fuel (SNF) and for handling commercial mixed oxide (MOX) fuel. DOE stated that it will submit an amendment request for the MCOs and the MOX fuel in obtaining authorization to receive and possess this waste. Therefore, the NRC staff proposes a condition of construction authorization that DOE shall not, without prior NRC review and approval, accept DOE spent nuclear fuel (SNF) in multicanister overpacks (MCOs) or commercial mixed oxide (MOX) fuel (SER Section 2.1.1.2.3.6.1).

## 2.3 Identification of Hazards and Initiating Events

SER Section 2.1.1.3 provides the NRC staff's review of DOE's identification of hazards and initiating events in both the surface and subsurface facilities of the GROA at Yucca Mountain during the preclosure period. The NRC staff focused its review on DOE's information identifying hazards and initiating events pertaining to the PCSA and the GROA design. Specifically, the NRC staff focused on (i) whether DOE adequately identified and provided systematic analysis of the potential naturally occurring and human-induced hazards and initiating events, including (i) associated probabilities of occurrence and (ii) whether DOE provided an adequate technical basis for either inclusion or exclusion of potential naturally occurring or human-induced hazards and initiating events.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.112(b) and 10 CFR 63.112(d) are met, subject to a proposed condition of construction authorization regarding flight restrictions and operational constraints used to limit aircraft hazards at the

GROA. The NRC staff finds that (i) the applicant adequately identified and provided systematic analysis of the potential naturally occurring and human-induced hazards and initiating events, including associated probabilities of occurrence, and (ii) the applicant provided adequate technical basis for either inclusion or exclusion of potential naturally occurring or human-induced hazards and initiating events in the PCSA (SER Section 2.1.1.3.3.1.3.3, and SER Section 2.1.1.3.3.1.3.4, respectively).

## 2.4 Identification of Event Sequences

SER Section 2.1.1.4 provides the NRC staff's review of DOE's information on identification of event sequences for the PCSA. The primary focus of Section 2.1.1.4 is for the NRC staff to assess DOE's methodology and technical bases for developing, quantifying, and categorizing event sequences used in the PCSA. The NRC staff focused its review on whether the (i) methodology is acceptable; (ii) event sequence development is based on consideration of relevant operational and site-specific natural hazards, reasonable combinations of initiating events, and is consistent with the facility description; (ii) reliability of the SSCs used to prevent or mitigate event sequences is consistent with the design information; and (iii) quantification of probability of occurrences of the event sequences and the categorization of event sequences are reasonable.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.112(b) are satisfied regarding the identification and categorization of event sequences for naturally occurring and human-induced hazards and initiating events at the geologic repository operations area event sequences.

## 2.5 Consequence Analysis

SER Section 2.1.1.5 provides the NRC staff's review of DOE's consequence analysis methodology and demonstration that the repository design meets 10 CFR Parts 20 and 63 radiation protection requirements. The NRC staff focused its review on DOE's information regarding (i) the methodology and input parameters used for the dose calculation, (ii) the consistency of source terms used in the dose calculation with those described in SAR Section 1.5, and (iii) the methodology for the worker and public dose determination.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the information on dose consequences for the GROA for construction authorization is adequate and satisfies 10 CFR 63.111(a), (b), and (c); 10 CFR 63.204; 10 CFR 20.1101(d); 10 CFR 20.1201(a); and 10 CFR 20.1301.

## 2.6 Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems

SER Section 2.1.1.6 provides the NRC staff's review of DOE's identification of important-to-safety (ITS) structures, systems, and components (SSCs); safety controls (SCs); and measures to ensure availability and reliability of the safety systems. The NRC staff focused its review on the DOE's PCSA that includes an analysis of the performance of the SSCs to (i) identify those SSCs that are important to safety, (ii) identify and describe the controls relied on to limit or prevent potential event sequences or mitigate their consequences, and (iii) identify measures taken to ensure the availability of safety systems, as required in 10 CFR 63.112(e).

The NRC staff reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e) are satisfied, subject to the proposed condition of the construction authorization regarding the design of the ITS safety interlock subsystems (SER Section 2.1.1.6.3.2.8.2.1). An adequate PCSA of the performance of the SSCs ITS has been provided. In particular, the NRC staff finds that (i) SSCs ITS are identified; (ii) criteria for categorization of the SSCs ITS are adequately developed and categorization of items is acceptable; (iii) controls that will be relied on to limit or prevent potential event sequences, or mitigate their consequences, are acceptable; and (iv) measures are adequate to ensure the availability and reliability of the SSCs ITS.

## 2.7 Design of Structures, Systems, and Components Important to Safety and Safety Controls

SER Section 2.1.1.7 provides the NRC staff's review of DOE's proposed design of ITS SSCs and SCs in the geologic repository operations area (GROA). The NRC staff focused its review on (i) whether DOE has provided an adequate description of the design of ITS SSCs and SCs, for both the surface and the subsurface facilities of the GROA, that satisfactorily includes the design bases, design criteria, and the relationship between design criteria and the preclosure performance objectives specified at 10 CFR 63.111(a) and (b); and (ii) the capability of the proposed design of ITS SSCs and SCs to perform their intended safety functions.

The NRC staff reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3), 10 CFR 63.112(e)(9), and 10 CFR 63.112(f) are satisfied, subject to proposed conditions of construction authorization that DOE shall not accept certain waste packages and canisters at the repository until DOE provides analyses to the NRC, for review and approval, that demonstrates that the waste packages and waste canisters are qualified for repository operations, either through a new analysis, or in demonstrating that the waste package and canister designs are enveloped by the PCSA (SER Sections 2.1.1.7.3.9.1 and 2.1.1.7.3.9.3.3).

The NRC staff also finds, with reasonable assurance, that DOE provided adequate description and discussion of the design of the SSCs ITS for the surface and subsurface GROA for (i) materials of construction of the GROA (including geologic media, general arrangement, and approximate dimensions), and codes and standards that DOE proposed to apply to the design and construction of the GROA; (ii) dimensions, material properties, specifications, analytical and design methods used along with any applicable codes and standards; (iii) design criteria used and their relationships to the preclosure and postclosure performance objectives for protection against radiation exposures and releases of radioactive material, numerical guides for design objectives, and identification of the design bases and their relation to the design criteria; and (iv) explosion and fire detection systems and appropriate suppression systems.

## 2.8 Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences

SER Section 2.1.1.8 provides the NRC staff's review of DOE's descriptions of its as low as is reasonably achievable (ALARA) program and the Operational Radiation Protection Program (RPP). The NRC staff focused its review on DOE's descriptions of the ALARA policy, design, and operational work practices for the GROA, relied upon to reduce doses to members of the public and occupational doses to workers with (i) the policy considerations, including its management commitment to maintain doses ALARA and the implementation of ALARA principles in the design process throughout the repository design and construction; (ii) the facility shielding design used to meet the ALARA requirements for normal operations and Category 1 event sequences; and (iii) the implementation of the ALARA principles into repository operations, including administrative controls to maintain doses ALARA and general operational guidelines through its Operational RPP.

The NRC staff reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(6) and 10 CFR 63.111(a)(1) are satisfied. Based on the information provided, the NRC staff has reasonable assurance that DOE will implement an RPP that will maintain occupational doses and public exposures below the applicable limits of 10 CFR Part 20. The operations at the GROA, through permanent closure, will comply with the ALARA requirements in 10 CFR Part 20.

## 2.9 Plans for Retrieval and Alternate Storage of Radioactive Wastes

SER Section 2.1.1.9 provides the NRC staff's review of DOE's description of its retrieval plan and alternate storage should retrieval become necessary. The NRC staff focused its review on DOE's waste retrieval plan to determine whether (i) the waste packages could be retrieved during the period of potential waste retrieval by reversing the operational procedure for waste emplacement, (ii) DOE identified a reasonable range of potential problems (off-normal scenarios) during retrieval, and (iii) DOE described approaches for restoring access to waste packages from potential off-normal conditions without physical damage or overheating of the affected waste packages.

The NRC staff reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(7) and 10 CFR 63.111(e) are satisfied because (i) DOE adequately described its plans for retrieval and provided details of the geologic repository operations area design that preserves the option to retrieve any or all of the emplaced waste; (ii) radiation safety, including implementation of ALARA principles, is built into the retrieval concepts; (iii) alternate storage sites of sufficient capacity are identified; and (iv) a reasonable schedule for a potential retrieval scenario is provided.

## 2.10 Plans for Permanent Closure and Decontamination or Decontamination and Dismantlement of Facilities

SER Section 2.1.3 provides the NRC staff's review of DOE's GROA design considerations and its plans to facilitate permanent closure and decontamination or the decontamination and dismantlement (PCDDD) of the GROA surface facilities. The NRC staff focused its review on DOE's information regarding the design considerations to facilitate PCDDD and its plans for the decontamination and dismantlement of repository surface facilities in the GROA.

The NRC staff reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(8) are satisfied because the applicant's plan describes the functions of design considerations that will facilitate permanent closure and decontamination or decontamination and dismantlement of surface facilities. The NRC staff also finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(22)(vi) are satisfied because the applicant has provided adequate plans for permanent closure and decontamination or decontamination and dismantlement of surface facilities.

# 3.0 Conclusions

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the U.S. Department of Energy's ("DOE" or the "applicant") Safety Analysis Report (SAR), Chapter 1: Repository Safety Before Permanent Closure and the other information submitted in support of its license application and has found that DOE submitted applicable information required by 10 CFR 63.21. The NRC staff has also found, with reasonable assurance, that subject to proposed conditions of construction authorization, DOE's design of the proposed geologic repository operations area (GROA) and preclosure safety analysis complies with the preclosure performance objectives at 10 CFR 63.111 and the requirements for preclosure safety analysis of the GROA at 10 CFR 63.112.

# 4.0 References

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# ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AC | alternating current |
| AF | aging facility |
| AFE | annual frequency of exceedance |
| ALARA | as low as is reasonably achievable |
| ANSI/ANS | American National Standards Institute/American Nuclear Society |
| AO | aging overpack |
| APE | annual probability of exceedance |
| ASD | adjustable speed drive |
| ASHRAE | American Society of Heating, Refrigerating, and Air Conditioning Engineers |
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society of Testing and Materials |
| ATHEANA | A Technique for Human Event Analysis |
| AWS | American Welding Society |
| BDBGM | beyond design basis ground motion |
| BWR | boiling water reactor |
| CCC | Center Control Center |
| CCCF | central control center facility |
| CDFM | conservative deterministic failure margin |
| CHC | cask handling crane |
| COF | coefficient of friction |
| CRCF | Canister Receipt and Closure Facility |
| CSNF | commercial spent nuclear fuel |
| CTCTT | cask tractor and cask transfer trailer |
| CTM | canister transfer machine |
| CTT | canister transfer trolley |
| DBGM | design basis ground motion |
| D/C | demand-to-capacity |
| DC | direct current |
| DCMIS | Digital Control Management Information Systems |
| DCP | Design Control Parameter |
| DIPA | double-interlock preaction |
| DOE | U.S. Department of Energy |
| DPC | dual purpose canister |
| DSEG | drip shield emplacement gantry |
| EBS | engineered barrier system |
| EC | electric combat |
| ECRB | enhanced characterization of the repository block |
| EDGF | emergency diesel generator facility |
| EPA | U.S. Environmental Protection Agency |
| EPRI | Electric Power Research Institute |
| EPS | emergency power systems |
| EPS | effective plastic strain |
| ESD | event sequence diagram |
| ESF | exploratory studies facility |
| ETF | Expended toughness fraction |

## ACRONYMS AND ABBREVIATIONS (continued)

| | |
|---|---|
| FAA | Federal Aviation Administration |
| FDH | fault displacement hazard |
| FE | finite element |
| GPS | Global Positioning Satellite |
| GROA | geologic repository operations area |
| HAZOP | hazard and operability |
| HCLPF | high confidence of low probability of failure |
| HEPA | high efficiency particulate air |
| HFE | human failure events |
| HLW | high-level radioactive waste |
| HLWRS | High-Level Waste Repository Safety |
| HMI | human–machine interface |
| HRA | human reliability analysis |
| HVAC | heating, ventilation, and air conditioning |
| IAEA | International Atomic Energy Agency |
| I&C | instrumentation and control |
| ICRP | International Commission on Radiological Protection |
| IEC | International Electrotechnical Commission |
| IEEE | Institute of Electrical and Electronics Engineers |
| IHF | Initial Handling Facility |
| ISG | interim staff guidance |
| ITS | important to safety |
| ITWI | important to waste isolation |
| JASPER | Joint Actinide Shock Physics Experimental Research |
| LA | license application |
| L/D | length-to-diameter |
| LATN | low altitude training and navigation |
| LLW | low-level radioactive waste |
| LLWF | low-level radioactive waste facility |
| LOSP | loss of offsite power |
| LPFs | leak path factors |
| MAPE | mean annual probability of exceedance |
| MCC | motor control centers |
| MCO | multicanister overpacks |
| MLD | master logic diagram |
| MOAs | military operations areas |
| MOX | mixed oxide |
| MRVs | multipurpose recovery vehicle |
| MTHM | metric tons of heavy metal |
| NARA | Nuclear Action Reliability Assessment |
| NASA | National Aeronautics and Space Administration |
| NEMA | National Electrical Manufacturers Association |
| NFPA | National Fire Protection Association |
| NNSS | Nevada National Security Site |
| NOAA | National Oceanic and Atmospheric Administration |
| non-ITS | not-important to safety |
| NRC | U.S. Nuclear Regulatory Commission |

## ACRONYMS AND ABBREVIATIONS (continued)

| | |
|---|---|
| NTS | Nevada Test Site |
| NTTR | Nevada Test and Training Range |
| OCB | outer corrosion barrier |
| ORPP | Operational Radiation Protection Plan |
| P&I | piping and instrumentation |
| P&IDs | piping and instrumentation diagrams |
| PCDDD | permanent closure and decontamination or for the decontamination and dismantlement |
| PCSA | preclosure safety analysis |
| PEFA | Passive equipment failure analyses |
| PFD | process flow diagrams |
| PFDHA | probabilistic fault displacement hazard analysis |
| PGA | peak ground acceleration |
| PGV | peak ground velocity |
| PLCs | programmable logic controllers |
| PMF | probable maximum flood |
| PMP | probable maximum precipitation |
| PRA | probabilistic risk analysis |
| PSCA | preclosure safety analysis |
| PSC | procedural safety control |
| PSHA | probabilistic seismic hazard analysis |
| PVHA | probabilistic volcanic hazard analysis |
| PWR | pressurized water reactor |
| QA | quality assurance |
| RAI | request for additional information |
| RF | Receipt Facility |
| RHH | repository host horizon |
| RMS | radiation/radiological monitoring systems |
| ROA | range of applicability |
| ROVs | remotely operated vehicle |
| RPCS | radiation protection and criticality safety |
| RPP | Radiation Protection Program |
| RVT | random vibration theory |
| SAR | Safety Analysis Report |
| SASW | spectral analysis of the surface wave |
| SCs | safety controls |
| SER | Safety Evaluation Report |
| SFPE | Society of Fire Protection Engineers |
| SFTM | spent fuel transfer machine |
| SNF | spent nuclear fuel |
| SONET | Synchronous Optical NETwork |
| SPM | site prime mover |
| SSCs | structures, systems, and components |
| SSHAC | Senior Seismic Hazard Assessment Committee |
| SSI | soil-structure interaction |
| STC | shielded transfer cask |

## ACRONYMS AND ABBREVIATIONS (continued)

| | |
|---|---|
| TAD | transportation, aging, and disposal |
| TEDE | total effective dose equivalent |
| TEV | transport and emplacement vehicle |
| TNT | Trinitrotoluene |
| TSPA | total system performance assessment |
| UHS | uniform hazard spectras |
| UL | Underwriter Laboratories |
| UPS | uninterruptible power supply |
| USL | upper subcritical limit |
| $V_p$ | compression wave velocity |
| $V_s$ | shear wave velocity |
| WHF | Wet Handling Facility |
| WP | waste package |
| WPTT | waste package transfer trolley |
| X/Q | atmospheric dispersion coefficients |
| YMRP | Yucca Mountain Review Plan |

# INTRODUCTION

Volume 2, *"Repository Safety Before Permanent Closure,"* of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Safety Analysis Report (SAR) the U.S. Department of Energy (DOE) provided in its June 3, 2008, license application (LA) submittal (DOE, 2008ab), as updated on February 19, 2009 (DOE, 2009av). The NRC staff also reviewed information DOE provided in response to the NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, this SER Volume 2 documents the results of the NRC staff's evaluation to determine whether the design of the proposed geologic repository operations area (GROA) for Yucca Mountain complies with the performance objectives and requirements that apply before the repository is permanently closed. These performance objectives and requirements can be found in NRC's regulations at 10 CFR Part 63, Subparts E and K.

Other portions of the NRC staff's safety review are documented in other SER volumes. SER Volume 1, NUREG–1949 (NRC, 2010aa) documents the results of the NRC staff's review of DOE's General Information. SER Volume 3 documents the results of the NRC staff's review and evaluation of the proposed repository design's compliance with the performance objectives and requirements that apply after the repository is permanently closed. SER Volume 4 documents the results of the NRC staff's review and evaluation of DOE's demonstration of compliance with administrative and programmatic requirements. SER Volume 5 documents the NRC staff's proposed conditions of construction authorization, and review and evaluation of probable subjects of license specifications.

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. Pursuant to 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, a period of operations, and termination of the license. The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The period of operations includes (i) the time during which emplacement would occur; (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable; and (iii) permanent closure. In addition, 10 CFR Part 63 represents a risk-informed, performance-based regulatory approach to the review of geological disposal. This risk-informed, performance-based regulatory approach uses risk insights, engineering analysis and judgments, performance history, and other information to focus on the most important activities and to focus the NRC staff's review on areas most significant to safety and performance. In conducting its review, the NRC staff was guided by the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) and the relevant supplements to the YMRP (i.e., ISG–01, "Review Methodology for Seismically Initiated Event Sequences;" ISG–02, "Preclosure Safety Analysis–Level of Information and Reliability Estimation;" ISG-03, "Preclosure Safety Analysis–Dose Performance Objectives and Radiation Protection Program;" and ISG-04, "Preclosure Safety Analysis–Human Reliability Analysis").

**Preclosure Performance Objectives and Requirements**

In its review of DOE's application, the NRC staff used a risk-informed and performance based review process and considered, among other things, whether the site and design comply with the performance objectives and requirements contained in 10 CFR Part 63, Subparts E and K.

In accordance with 10 CFR 63.21, the applicant must include in its SAR a preclosure safety analysis (PCSA). As described in 10 CFR 63.102(f), the PCSA identifies and categorizes event sequences, and identifies structures, systems, and components (SSCs) important to safety (ITS) and associated design bases and criteria. The PCSA is part of the risk-informed and performance-based review, which is described further in the following section. An event sequence, as defined in 10 CFR 63.2, is a series of actions and/or occurrences within the natural and engineered components of the facility that could potentially expose individuals to radiation. The applicant's PCSA must demonstrate that the repository, as proposed to be designed, constructed, and operated, will meet the specified radiological dose limits throughout the preclosure period. The applicant must also demonstrate that the GROA design will not preclude retrievability of the wastes, in whole or in part, from the underground facility where these wastes will be emplaced for permanent disposal (10 CFR 63.111).

**Risk-Informed, Performance-Based Review**

The PCSA quantifies GROA performance and is used to demonstrate compliance with the preclosure performance requirements in 10 CFR 63.111. The NRC staff evaluated DOE's PCSA using a risk-informed and performance-based review. A PCSA is a systematic analysis that answers three basic questions that are used to define risk: What can happen? How likely is it to happen? What are the resulting consequences? The applicant's PSCA includes a number of evaluations, such as identification of hazards and initiating event sequences; development and categorization of event sequences; failure mode and reliability assessments of SSCs; and SSCs' fragility assessments. Because the PCSA encompasses a broad range of technical subjects, the NRC staff used risk information throughout the review process to ensure that the NRC staff's review focused on significant items that could affect preclosure performance. YMRP Section 2.1.1 provides guidance to the NRC staff on how to apply risk information throughout its review of the applicant's PCSA.

**Recent Events**

Recent events in nuclear operations are also considered in the NRC staff's review of the DOE license application. The first is the March 11, 2011 earthquake and subsequent tsunami that resulted in extensive damage to the six-unit Fukushima Dai-ichi nuclear power plant in Japan. Following this event, the NRC took numerous actions in evaluating the event, and to prevent against such accidents occurring at U.S. nuclear power plants. In its review of the Yucca Mountain license application, the NRC staff considered insights from the NRC's analysis of the Fukushima Dai-ichi events as they may affect analyses of natural hazards and spent fuel handling operations at the proposed repository at Yucca Mountain, although the staff notes that there are significant differences between the proposed operations at the repository and a nuclear power plant. Most notably, the wet handling facility proposed for Yucca Mountain is not a storage pool and has a capacity necessary only to accommodate the loading of waste packages. In addition, the spent nuclear fuel proposed to be sent to Yucca Mountain would have experienced years of cooling prior to shipment, in contrast to spent fuel at reactors where the thermal demands can be much greater (i.e., more spent fuel and more recently discharged spent fuel, such was the case at Fukushima Dai-ichi). Thus, the NRC staff determined that many of the insights from the Fukushima accident do not directly impact the NRC staff's review of the wet handling facility at Yucca Mountain. However, consistent with insights gained from evaluating the Fukushima accident, the NRC staff evaluated the seismic hazard at Yucca Mountain based on new information developed since the application was submitted (SER Section 2.1.1.1.3.5.2.1). Additionally, the NRC staff's review considers other aspects of the proposed repository that relate to issues at Fukushima (e.g., the long term loss of electrical power (SER Section 2.1.1.3.3.1.3.5), emergency preparedness (SER Section 2.5.7), and the assessment of external hazards (SER Section 2.1.1.3.3.1).

The second set of events the NRC staff considered relates to two recent accidents at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, NM. These accidents include a salt haul truck fire in the underground facility on February 5, 2014, and the breach of at least one transuranic (TRU) waste container in the underground facility on February 14, 2014, which resulted in the release of a small amount radioactive material from the subsurface to the environment. As a result of these accidents, waste disposal operations at WIPP have been suspended. DOE is currently pursuing a recovery plan to safely resume waste emplacement at WIPP in the first quarter of calendar year 2016 (DOE, 2014ab). Although both WIPP and the proposed facility at Yucca Mountain represent geological repositories, there are significant differences between the two facilities. Principally, the differences in the wastes that are managed at WIPP (TRU waste), compared to those proposed for the Yucca Mountain repository (spent nuclear fuel and high-level waste), robustness of the waste packages (e.g., the waste package designs for Yucca Mountain have a wall thickness of 2.54 cm [1.0 in.] thick for the outer barrier and 5.08 cm [2.0 in.] for the inner barrier, which is more resilent to potential challenges to the structural integrity of the waste package than the steel drums used for the radioactive wastes at WIPP), as well as operational procedures. Although specific details of the recent events at WIPP do not inpact the NRC staff review of the Yucca Mountain license application, the NRC staff review did consider issues related to the WIPP events, such as: potential fire hazards in the subsurface (SER Section 2.1.1.4.3.2.1.3), equipment and facility design DOE will use to monitor and control dispersal of radioactive contamination (SER Section 2.1.1.6.3.2.4); DOE's operational plans, including maintenance, surveillance, and periodic testing (SER Section 2.5.6); and DOE's personnel qualifications and training requirements (SER Section 2.5.3.3).

**Review of the Applicant's Preclosure Safety Analysis**

The NRC staff's review of the applicant's PCSA included the site and design and the potential hazards, initiating events and event sequences (e.g., earthquake, aircraft crash, operational hazards, and human errors) and the potential radiological safety consequences. The following describes the NRC staff's review process in evaluating compliance with 10 CFR Part 63 preclosure requirements.

The NRC staff's review evaluated whether the applicant's PCSA contains sufficient information to satisfy applicable regulatory requirements, and whether the PCSA demonstrates that the repository would meet all performance objectives for the GROA through permanent closure.

In SER Section 2.1.1.1, the NRC staff evaluated the applicant's site description information to identify natural and human-induced hazards, focusing on those features, events, and processes that might affect the GROA design and preclosure safety. Next, the NRC staff evaluated the sufficiency of the applicant's descriptions of GROA surface and subsurface facilities to evaluate the applicant's PCSA and GROA design. This included evaluations of the applicant's description of SSCs, safety controls, equipment, and operational activities. In these evaluations, found in SER Section 2.1.1.2, the NRC staff focused on risk-significant operations, processes, and SSCs involving radioactive waste handling.

The NRC staff then evaluated DOE's identification of hazards and initiating events that could lead to an event sequence at repository facilities during the preclosure period in SER Section 2.1.1.3. The NRC staff's review of the applicant's identification of hazards and initiating events began with a systematic examination of the site, the design of the facilities, and the operations to be conducted at these facilities. This evaluation assessed the probability of the potential hazards, taking into account a range of uncertainties associated with data that support the applicant's probability estimations. The estimated probability of the initiating events was then used by NRC staff to analyze associated event sequences. Based on the identification of hazards and initiating events, the NRC staff evaluated DOE's information on identification of event sequences, which included initiating events and associated combinations of repository SSC failures (including human errors) that could potentially lead to the exposure of individuals to radiation. The NRC staff evaluated DOE's technical basis for developing, quantifying, and categorizing event sequences in SER Section 2.1.1.4.

The NRC staff also evaluated the consequence analysis the applicant conducted to support its PCSA in SER Section 2.1.1.5. The NRC staff reviewed the applicant's dose calculation methodology, atmospheric dispersion determination, assumptions and input parameters, source terms, and methodology for worker and public dose determinations.

The NRC staff then evaluated DOE's identification of important to safety (ITS) SSCs and procedural safety controls for reducing event sequences or mitigating dose consequences. This included evaluating criteria the applicant developed for identification of ITS SSCs and procedural safety controls, as well as the applicant's nuclear safety design bases for the ITS SSCs from the PCSA event sequence analyses. The NRC staff's review focused on how the applicant proposed to ensure the availability and ability of ITS SSCs to perform their intended safety function. This evaluation can be found in SER Section 2.1.1.6.

Evaluations of ITS SSC design, construction, and operation included (i) information relative to the codes and standards for design and construction of the GROA, (ii) design methodologies, (iii) design bases and design criteria, and (iv) design and design analysis. This evaluation is found in SER Section 2.1.1.7. The NRC staff also evaluated the consistency between the applicant's proposed design criteria and the GROA performance objectives in this section.

The NRC staff evaluated the applicant's Radiation Protection Program (RPP) to confirm that it would ensure compliance with applicable dose limits, as well as that the program would ensure that all doses are as low as is reasonably achievable (ALARA). The NRC staff also evaluated the facility shielding design for both normal operations and during event sequences. These evaluations can be found in SER Section 2.1.1.8.

**Plans for Retrieval and Alternate Storage of Radioactive Wastes**

NRC regulations at 10 CFR 63.21(c) require that the applicant include in its SAR a description of plans for retrieval and alternate storage of the radioactive wastes should retrieval become necessary. Section 63.111(e) requires that the applicant design its GROA to preserve the option of waste retrieval. The NRC staff evaluated the applicant's description of its retrieval plan in SER Section 2.1.2. The NRC staff also evaluated whether the GROA has been designed to preserve the option to retrieve any or all of the emplaced waste on a reasonable schedule. The applicant's description of an alternate storage plan that identifies a proposed alternate storage site, including the location, size, and storage operations, is also evaluated in this section.

**Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities**

NRC regulations at 10 CFR 63.21(c)(22)(vi) require the applicant to provide information concerning plans for permanent closure and plans for the decontamination or decontamination and dismantlement (PCDDD) of the surface facilities. NRC regulations further require the applicant to describe design considerations that facilitate the PCDDD of surface facilities. In SER Section 2.1.3, the NRC staff evaluated the applicant's design with respect to facilitating PCDDD, including facility history, dose modeling, facility radiological status, alternatives for decommissioning, ALARA, planned decommissioning activities, project management and organization, the health and safety program for PCDDD, an environmental monitoring and control program, the radioactive waste management program, radiation surveys, and the quality assurance program.

**Sections of the Preclosure Review**:

The individual sections documenting the NRC staff's review are

1. Site Description as It Pertains to Preclosure Safety Analysis (SER Section 2.1.1.1)

2. Description of Structures, Systems, Components, and Operational Activities as it Pertains to Preclosure Safety Analysis (SER Section 2.1.1.2)

3. Identification of Hazards and Initiating Events (SER Section 2.1.1.3)

4. Identification of Event Sequences (SER Section 2.1.1.4)

5. Consequence Analysis (SER Section 2.1.1.5)

6. Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems (SER Section 2.1.1.6)

7. Design of Structures, Systems, and Components Important to Safety and Safety Controls (SER Section 2.1.1.7)

8. Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences (SER Section 2.1.1.8)

9. Plans for Retrieval and Alternate Storage of Radioactive Wastes (SER Section 2.1.2)

10. Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities (SER Section 2.1.3)

**References**

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NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: U.S. Nuclear Regulatory Commission.

# CHAPTER 1

## 2.1.1.1 Site Description As It Pertains to Preclosure Safety Analysis

### 2.1.1.1.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.1 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the Department of Energy's (DOE or the applicant) Yucca Mountain site description as it pertains to DOE's preclosure safety analysis (PCSA) and design of the geologic repository operations area (GROA). PCSA is defined in 10 CFR 63.2 as a systematic examination of the site; the design; and the potential hazards, initiating events and event sequences and their consequences (e.g., radiological exposures to workers and the public). The applicant plans for a period of operations, also referred to as the preclosure period, of approximately 100 years (SAR Section 1.3.1). The preclosure period would consist of 50 years of waste emplacement, including an initial 24-year period of concurrent repository development and 50 years of postemplacement monitoring. The regulations in 10 CFR 63.112(b) require that the PCSA include an identification and systematic analysis of naturally occurring and human-induced hazards at the GROA, including a comprehensive identification of potential event sequences. The regulations in 10 CFR 63.112(d) require that the PCSA must also include the technical basis for either inclusion or exclusion of specific naturally occurring and human-induced hazards in the PCSA. The site description information contained in this SER chapter informs the NRC staff's evaluation of the applicant's assessment of initiating events from natural hazards in the PCSA contained in SER Section 2.1.1.3. In addition, many of the requirements for the structural design capacity of important to safety (ITS) structures, systems, and components (SSCs) are based on site characteristics evaluated in this SER chapter, such as the types of soils on which facilities would be built, meteorological conditions, or dynamic loads from seismic events. This design capacity information is evaluated in Section 2.1.1.7.3.1.1 of this SER.

DOE provided site characterization information in Safety Analysis Report (SAR) Section 1.1 (DOE, 2008ab) and in response to the NRC staff's requests for additional information (RAIs) (DOE, 2009ab,ap–au,bf,bg,eh–ej), which the NRC staff also evaluates in this section. In SAR Section 1.1, the applicant provided this information in a format that generally followed the NRC's staff's Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), Section 2.1.1.1, which is applicable to site information for the preclosure safety analysis, as shown in the Table on SAR Page 1.1-1. The SAR includes sections on Site Geography (Section 1.1.1); Regional Demography (Section 1.1.2); Local Meteorology and Regional Climatology (Section 1.1.3); Regional and Local Surface and Ground Water Hydrology (Section 1.1.4); Site Geology and Seismology (Section 1.1.5); Igneous Activity (Section 1.1.6); Site Geomorphology (Section 1.1.7); Geochemistry (Section 1.1.8); and Land Use, Structures and Facilities, and Residual Radioactivity (Section 1.1.9). In the SAR, the applicant included detailed information on (i) the site's natural features, including surface outcrops and subsurface bedrock; (ii) sediments and soils; (iii) rock fractures and faults; (iv) landforms; (v) surface and groundwater quantities and flow processes; (vi) chemistry and geochemistry of the rocks and water; (vii) earthquakes and active faulting; (viii) volcanic hazards; (ix) climatic history and weather conditions; (x) topography and land-use boundaries; (xi) current population and future population trends; and (xii) natural and man-made sources of radiation.

#### 2.1.1.1.2 Regulatory Requirements

The regulatory requirements applicable to the preclosure site description are in 10 CFR 63.21(c)(1) and 10 CFR 63.112(c).

The regulations in 10 CFR 63.21(c)(1) require that the SAR include a description of the Yucca Mountain site, with appropriate attention to those features, events, and processes of the site that might affect the design of the GROA and performance of the geologic repository. The description of the site must include information regarding features, events, and processes outside of the site to the extent the information is relevant and material to the safety or performance of the geologic repository. This information must include (i) the location of the GROA with respect to the boundary of the site; (ii) information regarding the geology, hydrology, and geochemistry of the site, including geomechanical properties and conditions of the host rock; and (iii) information regarding surface water hydrology, climatology, and meteorology of the site.

The regulations in 10 CFR 63.112(c) require that the preclosure safety analysis of the GROA include data pertaining to the Yucca Mountain site and the surrounding region, to the extent necessary, used to identify naturally occurring and human-induced hazards at the GROA.

The NRC staff reviewed the applicant's site information using the guidance and acceptance criteria identified in YMRP Section 2.1.1.1 (NRC, 2003aa). The acceptance criteria are as follows:

- The license application contains a description of the site geography adequate to permit evaluation of the PCSA and the GROA design.
- The license application contains a description of the regional demography adequate to permit evaluation of the PCSA and the GROA design.
- The license application contains a description of the local meteorology and regional climatology adequate to permit evaluation of the PCSA and the GROA design.
- The license application contains sufficient local and regional hydrological information to support evaluation of the PCSA and the GROA design.
- The license application contains descriptions of the site geology and seismology adequate to permit evaluation of the PCSA and the GROA design.
- The license application contains descriptions of the historical regional igneous activity adequate to permit evaluation of the PCSA and the GROA design.
- The license application provides analysis of site geomorphology adequate to permit evaluation of the PCSA and the GROA design.
- The license application contains site-sufficient geochemical information to support evaluation of the PCSA and the GROA design.
- The license application contains adequate evaluations of previous land use, impacts on existing structures and facilities, and the potential for exposures from residual radiation.

The NRC staff considered these acceptance criteria in its review of information provided by DOE. The staff focused its review on those aspects of the site description that could substantively affect the preclosure safety assessment, as determined by the NRC staff, and as discussed in detail in this section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses, direct observations of the physical and geological environment of the Yucca Mountain site, and numerous field surveys and other site activities.

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA. The applicability of such NRC guidance is discussed in greater detail in the sections where they were used as part of the application or the NRC staff's review.

### 2.1.1.1.3 Technical Review

The NRC staff organized its evaluation of the applicant's site description following YMRP, Section 2.1.1.1, which parallels the applicant's organization in SAR Sections 1.1.1 through 1.1.8. The NRC staff focused its review on the adequacy of the applicant's characterization of site information to (i) ensure that natural and human-induced hazards, which may initiate or be part of event sequences that impact the GROA, are sufficiently characterized for use in the PCSA and (ii) ensure that this information is appropriately used in the design of the GROA where engineered features can prevent or mitigate the effects or impacts from hazards. The review also provides the bases for the NRC staff's more detailed evaluation in later sections of this SER Volume 2, where the design of GROA operational facilities, systems, structures, and components are examined in greater detail.

In accordance with 10 CFR 63.21(a), DOE must submit an application that is "as complete as possible in the light of information that is reasonably available at the time of docketing." The NRC staff conclusions in this SER section are primarily based on the information that DOE provided in the SAR and in response to the NRC staff's RAIs. Because of the lapse in time between the license application submittal and the issuance of this SER volume, some information in the application does not reflect current circumstances. Site characterization information in the SAR was not updated. While many site characteristics (such as geology, geochemistry, or meteorology) are steady-state phenomena, which are unlikely to have changed significantly in the intervening time period, characteristics such as land use, existing structures, land ownership, and nearby facilities are not steady-state. Because DOE did not provide site characterization updates beyond those cited in this evaluation, the NRC staff proposes a condition of construction authorization. This condition of construction authorization requires DOE to confirm that its site characterization information and related analyses in the SAR submitted in accordance with 10 CFR 63.21(c)(1) continue to be accurate with respect to (i) site boundaries; (ii) man-made features; (iii) previous land use; (iv) existing structures and facilities; and (v) potential exposure to residual radioactivity. The NRC staff considers 90 days a reasonable amount of time for the applicant to confirm these items. The basis for this proposed license condition is documented throughout the NRC staff's evaluation in this section.

**Proposed Condition of Construction Authorization**

Within 90 days of issuance of construction authorization, DOE must confirm that its site characterization information and related analyses in the SAR submitted in accordance with

10 CFR 63.21(c)(1) continue to be accurate with respect to (i) site boundaries; (ii) man-made features; (iii) previous land use; (iv) existing structures and facilities; and (v) potential exposure to residual radioactivity. DOE must provide to the NRC written notification when its confirmatory analysis is complete. This notification must include, for NRC staff's verification, a copy of DOE's confirmatory analysis.

#### 2.1.1.1.3.1 Site Geography

In SAR Section 1.1.1, the applicant provided site geographic information to describe the location of the GROA, including the site boundary and prominent natural features that may be significant to the evaluations in the PCSA and to the design of the GROA. Locations and activities of man-made features that existed outside the controlled area at the time of the license application are also identified and described, including federal and military facilities, civilian and military airports, roads, railroads, and potentially hazardous commercial operations and manufacturing centers.

The planned man-made features to be constructed within the controlled area, which could potentially be important in the PCSA evaluations the same way as existing facilities, are described in SAR Sections 1.2 through 1.4, and evaluated as part of the NRC staff's PCSA and design reviews in SER Sections 2.1.1.2, 2.1.1.3, 2.1.1.4, and 2.1.1.7.

**Repository Boundaries**

SAR Figures 1.1-1 through 1.1-6 depict the proposed site boundary, preclosure controlled area, general environment, and location of the GROA at the time of license application. The repository would be located in Nye County, Nevada. The site boundary of the preclosure controlled area (also known as "proposed land withdrawal area") is the area that the DOE would control. The applicant also described the general environment and the protected and restricted areas of the GROA in SAR Section 1.1.1.1. In response to the NRC staff's RAIs regarding specific descriptions of the boundaries of the GROA and the preclosure controlled area, the applicant provided a description of the boundaries of the preclosure controlled area and the GROA using Public Land Survey System nomenclature (i.e., township, range, and section), as described in DOE (2009au, Enclosures 5 and 6).

In DOE Responses 7 and 8 (2009au), DOE addressed an RAI from the NRC staff concerning previous land use (see SER Section 2.1.1.1.3.9) impacts on the GROA site location and boundaries, as shown on SAR Figures 1.1-1 and 1.1-6. The applicant provided information to show that approximately 0.75 $km^2$ [182.5 acres] of patented mining claim (Patent 27-83-0002) area is private land excluded from the proposed land withdrawal area and is not part of the area over which DOE would control access. Additionally, DOE stated that it will update its controlled area boundary in SAR Figures GI 1-2 and 1-4 and SAR Figures 1.1-1 and 1.1-6 to reflect this information.

DOE identified three controlled access points to the surface GROA. The layout and the surface GROA for each of the planned phases of development are depicted in SAR Figures 1.1-2 and 1.1-3.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the proposed repository boundaries in SAR Section 1.1.1.1, references therein, and responses to RAIs (DOE 2009au, Enclosures 5, 6, 7, and 8). The NRC staff finds that DOE maps and information are acceptable, based on the NRC staff's review of the application and frequent visits to the Yucca Mountain site. SAR Figures 1.1-1, 1.1-2, and 1.1-3 delineate the site boundaries and GROA and are of sufficient detail and scale to permit review of the site boundary and the preclosure controlled area, access points, the general environment, and provide a detailed representation of the surface GROA, including its phased development. The NRC staff also compared the description of the GROA provided in DOE Enclosures 5 and 6 (2009au) with SAR Figures 1.1-1, 1.1-4, and 1.1-6, and found the boundaries as described in these SAR figures to be consistent with each other and with the NRC staff's understanding, based on its evaluations and site visits.

The NRC staff finds that DOE provided adequate information in its application and RAI response (DOE, 2009au) describing the site boundary, GROA, and the location of the land withdrawal area at the time of application. The NRC staff finds the maps and descriptions acceptable and adequate for use in the PCSA and to support the GROA design. Because DOE did not provide site characterization updates regarding land use beyond those cited in this evaluation, the NRC staff proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3. This condition of construction authorization would require DOE to confirm that site boundary information and related analyses in the SAR continue to be accurate.

**Natural Features**

In SAR Section 1.1.1.2, the applicant described the natural features within the preclosure controlled area, which is shown in SAR Figure 1.1-4. Prominent natural features, including the topography, stream channels, washes, and basin drainage in the vicinity of the GROA are described and are shown in SAR Figure 1.1-5. Using information from the Final Environmental Impact Statement (DOE, 2002aa), associated supplements, and references therein, the applicant concluded that there are no perennial or natural surface water features, including wetlands, on the Yucca Mountain site.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's descriptions of the natural features in SAR Section 1.1.1.2 and references therein. The NRC staff used publicly available maps (Carr, et al., 1996aa; Day, et al., 1998aa,ab; Potter, et al., 2002aa; Slate, et al., 1999aa; U.S. Geological Survey, 1961aa,ab,ac), satellite images, and first-hand experience obtained from NRC staff field investigations of the Yucca Mountain site to evaluate the applicant's information regarding site natural features. The NRC staff finds the maps shown in SAR Figures 1.1-4 and 1.1-5 used to depict this information are of appropriate scale and detail to permit evaluation of the site topography and surface water drainage patterns. Based on this information, the NRC staff finds the applicant's conclusion acceptable that there are no perennial or natural surface water features at Yucca Mountain. Therefore, the NRC staff concludes that the descriptions of natural features, particularly the surface water features, are adequate and acceptable to permit evaluation of these features in the PCSA and to support the GROA design.

**Man-Made Features**

In SAR Section 1.1.1.3, the applicant described the existing man-made features and facilities located outside the Yucca Mountain site and, in particular, within the abutting Nevada National Security Site (NNSS; formerly the Nevada Test Site, or NTS) to the east at the time of license application. These are depicted on maps in SAR Figures 1.1-6 through 1.1-10. The description included information regarding the use and construction of features and facilities that the applicant found may be significant for the review of the PCSA and GROA design, including the following: airspace and related facilities and activity; military, federal, and civilian airports and airfields; primary roads; potentially hazardous commercial operations and manufacturing centers; and electric power transmission lines. Additionally, the applicant used information included in the evaluation of hazard-initiating events due to industrial/military events (BSC, 2008an) on the Nevada Test and Training Range Chart (National Imagery and Mapping Agency, 2001aa), to conclude that there are no active, commercial passenger, or freight railroad lines within 32 km [20 mi] of the surface GROA.

**NRC Staff's Evaluation**

The NRC staff reviewed the descriptions of man-made features outside the Yucca Mountain repository site provided in SAR Section 1.1.1.3 and references therein. The NRC staff reviewed publicly available maps and satellite images of the site (National Imagery and Mapping Agency, 2001aa; U.S. Department of Transportation, 2009aa; U.S. Geological Survey, 1961aa,ab,ac) to independently evaluate the applicant's information. The NRC staff found that information DOE depicted in maps in SAR Figures 1.1-6 through 1.1-10 is adequate because the figures are consistent with publicly available information and are of sufficient detail to permit evaluation of the location and potential impacts of man-made features and facilities. The NRC staff independently evaluated the location of NNSS facilities indicated in SAR Figure 1.1-6 by comparing the applicant's information using the maps listed above. The NRC staff's complete evaluation of Industrial and Military Activity-Related Hazards is discussed in SER Section 2.1.1.3.3.1.3.4. The NRC staff finds that the information on railroad lines is sufficient, based on comparisons to the publicly available maps discussed above. Therefore, the NRC staff concludes that the descriptions of man-made features, particularly the facilities located at the adjacent NNSS, are adequate and acceptable to permit evaluation of these features in the PCSA and to support the GROA design. Because DOE did not provide site characterization updates regarding man-made features beyond those cited in this evaluation, the NRC staff proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3. This condition of construction authorization would require DOE to confirm that the characterization of man-made features and related analyses in the SAR continue to be accurate.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant adequately described information regarding Yucca Mountain site geography, including site boundaries, location of natural and man-made features, and relevant facilities outside the GROA. The NRC staff finds the maps provided by the applicant are of appropriate scale and detail to permit this evaluation and finds the descriptions to be complete and accurate. Therefore, the NRC staff finds, with reasonable assurance, that the applicant's information regarding pre-closure safety site geography is acceptable for use in the evaluations in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(i) and 10 CFR 63.112(c). The NRC staff also proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3, that would require DOE to confirm that its information on site boundaries and man-made features and related analyses in the SAR continue to be accurate.

### 2.1.1.1.3.2 Regional Demography

The applicant described the regional demography in SAR Section 1.1.2. DOE used this information to determine the location of members of the public to be included in the evaluations in the PCSA and to support the design of the GROA. The applicant provided the basic population distribution in the demographic study area it established based on Regulatory Guide 4.2 (NRC, 1976aa). DOE also described the population locations, regional population centers, and population projections for the 50-year preclosure period of waste emplacement described in the license application (2017-2067).

**Demographic Study Area**

DOE used census data from the U.S. Census Bureau, along with supplemental data from the states of Nevada and California, to determine the population distributions as a function of distance from the GROA. Other data used included electric utility data, economic and agricultural characteristics, and data acquired from census survey information. (BSC, 2003ah)

The applicant established the demographic study area following the guidance in Regulatory Guide 4.2 (NRC, 1976aa, Section 2.1). This area consists of an 84-km [52-mi] radial area, centered on Nevada State Plane coordinates Northing 765621.5 and Easting 570433.6, where the GROA is located. The area comprises parts of Clark, Esmeralda, Lincoln, and Nye Counties in Nevada, and Inyo County in California. The study area was divided into study area grid cells, for which the applicant estimated the 2003 resident population located in each study area grid cell and presented these estimates in SAR Table 1.1-2 and Figure 1.1-11. This information provided the baseline population distribution within the 84-km [52-mi] grid that the applicant used for population projection estimates for the 50-year period of waste emplacement.

The applicant did not identify any permanent residents closer than about 22 km [13.7 mi] of the GROA. The nearest resident population was located in the unincorporated town of Amargosa Valley. The closest year-round housing was at the intersection of U.S. Highway 95 and Nevada State Route 373, as presented in SAR Figure 1.1-11 and Table 1.1-2.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's demographic study data in SAR Section 1.1.2 and references therein and its methodology to establish the demographic study area. The NRC staff performed confirmatory independent calculations (NRC, 2014aa) to estimate DOE's baseline 2003 population distribution within 84 km [52 mi] of the GROA using the Nevada County population estimates from 2001–2004 (Nevada Small Business Development Center, 2014aa). The NRC staff's results are comparable to those of the applicant's baseline 2003 population distribution data presented in the SAR. The NRC staff also compared 2010 population distribution within 84 km [52 mi] of the GROA using the U.S. Census Bureau data (2010aa) with that of DOE's projected population distribution data and notes that DOE's data estimate is generally higher, which is conservative. The NRC staff finds that the applicant's methodology to establish the demographic study area is acceptable because it is consistent with the guidance in Regulatory Guide 4.2. While Regulatory Guide 4.2 was developed for use in the evaluation of nuclear power stations, the methodologies and conclusions in this regulatory guide are appropriate for use for analogous activities proposed for the GROA, and also tend to be more conservative in their assumptions and more protective of public health and safety than is required to ensure safety of the proposed preclosure facilities and associated activities. Here, the accurate characterization of regional demography is a process that is independent of the particular type of facility proposed; therefore, the NRC staff finds that the applicant's use of Regulatory Guide 4.2 is acceptable. The NRC staff also finds that DOE used appropriate census data and that the distribution estimates are reasonable, as confirmed by the NRC staff's independent confirmatory calculations.

### Population Centers

In SAR Section 1.1.2.2, the applicant listed the nearby Nevada population centers: Boulder City, Henderson, Las Vegas, Mesquite, and North Las Vegas in Clark County; Caliente, Alamo, Panaca, and Pioche in Lincoln County; Beatty, Gabbs, Manhattan, Pahrump, Round Mountain, Tonopah, and the town of Amargosa Valley in Nye County; and Goldfield and Silver Peak in Esmeralda County. The nearby California population centers are Bishop and Death Valley National Park in Inyo County. The closest large population center to the GROA identified by DOE was Pahrump, primarily in Nye County, and partly in Clark County, Nevada, 56 km [35 mi] southeast of the repository with a population of 24,631 in the year 2000.

### NRC Staff's Evaluation

The NRC staff reviewed the applicant's information in SAR Section 1.1.2 and references therein pertaining to population centers near the Yucca Mountain Repository. The NRC staff evaluated the applicant's information using independent sources of information, by comparing the population centers in county master plans (Lincoln County Board of Commissioners, 2007aa; Nye County Board of Commissioners, 2011aa; Inyo County Board of Supervisors, 2001aa; Clark County, 2009aa) to those identified by the applicant. The NRC staff finds that the applicant followed the guidance provided in Regulatory Guide 4.2 because the applicant identified all significant population centers within an appropriate demographic study area within 84 km [52 mi] and used population data consistent with other acceptable evaluations of demography and population centers in the repository area.

### Population Projections

The applicant's population distribution projections were developed by using the 2003 baseline population distribution presented in SAR Table 1.1-2 and then applying the same annual rate of growth or decline of respective county populations and data compiled and documented in BSC (2007bz). The annual rate of change for Nye County was taken from Nye County population projections the Nevada State Demographer's Office made for the period of 2003–2026; an assumed constant average annual growth rate of 1.4 percent was used from 2027–2067. The annual rate of change for Clark County was taken from Clark County population projections that the Center for Business and Economic Research made for the period of 2003–2035; an assumed constant average growth rate of 1.08 percent was used from 2035–2067 on the basis of constant growth rate between 2032 and 2035 (BSC, 2007bz).

The applicant based its projections for the annual rate of change for Inyo County in California on Inyo County population projections from 2000–2050 made by the Demographic Research Unit of the California State Department of Finance (BSC, 2007bz). Those rates include negative growth after 2020. On the basis of these decreasing population rates, an assumed constant average decline of 1.96 percent was used from 2030–2040; an assumed constant average decline rate of 1.12 percent was used from 2040–2050; an assumed constant average decline rate of 0.6 percent was used from 2050–2060; and the applicant assumed no change was applied from 2060–2067 on the basis of the assumption that no decline in population is expected beyond 2060.

The applicant also estimated projected populations in Nye and Clark Counties due to construction and operation of the proposed repository and the associated proposed railroad from Caliente, Nevada, to the repository and included them in the population distribution projection estimates within 84 km [52 mi] of the GROA (BSC, 2007bz). The estimated projected population within 84 km [52 mi] of the GROA was provided for each year from 2003–2017 in SAR Table 1.1-3 and for years

2017, 2020, 2030, 2040, 2042, 2050, 2060, and 2067 in SAR Table 1.1-4. The year 2042 was specifically included in the population projections because the applicant considered it the midpoint of the perceived 50-year operational period of 2017–2067. DOE also estimated the age group distribution for the projected population for preclosure operations (midpoint in 2042) and presented it in SAR Table 1.1-5. No population was observed for Lincoln and Esmeralda Counties within 84 km [52 mi] of the GROA for 2003; therefore, the applicant did not perform projection estimates for Lincoln and Esmeralda Counties.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's data, assumptions, and methodology used for the population distribution projections within 84 km [52 mi] of the GROA in SAR Section 1.1.2 and references therein. The NRC staff performed independent confirmatory comparisons (NRC, 2014aa) to estimate the population projections using Nevada County Population Estimates from 2013 to 2032 (Nevada Small Business Development Center, 2013aa) and Nevada County Population Projections from 2008 to 2028 (Nevada Small Business Development Center, 2014ab). The NRC staff finds that the applicant's growth rate assumptions are acceptable because they are based on state and county information, which is acceptable for use in these types of studies.

The applicant's projections are also comparable to more recent information (Nevada Small Business Development Center, 2013aa). The NRC staff's estimated population projection results are comparable to the applicant's presented population distributions within 84 km [52 mi] of the GROA, or slightly lower based on 2010 census data, which represents conservatism by the applicant. Therefore, the NRC staff concludes that DOE's population distribution projections are acceptable.

DOE did not address transient population estimates, which is recommended in Regulatory Guide 4.2, Section 2.1.2.3. Regulatory Guide 4.2, Section 2.1.2.3 recommends that the applicant provide transient population distribution within 16 km [10 mi]. Most of the area within 16 km [10 mi] of the GROA contains the preclosure controlled area and the NNSS, leaving an area less than 4 miles south and 8 miles east of the preclosure controlled area available for potential residents. The most recently available 2010 U.S. Census Bureau information is reported for the closest communities of Amargosa Valley, Crystal, and Beatty, which are located at least 16 km [10 mi] away from the GROA. Thus, because any transient populations would be centered in these communities, there is not a transient population within the 16-km [10-mi] limit specified in Regulatory Guide 4.2. Moreover, between 2005 and 2009, the 2010 census data indicates that less than 4 percent of the population located within these communities moved into or out of those communities (NRC, 2014aa). Therefore, the NRC staff does not consider transient populations to be significant in the characterization of site demographics and finds that it is reasonable that DOE did not address transient population estimates in its application.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant adequately described information regarding Yucca Mountain regional demography, population centers, and population growth. The NRC finds the population data used for determining current population centers and estimated population growth is from credible, publicly-available sources, and that the methodologies used adequately present population distributions as a function of distance from the GROA. The NRC staff finds, with reasonable assurance, that the applicant's information regarding preclosure regional demography is acceptable for use in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(i) and 10 CFR 63.112(c).

#### 2.1.1.1.3.3 Local Meteorology and Regional Climatology

The applicant described local meteorology and regional climatology conditions that could pose hazards to GROA facilities or repository safety during the preclosure period. This information, presented in SAR Section 1.1.3, is also used to develop design bases for structures, systems, and components (SSCs) at the site. Atmospheric conditions, such as atmospheric stability categories, average wind speeds, and prevailing wind direction, are also described in SAR Section 1.1.3. The applicant used this information in later sections of the SAR to evaluate the consequences of airborne radionuclide transport in hypothetical preclosure release scenarios.

**Data Collection Techniques and Summaries**

The applicant used 12 meteorological monitoring stations to characterize site meteorological conditions. The applicant stated that it used NRC Regulatory Guide 1.23, Meteorological Monitoring Programs for Nuclear Power Plants (NRC, 2007aa, Section C), as well as earlier versions of the Regulatory Guide, to design and operate the monitoring stations with respect to wind, temperature, humidity, and precipitation measurements. The stations, located throughout the GROA, include one 60-m [197-ft] tower site, eight 10-m [33-ft] tower sites, and three precipitation-only monitoring sites. Five tower sites were established in 1985, the remaining tower sites were established in 1992, and the three precipitation-only sites were established in 1999. The tower sites measure wind speed and direction, temperature, humidity, and precipitation. The applicant described the sensors used (BSC, 2007bs) and described how these sensors meet the accuracy and performance specifications of Regulatory Guide 1.23.

The information collected from 1994–2006 was provided in the applicant's report on local meteorology of Yucca Mountain (BSC, 2007bs) and summarized in the SAR. The summaries included mean monthly values as well as observed precipitation and temperature extremes. The applicant also described the data reduction techniques it used to calculate atmospheric stability and classify wind speed characteristics, according to atmospheric stability class, and described how these techniques meet the specifications of Regulatory Guide 1.23.

**NRC Staff's Evaluation**

The NRC staff examined the data collection techniques and summaries described in SAR Section 1.1.3 and references therein. Specifically, the NRC staff reviewed the locations of the tower and precipitation sites (SAR Figure 1.1-12) and regional sites (SAR Figure 1.1-13) using the guidance in Regulatory Guide 1.23 and determined that (i) these are located such that each type of primary geomorphic features of the Yucca Mountain site (i.e., ridgetop, major wash, minor wash, and flat) is represented by at least one monitoring site and (ii) the regional sites feature a variety of elevations and are located both upwind and downwind with respect to prevailing wind directions. Regulatory Guide 1.23 was developed for use in evaluating nuclear power plants; however, the methodologies and conclusions are applicable to the collection of meteorological data independent of the type of facility, and thus, the NRC staff concludes that use of this regulatory guide is acceptable here. Therefore, the NRC staff finds that the monitoring site locations provide meteorological data representative of the Yucca Mountain site consistent with guidance in Regulatory Guide 1.23, Section C.

The NRC staff reviewed the applicant's collection techniques by comparing the applicant's system-accuracy requirements for wind, temperature, humidity, and precipitation measurements summarized in SAR Table 1.1-9 with guidance in Regulatory Guide 1.23, Section C. The NRC staff finds that the applicant's collection techniques were based on accepted methods and

DOE's reported system-accuracy requirements for these parameters are consistent with this NRC guidance. In addition, the NRC staff finds that DOE's reported data-recovery rates meet or exceed the Regulatory Guide 1.23 values. Therefore, the NRC staff finds that the data collected by these methods is acceptable.

Further, the NRC staff evaluated the applicant's description in BSC Section 4.2 (2007bs) of the methods used to determine atmospheric stability and joint frequency distributions of wind speed and direction and finds they are consistent with guidance in Regulatory Guide 1.23, Section 2.2 because the applicant used an acceptable method to collect atmospheric stability and frequency distribution of wind speed and direction at different vertical heights. The use of the Pasquill Stability Classes is also consistent with Regulatory Guide 1.23, Section 2.2, and is therefore acceptable.

**Annual and Probable Maximum Precipitation**

DOE summarized the site precipitation data collected by the methods described above to characterize annual precipitation in SAR Section 1.1.3.2.1 and described the methodology used to estimate probable maximum precipitation in SAR Section 1.1.4.3.1. The applicant included site-specific precipitation data summaries for each precipitation station over the period of 1994 through 2006, that include (i) maximum hourly precipitation rate, (ii) maximum daily precipitation, (iii) average number of days with precipitation, (iv) annual average precipitation through 2006 for the set of meteorological and precipitation stations on both a monthly and annual basis, and (v) the annual average precipitation at Site 1. DOE noted that Site 1 is the location of the meteorological measurement station most representative of ambient weather conditions at the GROA (SAR Section 1.1.3.1.1). The applicant provided the maximum 24-hour precipitation totals for September 21 through 22, 2007, which the applicant described as the largest precipitation event reported at the site, in SAR Table 1.1-23; the largest reported 24-hour precipitation total among the 12 stations was 87.1 mm [3.4 in].

Following guidance for nuclear power plants specified in NUREG–0800 (NRC, 2007ak, Section 2.4.3), the applicant used a National Oceanic and Atmospheric Administration (NOAA) procedure (Hansen, et al., 1977aa) to estimate probable maximum precipitation. The probable maximum precipitation information is used to determine the flood hazards within the GROA. Hansen, et al. (1977aa, Chapter 4) describes a procedure based on scaling a standardized 1-hour storm on a reference 2.6-$km^2$ [1-$mi^2$] area to a standard 6-hour storm, adjusted to the desired basin area, as described in Hansen, et al. (1977aa, Figures 4.5, 4.7, and 4.9). This procedure uses historical records from meteorological stations across the Great Basin, including several stations in southern Nevada. In SAR Section 1.1.4.3.1, the applicant estimated values of the probable maximum precipitation to be 335 mm [13.2 in] for a 6-hour storm event for the basins encompassing the North Portal pad and 328 mm [12.9 in] for the basins encompassing the South Portal pad. For comparison, these 6-hour totals are approximately 3.8 times larger than the largest reported 24-hour precipitation total observed at any Yucca Mountain precipitation monitoring station.

To characterize snowfall at the site, DOE used data collected at the Desert Rock Airport Weather Service Observatory, approximately 45 km [28 mi] southeast of Yucca Mountain at an elevation of 1,006 m [3,301 ft] above mean sea level, with a maximum observed daily snowfall of 15 cm [6 in] and maximum monthly snowfall of 17 cm [6.6 in] during the period of record from January 1, 1983, through February 28, 2005.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's precipitation information in SAR Section 1.1.3 and references therein using the guidance in NUREG–0800 (NRC, 2007ak, Section 2.4.3). This revision to NUREG–0800 contains the NRC's most recent guidance on evaluating precipitation data.

The NRC finds that the precipitation data the applicant provided was collected and processed consistent with acceptable methods, and includes representative peak hourly and daily precipitation rates and described seasonal and interannual variation in precipitation consistent with the relevant guidance in NUREG–0800, Section 2.4.3.

The NRC staff finds that the applicant's estimated values for probable maximum precipitation are consistent with the procedure for a 16.8-$km^2$ [6.5-$mi^2$] watershed by obtaining the appropriate factors from the corresponding figures in Hansen, et al. (1977aa, Figures 4.5, 4.7, and 4.9) and multiplying them together to obtain 34 cm [13.2 in], a value which is consistent with the applicant's estimate for the basin encompassing the North Portal pad. The NRC staff finds that the applicant adequately estimated the probable maximum precipitation because the methodology that the applicant used is consistent with the NRC's guidance specified in NUREG–0800, Section 2.4.3.

The NRC staff finds that the applicant's estimate of a maximum daily snowfall of 15 cm [6 in] and a maximum monthly snowfall of 17 cm [6.6 in] is adequate because the methodology the applicant used to estimate the proposed maximum daily and monthly snowfall is consistent with NRC guidance specified in NUREG–0800, Section 2.4.3 (NRC, 2007ak).

**Severe Weather**

The applicant's assessment of severe weather was provided in SAR Section 1.1.3.6. This information was based on regional information and local or site weather data. The following severe weather types were included by the applicant: (i) tornadoes, (ii) thunderstorms and lightning strikes, (iii) sandstorms, and (iv) snowfall.

While the applicant also listed wind hazards from hurricanes as one of many potential external hazards for analysis in SAR Section 1.6.3.4.4, information concerning hurricanes was not discussed by the applicant. The NRC staff's evaluation of the wind hazard from hurricanes is presented in SER Section 2.1.1.3.3.1.3.2.

The applicant's discussion of sandstorms (dust storms) at the site (SAR Section 1.1.3.6.3) concluded that sandstorms would be unlikely because a wind speed of greater than 40 km/hr [25 mph]—rare at the site—would be needed to initiate them. Despite the conclusion that sand storms are unlikely, the applicant included them in the PCSA (SAR Section 1.6.3). The NRC staff evaluates the applicant's assessment of sandstorms in SER Section 2.1.1.3.3.1.3.5.

The applicant included a brief discussion of snowfall characteristics in the severe weather section of the SAR. Snowfall is included as part of the preceding discussion in SER Section 2.1.1.1.3.3 that evaluates the applicant's precipitation information.

## Tornadoes

The applicant described tornadoes as infrequent and weak in the Yucca Mountain region because of generally dry weather conditions and unfavorable terrain conditions. The applicant reported that three tornadoes have been observed in Nye County (SAR Section 1.1.3.6.1) during the period of 1950 through 2003. The applicant determined, however, that meteorological conditions favorable for tornado formation could exist at the site on rare occasions and therefore, the applicant stated that a tornado could initiate a detrimental event sequence that would need to be evaluated in the PCSA.

The applicant followed procedures described in Regulatory Guide 1.76 (NRC, 2007ai) and a DOE extreme hazard analysis (BSC, 2008ai) to develop design basis tornado characteristics. The applicant established design basis tornado parameters including a wind speed of 304 km/hr [189 mph], a pressure drop of 5.6 kPa [0.81 psi], and a rate of pressure drop of 2.1 kPa/s [0.3 psi/s] (SAR Section 1.1.3.6.1). The calculated wind speeds developed are based on a $1 \times 10^{-7}$ annual exceedance probability.

## NRC Staff's Evaluation

The NRC staff reviewed the applicant's characterization of tornadoes in SAR Section 1.1.3 and references therein using the guidance in Regulatory Guide 1.76 and finds that the applicant adequately characterized tornado characteristics and other high wind hazards at the site. Regulatory Guide 1.76 was developed for use in evaluating wind hazards for nuclear power plants. However, the methodologies and conclusions for evaluating extreme weather described in this regulatory guide are facility independent. Thus, the NRC staff finds that the hazard evaluations using this guidance are applicable to the analogous activities and facilities proposed for the GROA.

The applicant used appropriate site data to describe the past occurrence of tornadoes within Nye County and that meteorological conditions could exist that favor tornado formation. The NRC staff finds that the design basis wind speed of 304 km/hr [189 mph] is acceptable because it is higher than the regional value in Regulatory Guide 1.76, which was updated in 2007 to specify a maximum wind speed of 257 km/hr [160 mph]. Second, this updated wind speed in Regulatory Guide 1.76 has an annual exceedance probability of $1 \times 10^{-7}$ (the likelihood of this wind speed being larger in any year is smaller than $1 \times 10^{-7}$). This exceedance probability is one order of magnitude less likely than the $1 \times 10^{-6}$ probability for inclusion of the hazard in the PCSA. Therefore, the NRC staff finds that this DOE design basis is conservative and acceptable.

## Lightning

The applicant estimated lightning strike frequencies using annual cloud-to-ground lightning observations from 1991 through 1996 collected at the NNSS and warm-season cloud-to-ground lightning data in the vicinity of the NNSS from 1993 through 2000 (SAR Section 1.1.3.6.2). The Air Resources Laboratory and Special Operations and Research Division of NOAA collected these observations using an automated lightning-detection system. Measured annual flash density ranged from 0.06 to 0.4 strikes per $km^2$ [0.16 to 1.1 strikes per $mi^2$] per year. The applicant indicated that these observations are generally consistent with other estimates for southern Nevada (Randerson and Sanders, 2002). Therefore, the applicant determined that direct lightning strikes could initiate a detrimental event sequence that would need to be evaluated in the PCSA (SAR Section 1.6.3.4.6).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information in SAR Section 1.1.3, references therein, and other cited sources of information and finds the analysis of the frequency and distribution of lightning strikes within the GROA acceptable because the available lightning data are from credible sources, including the NOAA Air Resources Laboratory, and are consistent with each other. These data suggest that dozens of lightning strikes are expected within the GROA over the proposed100-year-preclosure period, which is consistent with DOE's characterization of lightning strikes.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant adequately described information on the Yucca Mountain site local meteorology and local climatology, including information characterizing the wind speed and direction, temperature, humidity, and precipitation. The applicant used technically acceptable instruments to collect the data and provided accurate summaries of the data, including annual and maximum precipitation data. The applicant also used acceptable methods to use this data and develop the probable maximum precipitation and adequately defined the type, frequency, magnitude, and duration of severe weather using acceptable regulatory guidance documents. Therefore, the NRC staff finds, with reasonable assurance, that the applicant's information regarding local meteorology and regional climatology is acceptable for use in the evaluations in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(iii) and 10 CFR 63.112(c).

#### 2.1.1.1.3.4 Regional and Local Surface and Groundwater Hydrology

The applicant provided regional and local surface and groundwater hydrological information in SAR Section 1.1.4. The surface GROA, situated on the east side of Exile Hill in Midway Valley at the eastern margin of Yucca Mountain, could be affected by water and debris flows emanating from the eastern slopes of Exile Hill during storm events. Therefore, the applicant estimated the probable maximum flood resulting from the probable maximum precipitation (see SER Section 2.1.1.1.3.3) to determine the extent of the hazard posed by flood waters at the GROA to include in the PCSA evaluations, and to support the design of the GROA by providing the flood water depths for flood protection structures necessary to protect the GROA from runoff and debris flows.

**Surface and Groundwater Hydrologic Features**

**Surface Water**

The applicant characterized the regional climate at Yucca Mountain and its vicinity as dry, semiarid because the site annual average precipitation is 125 mm/yr [4.9 in/yr] at a 1,500-m [4,921-ft] elevation, with infrequent regional rainstorms during the winter and localized thunderstorms during the summer. The streams in the Yucca Mountain vicinity are ephemeral, and no natural bodies of water or wetlands occur on the Yucca Mountain site. Winter storms and localized summer thunderstorms provide the main source of runoff. Flash flooding resulting from intense rainfall and runoff from localized convective storms or from high-intensity precipitation cells within regional storm systems constitute the major flood hazard at and near Yucca Mountain. The applicant summarized the flooding history in the Yucca Mountain area on the basis of both literature reviews and actual stream gauging records, as described in BSC (2004bj, Section 3.4.3).

If constructed, the Yucca Mountain surface GROA facilities would be situated on the east side of Exile Hill in Midway Valley at the eastern margin of Yucca Mountain. The applicant portrayed natural drainage channels near the Yucca Mountain site in SAR Figures 1.1-52 and 1.1-53. Fortymile Wash is the main natural drainage channel on the Yucca Mountain site. On the basis of site-specific climate (semiarid) and soil conditions (permeable surficial materials), the applicant determined that the pooling or ponding of large quantities of water on the surface is not likely to occur.

**Ground Water**

The applicant characterized the regional groundwater flow as occurring in an asymmetric radial flow pattern, flowing from recharge areas in mountains and other highlands toward Death Valley (SAR Section 1.1.4.2). The elevation of the surface GROA is 1,120 m [3,675 ft] above mean sea level, whereas the water table is approximately 730 m [2,395 ft] above mean sea level, or about 390 m [1,280 ft] below the surface (SAR Section 1.1.4.2.3). The minimum distance from the floor of the emplacement area to the top of the current water table is about 210 m [685 ft] in the northwestern part of the repository. The maximum distance to the water table is about 375 m [1,230 ft] in the southern part of the repository (SAR Section 1.1.4.2.3). Because the saturated zone is so far below the surface and subsurface facilities of the GROA, the applicant concluded that the saturated zone does not need to be considered for facility foundation or other aspects of building design. Perched water (entrapped water) has been identified in several boreholes (SAR Figure 1.1-56), however, they are located at depths of 100 to 200 m [328 to 656 ft] beneath the emplacement drifts. Because the perched water bodies are deep below the surface GROA and the emplacement drifts, the applicant concluded that they would not impact the facility foundations or other aspects of the building design.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of surface and groundwater hydrologic features in SAR Section 1.1.3 and references therein. The NRC staff finds the descriptions of the hydrologic features, including surface water drainage channels, runoff, the unsaturated zone, saturated zone, flash flood, and perched water acceptable because the information is consistent with the NRC staff's understanding of the hydrological systems at Yucca Mountain based on first-hand experience obtained from staff field investigations of the site during the pre-licensing period. The NRC staff finds that the applicant's conclusions that the saturated zone and perched water bodies need not be considered in the design of GROA facilities acceptable because it is consistent with the NRC staff's understanding of the groundwater system at Yucca Mountain (see SER Section 2.2.1.3.6). The NRC staff finds the applicant's regional flood history is adequately described because the applicant included complete information from a comprehensive literature review (SAR Section 1.1.3) of the drainage system patterns, paleo and historical surface water flow conditions, and historical flood occurrences and flood discharges in the Yucca Mountain area, which included supporting stream gauge measurements. The NRC staff finds that the flood history information is sufficient to support the probable maximum flood review in the next SER section.

**Probable Maximum Flood**

In SAR Section 1.1.4.3, the applicant provided a flood inundation analysis for the surface GROA. The applicant's analysis was conducted in two parts, the first considered a surface facility design without flood-inundation control measures, and the second part considered possible surface facilities to control flood inundation. The applicant's two-part analysis of the probable maximum flooding forms the basis for the applicant's assessment of preventive measures such as dikes and channels around facilities important to safety to control flooding.

The applicant used the U.S. Army Corps of Engineers HEC-1 (Version 4.0) software program to calculate probable maximum flood events (surface runoff and channel discharge) resulting from the probable maximum precipitation event (BSC, 2007db). The applicant used the U.S. Army Corps of Engineers HEC-RAS (Version 2.1) software programs to calculate flood depths that would occur during the probable maximum flood events produced by HEC-1 simulations. To perform simulations with HEC-1 and HEC-RAS, topography; hydraulic properties for subareas and channels, which are defined by the user of the models; and probable maximum precipitation values are required. The subareas the applicant defined for use in the probable maximum flood analysis models are depicted in BSC (2007db, Figure 6-1). The applicant used a 0.6-m [2-ft] elevation contour map to produce a digital elevation model of the study area, and obtained the length, slope, and channel dimensions using the topographic data for natural channels and engineering drawings for man-made channels for each of the defined subareas. The probable maximum precipitation value calculated by the applicant from precipitation data is also used for this flooding analysis (see SER Section 2.1.1.1.3.3 for the NRC staff's review of the applicant's probable maximum precipitation analysis).

**Subarea Hydraulic Properties**

The applicant used a unit hydrograph method to develop the probable maximum flood hydrograph. Determination of a unit hydrograph requires subarea size and time of runoff concentration. The applicant used a U.S. Bureau of Reclamation empirical formula to calculate the time of concentration for each subarea because it gives the smallest time of concentration, among common formulas available, as identified in BSC (2007db, Section 6.1.4) and, thus, the largest peak flow. The applicant assumed a uniform infiltration rate of 38.1 mm/hr [1.5 in/hr], which is lower than the lowest infiltration values obtained from *in-situ* infiltration tests conducted in the surrounding area (lower infiltration leads to greater runoff), as described in BSC (2007db, Section 6.1.4).

The applicant used a bulking factor of 10 percent to account for increased flow depths caused by the presence of entrained air, debris, and sediment load. Use of this bulking factor effectively increases the peak discharges by 10 percent in the probable maximum flood analyses. A literature review by the applicant suggested that flow bulking may not be a significant factor affecting probable maximum floods (BSC, 2007db), but the applicant chose to include the 10 percent bulking factor as a conservatism in its analysis.

**NRC Staff's Evaluation**

The NRC staff examined the division of subareas and probable maximum flood calculations, described by DOE in SAR Section 1.1.4.3 and references therein, using the guidance in NUREG–0800, Sections 2.4.1, 2.4.2, and 2.4.3. Based on the NRC staff's watershed modeling experience and knowledge, the NRC staff finds that the elevation contour of 0.6 m [2 ft] used to characterize the basin drainage is appropriate for characterizing the topography at the Yucca Mountain site because it results in model resolution that adequately captures the hydraulic properties for subareas and channels the applicant derived and used in the models. The NRC staff finds that the applicant appropriately applied a standard approach in developing a runoff hydrograph for HEC-1, an industry-standard code developed by the USACE for event-based rainfall-runoff analysis. The NRC staff also finds the subarea properties the applicant used in its HEC-1 and HEC-RAS model are reasonable and the assumptions are appropriate because they are based on applicable topographic and precipitation data and calculated using standard runoff concentration techniques commonly applied in surface water hydrology studies.

The NRC staff finds that the applicant assumed a reasonable infiltration rate for all subareas (Woolhiser, et al., 2006aa), which also represents a conservatism in the analysis because the assumed infiltration rate of 38.1 mm/hr [1.5 in/hr] is lower than measured infiltration rates. The NRC staff reviewed the inclusion of a bulking factor of 10 percent in the applicant's probable maximum flood analysis. Because of the lack of site-specific data for Yucca Mountain, the applicant considered a range of bulking factors between 4 and 10 percent to account for air, debris, and sediment entrained in the flood flow. For this range of values, the bulking factor of 10 percent represents a conservative flood analysis. The NRC staff finds the use of a bulking factor of 10 percent to be acceptable because the entrained air, debris, and sediment would be negligible compared to the large volume of water conveyed during the probable maximum flood event.

**Channel Hydraulic Properties**

Manning’s roughness coefficient is a channel property needed to calculate the hydraulic losses of fluid flow through a channel system required for the HEC-1 and HEC-RAS modeling. In a sensitivity study described in BSC (2007db, Section 6.1.5), the applicant considered the range of Manning’s coefficient for three flow conditions: (i) clear water flow, (ii) high sediment transport, and (iii) mudflow. The applicant used a Manning’s coefficient of 0.035 for the clear water channel flow condition, 0.09 for high sediment transport flow, and 0.16 for the mudflow. The values were selected on the basis of calibration studies provided in DOE (2009bf).

Results of the applicant’s sensitivity study showed that increasing the Manning’s coefficient from 0.035 to 0.09 resulted in a 2.4-m [8-ft] increase of predicted water surface elevation near the North Portal pad; increasing Manning’s coefficient further from 0.09 to 0.16, however, resulted only in a minimal additional increase of 0.15 m [0.5 ft] in predicted water surface elevation (DOE, 2009bf). For a probable maximum flood analysis, the applicant considered that the amount of clear water runoff would be large enough that a mudflow condition is unlikely to develop. Therefore, the applicant used a Manning’s coefficient of 0.09 in its probable maximum flood analysis, corresponding to the high sediment transport flow condition, because the applicant concluded that a mudflow was not likely to occur, and even if it did, the resulting increase in surface elevation would be negligible.

**NRC Staff’s Evaluation**

The NRC staff reviewed the procedure the applicant used to estimate the Manning’s coefficient of 0.09 for the probable maximum flood analysis in the documents supporting SAR Section 1.1.4 (BSC, 2007db; DOE, 2009bf). The NRC staff finds the values used by the applicant for the three cases of flow evaluated are acceptable because (i) they represent the full range of likely flooding conditions and (ii) they were selected on the basis of an acceptable calibration study performed by the applicant. The NRC staff finds that the sensitivity studies were conducted appropriately because the studies examined the changes in water surface elevation across the full range of applicable Manning coefficients and showed little change in water surface elevation for the case of mudflow, which the NRC staff finds is unlikely to occur in a flash flooding scenario, as discussed in the application. Therefore, the NRC staff concludes that the value used for Manning’s coefficient is acceptable.

**Results and Application of Probable Maximum Flood Analyses**

The applicant calculated that the probable maximum flood peak flow rate resulting from the HEC-1 model is 1,564 $m^3/s$ [55,240 cfs], as shown in BSC (2007db, Table 7-1). The applicant used this peak flow rate and estimated peak probable maximum flood flows from subareas and concentration points, as shown in BSC (2007db, Table 7-1), and flood inundation results for man-made channel segments, as shown in BSC (2007db, Tables 7-2 to 7-4). The evaluation included both a no-mitigation case and a mitigation case. For the no-mitigation case (evaluated first), the applicant assumed that the planned facilities upstream were not constructed and that no flood control measures were implemented. In SAR Figure 1.1-57, the applicant showed that in the no-mitigation case, the runoff from the probable maximum flood event would inundate the North Portal pad and important to safety (ITS) facilities in the vicinity of the North Portal. The applicant's calculations indicated that water would not overflow the South Portal pad or the planned North Construction Portal during an unmitigated probable maximum flood event.

In the second analysis, the flood mitigation case, the applicant showed in SAR Figure 1.2.2-7 that ITS structures, the North Portal, and the Aging Facility areas can be protected by reasonable engineered features, such as dikes, and drainage and diversion channels, and therefore, will include these features in PCSA evaluations and in the design of the GROA. The NRC staff reviews and evaluates the design of important to safety flood control features in SER Section 2.1.1.7.

The applicant also quantified the probable maximum flood in terms of annual exceedance probability (BSC, 2008ai). The frequency of the probable maximum flood is based on the combined probability of the probable maximum precipitation, antecedent moisture conditions, and the spatial and temporal distribution of the storm. The resulting probability is approximately $1.1 \times 10^{-9}$, which is less than the screening criteria of 1 in 10,000 before permanent closure. The flood flow rate of the million year return period flood is approximately 1,133 $m^3/s$ [40,000 cfs]. The NRC staff's review of the screening criteria for flood hazards is found in SER Section 2.1.1.3, Identification of Hazards and Initiating Events.

**NRC Staff's Evaluation**

The NRC staff reviewed the analysis of the probable maximum flood in SAR Section 2.1.1.7 and references therein. The NRC staff finds that the applicant's probable maximum flood analysis is adequate because (i) the methodologies employed in determining inputs to computer codes, including the unit hydrograph method, follow accepted professional practice in hydrological engineering; (ii) the software codes HEC-1 and HEC-RAS the applicant used are standard models employed to simulate flood analysis; (iii) the applicant's use of the unit hydrograph method and derivation of Manning's coefficient to derive the watershed and channel properties required for analyses are appropriate, as discussed in the previous subsection; and (iv) the applicant used appropriate input data for probable maximum flood simulation on the basis of site topography and probable maximum precipitation data that the NRC staff evaluated and found to be acceptable in SER Sections 2.1.1.1.3.1 and 2.1.1.1.3.3. The NRC staff also compared the applicant's results with those predicted by Bullard (1986aa), who computed flood potentials for 11 small drainage basins on Yucca Mountain for clear water flows, and found that the applicant's results for peak flow rate is about 20 percent higher than the maximum local probable maximum flood peak flow rate predicted by Bullard. The NRC staff finds that the applicant's HEC-1 probable maximum flood model simulation is comparable with this independent study (Bullard (1986aa). The NRC staff also finds that comparing the maximum flow rate of the Bullard study at the assumed bulking factor of 10 percent against the applicant's analysis, the Bullard results are still lower than that calculated by the applicant. This further supports the acceptability of the applicant's probable maximum flood analysis because it is conservative when compared to the Bullard study. The NRC staff also finds that the probabilistic evaluation of the probable maximum flood is acceptable because it adequately accounts for probabilistic estimates of maximum rainfall, antecedent moisture conditions, and spatial and the temporal distribution of storms.

**NRC Staff's Conclusion**

The NRC staff finds the applicant's description of the Yucca Mountain surface and groundwater hydrology adequately identifies features that are important to the PCSA and the GROA design. The NRC staff finds that DOE's probable maximum flood calculation and the analyses of proposed changes that could impact drainage features, specifically the need for flood control measures, are acceptable because (i) the applicant described the surface and groundwater features consistent with the NRC staff's knowledge gained through many field observations of the Yucca Mountain site, (ii) the applicant used appropriate modeling techniques and computer simulation programs to calculate the probable maximum flood; (iii) the calculation of the probable maximum flood is supported by sufficient and accurate precipitation and topographic data, and reasonable assumptions concerning flooding for the drainage basin based on historical information, and (iv) the two-part flood inundation analysis of the GROA is complete and adequate. Therefore, the NRC staff finds, with reasonable assurance, that the regional and local surface and groundwater hydrology information presented in the SAR is acceptable to perform the PCSA, to support the design of the GROA, and satisfies 10 CFR 63.21(c)(1)(iii) and 10 CFR 63.112(c).

### 2.1.1.1.3.5 Site Geologic and Geotechnical Engineering Conditions, and Seismology

The applicant provided information in SAR Section 1.1.5 and supplemental information related to site geology and seismology, particularly to identify naturally occurring hazards, for use in the PCSA and to support the GROA design. This information included descriptions of site geologic conditions, seismology and probabilistic seismic hazard, seismic site response modeling, site geotechnical conditions and stability of subsurface and surface materials, and fault displacement hazards; these are each discussed in the following subsections.

2.1.1.1.3.5.1 Site Geologic Conditions

In SAR Section 1.1.5, the applicant provided the geologic information from its site characterization investigations. In SAR Section 1.1.5.1, the applicant described the geologic site conditions of the rocks and alluvial deposits (sediments deposited by streams in valleys) on which the proposed surface GROA facilities are proposed to be built and into which waste packages would be placed in the underground (subsurface) GROA. The applicant also identified and described geologic structures, including faults, fractures, and the inclined layering of rocks, and characteristics of the rocks such as the degree of fusion of the rock matrix and relative abundance of lithophysae (voids in the rocks formed by volcanic-gas bubbles) that the applicant concluded would be likely to affect GROA mechanical and hydrologic properties and conditions. The NRC staff's review and evaluation of the Yucca Mountain site geologic conditions are described in the following subsections on the geology of the subsurface GROA (SER Section 2.1.1.1.3.5.1.1) and the geology of the surface GROA (SER Section 2.1.1.1.3.5.1.2).

#### 2.1.1.1.3.5.1.1 Geology of the Subsurface Geologic Repository Operations Area

The subsurface GROA is composed entirely of layered tuff (solidified erupted ash). The layers are inclined in an easterly direction, and they are fractured and faulted. The NRC staff organized its review and evaluation of GROA geology into evaluations of the site's stratigraphy and structural geology.

**Stratigraphy of the Subsurface GROA**

In SAR Section 1.1.5.1, DOE described the stratigraphy of Yucca Mountain as layered volcanic rocks that were erupted and deposited approximately 11 to 14 million years ago. The volcanic rocks consist primarily of tuffs that originated from large explosive volcanoes north of Yucca Mountain. The volcanic rock formations show widely varying thicknesses across Yucca Mountain, generally thicker to the north and thinner to the south. Rocks classified as the Paintbrush Group dominate the surface and subsurface at Yucca Mountain. These rocks are subdivided and labeled Topopah Spring Tuff, Pah Canyon Tuff, Yucca Mountain Tuff, and Tiva Canyon Tuff Formations, among others. The Topopah Spring Tuff Formation is a 12.8-million-year-old, mostly welded (dense, fused) tuff with a maximum thickness of approximately 380 m [1,247 ft].

The Topopah Spring Tuff contains the proposed repository host horizon (RHH), which consists of four zones where waste is proposed to be emplaced. These four zones, from bottom to top, are the lower nonlithophysal, lower lithophysal, middle nonlithophysal, and upper lithophysal zones. On the basis of its lithological studies of Yucca Mountain rocks, augmented by its studies of the rocks in the Exploratory Studies Facility (ESF) and Enhanced Characterization of the Repository Block (ECRB), DOE estimated that the two lithophysal zones in the RHH comprise approximately 85 percent of the waste emplacement area (SAR Section 1.1.5.3.1.1).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of stratigraphy of the subsurface GROA, including SAR Section 1.1.5.1, references therein, and the applicant's stratigraphical studies conducted in the Yucca Mountain region (BSC, 2004bi; Sawyer, et al., 1994aa). DOE's data are consistent with independent NRC studies derived from geologic maps; from observations of surface and subsurface rock exposures and alluvium; and from borehole logs, core samples, and three-dimensional computer scale-models (Waiting, et al., 2007aa; NRC, 1999aa). The NRC staff finds that DOE has adequately described the age of the rocks, rock layer stacking order, and thickness variations of the volcanic rocks because these descriptions are consistent with the NRC's independent studies.

**Structural Geology of the Subsurface GROA**

In SAR Section 1.1.5.1, DOE provided information on site structural geology and tectonics to describe past geologic hazards and potential future hazards caused by faulting, seismicity, rockfall, and volcanism. DOE also provided information on the structural geologic studies it conducted in the Yucca Mountain region (BSC, 2004bi; Day, et al., 1998aa). The principal geologic deformation features and processes that might affect the volcanic rocks at Yucca Mountain during the preclosure period are faulting and fracturing; which are evaluated next.

## Faulting

DOE located and characterized hundreds of faults within a 100-km [62-mi] radius of Yucca Mountain (BSC, 2004bi; Day, et al., 1998aa). DOE depicted the faults at Yucca Mountain on geologic maps and geologic cross sections in SAR Section 1.1.5.1.2 and in DOE (2009ar,bg). This information was used to identify faults that might affect the proposed repository site indirectly by generating earthquakes or directly by causing SSCs located sufficiently close to faults to slip, shear, or tilt. DOE described large faults, called block-bounding faults (e.g., the Solitario Canyon and Bow Ridge Faults), that control the structural framework of the site and intrablock faults (e.g., the Sundance and Ghost Dance Faults) that have been formed in response to strains developed in the faulted blocks resulting from slip of the block-bounding faults. DOE observed additional small-displacement faults and shear fractures in the ESF and ECRB.

The block-bounding faults are dominantly north-south-striking normal faults that dip moderately (30–60°) to steeply (60–90°) to the west, and separate 1 to 5-km [0.6 to 3.1-mi]-wide, and tilted blocks of gently (less than 30°) east-dipping volcanic rocks. DOE determined that displacement of such block-bounding faults could generate the largest displacement and vibratory ground motions (i.e., earthquakes) at the site.

DOE determined that the block-bounding faults were active during formation of the volcanic rocks that comprise the RHH (Paintbrush Group, 12.8 to 12.7 million years ago). Significant motion on the faults occurred about a million years later, after emplacement of the 11.6-million-year-old Rainier Mesa Tuff Formation. DOE provided further evidence that the block-bounding faults were reactivated in the Quaternary Period (less than 2 million years ago) and have the potential for significant future movement.

DOE stated that the regional east-west-directed extension of the Basin and Range Province, in which Yucca Mountain is located, is accommodated primarily by slip on block-bounding faults. DOE also observed greater crustal extension in the southern portion of the site than in the northern portion. DOE stated that the transition to greater extension in the south is marked by an increase in the number of fault splays off the block-bounding faults and an increase in displacement on faults such as the Solitario Canyon and Paintbrush Canyon faults.

In response to the NRC staff's RAIs, DOE described significant fault displacements and how such displacements were determined (DOE, 2009as). The main block-bounding faults that bound the subsurface GROA are the Solitario Canyon and Bow Ridge Faults to the west and east, respectively. DOE stated that the 60-m [197-ft] setback distance it established as design control parameter 01-05 was applied to these two faults as a postclosure criterion.

The applicant's proposed setback distance from this significant Quaternary fault is a prerequisite for completing the subsurface GROA design. The setback distance determination depends on the characterization of the main fault and splays and their displacements, width of fault damage zones, and attendant zones of influence (DOE, 2009as,bf). In particular, the location of the westernmost endpoints of emplacement drifts (and, therefore, the location and length of emplacement drifts) depends on the location of the west access main, which the applicant also proposes to setback from the Solitario Canyon fault (DOE, 2009bf). DOE estimated the setback distance for subsurface openings on the basis of the locations, strikes, and dips of known faults. DOE stated in SAR Section 1.1.5 that this information on which its estimates are based will be confirmed during excavation of the openings In particular, DOE stated that, "Emplacement drifts shall be located a minimum of 60 m from a Quaternary fault with potential for significant displacement" (SAR Table 1.9-9, DCP 01-05); "During construction activities (underground) in the vicinity of the Solitario Canyon fault, the location of the fault will be confirmed, and the condition of the rock near the fault will be examined" (DOE 2009bf, Enclosure 1; DOE 2009as, Enclosure 3); "A construction standoff [setback] will then be evaluated on observational data to confirm the design basis" (SAR Section 1.3.4.2.2).

DOE also stated that a setback distance of 60 m [197 ft] from Quaternary block-bounding faults with potential for significant displacement also provides a safety margin from preclosure fault displacement hazards. This is based on an analysis of displacement and stress adjacent to an active fault for displacements up to 1 m [3.3 ft]. The largest mean preclosure displacement on the Solitario Canyon fault is 32 cm [1 ft] with an annual exceedance probability of $10^{-5}$ (BSC, 2003aj). In addition to the hazard of direct fault displacement, the applicant determined that faults and their damage zones can disturb drift stability (SAR Section 1.3.4.2.2) and increase rockfall hazard (DOE, 2009bf). DOE stated that the 60-m [197-ft] setback of emplacement drifts from the Solitario Canyon fault is sufficient to mitigate this increased hazard and determined that this hazard is not present at the Bow Ridge Fault due to that fault's distance from the subsurface GROA (SAR Figure 2.2-12).

DOE determined that the subsurface GROA will only approach the Solitario Canyon fault (not the Bow Ridge Fault) closely enough for this 60-m [197-ft] setback to apply (SAR Figure 2.2-12). Because the proposed subsurface GROA will have its western boundary delimited by the subsurface trace of the Solitario Canyon fault at the level of the subsurface GROA, the component of the GROA that will be closest to the Solitario Canyon fault will be the perimeter access main (DOE, 2009bf). Further, because the access mains are subject to the 60-m [197-ft] standoff design control parameter 01-05 and because the applicant stated that waste emplacement will be located an additional 60 m [197 ft] from the access main as measured perpendicular to the access main (DOE, 2009bf), the closest a waste package can be to the Solitario Canyon fault would be 120 m [394 ft].

DOE expects to encounter faults during drift construction and recognizes the need to characterize their orientation, displacement, and widths of damage zone and zone of influence to assess the appropriate setback and predict the location of intersections in adjacent drifts [DOE (2009as, Section 1.2.3.1); SAR Table 5.10-3].

DOE determined that, for preclosure safety considerations, fault shear displacements of more than 3 m [10 ft] during a 100-year-preclosure period have annual exceedance probabilities of less than $10^{-6}$ and, therefore, found these displacements to not be important for its hazard analysis. Furthermore, narrow faults with observed total displacement of 2 m [6.7 ft] or less are estimated by DOE to have an annual probability of exceedance of less than $10^{-8}$ for future displacements of 3 m [10 ft].

**NRC Staff's Evaluation**

The NRC staff reviewed DOE's description of faulting of the subsurface GROA in SAR Section 1.1.5.1, references therein, and responses to RAIs (DOE, 2009ar,as,bg). The NRC staff also used its professional experience and knowledge gained from its own independent field, laboratory, and natural analog studies (Ferrill and Morris, 2001aa; Dunne, et al., 2003aa; Ferrill, et al., 1999ab; Stamatakos, et al., 2000aa; NRC, 2005aa). The NRC staff concludes that the applicant's cross sections covering the entire length and width of the subsurface GROA reasonably depict stratigraphic layering and faults because they are consistent with the NRC staff's independently derived knowledge of the subsurface GROA and are sufficient for the NRC staff's evaluation of the layout and design of emplacement drifts and other underground excavations. The cross sections also represent the ESF, ECRB, the elevation, relative angle of the planned repository underground excavations (i.e., tunnel, ramp, and emplacement drift), and the rock formations within which the excavations would take place.

The NRC staff finds acceptable the applicant's representation and interpretation of the spatial relationship between the major block-bounding faults that influence GROA design because DOE extended the geological cross sections beyond major faults and provided adequate supplemental explanations in response to RAIs (DOE, 2009ar,as,bg). Thus, the NRC staff concludes that DOE provided adequate information on the subsurface GROA structural geology. The NRC staff also finds that DOE's evaluation of block-bounding fault displacement and setback provided in DOE (2009as,bf) is acceptable because the applicant used an appropriate method to estimate the shortest distance from a waste package to a fault or fault zone.

The NRC staff finds adequate both (i) the analysis of fault displacement and rockfall hazard (BSC, 2003aa) the applicant utilized to establish the standard of a 60-m [197-ft] standoff from a Quaternary block-bounding fault with potential for significant displacement (i.e., the Solitario Canyon fault) and (ii) the applicant's stated approach to assess the significance of fault displacement in PCSA. The NRC staff concludes that these approaches are adequate based on the NRC staff's scientific and engineering judgment, and that they will enable DOE to clearly identify active faults that might affect the proposed repository site indirectly by generating earthquakes or directly by causing SSCs located sufficiently close to faults to slip, shear, or tilt, and thus to setback from these potentially active faults. On the basis of the NRC staff's review, as described previously, the NRC staff finds that the applicant provided adequate information on the subsurface GROA structural geology and the site faulting hazard for use in the PCSA and GROA design.

**Fracture Characteristics**

In SAR Section 1.1.5.1.3, the applicant characterized fractures of the rocks in the subsurface GROA. In SAR Section 1.1.5.1.3.3, the applicant described that fractures are found everywhere at Yucca Mountain, except in alluvium. Understanding the fracture characteristics in the different rock formations is important for the orientation, design, and construction of emplacement drifts and other subsurface structures and is also important to the design of ground-support systems (e.g., rock bolts, shotcrete) to stabilize emplacement drifts and ventilation shafts during the preclosure period. The applicant explained fracture formation and assessed its characteristics, including (i) orientation, (ii) dip angle, (iii) length, (iv) spacing, and (v) connectivity.

The applicant considered rockfall (spallation of tunnel wall rock blocks) and drift degradation (major tunnel collapse) to be "fracture hazards" controlled by aspects of the fracture networks measured in different RHH zones. The applicant considered the fracture hazard to drift degradation to be bounded by the hazard from seismic loading conditions, as described in SAR Sections 1.6 and 1.7 and evaluated by the NRC staff in SER Sections 2.1.1.1.3 and 2.1.1.1.4.

The applicant's description of fractures was based on data collected in the ESF and the cross drift (BSC, 2004al; Sweetkind, et al., 1997aa; Mongano, et al., 1999aa). Fractures in the two nonlithophysal zones, which make up approximately 15 percent of the proposed RHH, were formed during the early cooling of the tuffs. These fractures are longer than fractures in the two lithophysal zones, which make up 85 percent of the RHH.

DOE described the fractures in the upper lithophysal zone as having a predominantly north- and northwest-striking tectonic orientation, with spacing that ranges from 0.5 to 3 m [1.6 to 9.8 ft] and lengths of less than 3 m [10 ft]. The lower lithophysal zone—the rock layer proposed to contain most of the waste packages—has a few long fractures but many small fractures less than 1 m [3 ft] long, which are steeply dipping and have a spacing of a few centimeters [few inches]. The applicant characterized the middle nonlithophysal zone as a network of long, relatively closely spaced fractures that the applicant separated into four sets on the basis of orientation: two sets are subvertical with northwest- and northeast-striking orientations, the third set strikes to the northwest with a moderate dip, and the fourth set strikes northwest with a shallow dip.

For fractures in the lower nonlithophysal zone along the ECRB cross drift, the applicant identified three steeply dipping sets, with the most prominent striking northwest. The applicant also identified a northwest-striking, shallowly dipping set among the lower nonlithophysal zone fractures. The applicant indicated high fracture frequencies {19 to 24 fractures per each 3-m [10-ft] interval} in the lower nonlithophysal zone, similar to the intensities in the middle nonlithophysal zone.

The applicant described the zone of influence around faults (DOE, 2009bf). This zone is defined as the region near a fault where fracture intensity is increased or orientation changes. According to the applicant, the intensity of long fractures {greater than 1 m [3.3 ft]} correlates with rock type but not with proximity to faults. However, for shorter fractures, the applicant made four general observations on the zone of influence. First, the width of the zone of influence adjacent to a fault ranges from 1 to 7 m [3.3 to 23 ft]. Second, small displacement faults {1 to 5 m [3.3 to 16 ft]} have narrow zones of influence, whereas larger displacement faults have wider zones. Third, the zone of influence does not correlate with the depth below the ground surface. Fourth, the amount of observed deformation associated with a fault partly depends on the strata. Nonwelded tuffs are characterized by sharp faults and smaller zones of influence. Welded tuffs are characterized by less well-defined faults and relatively larger zones of influence.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on fractures in SAR Section 1.1.5.1, references therein, and associated RAIs. The NRC staff conducted independent analyses of surface fractures at Yucca Mountain (Dunne, et al., 2003aa) and subsurface fracture data for the RHH intervals (Smart, et al., 2006aa). The applicant's description of fracture orientations in the RHH intervals is consistent with the NRC staff's independent analyses, and the NRC staff finds that the applicant's fracture-orientation information is adequate to support its use in other SAR sections, the PCSA, and GROA design. For example, the results of staff fracture analyses (Smart, et al., 2006aa) show that the prevailing fracture orientations are consistent with the applicant's proposed alignment of emplacement drifts within the RHH, as identified in DOE (2009as, Enclosure 4) and SAR Section 1.3.4.2.3.

Fracture spacing and connectivity are considerations for design of ground support systems for safety during operations and are relevant to analyses of drift degradation, rockfall, and seepage (NRC, 2004ab, 2005aa; Ofoegbu, et al., 2007aa). The NRC staff analyses show that the applicant's characterization of fracture networks for use in the GROA design reflected several sampling biases (Smart, et al., 2006aa). The applicant's interpretation of fracture spacing and connectivity may not adequately capture the full range of uncertainties of these parameters. The NRC staff analyses show that the applicant overestimated fracture spacing at Yucca Mountain because of sampling biases (Smart, et al., 2006aa). Therefore, the NRC staff finds the applicant's analysis of fracture spacing and connectivity does not fully address all of the uncertainties in fracture spacing. However, the NRC staff notes that the applicant stated in the SAR that it will confirm the fracture parameter input to the design and performance to reflect actual field observations made during construction of underground openings. In particular, DOE stated in the SAR that rock conditions will be observed as emplacement drift boring is accomplished, including fracture characteristics such as orientation, spacing, length, intensity, and connectivity (PSC-25), in part to ensure that emplacement drifts are constructed nominally parallel with the design azimuth (70–80°) [SAR Table 1.9-9, DCP 01-08 and DCP 01-14; DOE (2009as, Enclosure 4)].

2.1.1.1.3.5.1.2 Geology of the Surface Geologic Repository Operations Area

In SAR Section 1.1.5.1.4, the applicant provided information on the stratigraphy and structural geology of the surface GROA. The surface GROA facilities would be constructed mainly in Midway Valley on alluvium. Characterization of alluvium and rock properties and conditions at the surface GROA are necessary for the design of facilities and their foundations and cut and fill slopes during construction. The applicant also used this information for analyses of potential hazards to the facilities such as earthquakes, surface faulting, landslides, and erosion of and deposition on the surface GROA.

**Stratigraphy of the Surface GROA**

In SAR Section 1.1.5.1.4, the applicant characterized the near-surface stratigraphy using geologic mapping, boreholes, test pits, trenches, and geophysical investigations (BSC, 2002aa; SNL, 2008af). The applicant determined that the surface GROA is underlain by tuff, partly covered with Quaternary-age alluvium, colluvium, and soil. The alluvium thickness varies from zero at the eastern base of Exile Hill to a maximum of approximately 61 m [200 ft] in the middle of Midway Valley.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's stratigraphic information in SAR Section 1.1.5.1 and references therein and finds that the applicant's general description and thicknesses of the bedrock formations and alluvium are adequate because they are consistent with the NRC staff's field observations and independent studies (Waiting, et al., 2007aa; NRC, 1999aa). Further discussion of the properties, variations, and thicknesses of the volcanic rocks and alluvium is provided in SER Section 2.1.1.1.3.5.4. Therefore, the NRC staff concludes that the applicant's information regarding the subsurface GROA structural geology is adequate to provide inputs to design and assess potential seismic and fault displacement hazards in the PCSA.

**Structural Geology of the Surface GROA**

The dominant geologic structural features of the surface GROA are north-south striking normal faults separating dipping rock layers. The applicant stated that Midway Valley is cut by several steeply dipping normal faults interpreted to offset (displace) the bedrock units but not the Quaternary alluvium. Exile Hill, the location of the North Portal, is bounded on the west by the west-dipping Bow Ridge Fault and on the east by the east-dipping Exile Hill fault. A north-northwest-striking, east-dipping fault, the Exile Hill fault splay crosses through the middle of the surface GROA. The Midway Valley fault underlies the northeastern portion of the surface GROA. Displacement on this north-northeast-striking, west-dipping normal fault in Midway Valley is estimated to be 40 to 60 m [131 to 197 ft] on the basis of gravity and magnetic surveys, but bedrock exposures of the fault north of Yucca Wash show 120 m [394 ft] of displacement. On the basis of geophysical data, the applicant also identified several potential additional faults with smaller displacements underneath the surface GROA (BSC, 2002aa; Keefer, et al., 2004aa). These geologic interpretations are depicted on geologic maps and geologic cross sections in SAR Section 1.1.5.1.4, Figures 1.1-64 through 1.1-67.

The applicant revised previous data and relocated the trace of the Bow Ridge Fault by about 100 m [330 ft] to the east and updated fault slip values (Orrell, 2007aa; SAR Figure 1.1-59). This relocation was based on 2006–2007 well-boring data. However, the applicant indicated that the revised fault location and slip information was consistent with previous interpretations because these values fall within the uncertainty ranges used by the applicant to estimate the probabilistic fault displacement and ground motion hazards. The applicant stated that during initial construction activities, the locations, widths, and age of displacement of damage zones from the Bow Ridge Fault, interpreted buried faults, and potential unknown faults in Midway Valley will be evaluated.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's faulting information described in SAR Section 1.1.5, references therein, SAR Figures 1.1-64 through 1.1-67, and responses to RAIs (DOE, 2009at,bg) and found the information to be adequate because the mapping sufficiently encompasses the surface GROA at a detailed scale of 1:12,000 and accurately depicts the topographic and surface geological features in relation to the major surface facilities. The information includes the description of the relocated Bow Ridge Fault (DOE, 2009at,bg). The NRC staff based its conclusion on technical knowledge gained through experience, including direct observations from site visits during the prelicensing period.

The NRC staff finds that the applicant adequately explained how the geological maps and cross sections were developed, as well as their intended use, limitations, assumptions, and associated uncertainties. To address any potential uncertainties in faulting interpretations, the applicant stated that if buried or unknown faults were encountered in the course of excavating for foundations, these faults would be investigated further to define the potential associated hazards (DOE, 2009bf). The NRC staff finds that the applicant's approach for further fault investigation during construction is adequate because it follows standard engineering practice to inform the design of a facility as new geotechnical and geological information becomes available during excavation and drilling activities conducted at the time of construction. Therefore, the NRC staff concludes that the applicant's information regarding the surface GROA structural geology is adequate to provide inputs to design and assess potential seismic and fault displacement hazards in the PCSA.

2.1.1.1.3.5.1.3 Fault Displacement Hazard Assessment

In SAR Section 1.1.5.2.4.1, the applicant described (i) the potential for fault displacement (the relative displacement of bedrock, sediment, and soils on opposite sides of a fault) that might adversely affect the surface and subsurface GROA; (ii) the probability that fault slip will exceed design specifications; and (iii) the expert elicitation process that led to the applicant's assessment. A fault that intersects the surface GROA could displace bedrock, sediment, or soil and thereby damage foundations of surface facilities by shearing or tilting them and disrupting surface drainage and erosion-protection features.  Also, fault displacement is a potential hazard to the subsurface GROA because it could damage or shear drifts or waste packages, trigger rockfall within the drifts and shafts, degrade drift walls and ground-support systems, or degrade other components of the engineered barrier system.

**Probabilistic Fault Displacement Hazard Assessment Methodology**

The applicant conducted a Probabilistic Fault Displacement Hazard Assessment (PFDHA) within the Probabilistic Seismic Hazard Assessment (PSHA) expert elicitation (CRWMS M&O, 1998aa).  The PFDHA relied on the same expert elicitation process as the PSHA; the NRC staff's evaluation of this process is in SER Section 2.1.1.1.3.5.2.  The applicant convened a panel of experts from the PSHA to develop probabilistic fault displacement hazard curves. These PFDHA curves are analogous to seismic hazard curves, in which increasing levels of fault displacements are computed as a function of the annual probability that those displacements will be exceeded.

The process the applicant followed in the PFDHA included an assessment of specific characteristics and uncertainties, including (i) identifying sources of fault displacement; (ii) evaluating the location, frequency, and size of displacements at selected points in the repository; (iii) evaluating displacements as a function of magnitude and distance; and (iv) integrating these data into a hazard curve that depicts possible fault slip as a function of annual exceedance probability.

To conduct the PFDHA, the applicant convened a panel of experts as described in SAR Section 2.2.2.1.1.1.  The expert panel consisted of six 3-member teams of geologists and geophysicists who developed probabilistic distributions to characterize potential fault displacements in the Yucca Mountain region.  The expert elicitation teams used two methods to generate fault displacement hazard curves, as applied in the PFDHA:  the displacement approach and the earthquake approach.  The displacement approach uses fault-specific data, such as cumulative displacement, fault length, paleoseismic measurements from fault trench studies, or data from records of earthquakes correlated with the known seismogenic faults.  The displacement approach relies on direct observational evidence of faulting.  The experts derived fault displacement and displacement probability over time directly from (i) paleoseismic displacement and recurrence rate data, (ii) geologically derived slip rate data, or (iii) scaling relationships that relate displacement to fault length and cumulative fault displacement.

The earthquake approach relates the frequency and magnitude of the faults' slip events to the frequency and magnitude of earthquakes on the seismic sources, as they were defined in the seismic-source models in the PSHA (CRWMS M&O, 1998aa).  The earthquake approach uses earthquake recurrence models from the seismic hazard analysis.  For this approach, the experts assessed three probabilities:  (i) the probability that an earthquake will occur; (ii) the probability that this earthquake will produce surface rupture on the source fault; and (iii) the probability that the earthquake will produce distributed surface displacements.

The probability that an earthquake will occur was derived from the frequency distribution of earthquakes for each source used in the seismic hazard assessment and based on geologic, historical seismic, or paleoseismic data.  The probability of surface rupture was determined by an analysis of historical earthquake and surface rupture data from the Basin and Range Physiographic Provinces and from focal depth calculations.  In the focal depth calculations, the size and shape of the fault rupture for each earthquake was estimated from empirical scaling relationships (e.g., Wells and Coppersmith, 1994aa). Depending on focal depth, the surface displacement (if any) along the fault was determined. The applicant introduced an additional variable that randomized the rupture along the fault length because the maximum surface displacement of a fault may not coincide with the location for which the fault displacement hazard curve is being generated (i.e., the demonstration point, as described in a following subsection). The probability of distributed faulting was determined from Basin and Range historical rupture data in which distributed faulting was mapped after an earthquake (e.g., Pezzopane and Dawson,1996aa) or through slip tendency analysis (Morris, et al., 1996aa).

**NRC Staff's Evaluation**

The NRC staff finds that the PFDHA methodology described in SAR Section 1.1.5.2 and references therein is acceptable for developing reasonable hazard estimates because it was derived from a formal expert elicitation that followed the guidance in NUREG–1563 (NRC, 1996aa). In addition, based on the NRC staff's observation of DOE's elicitation workshops and review of materials produced during the PFDHA, the NRC staff finds that the applicant's process was acceptable. The NRC staff also has extensive knowledge gained through experience evaluating geological evidence for recurrence and slip rates of faults that the applicant considered in the PFDHA (e.g., Stamatakos, et al., 2003aa). Based on this knowledge, the NRC staff concludes that the PFDHA captured the current scientific understanding of probabilistic fault displacement analyses and that the results represent the center, body, and range of viable interpretations, including uncertainty. The detailed NRC staff review of expert elicitation, as used by the applicant in the PFDHA, is provided in SER Section 2.5.4.

**Input Data and Interpretations**

The applicant's PFDHA integrated two data types: (i) faulting activity on mapped faults, defined by historic earthquakes or measured Quaternary fault displacements and (ii) faulting activity that may occur on unmapped faults or newly developed faults based on an assessment of the overall tectonic setting, *in-situ* stresses in the rock mass, geomechanical properties of the rock mass, and regional estimates of crustal strain. The applicant analyzed 100 earthquakes in the Basin and Range region to determine the relationships among the amounts and patterns of both principal and distributed fault displacements, the minimum magnitude at which an earthquake may produce surface faulting, and the maximum magnitude at which an earthquake does not displace the surface.

For the largest mapped faults at Yucca Mountain, the PFDHA curves were based on the same detailed paleoseismic and earthquake data used to characterize these faults as potential seismic sources in the PSHA. The expert elicitation relied on both anecdotal evidence and expert judgment to develop conceptual models of distributed faulting and to estimate the probabilities of secondary faulting along smaller faults and fractures in the repository (Youngs, et al., 2003aa; CRWMS M&O, 1998aa).

The applicant chose nine sites in and near Yucca Mountain as demonstration sites for the application of the PFDHA, as shown in SAR Table 1.1-67. These demonstration sites were selected to represent a range of faulting and related fault deformation conditions in the subsurface and near the proposed surface facility sites in the GROA, including large block bounding faults such as the Solitario Canyon and Bow Ridge Faults, smaller mapped faults within the repository footprint such as the Ghost Dance fault, unmapped minor faults near the larger faults, fractured tuff, and intact tuff. Individual PFDHA curves were developed to characterize fault displacements at each of the nine demonstration sites. Fault displacement curves for several of the nine demonstration sites are provided in SAR Figure 2.2-13.

Results of the PFDHA (CRWMS M&O, 1998aa) show that, except for the Bow Ridge and Solitario Canyon faults, mean fault displacements are estimated to be less than 1 m [3.28 ft] over the next 10 million years (SAR Table 2.2-15). Mean displacements for the demonstration sites within the current repository footprint [demonstration sites (v), (vii), and (viii)] do not exceed 0.40 m [1.3 ft] in 10 million years. For a 10,000-year period, mean displacements are calculated to be less than 0.01 m [0.03 ft] for all 9 demonstration sites (SAR Table 1.1-67).

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's information regarding site characterization input data and interpretations to the PFDHA in SAR Section 1.1.5.1, references therein, and supporting documents. The NRC staff also conducted independent analyses using slip tendency (Morris, et al., 1996aa) of faults within the Yucca Mountain region (Morris, et al., 2004aa). Based on the NRC staff's professional experience and knowledge gained from its own independent field, laboratory, and natural analog studies, the NRC staff finds that the input data to the PFDHA and the resulting interpretations are acceptable because (i) the broad collection of geological and seismological information allowed interpretations about fault displacement in the scientific community to be evaluated by the panel experts (the NRC staff evaluates DOE's expert elicitation process in SER Section 2.5.4), (ii) the fault displacement and earthquake approaches used by the experts to interpret the data and develop the fault displacement curves are acceptable because they are consistent with seismological theory and supported by geological observations, and (iii) the interpretations made by the expert panel are consistent with the NRC staff's independent evaluations of faulting (Ferrill and Morris, 2001aa; Dunne, et al., 2003aa; Ferrill, et al., 1999ab; Stamatakos, et al., 2000aa; NRC, 2005aa). On the basis of the NRC staff's review, the NRC staff finds acceptable DOE's methodology, input data, and probabilistic fault displacement hazard analyses. In particular, the NRC staff finds that the probabilistic estimates provided by the applicant in SAR Table 2.2-15 and SAR Table 1.1-67 are acceptable.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant's information on the site geologic setting, stratigraphy, and structural geology of the surface and subsurface at the Yucca Mountain Site is acceptable. The geologic information is based on adequate field characterizations and test results of the correct rock and stratigraphic layers where construction may take place and adequately represents features such as faulting and fracturing that may be important in the design of GROA facilities. The NRC staff finds the methodology, input data, and interpretations of the PFDHA to be adequate. Therefore, NRC staff finds, with reasonable assurance, that the information on surface and subsurface geology is adequate for use in the PCSA, supports the GROA design, and satisfies 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c), with respect to site geology.

2.1.1.1.3.5.2 Seismology

DOE investigated the geological, geophysical, and seismic characteristics of the Yucca Mountain region to obtain sufficient information to estimate how the site would respond to vibratory ground motions from earthquakes. In SAR Section 1.1.5.2, the applicant provided its description of site seismology. The applicant described its analysis of potential seismic hazards in SAR Section 1.1.5.2.4, the overall approach to developing a seismic hazard assessment for Yucca Mountain in SAR Section 2.2.2.1, and the conditioning (adaption or modification) of the ground motion hazard for seismic design at Yucca Mountain in SAR Section 1.1.5.2.5.1. Additional information was provided in DOE (2009ab, Enclosure 19), DOE (2009aq, Enclosures 6, 7, and 8), and references therein.

DOE's overall approach to developing a seismic hazard assessment for Yucca Mountain, including fault displacement hazards, as described in SAR Section 2.2.2.1, involved the following three steps:

1. Conducting an expert elicitation in the late 1990s to develop a PSHA for Yucca Mountain (CRWMS M&O, 1998aa). This assessment included a PFDHA that is discussed in SER Section 2.1.1.1.3.5.5. The PSHA was developed for a reference bedrock outcrop, specified as a free-field site condition with a mean shear wave velocity ($V_S$) of 1,900 m/sec [6,233 ft/sec] and located adjacent to Yucca Mountain. This value was derived from a $V_S$ profile of Yucca Mountain with the top 300 m [984 ft] of tuff and alluvium removed, as provided in Schneider, et al. (1996aa, Section 5).

2. Conditioning PSHA ground motion results to constrain the large low-probability ground motions to ground motion levels that, according to the applicant, are more consistent with observed geologic and seismic conditions at Yucca Mountain, as provided in BSC (2005aj, ACN02).

3. Modifying the conditioned PSHA results, using site-response modeling, to account for site-specific rock material properties of the tuff in and beneath the emplacement drifts and the site-specific rock and soil material properties of the strata beneath the GROA.

The applicant applied these three steps for seismic hazard assessment for preclosure seismic design and safety analyses as well as for its postclosure performance assessment. Moreover, many of the geological and geophysical data, conceptual and process models, and supporting technical analyses underpinning the applicant's conclusions in the SAR are common to the preclosure seismic design and safety analyses as well as the applicant's postclosure performance assessment calculations.

The first two steps described here are evaluated in this subsection of the SER. The third step involving site response modeling is evaluated in SER Section 2.1.1.1.3.5.3.

This three-step process to develop seismic inputs for the PCSA and preclosure seismic design involves a series of sub-steps that are described in this section. The process is linear, in that the outputs from earlier steps are used as inputs in subsequent steps. For this review, an intermediate evaluation is provided for each of these sub-steps regarding the adequacy of the applicant's results and methods. These intermediate evaluations are then consolidated in the NRC staff's conclusion.

2.1.1.1.3.5.2.1 Probabilistic Seismic Hazard Analysis (PSHA)

**Methodology**

The applicant conducted an expert elicitation on PSHA in the late 1990s (CRWMS M&O, 1998aa) on the basis of the methodology described in the Yucca Mountain Site Characterization

Project (DOE, 1997aa). The applicant stated that its PSHA methodology followed the guidance for expert elicitation described in the DOE-NRC-Electric Power Research Institute (EPRI)-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa).

To conduct the PSHA, the applicant convened two panels of experts as described in SAR Section 2.2.2.1.1.1. The first expert panel consisted of six 3-member teams of geologists and geophysicists (seismic source teams) who developed probabilistic distributions to characterize potential seismic sources in the Yucca Mountain region. These distributions included location and activity rates for fault sources, spatial distributions and activity rates for background sources, distributions of earthquake moment magnitude and maximum magnitude, and site-to-source distances. The second panel consisted of seven seismology experts (ground motion experts) who developed probabilistic point estimates of ground motion for a suite of earthquake magnitudes, distances, fault geometries, and faulting styles. These point estimates incorporated randomness and uncertainties that were specific to the regional crustal conditions of the western Basin and Range. The ground motion attenuation point estimates were then fitted to yield the ground motion attenuation equations used in the PSHA. The two expert panels were supported by technical teams from the applicant: (i) the U.S. Geological Survey and (ii) Risk Engineering, Inc. (1998aa). Both organizations provided the experts with relevant data and information; facilitated the formal elicitation, including a series of workshops designed to accomplish the elicitation process; and integrated the hazard results.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's PSHA methodology described in SAR Sections 1.1.5.2.4 and 2.2.2.1.1, references therein, and responses to RAIs, using the guidance provided in NUREG–1563. The NRC staff finds that the applicant's PSHA included the four basic elements of an expert elicitation, as described in Budnitz, et al. (1997aa). These are (i) identification of seismic sources, such as active faults or seismic zones; (ii) characterization of each of the seismic sources in terms of their activity, recurrence rates for various earthquake magnitudes, and maximum magnitude; (iii) ground motion attenuation relationships to model the distribution of ground motions that will be experienced at the site when a given magnitude earthquake occurs at a particular source; and (iv) incorporation of the inputs into a logic tree to integrate the seismic source characterization and ground motion attenuation relationships, including associated uncertainties. In this methodology, each logic tree pathway represents one expert's weighted interpretations of the seismic hazard at the site. The computation of the hazard for all possible pathways results in a distribution of hazard curves that is representative of the seismic hazard at a site, including variability and uncertainty.

In addition, the NRC staff observed all expert elicitation meetings and reviewed summary reports of those meetings as they were produced. On the basis of these reviews, including the evaluation with respect to Budnitz, et al. (1997aa) and the NRC staff's direct observations of the expert elicitation process, the NRC staff concludes that the applicant's elicitation for the PSHA is consistent with the implementation guidance for conducting an expert elicitation described in NUREG–1563. Therefore, the NRC staff finds that the applicant's implementation of the PSHA expert elicitation is adequate to develop estimates of seismic hazards for use in the PCSA and GROA design. The NRC staff notes that the applicant's PSHA methodology is also consistent with updated NRC guidance on how to implement the Senior Seismic Hazard Assessment Committee (SSHAC) guidelines in NUREG–2117 (NRC, 2012aa), as described in the NRC staff's evaluation of the DOE PSHA expert elicitation provided in SER Section 2.5.

**Input Data and Interpretations**

During the expert elicitation, the applicant's seismic source teams considered a range of information from many resources, including the applicant, the U.S. Geological Survey, project-specific Yucca Mountain studies, and information published in the scientific literature. This information is presented in SAR Sections 1.1.5.2 and 2.2.2.1.1 and includes Figures 1.1-68 through 1.1-94 and Table 1.1-65. This information included (i) data and models for the geologic setting; (ii) seismic sources and seismic source characterization, including earthquake recurrence and maximum magnitude; (iii) historical and instrumented seismicity, as outlined in CRWMS M&O Appendix G (1998aa); (iv) paleoseismic data (Keefer, et al., 2004aa); and (v) ground motion attenuation (e.g., Spudich, et al., 1999aa). The applicant also supported the PSHA with a broad range of data, process models, empirical models, and seismological wave propagation theory (CRWMS M&O, 1998aa). The expert panels built their respective inputs to the PSHA on the basis of this information and information they received during the elicitation meetings (CRWMS M&O, 1998aa). The resulting set of hazard curves were intended to provide the applicant with sufficient representation of the seismic hazard for use in the PCSA and GROA design.

The applicant expressed the PSHA curves in increasing levels of ground motion as a function of the annual probability that the ground motion will be exceeded. These curves are developed for the bedrock conditions with a mean $V_S$ of 1,900 m/sec [6,233 ft/sec] located adjacent to Yucca Mountain, as described previously in this section, and they include estimates of uncertainty (see SAR Figure 1.1-74 for an example of one of the applicant's seismic hazard curves). The SAR provided PSHA results on horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and peak ground velocity (PGV).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's PSHA input data and interpretations, as described in SAR Sections 1.1.5.2 and 2.2.2.1.1, references therein, and responses to RAIs. The NRC staff concludes that the applicant adequately developed the geological, geophysical, and seismological information necessary to support the expert elicitation. This conclusion is based in part on the NRC staff's evaluations in NUREG–1762 (NRC, 2005aa), where the NRC staff found that the applicant's information was consistent with site conditions at Yucca Mountain. This conclusion is also based on the NRC staff's first-hand knowledge of the geology and seismic characteristics of the Yucca Mountain region, which includes more than a decade of independent geological and geophysical research and study (e.g., Ferrill, et al., 1996aa,ab; Stamatakos, et al., 1998aa; Waiting, et al., 2003aa; Gray, et al., 2005aa; Biswas and Stamatakos, 2007aa). The NRC staff also finds that the resulting suite of ground motion hazard curves; horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and PGV are adequate because they are consistent with NRC guidance in Regulatory Guide 3.73 (NRC, 2003ae) and Regulatory Guide 1.208 (NRC, 2007ah). Although these regulatory guides were developed for other types of NRC-regulated facilities (e.g., nuclear power plants and interim spent fuel storage facilities), they are applicable here because (i) the seismic hazard assessment is independent of the type of potentially affected facility and (ii) the methodologies and conclusions in these Regulatory Guides are generally applicable to analogous activities proposed for the GROA.

The NRC staff also reviewed additional geological, geophysical, and seismological information in Wernicke, et al., (2004aa) and Hanks, et al., (2013aa), which were developed after the DOE

PSHA elicitation was performed. Wernicke, et al., (2004aa) provided updates to the Global Positioning Satellite (GPS) data for Yucca Mountain to include data from a continuously operating network. These results showed that the anomalously large crustal strain rates indicated by GPS results (Wernicke, et al., 1998aa) considered in the PSHA were in part transient strains associated with the 1992 Little Skull Mountain earthquake and not indicative of increased seismic hazard at the site. Results in Hanks, et al. (2013aa) are based on two studies: one on the physical limits of ground velocities that the lithology at Yucca Mountain could have experienced since deposition, based on the physical limits of rock strength, and a second detailed analysis of the age, distribution, and geometries of precariously balanced rocks along the steep hill slopes in the Yucca Mountain region. Both the physical limits and precarious rock studies in Hanks, et al. (2013aa) suggest upper limits on the amplitudes of earthquake ground motions that occurred in the geological past at Yucca Mountain. These results, thereby, constrain the upper limits of the PSHA at low annual exceedance probabilities and suggest that extremely large ground motions at low annual exceedance probabilities in the DOE PHSA are conservative. These new results, therefore, further support the NRC staff's conclusion that DOE's probabilistic seismic hazard analysis input data and interpretations are adequate. On the basis of its detailed understanding of the Yucca Mountain geology, the NRC staff concludes that new geological and seismological information would not substantially alter the PSHA results, with the exception of over estimation of ground motions at low annual exceedance probabilities, which is described in the following section regarding conditioning of low probability ground motions.

**Conditioning of Ground Motion Hazard**

DOE provided in SAR Section 1.1.5.2.5.1 the conditioning of ground motion hazard at the reference bedrock outcrop where the PSHA was developed. Since completion of the PSHA in 1998, several studies and reports, including ones from the NRC staff (NRC, 1999aa), the Nuclear Waste Technical Review Board Panel on Natural System and Panel on Engineered Systems (Coraddini, 2003aa), and DOE itself (BSC, 2004bj) questioned whether the very large ground motions the PSHA predicted at low annual exceedance probabilities (below ~$10^{-6}$/yr) were physically realistic. DOE stated that these ground motion values are well beyond the limits of existing earthquake accelerations and velocities from even the largest recorded earthquakes worldwide. They are deemed physically unrealizable because they require a combination of earthquake stress drop, rock strain, and fault rupture propagation that cannot be sustained without wholesale fracturing of the bedrock, which is not observed at Yucca Mountain (Kana, et al., 1991aa).

For Yucca Mountain, however, the seismic hazard curves were extrapolated to estimate ground motions with annual exceedance probabilities as low as $10^{-8}$ (SAR Section 1.1.5.2.5.1). At these low probabilities, the seismic hazard estimates are driven by the tails of the untruncated Gaussian distributions (the tail is not defined by the data, but by the assumed distribution) of the input ground motion attenuation models (Bommer, et al., 2004aa). As Anderson and Brune (1999aa) pointed out, overestimates of the hazards may also arise because of the way in which uncertainty in ground motion attenuation from empirical observations or theory is distributed between aleatory and epistemic uncertainties.

To account for these large ground motions, DOE modified or conditioned the hazard using both a shear-strain-threshold approach and an extreme-stress-drop approach, as described in SAR Section 1.1.5.2.5.1. The applicant used these two independent methods for conditioning the PSHA results to make the seismic hazards consistent with the geologic setting of Yucca Mountain. The first method in the SAR used geological observations at the repository level to develop a limiting distribution on shear strains experienced at Yucca Mountain (BSC, 2005aj). The second method in the SAR used expert judgment (BSC, 2008bl) to develop a distribution of extreme stress drop in the Yucca Mountain vicinity. The distribution is based on available data (stress drop measurements and apparent stress drops from laboratory experiments) and interpretations. As discussed in SAR Section 1.1.5.2.5.1 and BSC (2008bl), the applicant conducted conditioning using the shear-strain-threshold and extreme-stress-drop approaches in series (the combined conditioning) because these two methods are independent.

Rather than reconvene the PSHA expert elicitation and redo the hazard analysis, DOE chose to treat the issue as part of the ground response analysis. Accordingly, DOE's second step in developing ground motion inputs for analyses, after the development of PSHA, was to condition the ground motion hazard. This second step in the three-step DOE process includes information on the level of extreme ground motion that is consistent with the geological setting of Yucca Mountain. Conditioning of the ground motion hazard is a unique study developed for the Yucca Mountain project.

The unconditioned hazard curve DOE developed, which is the annual probability of exceedance (APE) as a function of ground motion, is convolved with the distribution of extreme ground motion for the reference bedrock outcrop to produce the conditioned ground motion hazard of the same bedrock outcrop. The impact of conditioning at higher probabilities is less significant and increases as the probability of exceedance decreases (i.e., annual probabilities of exceedance of $10^{-5}$, $10^{-6}$, $10^{-7}$, and $10^{-8}$) (SAR Section 1.1.5.2.5.1). SAR Figures 1.1-79 and 1.1-80 compared the unconditioned and conditioned peak ground accelerations (PGAs) and PGV mean hazard curves for the reference bedrock outcrop.

For the extreme-stress-drop approach, BSC (2008bl, Appendix A) outlined the workshops, which included presentations, discussions, and assessments that were conducted to develop the expert judgment. The stress-drop data from the United States and other countries were used in the expert elicitation. The parameter variability involved in the empirical ground motion attenuation relationship and numerical simulations of ground motions that the experts relied on was included in the conditioning. Variability in velocity profile, stress drop, source depth, and kappa (the site- and distance-dependent parameter representing the effect of intrinsic attenuation of the wave field as it propagates through the crust from source to the receiver) were considered in the modeling to map the stress drop into ground motion distribution.

In response to the NRC staff's RAIs (DOE, 2009aq), DOE provided information explaining its application of the two methods in series where the output of the extreme-stress-drop conditioning becomes the input for the shear-strain-threshold approach. In the RAI responses, DOE also clarified and updated the formulations for the two conditioning methods, as described in BSC (2008bl, Appendix A).

**NRC Staff's Review**

The NRC staff reviewed the applicant's methods for conditioning PSHA results in SAR Section 1.1.5.2.5.1 and the applicant's responses to the NRC staff's RAIs (DOE, 2009aq) to evaluate whether the applicant's two independent conditioning methods are adequate. The NRC staff finds that the shear-strain-threshold approach is adequate because it follows appropriate mechanical, material, and seismological principles and is based on laboratory rock mechanics data and corroborated by numerical modeling.

The NRC staff finds that the extreme stress drop method is adequate because it is supported by observations from worldwide earthquake recordings (SAR Section 1.1.5.2.5.1). These earthquake observations were used by the applicant's experts to develop limits on stress drop.

The NRC staff also finds acceptable that the DOE applied these two methods in series because, as DOE described in SAR Section 1.1.5.2.5.1 and its RAI response (DOE, 2009aq), they are independent from each other. The NRC staff, therefore, concludes that applying both in series would not duplicate or double count their respective effects on conditioning the hazard curve. Moreover, the NRC staff notes that the shear-strain-threshold approach has less of an effect on reducing the hazard as compared to the extreme-stress-drop approach. For example, for an APE of $1 \times 10^{-8}$, the shear-strain-threshold conditioned PGV hazard is reduced from 1,200 cm/sec to about 1,100 cm/sec [472 to 433 in/sec] or about 10 percent; the stress-drop-conditioned PGV hazard is reduced from 1,200 cm/sec to about 480 cm/sec [472 to 189 in/sec] or about 60 percent, as identified in BSC (2008bl, Section A4.5.1).

The NRC staff also finds that the final conditioned ground motion levels at very low APE are conservative when compared with the observed worldwide strong motion data, which include records from earthquakes much stronger than those expected in the Yucca Mountain region. DOE assumed that stress drops from earthquakes in the Yucca Mountain region will remain consistent during the next 1 million years. The NRC staff finds that the applicant's assumption that the tectonic setting and therefore the stress drops of earthquakes from the existing faults at Yucca Mountain are not going to change significantly in the next 1 million years is also reasonable on the basis of the NRC staff's understanding of the seismotectonic history of the Yucca Mountain region (NRC, 2005aa).

2.1.1.1.3.5.2.2 Seismic Site Response Modeling

The applicant provided information in SAR Section 1.1.5.2.5.2 on how the surface and subsurface GROA might behave if the site was subjected to seismic loads. Seismic site response modeling is the last step in the development of seismic inputs for preclosure seismic design and PCSA.

To address the effects of earthquakes at the site for the preclosure period, the applicant provided information in the following areas: (i) site-response modeling methodology; (ii) geophysical information to develop compression wave velocity ($V_P$), shear wave velocity ($V_S$), and density profiles; (iii) geotechnical information used to develop dynamic material properties; and (iv) development of seismic design inputs. The NRC staff's review of the applicant's information and analyses within these four topical areas follows.

**Overall Approach to Site-Response Modeling**

In SAR Section 1.1.5.2.5.2, the applicant discussed how the various types and thicknesses of rocks, alluvium, and soils that comprise the GROA and the site would likely respond to earthquake ground motions. The results of site-response modeling included quantifying the amplification or damping factor of ground motion at or near the location of SSCs and determining any vertical-to-horizontal motion ratio variance from place to place (factors and ratios are important to the design of earthquake-resistant facilities). The applicant used the site-specific ground motion curves that are consistent with the conditioned PSHA ground motion hazard curves, which staff have reviewed in the previous SER section and found to be acceptable.

The applicant used Approach 3 from NUREG/CR–6728 (McGuire, et al., 2001aa) for its site-response modeling used to develop hazard-consistent, site-specific ground motion spectra (the spectra consistent with the annual probability of exceedance). In this approach, the site-specific hazards are calculated using rock hazards and the site-response modeling (see SER Section 2.1.1.1.3.5.3.1). Two frequency ranges (1–2 and 5–10 Hz) are covered in this approach to accommodate the magnitude distributions of design earthquakes. In Approach 3, the results are averaged to take into account model uncertainty in the site-response inputs (SAR Section 1.1.5.2.5.3).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's overall approach to site-response modeling presented in SAR Section 1.1.5.2.5.2 and references therein using the guidance in NUREG/CR–6728. There are five approaches (1, 2A, 2B, 3, and 4) described in NUREG/CR–6728 to conduct this analysis. The approaches are each applicable under certain circumstances, according to available data and information. The site-specific data and information needed increase with each successive approach, and the resulting analyses yield increasing levels of accuracy. Approach 4 requires site-specific soil attenuation relations that are based on detailed observations of earthquake data, which were not available for Yucca Mountain. The applicant adopted Approach 3 from NUREG/CR–6728 for preclosure site-response analyses, which the staff concludes is adequate because it is one of the approaches recommended in NUREG/CR–6728 for site-response modeling. The NRC staff concludes that Approach 3 is also adequate because the two frequency ranges (1–2 and 5–10 Hz) used in the calculations of input control motions conform to NRC guidance provided in Regulatory Guide 1.208 (NRC, 2007ah). The use of these frequency ranges for input control motions is also consistent with Regulatory Guide 1.208.

**RVT-Based Point-Source Equivalent-Linear Site-Response Modeling**

The applicant relied on a Random Vibration Theory (RVT)-based point-source equivalent-linear site-response model to perform the site-response calculation in the adopted Approach 3 discussed above. The RVT-based point-source model produces amplification factor transfer functions, which model the nonlinear amplification behavior of the site tuff and alluvium (BSC, 2004aj, 2008bl). This is described in the applicant's ground motion report, BSC (2004aj, Section 6.1.1). The important aspects of this model, which the applicant validated, are (i) description of the earthquake source (point source v. finite source), (ii) assumed behavior of the rock and soil (equivalent linear v. nonlinear), and (iii) dimensionality of the model (one, two, or three dimensions).

**NRC Staff's Evaluation**

The NRC staff reviewed the RVT-based point-source model combined with one-dimensional equivalent-linear site-response in SAR Section 1.1.5.2, references therein, and supporting documentation (BSC, 2004aj, 2008bl). The NRC staff review focused on the applicability and accuracy of the model to develop earthquake ground-motion input for the PCSA and GROA design. The applicant established the applicability of this site-response model for developing the ground motions for preclosure at Yucca Mountain on the basis of prior published studies and well-documented validations that compare the model's predictions with observed data and alternative models (such as the nonlinear and two-dimensional models). The simplification and approximation of the model the applicant made included choosing point source over finite source, stochastic over deterministic for the source modeling, and one-dimensional over two-dimensional or three-dimensional equivalent-linear over nonlinear for the site-response modeling.  The NRC staff concludes that the applicant adequately justified these simplifications and approximations through validation results, which showed the model predictions having near-zero bias and low variability compared with observations (BSC, 2004aj; 2008bl).  The model parameter uncertainties and the geotechnical data, such as the material dynamic properties' uncertainties and the velocity profiles, were adequately incorporated in the model. This modeling approach has been well tested and validated by the seismic community for decades and has been adopted by the NRC for reevaluating the site responses at all US power plants following the accident at the Fukushima Da'ichi nuclear power plant in Japan (EPRI, 2013aa).

In addition, the NRC staff conducted independent calculations using velocity profiles and material properties similar to the applicant's with the software package SHAKE2000 (Ordonez, 2006aa), which is also a one-dimensional equivalent linear model, to calculate the amplification factors between the output surface ground motion and the input outcrop ground motion.  The NRC staff calculations (Gonzalez, et al., 2004aa) are consistent with the applicant's results shown in BSC (2008bl, Figures 6.5.2-1a to 3d).

The NRC staff concludes that the applicant provided evidence from independent researchers (BSC, 2008bl) that strong two- and three-dimensional effects are not significant at Yucca Mountain.  Two- and three-dimensional effects arise when deep alluvial basins are present and the seismic sources are dominated by low-frequency ($\leq$ 0.5 Hz) energy. Geotechnical data collected at Yucca Mountain (BSC, 2002aa) do not show evidence for these conditions.  Therefore, the NRC staff finds that the one-dimensional RVT-based model is adequate for modeling the Yucca Mountain site.

#### 2.1.1.1.3.5.2.3 Geophysical Information to Develop Compression Wave Velocity, Shear Wave Velocity, and Density Profiles

As part of site characterization activities, the applicant collected geotechnical and geophysical data across the GROA and in the repository block (the tilted section of welded and nonwelded tuff situated between the Solitario Canyon and Bow Ridge Faults).  These data, described in SAR Section 1.1.5.3, were used to develop necessary inputs for the seismic site-response modeling.  The applicant's information included the following:  (i) depth to the alluvium-tuff contact; (ii) subsurface configuration of volcanic strata and subsurface location of faults; (iii) $V_S$ and $V_P$ profiles; (iv) density; and (v) dynamic material properties (shear modulus and damping ratios) obtained from geophysical measurements in boreholes, surface geophysical measurements, and dynamic laboratory testing from combined resonant column and torsional shear experiments.  These geotechnical properties influence how the seismic energy is attenuated or amplified through the soil and near-subsurface strata at the site.  In SAR Section 1.1.5.2.7.2, the applicant described the methodology and site characterization studies used to develop this information.

The applicant collected data from 89 exploratory boreholes and surface wave survey lines across the site to develop depth to the base of the alluvium and the $V_S$, $V_P$, and density profiles for the surface GROA.  The applicant used several standard methods to obtain the data: (i) conventional downhole logs, including gamma ray logs to obtain density information; (ii) downhole suspension surveys; and (iii) spectral analysis of the surface wave (SASW) profiles.  Data collection can be organized within three periods of data collection activities: (i) prior to 2005, (ii) the 2005–2006 campaign, and (iii) the 2006–2007 campaign.  These three campaign periods reflect additional data needs associated with revisions of the GROA design during the prelicensing period.

The NRC staff organized its review as follows: (i) evaluation of the applicant's alluvium thickness calculations, (ii) $V_S$ of the subsurface strata, (iii) primary wave velocities of the subsurface strata, and (iv) density profiles. These four properties of the bedrock and alluvium are used in the applicant's one-dimensional site-response models. The NRC staff's review of the dynamic material property information is provided in SER Section 2.1.1.1.3.5.3.3.

**Alluvium Thickness Calculations**

The applicant identified alluvium thickness as an important factor in developing a suite of representative profiles used in its one-dimensional site-response models. The applicant developed a contour map of the depth to the alluvium-tuff contact (SAR Figure 1.1-130) on the basis of data from the boreholes drilled during the pre-2005 and 2005–2006 campaigns, as well as data from 23 of the 43 boreholes from the 2006–2007 campaign. The applicant provided the information from all 43 boreholes drilled during the 2006–2007 campaign in DOE (2009ap).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on alluvium thickness provided in SAR Section 1.1.5, references therein, and RAI responses (DOE, 2009ap). Alluvial thickness is important because of the strong impedance contrast between alluvium and bedrock and because the overall thickness of the alluvium has the greatest influence on surface ground motion calculated by the site response. The staff review included checking the modeled alluvium thicknesses in the contour map (SAR Figure 1.1-130) against recorded alluvium thickness from selected borehole logs. Additionally, the NRC staff independently evaluated alluvium thicknesses using information provided by the applicant (DOE, 2009ap) from the 20 boreholes from the 2006–2007 campaign that the applicant did not use to develop the contour map. The NRC staff finds that the contour map adequately represents the observed alluvium thickness for most of the GROA surface facility sites. Discrepancies between the applicant's alluvium thickness model and observations of alluvium thickness from the borehole data can be as large as 12 m [40 ft]. For example, the observed thickness of alluvium in borehole RF 94, as indicated in the information provided in DOE (2009ap), was 43 m [141 ft], but the location of this well on SAR Figure 1.1-130 indicated an alluvium thickness of more than 55 m [180 ft]. However, as discussed next in the NRC staff's evaluation of the applicant's seismic velocity profiles for the surface GROA, these uncertainties in alluvium thickness are bounded by the applicant's representative base case $V_S$ profiles, which included profiles with as much as 61 m [200 ft] of alluvium, 6 m [20 ft] thicker than those indicated on the applicant's alluvium thickness map. By developing the model with the thickest possible alluvium, in this case 61 m [200 ft], the applicant derived the site-response amplifications that bound site-response values compared to models with less conservative alluvium thicknesses (the thicker the alluvium, the greater the amplification of seismic energy). Thus, the NRC staff concludes that the applicant has adequately included information on alluvium thickness to develop representative soil profiles sufficient for use in its one-dimensional site-response models.

**Shear Wave Velocity**

The applicant described how $V_S$ profiles were obtained from a range of techniques, including SASW, downhole seismic velocity surveys, suspension logging surveys, sonic velocity logging, and vertical seismic profiling in SAR Section 1.1.5.3.1.3.1. Of these, the applicant relied on the borehole and SASW methods to develop profiles for site-response models because the borehole-based techniques provided reliable information on velocities in the immediate vicinity of the borehole. SASW surveys complemented the borehole-based measurements and provided information on the average $V_S$ over a larger volume of the subsurface.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's information in SAR Section 1.1.5.3 and references therein, including the application of the SASW methodology, which was used by the applicant to acquire much of the $V_S$ data used in the site-response calculations. $V_S$ velocity profiles are important components of the site-response models because they define the acoustic impedance contrasts between strata layers. Larger acoustic impedance contrast between the strata layers causes the seismic wave amplitude to change as it passes through the strata. The NRC staff also reviewed the applicant's site data and information collected prior to 2005, as documented in Gonzalez, et al. (2004aa). The NRC staff finds the applicant's use of the SASW methodology adequate for the following reasons. The SASW method has yielded similar results when compared to conventional downhole testing at numerous sites (e.g., Brown, et al., 2002aa). The NRC staff's comparisons (e.g., Gonzalez, et al., 2004aa) of the downhole and SASW measurements at Yucca Mountain show they are consistent with each other (within one-sigma statistical measurement uncertainties). Moreover, the number and spatial distribution of SASW profiles, supported by borehole information, cover the entire area of the GROA, the crest of Yucca Mountain, and the ESF and cross drift, which the NRC staff finds sufficient to characterize the full range of $V_S$ for the site. Because the NRC staff finds the methods the applicant used and the spatial coverage to be adequate, the NRC staff concludes that the applicant has collected sufficient information on the $V_S$ of the rocks and alluvium at Yucca Mountain to develop adequate site-response models.

**Compression Waves**

The applicant described development of $V_P$ information used for its site-response models in SAR Section 1.1.5.2.7.2 and in the applicant's supplemental ground motion input document (BSC, 2008bl). According to the applicant, $V_P$ values were developed from a combination of direct measurements and derived values on the basis of $V_S$ and Poisson's ratio. Initial measurements of $V_P$ were made in the 15 boreholes drilled in 2000 and 2001. These $V_P$ values were then used with $V_S$ from the same boreholes to generate smoothed Poisson ratio curves. These smoothed Poisson ratio curves were extrapolated to greater depths on the basis of vertical seismic profiling data. The smoothed and extrapolated Poisson ratio curves were then combined with $V_S$ profiles to recompute the $V_P$ profiles. These recomputed $V_P$ profiles were used in the site response analysis and to support average Poisson ratio values for the Calico Hills Formation and Prow Pass Tuff. In addition, the applicant developed sensitivity analyses (DOE, 2009aq), which showed that the seismic hazard at the surface GROA and in the repository are insensitive to uncertainties in both Poisson's ratio and $V_P$.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's information in SAR Section 1.1.5.2.7.2, references therein, and supporting documentation, including DOE (2009aq). The applicant made direct measurements of $V_P$ at 15 boreholes and calculated interpolated $V_P$ values in combination with other geotechnical information (e.g., Poisson's ratio). The NRC staff finds that the use of interpolated $V_P$ values is appropriate because of the well-established theoretical relationships between $V_S$, $V_P$, and Poisson's ratio and the lack of hazard sensitivity to uncertainties in both Poisson's ratio and $V_P$. Therefore, the NRC staff concludes that DOE has developed adequate information on $V_P$ of the rocks and alluvium at Yucca Mountain to develop adequate site-response models.

**Density**

The applicant provided information on bulk density of rocks and alluvium beneath the surface GROA in SAR Section 1.1.5.3.2.3, and in BSC (2002aa). The applicant determined the bulk density of the rocks in the repository and alluvium beneath the surface GROA using both field gamma-gamma measurements (a type of geophysical assay) and laboratory measurements from core samples.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's use of gamma-gamma measurements and core samples provided in SAR Section 1.1.5.3.2.3 and in BSC (2002aa) to determine bulk density of the rocks in the repository and alluvium beneath the surface GROA. The NRC staff finds that the bulk density is important because it influences the site-response modeling, especially damping of the seismic energy. The NRC staff finds that the measurements and core samples are acceptable because they follow standard industry practice. The NRC staff compared the applicant's site data and information, as described in SAR Section 1.1.5.3.2.3, with more recent measurements of density for core samples from the Topopah Spring Tuff provided in SNL (2008af). The NRC staff concludes that the field sample data provided in SNL (2008af) is consistent with the applicant's initial data and that the values provided in the SAR are appropriate. Therefore, the NRC staff concludes that the applicant's information on the bulk density of the rocks and soil at Yucca Mountain is adequate to develop site-response models for use in the PCSA and GROA design.

**Seismic Velocity Profiles for Surface GROA**

The development of seismic velocity profiles as input to the seismic site-response model for the surface GROA is described in SAR Section 1.1.5.2.7.2, with additional detailed information in BSC (2008bl), SNL (2008af), and in the applicant's response to the NRC staff's RAI (DOE, 2009aq). On the basis of the available velocity data and site geology, the applicant developed 13 base-case velocity profiles for the surface GROA to fully capture the variability and uncertainty of the site. To capture the randomness of the site response, the applicant used each base-case profile as the basis for stochastically generating 60 randomized profiles that remain consistent with the given uncertainty and the mean profile. A site-response model is generated for each of the 60 velocity profiles, and the resulting seismic response spectra or amplification transfer functions are averaged to determine the mean response spectra and its associated uncertainty. This process is repeated using a suite of input ground motions that correspond to a range of exceedance probabilities in the PSHA to develop representative surface hazard curves. To capture the spatial variability of the site, including differences across the Exile Hill fault splay or variability in the stiffness of the underlying tuff, the applicant enveloped the site-specific hazard results to develop a single hazard curve for the entire surface GROA.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's information in SAR Section 1.1.5.2.7.2, references therein, and supporting documentation, including DOE (2009aq) by performing confirmatory calculations of the one-dimensional linear-equivalent site-response modeling (Stamatakos, 2014aa). These calculations were for 26 borehole-specific lithologic profiles throughout the GROA using the SHAKE2000 code, which is a well-established industry code for site-response modeling. Mean transfer functions based on the individual profiles for each of the additional 26 boreholes are bounded by the applicant's site response model. These results are also consistent with the NRC staff's earlier evaluation of the applicant's site data provided in Gonzalez, et al. (2004aa). In Gonzalez, et al. (2004aa) the NRC staff performed a similar one-dimensional site-response evaluation using data from the initial 15 site-response boreholes drilled within the GROA. Results of the staff's independent calculations showed that the applicant's approach captures both the randomness and uncertainty of the site velocity measurements, as well as the spatial variability of the site conditions, including spatial variations in the thickness of alluvium. All of the NRC staff's independent, one-dimensional profiles result in site amplification curves that fall within the applicant's distribution. Because of these results, the NRC staff finds that the site hazard curves are conservative because they are based on an envelope of the individual site-specific hazard curves. Therefore, the NRC staff concludes that the applicant developed sufficient information and an acceptable approach to develop adequate velocity profiles for seismic site-response models. These models are adequately representative of site conditions for use in PCSA and for the GROA design for the development of the field-free uniform hazard spectra, as described in SER Section 2.1.1.1.3.5.2.5.

**Seismic Velocity Profiles for Subsurface GROA**

As described in SAR Section 1.1.5.2.7.2, seismic profiles for the repository block were derived from 21 SASW profiles from 2004–2005 together with the SASW data from the 2000–2001 campaign. $V_S$ values varied spatially within the ESF and ECRB. DOE determined that these variations coincided with lateral changes in rock conditions, such as variations in lithology, stratal contacts, or the degree of fracturing in the tuffs. As a result, the applicant developed four separate velocity profiles to represent a central "stiff" zone and three relatively "softer" zones. Similar to the methodology for the surface GROA, the applicant developed a suite of site-response models that were combined to produce representative hazard curves for the repository block.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's methods and information on one-dimensional linear-equivalent site-response modeling described in SAR Section 1.1.5.2.7.2 and references therein. The NRC staff finds the seismic velocity profiles for the subsurface GROA to be adequate because the approach used to develop velocity profiles for the repository block parallels the approach the applicant used for the surface GROA. Similar to the evaluation of the applicant's velocity profiles for the surface GROA, the NRC staff concludes that the information and approach the applicant used are sufficient to develop adequate velocity profiles for seismic site-response models. These models are adequately representative of site conditions in the repository block for use in the PCSA and the GROA design.

2.1.1.1.3.5.2.4 Geotechnical Information Used to Develop Dynamic Material Properties

The applicant provided information on the dynamic properties of the site materials (rocks and soils) across the GROA and the repository block in SAR Sections 1.1.5.2.7.2 and 1.1.5.3.2.6.3. The dynamic properties of the alluvium and rock underlying the site are a component of the applicant's calculation to estimate the vibratory ground motion at the surface. The normalized shear moduli and damping ratios of rock and alluvium control the propagation of ground motion through the geologic medium in the applicant's site-response analysis. The applicant derived these values from experiments conducted over the past two decades. The applicant detailed descriptions of the data acquisition activities in BSC (2002aa). Both resonant column and torsional shear tests were performed in a sequential series on the same specimen over a shear strain range from about $10^{-4}$ percent to $10^{-1}$ percent (BSC, 2002aa, 2004aj; SNL, 2008af).

**Normalized Shear Modulus and Damping**

The applicant provided normalized shear modulus and material damping values used to assess the ground response at the surface from a controlled ground motion at the rock outcrop level [SAR Section 1.1.5.3.2.6.3 and BSC (2008bl, Section 6.4.4)]. Additional information needed for shear modulus and damping value reductions for each rock layer and alluvium present at the site is available in BSC (2004aj) and SNL (2008af). BSC (2004aj, Section 6.2.4) described the original experimental results of normalized shear modulus and damping curves for alluvium and tuff samples obtained from boreholes near the North Portal and waste-handling building areas. SNL (2008af) reported results of testing tuff samples from 2004 through 2006. These samples are from the major geologic units above, at, and below the waste emplacement level. These normalized shear modulus and damping curves in SNL (2008af), originally developed in BSC (2004aj), include the effects of confining pressure.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on normalized shear modulus and damping provided in SAR Section 1.1.5.3.2.6.3, references therein, and BSC (2008bl, Section 6.4.4). The NRC staff concludes that the applicant appropriately tested samples of alluvium from the surface facilities area and tuff from the repository block to determine the normalized shear modulus and damping ratio curves at different shear-strain levels. Because the applicant tested samples of tuff from the complete range of tuff strata at Yucca Mountain, including the repository horizon, the NRC staff concludes that the applicant adequately characterized the range of dynamic material properties at the site. The NRC staff concludes that, although some of the available data from the repository block for Tiva Canyon tuff and Yucca Mountain tuff samples in BSC (2004aj) are unqualified under the applicant's quality assurance program, as outlined in BSC (2008bl, Section 6.4.4.2), results from qualified tests (SNL, 2008af) from the same area corroborate the curves developed in BSC (2008bl).

The NRC staff also concludes that the applicant used appropriate methodologies to characterize the dynamic material properties, namely, normalized shear modulus and damping ratios, for both alluvium and rock strata lying underneath the repository area for the following reasons: (i) the applicant used acceptable industry standard guidance provided in EPRI (1993ab), which recommended using these properties to model the behavior of the geologic units to estimate the ground motion at the surface; (ii) DOE's results obtained for both normalized shear modulus and damping ratio adequately represent the characteristics of both alluvium and rock at the repository area for a shear strain about $10^{-4}$ percent to $10^{-1}$ percent; (iii) scatter of the experimental data for both normalized shear moduli and damping ratios follows the idealized shape of the cohesionless soil curve, as given in EPRI (1993ab). For these reasons, the NRC staff finds that the use of this “type curve” shape is reasonable to represent both tuff and alluvium response.

The NRC staff concludes that uncertainty exists in both curves, as indicated by the scatter of the experimental data. The applicant used two sets of mean normalized shear modulus and damping ratio curves developed for both tuff and alluvium to bound that uncertainty. The applicant extended the curves for both alluvium and rock at shear strains larger than 0.1 percent. This extension was conducted using the curve for cohesionless soil as a guide, in addition to engineering judgment. The NRC staff finds that this extension is acceptable because the applicant demonstrated that the data trend is consistent with the cohesionless soil curve in EPRI (1993aa). Therefore, the NRC staff concludes that the applicant has adequately characterized the normalized shear modulus and damping, including uncertainty, for use in the seismic hazard assessment in the PCSA and GROA design.

2.1.1.1.3.5.2.5 Seismic Design Inputs

The applicant’s development of the seismic design inputs was provided in SAR Sections 1.1.5.2.5.3, 1.1.5.2.5.4, 1.1.5.2.5.5, and 1.1.5.2.5.6. The applicant developed site-specific hazard curves for various combinations of velocity profiles, dynamic material property curves, and alluvium thicknesses (SAR Section 1.1.5.2.5.3). The hazard curves represent uncertainties in averaged velocity and dynamic properties. However, hazard curves for different cases representing observed variability in site properties that include the various depths of alluvium are combined, and a single curve is developed that envelopes all the individual cases (i.e., the single enveloping curve bounds all the input data). For the repository block, hazard results for two velocity profiles (northeast and south of the Exile Hill Fault splay) were enveloped. These two velocity profiles were used to account for the significant difference in alluvial thickness juxtaposed across the fault. This process incorporates uncertainty in hazard curve development. On the basis of the final hazard curves for the surface and subsurface GROA, the applicant provided design response spectra, time histories, and strain-compatible soil properties that are used to calculate the potential seismic hazards at the GROA and inputs into the GROA design. The applicant developed vertical hazard curves by applying distributions of vertical-to-horizontal 5 percent damped response spectral ratios to the site-specific horizontal hazard curves. The applicant followed NUREG/CR–6728 (McGuire, et al., 2001aa) to develop these ground motion inputs.

**NRC Staff’s Evaluation**

The NRC staff reviewed the information on seismic design inputs provided in SAR Sections 1.1.5.2.5.3, 1.1.5.2.5.4, 1.1.5.2.5.5, 1.1.5.2.5.6, and references therein, to evaluate the adequacy of the methods to develop site-specific ground motion parameters, which included (i) site-specific hazard curves and uniform hazard spectra, (ii) design-response spectra (5 percent damped), (iii) scaled earthquake time histories, and (iv) strain-compatible soil properties. The NRC staff finds the applicant’s approach acceptable because the analyses followed the recommended fully probabilistic Approach 3 (NUREG/CR–6728) to develop the site-specific hazard curves for the surface and subsurface GROA for the horizontal motions. In addition, the applicant used an averaging process to account for uncertainties and an enveloping process to accommodate spatial variability of the alluvium thickness across the site. The NRC staff determines that the applicant’s predicted ground motions are acceptable because they were developed with appropriate inputs to the validated site-response model and incorporated uncertainties. The NRC staff also finds that those final ground motion results are conservatively high at low annual exceedance probabilities ($<10^{-6}$) compared with the available worldwide strong motion data, which include records from earthquakes much greater than those expected in the Yucca Mountain region. The NRC also notes that the newly developed design spectra (BSC, 2008bl) supplement the 2004 version (BSC, 2004aj). The conditioning of PSHA results was also applied in the new derivation.

**NRC Staff's Conclusion**

The NRC staff finds that the seismic information for the Yucca Mountain site is acceptable because it has been properly characterized by the applicant through the use of a PSHA, which is considered state-of-the-art for determining the potential hazards from seismic events for nuclear facilities.

The NRC staff finds the PSHA methodologies and input data and interpretation of the PSHA and conditioning of PSHA results for the Yucca Mountain site to be adequate. The NRC staff finds that the applicant relied on the collective judgment of established experts, followed an acceptable procedure to elicit and document the experts' conclusions, and supported the expert elicitation with sufficient technical and scientific information. Also, the NRC staff finds that new information about the seismic hazards at Yucca Mountain published since DOE completed its expert elicitation, as presented in Hanks, et al., (2013aa), suggests that the DOE PSHA provided in the SAR is conservative at low annual exceedance probabilities, further supporting the NRC staff finding that the probabilistic seismic hazard analysis is acceptable.

The NRC staff also finds that the applicant used appropriate methods and information to perform seismic site-response modeling. In particular, the NRC staff finds that the RVT-based point-source/one-dimensional equivalent-linear site-response model combined with Approach 3 and the conditioned hazard is adequate for use in characterizing the seismic ground motions at Yucca Mountain. Further, the NRC staff reviewed the applicant's description of geophysical information to develop $V_P$, $V_S$, and density profiles; the NRC staff finds that the applicant provided sufficient information about these geotechnical properties of the subsurface materials to develop adequate seismic site-response models. The NRC staff finds the applicant adequately accounted for uncertainty and variability in these parameters across the GROA through use of bounding analyses. The NRC staff finds that application of these results in the applicant's site-response models is sufficient to develop an adequate seismic hazard for the surface and subsurface GROA as input to the seismic design and PCSA. Moreover, the NRC staff finds that the applicant adequately described the basis for the development of the dynamic material properties used in its site-response calculations.

Therefore, the NRC staff finds, with reasonable assurance, that the information the applicant provided on seismicity at the Yucca Mountain site for the preclosure period is acceptable for use in the evaluations in the PCSA and to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(i) and (ii) and 10 CFR 63.112(b) and (c) with respect to site seismology.

2.1.1.1.3.5.3 Site Geotechnical Conditions and Stability of Subsurface Materials

The applicant described the geotechnical properties and conditions of the repository site for use in the PCSA and GROA design in SAR Section 1.1.5.3. The applicant described the types and geometrical configuration of subsurface materials (rocks and soil) at the site and mechanical properties of these materials. These properties are used to evaluate the stability of subsurface materials. On the basis of the applicant's information in SAR Section 1.1.5.3 and the applicant's responses to the NRC staff's RAIs, the NRC staff organized its review of the site geotechnical conditions and stability of subsurface materials into (i) types and geometrical configurations of subsurface materials at the surface GROA, (ii) inputs to analyses of stability of subsurface materials present beneath the surface facilities at the GROA, and (iii) geotechnical conditions of the materials present in the subsurface GROA.

2.1.1.1.3.5.3.1 Types and Geometrical Configuration of Subsurface Materials at the Surface GROA

The applicant provided information pertaining to the nature and configuration of subsurface materials (rocks and soils) at the surface facility GROA in SAR Section 1.1.5, including Figures 1.1-2, 1.1-130, 1.1.59, and 1.1-60. The applicant conducted geological and geophysical studies at the site, including geologic mapping of outcrops, characterization of cuttings from geophysical testing boreholes, observations in test pits and trenches, and surface- and borehole-based geophysical testing. The applicant concluded that the surface facility site is underlain by Quaternary alluvium and colluvium up to 61 m [200 ft] thick, which overlies a sequence of volcanic tuff, as shown in SNL (2008af, Table 6.2-1). The applicant stated that the tuff is much stronger than the alluvium and poses no constraints on site development because tuff deformation would be much smaller than the alluvium deformation. Thus, the applicant focused its analysis on the deformation of alluvium rather than tuff.

As shown in SAR Figure 1.1-130, the alluvium thickness varies in the east-west direction from none at the base of Exile Hill to a thickness of approximately 9.1 m [30 ft] at the west boundary of the proposed Initial Handling Facility, increasing to approximately 61 m [200 ft] thick in the middle of Midway Valley near the location of the easternmost proposed Canister Receipt and Closure Facility. The applicant interpolated variations of the alluvium thickness from borehole data, as described in SNL (2008af). The applicant described the alluvium as soil material consisting of interbedded calcite-cemented and noncemented, poorly sorted, coarse-grained gravel with sand and some fine-sized particles, cobbles, and boulders (BSC, 2007bq). The alluvium in the area of the North Portal is overlain by up to 9.1 m [30 ft] of nonengineered fill that will be replaced with engineered fill as part of surface facilities construction.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information pertaining to the types and configuration of subsurface materials at the surface facility site in SAR Section 1.1.5.3, and references therein, to assess the adequacy of the applicant's geotechnical studies at the site in determining the stratigraphy of the subsurface conditions at the GROA site. The NRC staff concludes that the applicant's site characterization from test pits and cutting samples from boreholes is adequate for use in engineering evaluations in the PCSA because the information was obtained through appropriate site investigations using geologic and geophysical techniques consistent with Regulatory Guide 1.132 (NRC, 2003ag). These techniques are commonly used for geological and geophysical investigations (NRC, 2003ag). Regulatory Guide 1.132 was developed for use in evaluating nuclear power plants; however, the methodologies and conclusions are generally applicable to site characterization and are appropriate for use in characterizing the GROA.

2.1.1.1.3.5.3.2 Inputs to Analyses of Stability of Subsurface Materials at the Surface GROA

**Shear Strength of the Alluvium**

The applicant provided information on the stability of the subsurface materials (rocks and soils) beneath the surface GROA in SAR Sections 1.1.5.3.2.3 and 1.1.5.3.2.4. The applicant's assessment of the performance of surface facility structures assumed that the subsurface materials supporting the foundations will be stable during the preclosure period and undergo only elastic deformations when subjected to static and seismic loading (BSC, 2007ba). To support this assumption, the applicant provided information pertaining to the allowable bearing capacity of the alluvium, which the applicant calculated using the estimated shear strength of the alluvium.

To estimate the shear strength of the alluvium, the applicant conducted laboratory and field investigations to measure the relative density of the alluvium. The applicant used relative density in empirical relationships to estimate shear strength (BSC, 2002ab). To obtain relative density, the applicant first determined the bulk density of the alluvium from geophysical measurements in seven boreholes (BSC, 2002aa,ab) and water-replacement and sand-cone density tests in test pits. Then, maximum and minimum density indices were measured in the laboratory from samples obtained from the test pit locations. Relative density of the alluvium was then calculated from the minimum and maximum density indices and *in-situ* measured bulk densities. For the range of relative densities considered, the applicant determined that the angle of internal friction ranged from 33° to 52°. DOE proposed an internal friction angle of 39° with zero cohesion to represent the shear strength parameters of the alluvium at the site (SAR Section 1.1.5.3.2.4). The applicant justified the use of 39° based on several correlations that showed that the proposed value was between the low and the mean of the test data.

The applicant stated that an internal friction angle of 39° with zero cohesion represents the shear strength parameters of the alluvium at the site. The applicant concluded, and provided additional information in DOE (2009bg,aq,eh) to support its conclusion that this value is appropriate and may be conservative because, at the scale of building foundations, the very large volume of alluvial material exhibits a behavior that can be conservatively represented by average laboratory and field test results. The applicant further supported its analyses based on additional measurements of relative density (SAR Section 1.1.5.3.2.3, SAR Table 1.1-85). Further, the applicant explained that the effects of cementation were implicitly included in the analysis of field measurements of shear wave velocity ($V_S$) used to assess potential settlement (DOE, 2009aq; BSC, 2007bq). The applicant also stated that although the alluvium is laterally discontinuous and layered over small scales, when considered at a large scale, for example averaging data across the GROA, which is more appropriate for evaluating these building foundations, the alluvial material can be accurately represented as homogeneous.

**NRC Staff's Evaluation**

The NRC staff reviewed the information in SAR Sections 1.1.5.3.2.3 and 1.1.5.3.2.4, references therein, and responses to RAIs (DOE, 2009bg,aq,eh) on the stability of the subsurface materials at the surface GROA, and evaluated the field testing procedures and the applicant's use of those results and the applicant's empirical relationships to determine the shear strength parameters of alluvium. The NRC staff finds that the applicant appropriately used *in-situ* and laboratory and field test procedures to determine geotechnical parameters of the alluvium because these tests and procedures are generally accepted and used in the geotechnical engineering profession.

The NRC staff also concludes that the empirical methods used to correlate field measurements with shear strength parameters (e.g., the angle of internal friction) are adequate because the methodology used by the applicant is widely accepted by the geotechnical community. The NRC staff finds that the DOE-proposed value of angle of internal friction in the SAR is conservative because this value assumes cohesionless soils and uniform soil strength with depth, where there is evidence that shear strength may increase with depth, as shown in SAR Figures 1.1-133 and 1.1-144, and evidence that some of the alluvium is cemented or partially cemented and thus may be stronger than assumed. Therefore, the NRC staff finds that the proposed 39° angle of internal friction, in combination with zero cohesion, is acceptable for use in foundation design.

The NRC staff notes that additional geotechnical characterization will be performed by DOE prior to the construction of the surface facilities. As is typical for construction of large-scale buildings in alluvial basins (such as the numerous large DOE facilities constructed across the NNSS), characterizations of the bearing capacity and spatial distribution of cemented alluvium across the site and over the zone of influence of the foundation loadings will be evaluated in further detail through systematic measurements of alluvial thickness and distribution of cementation. These characterizations will be conducted in accordance with appropriate industry codes and standards.

In summary, the NRC staff concludes that for the purpose of the PCSA and GROA design, the applicant's information on the shear strength of alluvium is adequate to assess the engineering design and performance of the foundations of surface facilities.

**Compressibility of Low-Density Tuff**

The applicant stated in SAR Section 1.1.5.3.2.6.2.1.1 that the presence of low-density bedded tuff could potentially affect the engineering performance of structures if the low-density tuff is more compressible than the overlying alluvium. However, DOE provided shear wave velocity data (DOE, 2009aq) to show that the low-density tuffs and other tuffs directly underlying the alluvium at the surface facility site have shear wave velocity ($V_S$) equal to or greater than that of the alluvium, indicating that the low-density tuff is stronger and less compressible than the alluvium, and therefore does not affect potential deformation of surface facility SSCs.

**NRC Staff's Evaluation**

The NRC staff reviewed the test results provided in SAR Section 1.1.5.3.2.6.2.1.1, references therein, and responses to RAIs (DOE, 2009aq), and, based on the shear wave velocities of the tuff compared to the alluvium, the NRC staff concludes that the results from DOE's geophysical tests sufficiently demonstrate that the tuff material is stronger and less compressible than alluvium and, thus, deformation of the tuff will not have any significant effect on the stability of subsurface materials underlying the proposed surface facility structures.

**Allowable Bearing Pressure and Settlement of Foundations of Surface GROA Facilities**

The applicant provided information pertaining to allowable bearing pressure for the foundations of the surface facility structures (BSC, 2007bq). The applicant determined the allowable bearing pressure for three conditions using shear strength of alluvium on the basis of an internal friction angle of 39: (i) square and strip footings with no limit on settlement, as described in BSC (2007bq, Figure B6-2); (ii) square and strip footings with settlement limited to 12.7 mm [0.5 in], as shown in BSC (2007bq, Figures B6-7, B6-8, and B7-2); and (iii) square and strip footings with settlement limited to 25.4 mm [1.0 in], as outlined in BSC (2007bq, Figures B6-13 and B7-1). The applicant's results for condition (i) determined potential limits on foundation loading without causing a generalized shear failure of the subsurface materials (i.e., rotational failure of the foundation and underlying materials). Conditions (ii) and (iii) determined potential limits on foundation loading without causing excessive settlement due to localized shear failure of the subsurface materials. The allowable bearing pressure from condition (i) increased as the footing width increased. Conditions (ii) and (iii), in contrast, yielded results of allowable bearing pressure that decreased as the footing width increased, but approached a minimum value for large footing widths.

In response to an RAI (DOE, 2009ei), the applicant stated that its approach to design mat foundations at the surface facility is based on a finite element (FE) model in which the subsurface material is represented by soil springs and the resulting deformation is used to calculate foundation pressures and settlements. The FE model is also used to check the calculated pressures and settlements. The applicant calculated the bearing pressure from empirical relationships as described in Terzaghi, et al. (1996aa, Section 50.2) using relative densities measured at the site. For footings up to 9.1 m [30 ft] wide, the allowable bearing pressure in BSC (2007bq, Figures B7-1 and B7-2) was controlled by settlement criteria of 25.4 and 12.7 mm [1.0 and 0.5 in], respectively. Similarly, the applicant performed an FE analysis to design mat foundations for the design-basis seismic load.

In BSC (2007bq), the applicant analyzed elastic settlement of a 91.4 by 122.9-m [300 by 400-ft] mat on a 36.6-m [120-ft]-thick alluvium surface, subjected to normal loading conditions (dead load plus live load) of 144, 239, and 335 kPa [3, 5, and 7 ksf]. BSC (2007qq, Table B7-1) presented the calculated settlements, ranging from 5 to 76 mm [0.2 to 3.0 in] for the range of load. The applicant provided the following in DOE (2009ei): (i) Table 1, presenting new results of total and differential settlements for the static loading conditions (normal load) for all potential ITS structures; (ii) Figure 1, showing calculated allowable bearing capacity for foundations up to 92.6 m [300 ft] wide, which is similar to BSC (2007qq, Figure B6-2) for rotational shear failure of foundation material discussed previously; and (iii) Figure 2, showing allowable bearing pressure for foundations up to 92.6 m [300 ft] wide, which would limit the settlement to 50 mm [2 in]. On the basis of DOE (2009ei, Figure 2), limiting the settlement to 50-mm [2-in] criteria for large mat foundations, DOE proposed an allowable bearing pressure of 479 kPa [10 ksf] for normal loading conditions. The applicant provided the rationale for a settlement limit not to exceed 50 mm [2 in] for large mat foundations of ITS structures based on technical literature (DOE, 2009ei, Section 1.3,). For extreme loading conditions, such as a design basis seismic event, the applicant proposed an allowable bearing capacity of 2,394 kPa [50 ksf] from DOE (2009ei, Figure 1) on the basis of rotational shear failure of the foundation material criterion with no consideration of settlement criterion. DOE (2009ei, Table 1) lists the average foundation pressure for various potential ITS structures under normal loading to range from 81 to 225 kPa [1.7 to 4.7 ksf], which is below the recommended allowable bearing pressure of 479 kPa [10 ksf] for normal loading.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's methodology and data used to determine allowable bearing pressure for surface GROA facility foundations provided in SAR Sections 1.1.5.2 and 1.1.5.3, references therein, and responses to RAIs (DOE, 2009ei). The NRC staff finds that

DOE's general approach to mat foundation design is acceptable because it is based on standard methods used in foundation engineering and documented in standard engineering texts and handbooks. The NRC staff finds that the applicant's analysis method and proposed allowable bearing pressure for footing and mat foundations are acceptable because they were determined by methods commonly used in the geotechnical engineering profession and are further supported by numerical analyses. The NRC staff finds that, while there might be some uncertainty associated with computation of shear strength parameters by correlating relative density to the angle of internal friction, and that such correlations could impact the estimated bearing capacity of alluvium, the NRC staff also finds that high factors of safety in the applicant's use of well-established conventional methods of static bearing capacity analysis adequately account for such uncertainties. For example, the allowable bearing capacity estimated by DOE is approximately 30 to 100 times the calculated average foundation pressure for the cases presented in the RAI response (DOE, 2009ei), indicating significant conservatism.

For large mat foundations, the allowable bearing pressure is controlled by settlement criterion rather than shear failure criterion. Therefore, the bearing capacity calculated for rotational shear of the foundation material alone may not control the design of large mat foundations. The NRC staff finds that the methodology the applicant used to calculate elastic settlement for normal load conditions is acceptable because the calculated elastic settlement of the mat foundation was based on a range of uniform loads, a uniformly thick alluvium layer, and elastic moduli of soil calculated from shear wave velocity ($V_S$) data. The applicant presented these calculated settlements at the center and corners of the mat foundation for the range of loads (BSC, 2007bq).

The applicant provided the rationale for limiting settlement to 2 inches for large mat foundations in DOE (2009ei). The NRC staff finds that this rationale is acceptable because the calculations are based on empirical relationships. The empirical relationships were developed from case histories of the performance of large mat foundations on granular soils. This methodology is widely accepted in the geotechnical engineering profession. Therefore, the applicant's proposed allowable bearing pressure of 479 kPa [10 ksf], based on limiting settlement for normal loading, is acceptable. DOE calculated estimated settlements and showed that they are well within the maximum allowable settlement of 5 cm [2 in] established above. In addition to the maximum settlement, DOE also analyzed the potential for differential settlement, which is crucial for foundation performance. The NRC staff finds that, as presented in the RAI response, the estimated differential settlement is about half of the estimated maximum settlement. This is consistent with the general design criteria used by foundation engineers and also meets DOE's proposed design goal. The range of estimated differential settlements provided in the RAI response is acceptable, as it is well within the normal values for the design of footings and mat foundations.

For seismic loading conditions, the applicant proposed an allowable bearing pressure of 2,394 kPa [50 ksf], based on laboratory and field test data that yielded an internal friction angle of 39 degrees. This analysis considers rotational shear failure of the foundation material, which does not include a limit on foundation settlement (DOE, 2009ei). The applicant designed the mat foundation, considering design-basis seismic loads, using a FE analysis where the alluvium under the mat foundation was modeled as a soil spring. The FE modeling analysis represents the alluvium as a soil spring using shear modulus calculated from shear wave velocity data. Representing the alluvium as a spring assumes that the alluvium will not undergo localized shear failure, will respond linearly, and will not deform significantly under the imposed loading. This analysis yielded deflection of the mat and a resulting bearing pressure at nodal points of the FE mesh. Based on this analysis, the applicant determined an allowable bearing pressure of 2,394 kPa [50 ksf] for mat foundation (DOE, 2009ei, Figure 1). The NRC staff finds that these considerations are part of standard elastic analyses in engineering practice and are therefore acceptable.

In order to evaluate the impact of potential uncertainties in the shear strength parameters used by DOE in estimating the ultimate bearing capacity under dynamic conditions, the NRC staff evaluated the significance of uncertainties in estimated friction angle of the alluvium. For example, the NRC staff reviewed design charts developed for footings by Al-Karni, (1994aa), which show that friction angle is the least sensitive of all parameters considered in the study. Design curves generated in this reference showed a very small impact on seismic safety factor when friction angle was varied between 0° and 40° for a given value of horizontal ground acceleration. Therefore, the NRC staff concludes that any inaccuracy introduced in computing the factor of safety against bearing capacity failure under seismic conditions as a result of potential uncertainty in the estimated friction angle would not be significant. Further, the NRC staff took into consideration that, in practice, compared to static bearing capacity computations, lower factors of safety would be acceptable under extreme loading conditions (such as under low probability seismic events). However, given the shear strength parameters of alluvium, the NRC staff concludes that sufficient factors of safety (well above 1.0) will be maintained even under extreme loading conditions. In addition, the staff notes that construction practice would dictate removal of any loose surficial material and its replacement with engineered compacted fill prior to foundation construction to achieve a desired friction angle to ensure adequate bearing capacity both under static and dynamic conditions. Therefore, the NRC staff finds the applicant's calculated allowable bearing pressures of 2,394 kPa [50 ksf] for extreme loading conditions and 479 kPa [10 ksf] for normal loading are acceptable, as they are adequately supported by both empirical methods and FE analyses.

In summary, the NRC staff concludes that for the purpose of the PCSA and GROA design, the applicant's information is adequate to assess the engineering design and performance of the foundations of surface facilities both under normal and seismic conditions.

**Stability of Slopes**

The applicant stated in SAR Section 1.1.5.3.2 that engineered slopes will be constructed as part of the design of the aging pads and transportation routes linking the aging pads to other surface facilities (SAR Figure 1.1-129). In particular, excavations for the aging pads could expose the alluvium in a cut slope up to approximately 10 m [33 ft] high, and transportation routes could involve cut-and-fill slopes (SAR Figure 1.2.7-2). The applicant stated that design and construction of these cut-and-fill slopes will be accomplished to ensure that slopes will not fail under static and seismic loads.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on stability of the cut-and-fill slopes in SAR Section 1.1.5.3.2.1, references therein, and responses to the NRC staff's RAIs (DOE, 2009aq,eh). The NRC staff finds the assessment of the stability of the slopes is acceptable because the applicant stated it will ensure the stability of the slopes as part of the detailed design (DOE, 2009aq,eh). Further, the applicant also stated that aging pads will be built on terraces to minimize the amount of cut-and-fill, and any cut-and-fill slopes will not have a steeper slope than 0.5. The applicant provided a stability analysis that indicates the alluvial slopes will be stable under seismic loading conditions (DOE, 2009ej) and, based on the NRC staff's review of DOE's analyses and the NRC staff's knowledge, the NRC staff finds the stability analyses to be acceptable because they are based on well-known methods of stability analysis, which are accepted by the geotechnical engineering profession. Therefore, the NRC staff concludes that, for the purpose of providing inputs to the PCSA and the GROA design, the applicant's information is adequate.

#### 2.1.1.1.3.5.3.3 Geotechnical Conditions at the Subsurface GROA

The applicant provided information pertaining to geotechnical conditions at the subsurface GROA in SAR Sections 1.1.5.3 and 2.3.4.4.2.1. The applicant detailed the type and configuration of repository host horizon (RHH) strata and parameters describing the material properties of these rocks needed for an engineering analysis.

**Repository Host Horizon (RHH) Materials**

In SAR Section 1.1.5.3.1.1, the applicant stated that the repository emplacement areas will be located approximately 300 to 400 m [984 to 1,312 ft] below the ground surface within several subunits of the crystal-poor member of the Topopah Spring Tuff (SAR Figure 2.3.4-21). The applicant explained that the repository host rock includes lithophysal and nonlithophysal subunits; the nonlithophysal subunits comprise approximately 15 percent of the emplacement area and the lithophysal subunits approximately 85 percent, with approximately 80 percent within the Lower Lithophysal subunit (SAR Figure 2.3.4-22). The applicant explained in SAR Section 1.1.5.3.1.1 that the lithophysal and nonlithophysal rock types are compositionally similar but have different physical, thermal, and mechanical properties because of the differences in their internal geologic structures.

The applicant stated that the nonlithophysal rocks are hard, strong, fractured rock masses, whereas the lithophysal rocks are more deformable with lower compressive strength than the nonlithophysal rocks. According to the applicant, the lithophysal rocks contain macroscopic voids (i.e., lithophysae); these resulted from gas that was trapped when magma cooled to form volcanic tuff, with the volume fraction of lithophysae in the range of 10 to 30 percent. The Lower Lithophysal subunit is heavily fractured with small-scale {lengths smaller than 1-m [3.3-ft]} fractures. The applicant explained that the rock-mass strength and stiffness of nonlithophysal units are controlled by the mechanical properties and behavior of existing fractures, whereas the rock-mass strength and stiffness of lithophysal units are controlled by the lithophysal porosity and density of small-scale fractures.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on RHH materials provided in SAR Section 1.1.5.3.1.1 and references therein, and evaluated the data collected by geotechnical and geophysical site studies conducted by the applicant. The NRC staff finds that the applicant's description is acceptable because the characterization of materials that constitute the repository horizon was based on geologic studies performed at the site using standard practice and geotechnical techniques such as stratigraphic measurements, petrographic examinations, geomechanical investigations, and geochemical assays. On the basis of these studies, the applicant described the expected locations of stratigraphic contacts and estimated the percentage occurrence of each rock type within the repository horizon. Therefore, the NRC staff finds that the geotechnical information the applicant provided to describe the RHH and materials is adequate for use in the PCSA and the GROA design to assess the geotechnical characteristics of the subsurface facility design.

**Mechanical Properties of Nonlithophysal Rocks**

The applicant used a variety of geological and geomechanical analyses to characterize the mechanical properties of nonlithophysal rock. These included detailed line surveys and full-periphery geologic mapping of the ESF and ECRB cross drift (SAR Section 1.1.5.3.1.2.1), unconfined and triaxial compression tests of intact rock specimens (SAR Section 1.1.5.3.1.2.2.1), and direct shear and rotary shear tests of fracture surfaces (SAR Section 1.1.5.3.1.2.1). The applicant also used well-established empirical rock mass classification systems (SAR Section 1.1.5.3.1.2.1) to determine rock mass quality designations (an indirect measure of strength of rock mass) and calculate values of rock-mass strength and stiffness parameters (BSC 2007be). SAR Table 1.1-82 summarized the rock-mass strength and stiffness of the nonlithophysal rock units.

The applicant provided an engineering characterization of the nonlithophysal rock units that the applicant encountered in the ESF tunnel and used this information (SAR Table 1.1-82) as the basis for assessing the engineering behavior of nonlithophysal rock in the entire repository block.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on the mechanical properties of the nonlithophysal rocks provided in SAR Section 1.1.5.3.1.2.1 and references therein to evaluate the adequacy of the applicant's analysis of field-observed and laboratory-test data and determination of the strength and stiffness of nonlithophysal rock units. The NRC staff concludes that the characterization of mechanical properties of the nonlithophysal rocks in the GROA, for the purpose of design and for use in the PCSA, is adequate because the characterization is based on site-specific data and analyses using techniques that are well established in geotechnical engineering practice. The applicant's characterization also conforms to the NRC staff's direct observations of nonlithophysal rock units at the Yucca Mountain site based on numerous site visits and workshops held during the prelicensing period.

**Mechanical Properties of Lithophysal Rocks**

The applicant used a variety of geological and geomechanical analyses to characterize the mechanical properties of lithophysal rock (SAR Section 2.3.4.4). The applicant tested 29 large-diameter specimens from the lithophysal rock units and used the results to group the rock mass into 5 categories based on the values of strength and elastic stiffness, as identified in SAR Section 2.3.4.4.2.3.3.4 (BSC, 2004al). The applicant also used lithophysal porosity data from the ECRB cross drift to define ranges of porosity for the five rock mass categories (SAR Figure 2.3.4-29) and examined relationships between strength and elastic stiffness of lithophysal rock using numerical model calculations (SAR Figure 2.3.4-30, BSC 2007be). The rock-mass strength and stiffness of the lithophysal rock units were summarized in SAR Table 2.3.4-16 and Figure 2.3.4-30.

The applicant tested large-diameter lithophysal rock specimens to determine their strength and stiffness (SAR Table 2.3.4-16 and SAR Figure 2.3.4-30). Six of the 29 specimens in BSC (2007be, Table 6-69) were from the Lower Lithophysal subunit, and the other 23 were from the Upper Lithophysal subunit. Laboratory results were augmented by numerical modeling simulations to develop these strength and stiffness values (SAR Figure 2.1.4-30).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information in SAR Section 2.3.4.4 and references therein, regarding the mechanical properties of lithophysal rocks. The NRC staff finds the information acceptable because the applicant provided mechanical properties of lithophysal rock on the basis of site-specific data and analysis of the data using techniques that are well established in the geotechnical engineering profession. The NRC staff notes that the 29 specimens the applicant tested are shorter than the minimum length of specimens for unconfined compression testing, as recommended by International Society for Rock Mechanics Commission on Testing Methods (1981aa, p. 113). Six of the specimens had a length-to-diameter (L/D) ratio of 1.0 to 1.5, and 23 had an L/D ratio of 1.7 to 2.1. Therefore, the values of L/D ratio for the specimens are smaller than the recommended value of 2.5 to 3.0. The NRC staff reviewed a relationship suggested in Jaeger and Cook (1979aa, p. 144) that indicates the deviation from the recommended L/D ratio implies the test results could overestimate the strength of the tested rock by approximately 2 to 20 percent. However, the results of the applicant's numerically simulated testing indicated that uncertainties in the strength and stiffness data are encompassed by the upper and lower bounds that the applicant defined in SAR Figure 2.3.4-30. Furthermore, the NRC staff's confirmatory calculations (Ofoegbu, et al., 2007aa, p. 3-5) indicate that the upper and lower bounds the applicant defined agree with bounds, based on 95-percent confidence limits. Therefore, the NRC staff concludes that the characterization of mechanical properties of the lithophysal rocks in the subsurface GROA is sufficient for use in the PCSA and for GROA design.

**Other Geotechnical Properties at the Subsurface GROA**

The applicant conducted a variety of laboratory and field tests to determine the thermal properties (i.e., thermal conductivity, thermal expansion coefficient, and heat capacity) for lithophysal and nonlithophysal rock (SAR Section 1.1.5.3.1.2.3). The applicant evaluated *in-situ* stress based on two hydraulic fracturing tests (SAR Section 1.1.5.3.1.2.4) and seismic velocities using downhole and surface-based geophysical testing (SAR Section 1.1.5.3.1.3.1). The applicant determined dynamic properties, such as shear modulus and damping ratio, from laboratory testing (SAR Section 1.1.5.3.2.6).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information in SAR Section 1.1.5.3.1 and references therein regarding other properties of lithophysal and nonlithophysal rocks. The NRC staff finds that the information the applicant provided for the *in-situ* stress and thermal and dynamic properties of lithophysal and nonlithophysal rock is adequate to characterize the properties of the rocks encountered in ESF and ECRB because the applicant used standard techniques for the laboratory and field tests and provided information to define potential uncertainties in the test results. The NRC staff further concludes that the information is sufficient for use in the PCSA and GROA design to assess the performance of subsurface GROA.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant adequately described information regarding the geotechnical conditions and stability of surface and subsurface materials at the GROA. The NRC staff finds the applicant has adequately identified the types and configuration of subsurface materials at the GROA that will be important in assessing the engineering design and performance of the foundations of surface facilities, has characterized the subsurface materials adequately based on longstanding standard field and laboratory tests, and has provided adequate characterizations of important ranges of geotechnical parameters based on standard geotechnical methodologies, geomechanical and geotechnical analyses, and standard interpretations of these analyses based on field and laboratory data. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's information regarding site geotechnical conditions and stability of subsurface materials is acceptable for use in the evaluations in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c).

#### 2.1.1.1.3.6 Site Igneous Activity

The applicant provided information in SAR Sections 1.1.6, 2.2.1, and 2.3.11 on the known intrusive and extrusive (volcanic) igneous activities in the Yucca Mountain region as they pertain to the PCSA and GROA design. The applicant also conducted a probabilistic volcanic hazard analysis (PVHA). The applicant indicated that volcanic activity has occurred in the tectonically active Yucca Mountain region and could occur again in the future. On the basis of this, igneous activity may possibly affect GROA design and preclosure repository performance, as discussed below.

In SAR Section 1.1.6, the applicant assessed the location and magma types of past volcanism in the Yucca Mountain region, described the characteristics of basaltic volcanism in the region, and presented evidence for simultaneous seismic activity and volcanic eruption. This information is described within the context of the applicant's PVHA, undertaken in 1996 (CRWMS M&O, 1996aa). The applicant assessed the potential hazard and possible effects from volcanic ash fall in the preclosure period. The probability of a recurrence of igneous activity is compared to an event criterion of less than a 1 in 10,000 chance of an occurrence in the 100-year-preclosure period, or $1 \times 10^{-6}$ per year. Aspects of the hazard that igneous activity poses to repository performance in the postclosure period are evaluated in SER Sections 2.2.1.2.2 and 2.2.1.3.10.

In this SER section, the NRC staff's evaluation of the information presented in SAR Section 1.1.6 is coordinated with that of SER Sections 2.5.4, 2.2.1.2.1, 2.2.1.2.2, and 2.1.1.7 because these sections also pertain to aspects of possible future igneous activity at the repository site. The NRC staff's review of the risk that igneous activity poses in the preclosure period also relies upon information in SAR Section 5.2.1.5; in SAR Sections 1.2.2.1.6.5, 1.6.3.4.2, and 2.3.11; on relevant applicant-provided reports; and on publicly available information. This review concentrates on volcanic (extrusive) surface activity as it is more likely to affect the repository surface facilities and operations in the preclosure period than an intrusive event (where magma does not reach the surface), even though the applicant concluded that the probability of a future eruption within the preclosure period is extremely low (less than $1 \times 10^{-6}$ per year).

##### Magma Types, Location, Style, and Timing of Igneous Activity in the Yucca Mountain Region

##### Rhyolitic Igneous Activity

The applicant determined the age range for the major explosive volcanic flare-up that formed the rhyolitic (silicic) ash-flow tuffs of Yucca Mountain, the host rocks for the repository, to be between 13 and 10 million years ago (SAR Section 2.3.11.1). The applicant found that there has been a long time gap between the cessation of these large-scale, caldera-forming explosive eruptions and the present day (BSC, 2004bi). During this time, no further rhyolitic activity has occurred in the Yucca Mountain area of the Basin and Range Province. Considering the brief duration of the preclosure period (i.e., 100 years from the time of license application), the applicant considered the chance of this type of volcanic activity recurring within that timeframe to be exceedingly small (BSC, 2003ae).

**NRC Staff's Evaluation**

The NRC staff reviewed the information on rhyolitic igneous activity that the applicant provided in SAR Sections 2.3.11 and references therein. The NRC staff also reviewed the information in Detournay, et al. (2003aa) and finds that the applicant adequately considered explosive rhyolitic magmatic activity near the Yucca Mountain site. Specifically, DOE's assessment provided an adequate basis for its determination of the likelihood of future explosive volcanic activity of rhyolitic magma at the site because DOE showed that such activity is not expected to recur in the area of Yucca Mountain within the next 1 million years.

**Basaltic Igneous Activity**

With regard to the smaller-scale basaltic igneous activity, the applicant presented evidence that basaltic eruptions and intrusions in the Yucca Mountain region have fallen into 2 major time periods or phases: (i) from 11 million to 8 million years ago and (ii) beginning about 4.6 million years ago and continuing to the latest eruption ~77,000 years ago at the Lathrop Wells volcano. The latter phase consisted of at least six volcanic events, based on age-dated, surface-exposed eruption products (cones and lavas) and can be further subdivided into two episodes: an older, Pliocene-age episode (volcanoes 4.6 to about 3 million years old) and a younger, Quaternary-age episode (volcanoes approximately 1 million years old or less) (SAR Table 2.3.11-2).

Igneous features buried by alluvium and located by geophysical surveys during DOE-conducted studies have also been documented in the region; the youngest of these is approximately 3.8 million years old (see also SAR Section 2.3.11.1). While more than 10 of these buried igneous features are known, the applicant concluded that their presence does not significantly increase the future probability of an eruption at the repository site on the basis of the number of post-Pliocene igneous events (BSC, 2004af).

The applicant focused on the young (post-Pliocene) basaltic volcanic deposits, lavas, and intrusions because they can be used to determine the type and style of volcanism that has occurred most recently and that may recur in the near future. Furthermore, several of the Quaternary-age volcanoes that lie in the Crater Flat Basin are the closest located basaltic igneous features to the repository site, approximately 7 km [4.5 mi] away. The applicant also determined that the volumes of basaltic magma erupted in the Yucca Mountain region are very small (on a comparative global scale), that the youngest phase of igneous activity has featured the smallest eruptions, and that activity has generally decreased in volume over time.

**NRC Staff's Evaluation**

The NRC staff reviewed the information about basaltic igneous activity provided in SAR Section 2.3.11 and references therein by conducting independent confirmatory studies to evaluate the style and frequency of past basaltic volcanism in the Yucca Mountain region (Hill and Connor, 2000aa; Conner, et al., 2000aa; Stamatakos, et al., 2007aa). On the basis of the NRC staff's studies and consideration of the available information the applicant presented, such as the type and number of basaltic volcanoes and their ages, the NRC staff concludes that the applicant's approach to assessing the basaltic igneous activity in the area around the repository is acceptable. Moreover, the NRC staff concludes that the applicant's assessment for preclosure provides a reasonable basis to support its determination that the likelihood of future basaltic magmatic activity in the area around the proposed repository is low (less than $1 \times 10^{-6}$ per year), including intrusive and volcanic events, and if an eruption were to occur, the volumes of magma would be small. This conclusion is based on the NRC staff's independent analyses (Conner, et al., 2000aa; Hill and Connor, 2000aa) and on other peer-reviewed published information on the nature and timing of eruptions of the Yucca Mountain basaltic volcanoes (Valentine and Perry, 2006aa, Valentine and Perry,2007aa; Valentine, et al., 2007aa).

**Relationship between Seismic and Igneous Activity**

In SAR Section 1.1.6.1.2, the applicant identified the possibility that rising magma could trigger seismic activity. The applicant also described the occurrence of patches of basaltic ash particles showing signs of minimal abrasion in some alluvium horizons and ground cracks (fissures) exposed in paleoseismic trenches that the applicant excavated to investigate faults. The applicant concluded that the cracks were caused by faulting. The applicant indicated that several such ash occurrences found in trenches dug across the Solitario Canyon Fault near Yucca Mountain were from the eruption of the youngest (~77,000 years old) Lathrop Wells volcano. On the basis of these observations, the applicant suggested that Lathrop Wells volcanic activity and faulting were linked.

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant provided in SAR Section 1.1.6.1.2 and references therein to evaluate the relationship between seismic and igneous activity near the proposed repository site. The NRC staff notes that information the applicant provided concerning the possibility of a causal relationship between volcanism and seismicity [i.e., the ground cracks (an earthquake hazard) and an ash-producing eruption (an igneous hazard) were caused by the same earthquake at the same time] is inconclusive. This is because, in the case of faulting occurring before or during ash deposition, new fresh ash could be swept into fractures by wind or water movements. Alternatively, if the ash predated faulting, the disturbed ash could similarly work its way down into fractures. The possibility of a causal relationship between volcanism and seismicity has not been conclusively demonstrated by the applicant. However, the NRC staff concludes that any potential relationship between seismic activity and igneous activity is appropriately accounted for by the PVHA, PVHA-U, and PSHA estimates (CRWMS M&O, 1996aa; SNL, 2008ah; CRWMS M&O, 1998aa) for both future igneous and seismic activity. This is because whether a causal relationship exists or not, the current estimates of the probability of occurrence already include both the paleoseismic events mapped on the Solitario Canyon Fault and the Lathrop Wells volcano. The NRC staff finds that any potential relationship between seismic and igneous activity is not significant for risk to preclosure performance because the probability of igneous activity does not exceed $1 \times 10^{-6}$ per year, as discussed in the next subsection.

**Probabilistic Igneous Hazard Analysis**

In SAR Section 1.1.6.2, the applicant assessed the likelihood of future basaltic igneous activity in the repository area, together with an estimate of the uncertainty associated with that probability, by relying upon the result of a PVHA (CRWMS M&O, 1996aa; BSC, 2004bi) and the information provided in the applicant's external events hazard screening analysis in BSC (2008ai, Section 6.3).

During the 100-year-preclosure period, the probability of future igneous activity affecting the repository was compared to a criterion of less than a 1 in 10,000 chance of an event occurring during the 100-year-preclosure period (i.e., $1 \times 10^{-6}$ per year), as identified in SAR Table 1.6-1. On the basis of the applicant's PVHA, the applicant determined (i) the mean annual frequency of the likelihood of a basaltic dike intruding the underground repository as $1.7 \times 10^{-8}$ (see also BSC, 2004af) and (ii) the mean conditional annual frequency of occurrence of one or more volcanic eruptive centers (i.e., an intrusive dike that reaches the surface and leads to an eruption) within the subsurface facility ranges from $4.8 \times 10^{-9}$ to $1.3 \times 10^{-8}$ (SAR Section 2.3.11). The applicant performed other evaluations that supported these values, which the NRC staff reviewed as part of the postclosure review in SER Section 2.2.1.2.2).

These values indicate the general range of probabilities the applicant determined for an igneous intrusion into, and a volcanic eruption within, the subsurface GROA. Actual probability values applicable to the preclosure period were discussed in the applicant's external events hazard screening analysis in BSC (2008ai, Section 6.3) as lower than $10^{-6}$ per year. An applicant-conducted PVHA update (SNL, 2008ah) made similar conclusions about the annual probability of future intrusive and volcanic activity at the repository site.

In general, the probability of a dike intruding the repository, according to the applicant's igneous consequence peer review panel (BSC, 2003ae), ranged between $1 \times 10^{-8}$ and $1 \times 10^{-7}$. Therefore, the applicant stated that the likelihood of future igneous activity directly impacting the subsurface repository site during the preclosure period is much lower than $1 \times 10^{-6}$ per year.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information about probabilistic igneous hazard analyses in SAR Section 1.1.6 and references therein by conducting independent studies of the nature and frequency of past basaltic volcanism in the Yucca Mountain region (Hill and Connor, 2000aa; Conner, et al., 2000aa; Stamatakos, et al., 2007aa). The NRC staff concludes that the applicant correctly assessed the probability of future basaltic volcanism and, on the basis of the NRC staff's independent studies, that DOE probability estimates are sufficiently low to exclude volcanic hazard assessment from the PCSA and from consideration in the GROA design.

**Potential Hazard from Ash Fall from Distant Active Volcanoes and Volcanic Fields in the Region**

The applicant described the potential effects of fallout of volcanic ash (tephra) on the GROA (SAR Section 1.1.6.3; BSC, 2008ai). The applicant concluded that future volcanic ash falls that may impact the proposed repository site could come from active volcanoes far from the Yucca Mountain region, such as from California, and also from the local fields of basaltic volcanic activity described in the previous section. The applicant considered past volcanic activity from distant sources over a time scale of 100,000 years because this time period captures many small volume eruptions from distant, active volcanic source areas, such as small explosive eruptions from rhyolitic volcanoes in California (SAR Section 1.1.6.3; DOE, 2009ap).

The applicant determined that these small explosive rhyolitic eruptions in California would deposit less than 1 cm [0.4 in] of ash over the Yucca Mountain region if future activity of the most likely volume and type occurred. Perry and Crowe (1987aa) stated that even the most likely potential distal activity has less than a 1 in 10,000 chance of occurring within the preclosure period for surface activities (DOE, 2009ap). Further, the applicant recognized that this type of activity would deposit less ash fall at the repository site than basaltic volcanoes located closer to the proposed repository site (SAR Section 1.1.6.3).

**Potential Ash Fall from Distant (Caldera) Volcanoes**

The applicant considered the ash-fall hazard posed by extremely rare distal explosive eruptions, such as large caldera-forming events at Yellowstone (Wyoming) and Long Valley (California) that occurred within the past 1 million years. In the past, such eruptions have deposited ash falls up to a few tens of centimeters [~10–20 in] in the Yucca Mountain area, as described in Perry and Crowe (1987aa, p. 12). However, on the basis of present knowledge of the Yellowstone and Long Valley magma systems, the likelihood of ash fall from Yellowstone or Long Valley onto Yucca Mountain was estimated at less than a $1 \times 10^{-6}$ per year probability of recurrence (DOE, 2009ap).

**Potential Ash Fall from Nearby (Basaltic) Volcanoes**

The applicant presented information that showed that ash fall from nearby future basaltic eruptions in the Southwest Nevada Volcanic Field, similar to the Lathrop Wells volcano (located south of the repository site), would deposit a range of ash thicknesses from 0.5 to 3 cm [0.2 to 1.2 in] on the repository site. This thickness also is greater than the potential ash-fall thickness from small rhyolitic volcanoes in California. The applicant found the average probability of recurrence of basaltic volcanism that could deposit a few centimeters of ash on the repository site in the preclosure period was low and, on the basis of the applicant-conducted PVHA (CRWMS M&O, 1996aa), concluded that it was less than $10^{-6}$ per year, as described in SAR Section 1.6.3.4.3 and BSC (2008ai, Section 6.3).

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant provided regarding potential ash fall in SAR Section 1.6.3.4.3, and references therein, using its own independent field observations and estimations of likely ash-fall thicknesses (e.g., Hill and Connor, 2000aa; Conner, et al., 2000aa; Stamatakos, et al., 2007aa), as well as NRC staff knowledge and experience. The NRC staff concludes that the applicant adequately determined the ash-fall hazard to Yucca Mountain from the distant calderas of Yellowstone and Long Valley and that the hazard has less than a 1 in 10,000 chance of recurring within the 100-year-preclosure period. The NRC staff also concludes that exclusion of ash fall from nearby basaltic volcanoes is acceptable because the probability of these types of eruptions is less than $1 \times 10^{-6}$.

**NRC Staff's Conclusion**

The NRC staff finds the applicant adequately described information regarding past and possible future volcanism in the Yucca Mountain area, including the age, timing, and location of past igneous intrusion events. The NRC staff finds that the applicant adequately considered the possibility of future igneous activities, including volcanic eruption, subsurface magmatic activity, and volcanic ash fall and flow affecting the site, relevant to the duration of the preclosure period and based on the knowledge of past volcanic events. The NRC staff finds the applicant's information is consistent with the NRC staff's knowledge based on numerous site visits and with independent analyses conducted on igneous intrusion events. Therefore, the NRC staff finds, with reasonable assurance, that the applicant's information regarding regional igneous activity is acceptable for use in the evaluations in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c).

#### 2.1.1.1.3.7 Site Geomorphology

In SAR Section 1.1.7, the applicant assessed geologic landforms and geomorphic processes that might influence evaluations in the PCSA and affect structures or operations at the GROA during the preclosure period. These geologic processes may significantly alter surface topography and include erosional and depositional processes, such as running water, wind, rock weathering, and soil development. The applicant assessed the site's landscape response to climate change and erosional and depositional processes.

**Geomorphic Information and Tectonic Activity**

**Erosion, Erosion Rates, and Deposition**

The applicant conducted geomorphic studies in the Yucca Mountain region to characterize the site, as described in BSC (2004bi, Section 3). On the basis of these studies, the applicant described Yucca Mountain as a series of north-trending ridges and valleys separated by high-angle faults. The fault blocks are tilted eastward, such that the west-facing slopes are generally high, steep, and straight, in contrast to the gentler and commonly deeply dissected, east-facing slopes. The applicant's mapping and trenching studies identified some faults that were active during the Quaternary Period (over the approximately last 2 million years) and were exposed at the surface. The applicant observed slopes with flat eroded surfaces covered by a thin veneer of alluvium and colluvium (indicating stable or balanced transport processes on hill slopes), as well as many angular ridges, narrow and V-shaped valleys, and some steep hill slopes and fault scarps not yet eroded. The applicant concluded that these geomorphic observations support a slow rate of erosion for the region.

Additionally, the applicant presented geomorphic information related to volcanism in the Yucca Mountain region that it used to determine erosion rates. The applicant examined cinder cones (also known as scoria cones) and their associated basaltic lava flows in Crater Flat. The degree of cinder cone erosion correlates with age. Cones that formed ~77,000 to 1 million years ago in Crater Flat are only slightly eroded, whereas those that formed approximately 3.7 million years ago are deeply eroded, exposing internal dikes. Such information is evidence of low erosion rates in Crater Flat.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of erosional and depositional processes and landforms at and near the repository site, relevant to the preclosure period, in SAR Section 1.1.7 and references therein. The NRC staff also conducted field observations of faults and erosion of the ~77,000-year-old Lathrop Wells cinder cone, as well as analogous geologic sites, during independent structural geology and volcanology studies in the Yucca Mountain region (Conner, et al., 2000aa; Hill and Connor, 2000aa). At the Lathrop Wells cinder cone, the NRC staff identified evidence for limited amounts of erosion, including shallow dissection of the cone flanks, modest expansion of the flanking (neighboring) debris apron (deposits) by slope wash and mass wasting, rounding of the crater rim, and partial infilling of the summit crater.

Also during field observations, the NRC staff observed 25-m [82-ft]-deep gullies incised into the sand ramps banked against the west slope of Busted Butte. Sand ramps at Busted Butte and in southeastern Midway Valley consist of wind-blown and hill slope deposit sequences. The NRC staff concludes that hill slope erosional processes were slow acting during the last half of the Quaternary Period. This is based on evidence of the effect of rare debris-flow-stripping events on the hill slopes around Midway Valley, the preservation of essentially unconsolidated sandy sediments on Yucca Mountain and Busted Butte hill slopes, and the exposure ages of hill slope boulder trains, among other indicators. The NRC staff finds that the applicant's geomorphic investigations and descriptions are acceptable because the applicant obtained rates of erosion on the basis of fault-scarp erosion of known ages, ages of boulders on hill slopes, erosion rates of cinder cones of known ages, and analyses of stream incisions and alluvial surfaces using standard and reasonable methods of analyses. The applicant's investigations and descriptions are consistent with the NRC staff's independent observations and analyses of geomorphic processes, landforms, and erosion rates (Ferrill, et al., 1996aa).

As part of its independent analyses, the NRC staff identified potential neotectonic (Quaternary Period and recent) movements in the lower reaches of Fortymile Wash that have influenced erosional and depositional processes in that area since the latter part of the Quaternary Period (McKague, et al., 2006aa; Sims, et al., 2008aa). The effects on the landscape are at lower elevation than, and beyond the boundary of, the GROA. The NRC staff finds that the continuing aggradation and slow westward migration of the lower part of Fortymile Wash is not a geomorphic hazard to the GROA or preclosure operations because the effects of sedimentation and lateral migration cannot impinge on the distant GROA within a period of hundreds of years.

Therefore, the NRC staff finds that the applicant's assessment of the potential erosion of the land surface, aggradation of stream valleys, and mass wasting or rapid fluvial degradation in channels and interfluves (land separating adjacent stream channels) during the preclosure period is adequate for the applicant to use in its PCSA and to support the GROA design.

**Variability of Quaternary Processes**

In SAR Section 1.1.7.2, the applicant described how climate variability during the Quaternary Period affected landforms and rates of erosional and depositional processes in the Yucca Mountain region. The model adopted by the applicant for landscape response is area-specific and builds upon a general semiarid landscape model, as described in BSC (2004bi, Section 3). Under present conditions, according to the applicant, most runoff takes place during infrequent, intense, short-duration summer thunderstorms. This process activates unconsolidated slope material to produce debris flows. The applicant indicated in BSC (2004bi, Section 3) that such debris flows are infrequent events. As an example, the applicant described the 1984 debris flow triggered on Jake Ridge, located approximately 6 km [3.7 mi] northeast of the Yucca Mountain crest. The recurrence interval of a mass-wasting event of this magnitude was estimated by the applicant to be much longer than 500 years (BSC, 2004bi).

The applicant concluded that over the next 10,000 years, under climatic conditions similar to the present, the current rates of sediment accumulation around Yucca Mountain should continue. This accumulation rate is aggradational (positive) rather than degradational (negative or eroding) and consists of a slow buildup of sediment on valley floors from alluvium, dust deposition (e.g., Reheis and Kihl, 1995aa), and occasional debris flows such as the Jake Ridge event (BSC, 2004bi). Unless a future change in climate occurs, the aggradation will continue at slow rates. If a climatic change toward wetter conditions occurs, the applicant concluded that degradation may overtake aggradation; however, it would take substantially more than 10,000 years for erosion to remove alluvium and start eroding bedrock in the valleys above the underground repository within Yucca Mountain (BSC 2004bi).

**NRC Staff's Evaluation**

The NRC staff reviewed the information in SAR Section 1.1.7 and the applicant's cited published information on the geomorphological processes using its own knowledge derived from field observations of the depositional processes in the Yucca Mountain region. The NRC staff finds that the applicant's description of these surface features and rates of erosional and depositional processes as aggradational and slow is consistent with the climate setting of the region, and they are consistent with the NRC staff's observations from numerous site visits during the prelicensing period. The NRC staff also finds that the applicant's description of future depositional processes as degradational and its conclusion that this process could not take place quickly enough to remove alluvium and start eroding bedrock at Yucca Mountain within the preclosure period is reasonable, based on the NRC staff's observations from numerous site visits and the NRC staff's knowledge of similar degradational processes.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant adequately described information regarding Yucca Mountain site geomorphology, including descriptions of landforms and erosional and depositional processes, and whether site structures or operations could be affected by a geomorphic hazard. The NRC staff finds that the information is consistent with the information in publicly available literature on the subject and with the staff's own independent field observations and evaluations. The NRC staff finds that the applicant's conclusions are based on geomorphic studies using standard techniques and methodologies to determine ages of rocks, erosion rates, and stream incision bed formation. Therefore, the NRC staff finds, with reasonable assurance, that the applicant's information regarding site geomorphology is acceptable for use in the evaluations in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c).

### 2.1.1.1.3.8 Site Geochemistry

The applicant described Yucca Mountain site geochemistry applicable to the preclosure period in SAR Section 1.1.8. The applicant cited BSC (2004bi, Sections 3.3.5.1 and 5.2.2) for details of subsurface water chemistry and the geochemistry of rock units associated with the subsurface GROA. SAR Sections 2.3.3 and 2.3.5 contain information about site geochemistry, including porewater geochemistry, evolution of porewater chemistry at elevated temperatures, past hydrothermal alteration of the host rock, distribution and reactivity of minerals in the rock units, dust deliquescence, and the composition of airborne dust particles that may accumulate in the repository drifts. The applicant stated that the site geochemistry and geochemical processes discussed in SAR Section 1.1.8 are unlikely to affect safety of the facilities and operations of the GROA during the preclosure period.

The applicant stated that elevated temperatures, which are due to heat output from waste packages emitted by the waste forms, will be at maximum values during the preclosure period (SAR Sections 1.1.8.1 and 1.1.8.4.2) while the packages are being placed into the repository drifts. Subsurface repository construction will introduce dust, chemical residues from construction activities, and other anthropogenic (man-made) materials as potential chemical reactants in the repository drifts. In describing preclosure site geochemistry, the applicant focused on characteristics of the near-field environment (i.e., the excavated repository drifts and adjacent host rock) and how preclosure activities would affect near-field geochemical conditions. In SAR Section 1.1.8, the applicant identified four factors associated with the preclosure period that would modify current geochemical conditions in the near-field environment: elevated temperatures, gamma radiation, underground construction activities, and underground forced ventilation. The applicant stated that the potential geochemical effects would be mitigated by the applicant's preclosure facility design, which calls for continuous forced ventilation by fans in the subsurface GROA (SAR Section 1.3.5). The applicant stated that the subsurface ventilation system, which circulates air for workers and removes decay heat from the waste packages, plays an important safety role and conforms to the regulatory requirements of the Occupational Safety and Health Administration and the Mine Safety and Health Administration (SAR Section 1.3.5).

The preclosure effects on geochemistry from elevated temperatures, gamma radiation, and construction activities, as well as the effects of the subsurface ventilation system on geochemistry, are further evaluated in the following sections.

**Elevated Temperature and Ventilation Effects**

The applicant assessed how elevated temperatures could affect site geochemistry by modifying mineral dissolution, alteration, and precipitation reactions between rocks and the water in pore spaces and fractures, and how this could affect preclosure safety (SAR Section 1.1.8.1). Water–rock interactions, if extensive, have the potential to modify (i) physical and chemical properties of the near-field rock mass (SAR Sections 2.3.3 and 2.3.5) and (ii) the composition of water that may later enter the repository drifts as seepage after the temperatures decrease (SAR Section 2.3.5). The applicant included the modified composition of seepage into repository drifts due to elevated temperatures (SAR Section 2.3.5.3) in postclosure performance assessment calculations because the water chemistry potentially affects the corrosion rates of engineered barrier materials. The applicant stated that during the preclosure period, the continuous forced ventilation of hot, dry air in the repository drifts would limit any geochemical effects by reducing the availability of water in the drifts resulting in a region of dry rock, referred to as the dryout zone, that would extend several meters into the surrounding rock from the drift walls (SAR Section 1.1.8.1). In particular, the applicant assumed that the preclosure forced ventilation system would limit geochemical interactions between rocks and water in the near field by (i) lowering the relative humidity and overall temperature in the near-field environment and (ii) drawing water vapor out of the rock, then out of the repository, instead of allowing the water vapor to condense in the host rock as would happen for postclosure near-field conditions. To support the technical basis for a preclosure dryout zone in the wall rock, the applicant cited field observations of wall rock dewatering due to forced ventilation in the ESF under ambient conditions (SAR Section 2.3.3) and thermal-hydrologic-chemical and seepage evaporation modeling analyses, as described in BSC (2004bg, Section 6.6) and SNL (2008aj, Section 7.5.2).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of site geochemistry in SAR Section 1.1.8 and references therein. The NRC staff evaluated how DOE addressed the potential for geochemical interactions, including corrosivity, in the near-field host rock under current conditions and subject to elevated temperatures and ventilation effects during the preclosure period. First, the NRC staff evaluated the applicant's description of near-field geochemistry of the composition of subsurface waters, specifically matrix porewaters, and host rock mineralogy.

Based on the NRC staff's understanding of the Yucca Mountain natural system, as obtained from extensive prelicensing experience, laboratory analysis, and field sampling, the NRC staff finds that the applicant's descriptions are adequate. Further, based on the NRC staff's understanding of coupled heat transfer processes in unsaturated tuffs and dryout zones in ventilated excavations, the NRC staff finds that, because of the elevated temperature and very low relative humidity, the amount of water available to enter the drifts or cause geochemical interactions with host rock, such as formulation of corrosive waters, would be negligible. Therefore, the NRC staff concludes that the applicant adequately described site geochemistry as it relates to the PCSA and the GROA because the applicant (i) appropriately identified dissolution, alteration, and precipitation of minerals as the main geochemical interactions potentially affecting the repository host rock and near-field water chemistry during the preclosure period and (ii) adequately described how forced ventilation during the preclosure period would limit geochemical interactions at elevated temperatures in the rocks around the drifts. The NRC staff further evaluated the applicant's characterization of geochemical composition of subsurface matrix and pore waters and geochemical composition of the rock strata in SER Section 2.2.1.3.3.

**Gamma Radiation and Ventilation Effects**

In describing the geochemistry of the Yucca Mountain site for preclosure conditions, the applicant stated that radiation emitted by the emplaced waste packages would be at maximum values during the preclosure period and might affect water–rock interactions in the repository near field (SAR Section 1.1.8.4.2). The applicant conducted irradiation experiments to investigate radiation effects on repository host rock and observed that even at much higher doses than anticipated in the repository, gamma radiation damage was limited to small changes in the mechanical properties of minerals due to the radiolysis of water in the rock samples. The applicant cited field observations and coupled heat transfer modeling analyses to support the assumption that forced ventilation and elevated temperatures during the preclosure period would limit the availability of water for radiolysis in the repository near field. The applicant concluded that radiation was not important in terms of preclosure site geochemistry because (i) even at maximum field strength, the gamma radiation would penetrate no more than a few centimeters [inches] into the repository host rock and (ii) the scarcity of water in the rocks within the dryout zone resulting from ventilation effects would greatly reduce any geochemical interactions caused by radiolysis.

**NRC Staff's Evaluation**

The NRC staff reviewed the information about geochemical radiation effects and radiolysis experiments provided by the applicant in SAR 1.1.8.4.2, and references therein, to determine how the applicant related the potential effects of gamma radiation to the geochemistry of the repository near-field environment. On the basis of the NRC staff's understanding of Yucca Mountain site conditions, radiation physics, coupled heat transfer processes in unsaturated tuffs, and the formation of dryout zones in ventilated excavations, the NRC staff concludes that the applicant's description of preclosure radiation effects in the context of site geochemical characteristics is adequate. The NRC staff also finds acceptable the applicant's conclusion that gamma radiation would not have an important effect on the near-field rocks, because the experiments resulted in negligible impacts even though the applicant used radiation levels that were much higher than expected for the preclosure period. Moreover, the NRC staff finds that the experimental factors were conservative because they did not include the effects of attenuation of radiation by additional shielding that otherwise would be provided in a repository setting by the waste packages and casks. The NRC staff finds that the applicant's description of how the geochemical effects of radiolysis of porewater in drift walls would be minimized by the presence of a dryout zone is acceptable because it is consistent with the NRC staff's understanding of expected conditions in the subsurface facility during the preclosure period, based on the NRC staff's knowledge gained through experience and literature reviews (e.g., Tsang, 2012aa; Gascoyne, et al., 1996aa).

**Construction Activities and Ventilation Effects**

The applicant stated that subsurface construction activities, including excavation of the repository, will introduce rock dust and limited amounts of anthropogenic materials in the drifts (e.g., explosives residue, diesel exhaust, lubricants, coolants, and solvents) during the preclosure period. As described in SAR Sections 1.1.8.3 and 1.1.8.4.2, these materials could serve as potential geochemical reactants, particularly if particles settled on waste package surfaces and affected metal corrosion rates. The applicant also identified atmospheric dust, brought into the repository by the forced ventilation system, as a potential source of geochemical reactants on metal surfaces during the preclosure period.

The applicant concluded that the presence of hot, dry air in the drifts from the continuous forced ventilation system would limit any geochemical interactions for several reasons: (i) during the preclosure period, salts produced by evaporation of porewater would precipitate within the rock dryout zone instead of on drift walls, thereby limiting the salt crystals' mobilization as dust particles in the drift; (ii) any potential seepage of water into the drift during preclosure would be limited by the presence of the dryout zone in the rock around the drift and by the tendency of water in unsaturated rocks to divert around large openings such as the repository drifts; (iii) elevated temperatures in the drifts would cause any potentially corrosive ammonium salts to volatilize and be carried away by the preclosure ventilation system; and (iv) the removal of moisture by the preclosure ventilation system would lower the relative humidity in the drifts that otherwise might contribute to the corrosion of metals in humid air or absorption of water vapor by salts on container surfaces (SAR Section 1.1.8.3).

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant provided about construction activities and ventilation effects in SAR Section 1.1.8 and references therein and finds that the applicant provided sufficient information to support an evaluation of how these site-specific geochemical components may contribute to the corrosivity of water in the repository near-field environment during the preclosure period. The SAR discusses the proposed ventilation system in detail in BSC (2004bg). The NRC staff's evaluation of this system is in SER Section 2.2.1.2 (Subsection 2.1.1.2.3.2.4). On the basis of the NRC staff's understanding of coupled heat transfer processes in unsaturated tuffs and the formation of dryout zones in ventilated excavations (e.g., Tsang, 2012aa; Gascoyne, et al., 1996aa), the NRC staff concludes that the applicant has adequately described the limited geochemical effects of dust during the preclosure period, namely that the presence of elevated temperatures and the use of forced ventilation in the drifts would minimize the availability of water to react with the dust particles and other materials. Given the elevated drift temperatures expected during preclosure, the volatility of ammonium salts, and the use of forced ventilation during the preclosure period, the NRC staff concludes that the applicant has acceptably described the limited contributions of ammonium salts to the corrosivity of water in the near-field environment.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant sufficiently described Yucca Mountain site geochemistry to support the identification of naturally occurring or human-induced effects in the context of preclosure repository safety. The NRC staff finds that the applicant provided an adequate technical basis for the description of existing site geochemical conditions and potential changes in geochemistry during the preclosure period. The NRC staff finds the applicant adequately demonstrated that the preclosure ventilation system and heat output from the waste packages would reduce the availability of water in the rock, thereby limiting geochemical reactions such as (i) the precipitation, dissolution, or alteration of minerals in the near-field rock; (ii) changes in mineral structure or porewater geochemistry due to radiolysis; and (iii) the potential corrosive effects of salts on waste package surfaces. Therefore, the NRC staff finds, with reasonable assurance, that the applicant's information on site geochemistry is acceptable to use in the evaluations in the PCSA, to support the GROA design, and satisfies 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c).

### 2.1.1.1.3.9 Land Use, Structures and Facilities, and Residual Radioactivity

The applicant provided the following information in SAR Section 1.1.9 to determine potential human-induced hazards at the site that could impact the GROA and, therefore, needed to be evaluated in the PCSA and considered in the design: (i) previous land uses to identify potential land use conflicts, (ii) whether existing structures or facilities are likely to interfere with planned preclosure activities, and (iii) the potential for exposures to the public or workers from residual radiation within the land withdrawal area.

**Previous Land Use**

The applicant summarized previous land uses in SAR Sections 1.1.9.1 and 1.1.9.2 for the proposed land withdrawal area of 59,500 ha [147,000 acres] and in the vicinity of the proposed land withdrawal area. Historically, the land has been under the federal control of DOE, the U.S. Bureau of Land Management, and the U.S. Air Force, and potential land use conflicts would have been among these agencies. The applicant described various military, government, and commercial land uses that occurred prior to submission of the license application. At the time of the application, the applicant stated the land on which the GROA will be located is covered by legal interests, which take the form of two rights-of-way, an administrative land withdrawal, and a public land order (SAR Section 5.8.1.1).

The applicant identified existing mining claims located just outside and just within the southern boundary of the proposed preclosure controlled area. The applicant also identified and described both patented and unpatented mining claims located about 15 km [9 mi] south of the proposed GROA (SAR Sections 2.1.1.1.3.1, 5.8.2.2.1, and 5.8.2.2.2). In addition to the mining claims, the applicant identified a borrow pit (an area where soil material has been dug for use in another location) located within the proposed withdrawal area.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information in SAR Section 1.1.9 and references therein on previous land uses. The NRC staff finds that, at the time of license application, the applicant provided sufficient information on land use to identify previous and present uses of the land and the potential for conflicts with the use of the land for the GROA. The NRC staff finds this information sufficient because the NRC staff evaluated previous land usage at the site by reviewing the information from multiple publicly available sources (which are identified in the evaluation of site geography in SER Section 2.1.1.1.3.1). In DOE (2009au, Enclosures 7 and 8), the applicant stated that it will revise GI Figures 1-2 and 1-4 and SAR Figures 1.1-1 and 1.1-6 to update its controlled area boundary depicted to show that the approximately 0.8 $km^2$ [200 acres] of the mining claim acreage at the Lathrop Wells cinder cone (U.S. Patent 27-83-0002) area is private land that is excluded from the proposed land withdrawal area. While this correction affects the NRC staff's evaluation of the boundaries of the GROA (as discussed in SER Section 2.1.1.1.3.1), it does not impact the review considerations in this section, as the designation of the previous land use (mining claim) has not changed. Therefore, the NRC staff finds that the applicant has provided acceptable information to determine that, based on the information provided in the license application, there are no conflicts with the land use that need to be included as potential initiating events in the evaluations in the PCSA and to support the design of the GROA. Because DOE did not provide site characterization updates regarding previous land use beyond those cited in this evaluation, the NRC staff proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3. This condition of construction authorization would require DOE to confirm that land use characterization information and related analyses in the SAR continue to be accurate.

The NRC staff notes that additional and related information regarding land use, control, and ownership is provided by the applicant in SAR Sections 5.8.1 and 5.8.2.2. This information is reviewed by the NRC staff in SER Sections 2.3.8 and 2.3.9 in Volume 4.

**Existing Structures and Facilities**

The applicant provided a summary, including location maps, of the existing structures and facilities in the proposed land withdrawal area at the time of application in SAR Section 1.1.9.3. The applicant noted that there were no civilian facilities within the GROA. Because the land had been under federal control for many years, the only nongovernment facilities located within the proposed land withdrawal area (preclosure controlled area) were water wells associated with the Nye County Early Warning Drilling Program. As noted previously, these facilities were within the analyzed proposed land withdrawal area but outside of the GROA. Access roads from U.S. Highway 95 are short and terminate at these facilities.

All other existing surface structures and facilities noted in the information provided were associated with federal government activities, including surface facilities to support site characterization activities and environmental monitoring activities at Yucca Mountain. The applicant noted that these existing structures and facilities are subject to being replaced during construction activities at the GROA, in accordance with planned repository structures and facilities described in SAR Section 1.2.

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant provided on existing structures and facilities in SAR Section 1.1.9 and references therein. The staff finds that the applicant's information, at the time of license application, is adequate because the descriptions were verifiable and consistent when compared with other publicly available information (which are discussed in the evaluation of site geography, SER Section 2.1.1.1.3.1), and sufficiently comprehensive to determine impacts on or from these structures and facilities on GROA structures and facilities, and for use in the PCSA and in support of the GROA design. Because DOE did not provide site characterization updates regarding existing structures and facilities beyond those cited in this evaluation, the NRC staff proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3. This condition of construction authorization would require DOE to confirm that its structure and facility characterization information and related analyses in the SAR continue to be accurate.

**Potential Exposure to Residual Radioactivity**

**Radiological Surveys**

The applicant relied on several aerial radiological surveys, described in SAR Section 1.1.9.4, to determine whether there was residual radioactivity that could contribute to worker and public radiation exposures at the Yucca Mountain site. Two aerial surveys, performed in 1970 and 1976 [detailed in Hendricks and Riedhauser (2000aa) and Tipton (1979aa)], included the proposed land withdrawal area along Fortymile Canyon, which includes Fortymile Wash. Other surveys DOE relied on included Area 25 of the NNSS, as described in Hendricks and Riedhauser (2000aa) and Lyons and Hendricks (2006aa, Section 6.8). Area 25 is located east of the proposed Yucca Mountain site, and portions of Area 25 are within the withdrawal area. These surveys did not detect man-made radioactivity within the proposed land withdrawal area or GROA that could be identified through an aerial survey.

During a radiological survey DOE conducted in 1991 at reclamation trial area number 3 on the east side of Fortymile Wash (on the NNSS and within the proposed land withdrawal area), an isolated piece of radioactive material was identified that was believed to be present from previous NNSS operational activities. The material was recovered and removed (Sorensen, 1991aa).

A 2006 radiological aerial survey conducted by DOE (Lyons and Hendricks, 2006aa) examined the proposed land withdrawal area and the section of Area 25, located more than 8 km [5 mi] from the GROA, where nuclear rocket testing activities were performed. The survey did not detect any regions of anomalous activity within the proposed land withdrawal area in Area 25. However, five sites of man-made radiological activity were detected outside of the proposed land withdrawal area in Area 25.

**Emissions from the Nevada Nuclear Security Site**

In SAR Section 1.1.9.4, the applicant identified several sources of emissions at the NNSS during calendar year 2005 that could potentially result in exposure to the public and workers in the proposed land withdrawal area. These sources included a very small amount (less than 1 mCi) of tritium gas that is released to the environment when tritium monitors are calibrated. Other sources of tritium included evaporation of tritiated water from containment ponds, evaporation and transpiration of tritiated water from soil and vegetation at sites of past nuclear tests and from the Radioactive Waste Management Sites, and evaporation of tritiated water from a sewage lagoon. In addition to tritium, resuspension of plutonium and americium from soil contaminated by past nuclear testing continued to contribute to radioactive emissions.

The applicant relied on the NNSS air sampling stations that are required to monitor for radioactive airborne particulate and tritium contamination for data on the levels of contamination of these constituents that could impact the GROA and that would need to be included in the PCSA. Six of the sampling locations are near the boundaries and at the center of the NNSS, as outlined in Wills (2006aa, Section 3.1). The applicant estimated total tritium activity from all sources to be 6,290 GBq [170 Ci] in 2005. Activity of Pu-239/Pu-240 and Am-241 totaled 11 GBq and 1.7 GBq [0.29 and 0.047 Ci], respectively, as shown in Wills (2006aa, Table 3-13).

**Offsite Monitoring of Releases**

At the time of the license application, offsite releases of radioactive material from the NNSS were monitored using a monitoring network operated by the Community Environmental Monitoring Program and coordinated by the Desert Research Institute (SAR Section 1.1.9.4). One of the air sampling stations of this network that measures radionuclide air concentrations from the NNSS was located at the southern boundary of the proposed land withdrawal area. The applicant found that no airborne radioactivity related to historic or current NNSS operations and no man-made, gamma-emitting radionuclides were detected in any of the samples from the particulate air samplers during 2005, as detailed in Willis (2006aa). On the basis of these measurements, the applicant determined that the concentrations in 2004 and 2005 were less than 1 percent of the compliance levels for the national emission standards for hazardous air pollutants (SAR Appendix E, Table 2).

**Residual Radioactivity from Previous Land Uses**

In SAR Section 1.1.9.1.3 and 1.1.9.1.4, the applicant describes two locations of residual radioactivity within the proposed land withdrawal area from previous land uses. These locations are not within the boundaries of the GROA. One was the Army Ballistics Research Laboratory Test Range, located in the southeast corner of the proposed land withdrawal area at a distance of more than 16 km [10 mi] from the GROA. It was used for multiple open-air tests of depleted-uranium munitions. According to the information DOE provided in the application, the Army Ballistics Research Laboratory Test Range site implemented operational procedures that minimized leaving residual radioactivity at the facility, including (i) removing the remains of the depleted-uranium munitions after tests were conducted and (ii) removing and disposing of the depleted-uranium contaminated soil at an approved low-level radioactive waste management site. Additionally, the area used for tests was posted and fenced off, in accordance with DOE radiological protection requirements in 10 CFR Part 835.

The other site of residual radioactivity noted by the applicant was borehole USW G-3, located on the crest of Yucca Mountain. It contains a Cs-137 source that was lost on January 26, 1982, from a logging tool during cementation activities in the borehole. The source is thought to be encased in concrete between 38 and 39 m [125 and 128 ft] below ground surface. The borehole has been capped at the surface and posted and fenced as an underground radioactive material area in accordance with 10 CFR Part 835, as described in DOE (2001aa, Section 2.2.1.5).

The applicant stated that any residual radioactivity within the proposed land withdrawal area from either of these sites will make a negligible contribution to worker and public radiation exposure at the GROA.

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant presented in SAR Section 1.1.9 on potential exposure to residual radiation and finds that the applicant's data identifying residual radioactivity at the Yucca Mountain site are adequate, as of the time of application, to determine the potential for exposure to workers and the public because surveys were completed that would have identified any residual radioactivity from previous land uses. The NRC staff concludes that the emissions from the NNSS were fully characterized at the time of application because of the mandatory reporting requirements for the operator of the NNSS site. Further, the NRC staff finds this information to be consistent with the 2006 NNSS Environmental Report (Willis, 2006aa).

The NRC staff concludes that, as of the time of application, the offsite monitoring data fully characterized any offsite sources that could contaminate the Yucca Mountain site. The NRC staff also concludes that the applicant adequately identified the locations and source strengths of residual radioactivity from previous land uses near, but not in, the land withdrawal area. The applicant's radiation surveys included the entire land withdrawal area and would have detected residual radioactivity that could result in a significant dose to workers or the public. On the basis of the location and known source strength of the identified residual radioactivity, the NRC staff concludes that the data are sufficient to evaluate the contribution to worker and public radiation exposure from residual radioactivity. The NRC staff also finds, on the basis of the small quantity, low strength, and condition (capped and buried in the case of the sealed source) of sources that could contribute radioactivity and the distance of the sources from the activities DOE proposed to carry out at the GROA, that residual radioactivity would make a negligible contribution to worker and public radiation exposure. Because DOE did not provide site characterization updates regarding potential exposure to residual radioactivity beyond those cited in this evaluation, the NRC staff proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3. This condition of construction authorization would require DOE to confirm that its characterization information and related analyses regarding the potential exposure to residual radioactivity in the SAR continue to be accurate.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant (i) adequately described information regarding previous and ongoing land uses and existing structures and facilities in the vicinity of the Yucca Mountain site, (ii) adequately evaluated whether site structures or operations could be affected by potential land use conflicts, and (iii) adequately characterized any residual radioactivity within the proposed land withdrawal area (including the preclosure controlled area) to determine the potential for exposure to workers and the public. The NRC staff finds the information sufficient because the NRC staff evaluated previous land uses and radioactive contamination from multiple publicly available sources, that multiple surveys of radioactive contamination were comprehensive to find all previous sources of contamination, that previous and current site monitoring is thorough enough to identify any previous sources of contamination, and that the analysis to determine conflicts in land use or impacts from previous contamination are valid. Therefore, the NRC staff finds, with reasonable assurance, that the applicant's information regarding previous land use, existing structures and facilities, and potential residual radioactivity, is acceptable for use in the evaluations in the PCSA, in support of the GROA design, and satisfies 10 CFR 63.112(c). The NRC staff also proposes a condition of construction authorization, as stated in SER Section 2.1.1.1.3, that would require DOE to confirm that land use, existing structures and facilities, and residual readioactivity information and related analyses in the SAR continue to be accurate.

### 2.1.1.1.4 Evaluation Findings

The NRC staff reviewed the applicant's SAR and other information submitted in support of the license application relevant to site characteristics of the Yucca Mountain site important to the preclosure safety of the facility and the GROA design, and concludes, with reasonable assurance, that the relevant requirements of 10 CFR 63.21(c)(1)(i–iii), 10 CFR 63.21(c)(15), 10 CFR 63.112(b) and 10 CFR 63.112(c) are met. Specifically, the NRC staff finds that

- The license application adequately described the site geography, including the location of the GROA, with respect to the boundary of the site [10 CFR 63.21(c)(1)(i); 10 CFR 63.112(c)].

- The license application adequately described the regional demography, including information regarding the location of the local human population and its distribution to support a PCSA and GROA design [10 CFR 63.21 (c)(1)(i); 10 CFR 63.112(c)].

- The license application adequately described the local meteorology and regional climatology, including information to support a PCSA and GROA design [10 CFR 63.21(c)(1)(iii); 10 CFR 63.112(c)].

- The license application adequately described the local and regional surface and groundwater hydrology, including information to support a PCSA and GROA design, the probable maximum precipitation, and the probable maximum flood [10 CFR 63.21(c) (1)(iii); 10 CFR 63.112(c)].

- The license application adequately described the site geology and seismology to support a PCSA and GROA design [10 CFR 63.21(c)(1)(ii); 10 CFR 63.112(c)].

- The license application adequately described the historical and regional igneous activity to support a PCSA and GROA design, including the PVHA and consideration of amount and frequency of potential future ash falls on or near the site [10 CFR 63.21(c)(1)(ii); 10 CFR 63.112(c)].

- The license application adequately described site geomorphology to support a PCSA and GROA design, including descriptions of landforms and erosion and depositional processes, and whether a geomorphic hazard could affect site structures or operations [10 CFR 63.21(c)(1)(ii); 10 CFR 63.112(c)].

- The license application adequately described site geochemistry conditions to support a PCSA and GROA design [10 CFR 63.21(c)(1)(ii); 10 CFR 63.112(c)].

- The license application adequately described land use, structures, and facilities and residual radiation, including information on previous land use, potential impacts on existing structures and facilities, and potential for exposures from residual radiation to support a PCSA and GROA design [10 CFR 63.112(c)].

**Proposed Condition of Construction Authorization**:

Within 90 days of issuance of construction authorization, DOE must confirm that its site characterization information and related analyses in the SAR submitted in accordance with 10 CFR 63.21(c)(1) continue to be accurate with respect to (i) site boundaries; (ii) man-made features; (iii) previous land use; (iv) existing structures and facilities; and (v) potential exposure to residual radioactivity. DOE must provide to the NRC written notification when its confirmatory analysis is complete. This notification must include, for NRC staff's verification, a copy of DOE's confirmatory analysis.

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# CHAPTER 2

## 2.1.1.2 Description of Structures, Systems, Components, Equipment, and Operational Process Activities

### 2.1.1.2.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.2 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the United States Department of Energy's ("DOE" or "applicant") description of structures, systems, and components (SSCs); safety controls (SCs); equipment; and operational process activities, both Important to Safety (ITS) and not important to safety (non-ITS) in surface and subsurface facilities of the geologic repository operations area (GROA) for the application to receive a construction authorization under 10 CFR Part 63. The primary focus of Section 2.1.1.2 is for the NRC staff to assess the acceptability of the applicant's information related to descriptions of the design of SSCs, SCs, equipment, radioactive wastes to be disposed, and operations of the GROA facility and preclosure safety analysis (PCSA). The PCSA is used to identify Important to Safety (ITS) SSCs, SCs, and equipment that must perform their functions to comply with the preclosure performance objectives. For the NRC to issue a construction authorization, under 10 CFR Part 63, the applicant needs to provide, in part, adequate information for the staff to determine there is a reasonable assurance that the types and amounts of radioactive materials described in the application can be received and possessed in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. This determination on construction authorization does not require the applicant to finalize the design and operations for the entire facility, but addresses those items that may significantly influence the final design. The NRC staff's review to evaluate compliance with 10 CFR Part 63, a risk-informed, performance–based regulation, is commensurate with the safety significance of the SSCs and focuses on ITS items.

The applicant's description of the design of SSCs, SCs, equipment, and operational processes includes (i) civil and structural systems; (ii) mechanical systems; (iii) electrical power systems; (iv) heating, ventilation, and air conditioning (HVAC) systems; (v) radiation/radiological monitoring systems (RMS); (vi) types of radioactive waste; (vii) waste containers; (viii) instrumentation and control systems; and (ix) operation of the facilities.

The NRC staff evaluated the information in Safety Analysis Report (SAR) (DOE, 2008ab) Sections 1.2 through 1.14, 5.5, 5.6, and supporting documents, including the applicant's responses to the NRC staff requests for additional information (RAIs) (DOE, 2009ab,av,bb,ct,dh,di,dk-dx,dz,ea-eg,gk,gl,gu,gw 2008ab).

### 2.1.1.2.2 Regulatory Requirements

The regulatory requirements for the description of the design of SSCs, SCs, equipment, and operational process activities are in 10 CFR 63.21(c)(2), 63.21(c)(3)(i), and 63.21(c)(4). These regulations require that the SAR describe and discuss (i) structures, including general arrangement and dimensions; (ii) material properties and specifications; (iii) analytic and design methods, including applicable codes and standards; and (iv) kind, amount, and specifications of the radioactive material proposed to be received and possessed at the geologic repository operations area. The information provided by the applicant must satisfy the PCSA requirements identified in 10 CFR 63.112(a).

The NRC staff evaluated the descriptions of the design of SSCs, SCs, equipment, and operational process activities in the applicant's SAR using the guidance in Yucca Mountain Review Plan (YMRP), Section 2.1.1.2 (NRC, 2003aa). The relevant acceptance criteria in YMRP Section 2.1.1.2.3 are:

- The license application contains a description of the location of the surface facilities and their designated functions sufficient to permit evaluation of the preclosure safety analysis and the geologic repository operations area design.

- The license application contains descriptions and design details for structures, systems, and components, and equipment of the surface facilities sufficient to permit evaluation of the preclosure safety analysis and the geologic repository operations area design.

- The license application contains descriptions and design details for structures, systems, and components, and equipment of the subsurface facility sufficient to permit evaluation of the preclosure safety analysis and the geologic repository operations area design.

- The license application describes the characteristics of the spent nuclear fuel and high-level radioactive waste sufficient to permit evaluation of the preclosure safety analysis and the waste package design.

- The license application provides a general description of the engineered barrier system and its components sufficient to support evaluation of the preclosure safety analysis and the engineered barrier system design.

- The description of the operational processes to be used at the geologic repository operations area is sufficient for review of the preclosure safety analysis (PCSA).

In addition to reviewing the descriptions and discussions of the design of SSCs, SCs, equipment, and operational process activities mentioned before, the NRC staff reviewed the design of underground openings, subsurface ventilation system, and invert structure and rails in this SER section instead of in SER Section 2.1.1.7, Design of Structures, Systems, and Components Important to Safety and Safety Controls. This is because SER Section 2.1.1.7 only focuses on the design of SSCs ITS and safety controls. The underground openings, subsurface ventilation system, and invert structure and rails are non-ITS, but the applicant stated that it will rely on them to perform functions important to subsurface facility operations relevant to the applicant's demonstration of compliance with NRC regulations. The designs of the subsurface ventilation system and invert structure and rails provided by the applicant are evaluated for compliance with 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), 10 CFR 63.111(e), 10 CFR 63.112(a), and 10 CFR 63.112(f). Section 63.111(e) requires that the geologic repository operations area be designed to preserve the option of waste retrieval, and 10 CFR 63.112(f) requires that the description and discussion of design include design bases and their relation to design criteria. The design of underground openings provided by the applicant is evaluated for compliance with 10 CFR 63.111(d), 10 CFR 63.111(e), 10 CFR 63.112(a), and 10 CFR 63.112(d). Section 63.111(d) requires that the geologic repository operations area be designed to permit implementation of a performance confirmation program, and 10 CFR 63.112(d) requires that a technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards be included in the PCSA.

In addition to the guidance in YMRP Section 2.1.1.2, the NRC staff used the guidance in YMRP Section 2.1.1.7.3 (NRC, 2003aa), where applicable, to evaluate the design information for underground openings, subsurface ventilation system, and invert structure and rails provided by the applicant. The relevant acceptance criteria in YMRP Section 2.1.1.7.3 are as follows:

- The relationship between the design criteria and the requirements specified in 10 CFR 63.111(a) and (b), the relationship between the design bases and the design criteria, and the design criteria and design bases for structures, systems, and components important to safety are adequately defined.

- The geologic repository operations area design methodologies are adequate.

- The design assumptions, codes, and standards used for the design of subsurface facility structures, systems, and components important to safety are acceptable.

- The design of subsurface operating systems is adequate.

- The materials and material properties used for the subsurface facility design are appropriate.

- The design analyses use appropriate models and site-specific properties of the host rock and consider spatial and temporal variation and uncertainties in such properties.

- The design of ground support systems is based on appropriate design methodologies and interpretations of modeling results.

- The subsurface ventilation systems are adequately designed.

- An adequate maintenance plan exists for subsurface facility structures, systems, and components, equipment, and controls important to safety.

- The waste package and engineered barrier system structures, systems, and components and their controls are adequately designed.

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA. The applicability of such NRC guidance is discussed in greater detail in the sections where they were used as part of the application or the NRC staff's review.

### 2.1.1.2.3 Technical Review

The structure of Safety Evaluation Report (SER) Section 2.1.1.2 follows the review guidance provided in YMRP Section 2.1.1.2 for evaluation of the applicant's description of design of the GROA facilities and in YMRP Sections 2.1.1.2 and 2.1.1.7 for evaluation of design of the subsurface ventilation system, invert structure and rails, and underground openings. The applicant provided the information on description and design of SSCs, SCs, and equipment in Safety Analysis report (SAR) Sections 1.2 through 1.14. SER Section 2.1.1.2.3.1 discusses the location and functions of surface facilities. SER Section 2.1.1.2.3.2 covers the SSCs, SCs, equipment, and utility systems for the surface facilities. The main surface facilities include the receipt facility (RF), initial handling facility (IHF), canister receipt and closure facility (CRCF), wet handling facility (WHF), and the aging facility (AF). SER Section 2.1.1.2.3.3 details the SSCs, equipment, and utility systems for the subsurface facilities. SER Section 2.1.1.2.3.4 describes high-level radioactive waste (HLW) characteristics. SER Section 2.1.1.2.3.5 covers the engineered barrier system (EBS) components (e.g., drip shield, waste package), spent nuclear fuel (SNF) waste canisters, and overpacks. SER Section 2.1.1.2.3.6 covers the operational processes associated with the geologic repository operations area (GROA) and reviews the communication, instrumentation, and control systems for both surface and subsurface facilities. SER Section 2.1.1.2.3.7 reviews the design of the subsurface ventilation system, invert structure and rails, and underground openings of the subsurface facility.

SER Section 2.1.1.2 provides the NRC staff's review of the applicant's (i) description of the SSCs, SCs, and equipment; (ii) description of GROA operational activities; (iii) drawings and figures showing basic geometry and dimensions; and (iv) information on materials. The NRC staff reviewed the functions of the SSCs and equipment in the context of operations and any interaction with other SSCs. For the important to safety (ITS) SSCs, the NRC staff evaluated whether the codes and standards the applicant proposed for the design are applicable. The NRC staff evaluated the adequacy of the information in the SAR on the design description and functions of the SSCs, SCs, and equipment in the context of operations, and its use in the review of the applicant's PCSA results and design. In accordance with the YMRP guidance, the level of NRC staff review to evaluate compliance with 10 CFR Part 63, which is risk informed and performance based, is commensurate with the safety significance of the SSCs.

#### 2.1.1.2.3.1 Description of Location of Surface Facilities and Their Functions

The applicant provided an overview of the surface facilities and their associated operations in SAR Section 1.2.1. Information provided in the Yucca Mountain Repository License Application General Information Volume, Section 1.1 presented a general description of the proposed geologic repository at Yucca Mountain, location of the geologic repository operations area, and information on the proposed activities at the site. General Information Figures 1-4 and 1-6 showed the boundary of the controlled area for the preclosure phase of the project and planned layout of the surface facilities and their relative locations with respect to the site boundary. The surface facilities will include waste handling facilities, surface transportation network, balance-of-plant facilities, flood control features, and support systems. The waste handling facilities will include the initial handling facility (IHF), canister receipt and closure facility (CRCF), wet handling facility (WHF), receipt facility (RF), and aging facility (AF). The structures of the IHF, CRCF, WHF, and RF were designated as important to safety (ITS). The applicant designated aging pads of the AF to be ITS structures. Applying the 10 CFR Part 63 requirements for identifying ITS SSCs, the applicant identified the following surface structures as non-ITS: central control center facility (CCCF), emergency diesel generator facility (EDGF), cask receipt security station, and the low-level radioactive waste facility (LLWF). The applicant proposed an ITS system of dikes (levees) and ditches to prevent inundation of surface facilities from a potential probable maximum flood. SAR Sections 1.2.3 to 1.2.7 described the design and functions of the surface waste handling facilities. Descriptions of the balance-of-plant facilities were given in SAR Table 1.2.8-1.

SAR Figure 1.1-2 showed the GROA surface facilities within the restricted area boundary. The GROA site plan (SAR Figure 1.2.1-1) showed the location of major surface facilities, the aging pads, and the balance-of-plant facilities in relation to the North Portal. SAR Figure 1.2.2-7 showed the general layout of the flood control structures. SAR Figure 1.2.1-2 provided further details, such as the locations and orientations of the structures with respect to the North Portal. SAR Figure 1.2.1-4 addressed the sequence of movement of HLW at the GROA surface facilities. SAR Section 1.2.1.2 identified and discussed the primary functions of the major surface facility structures. The function of each waste handling facility is described in SAR Sections 1.2.3 through 1.2.7 and is discussed next.

The IHF will receive transportation casks containing naval spent nuclear fuel (SNF) or HLW canisters and prepare the casks for unloading. The operations in the IHF will place these canisters into the waste package, close the waste package, and load the waste package to a transport and emplacement vehicle (TEV) for transporting to the subsurface for emplacement in a drift. The other facilities used to load waste packages are the three CRCF facilities.

Each CRCF will receive and unload transportation casks containing transportation, aging, and disposal (TAD) canisters and HLW and DOE SNF canisters. The TAD canisters may also be received in aging overpacks. The canisters will be transferred to the waste packages, and the waste packages will be placed in the TEV for transporting to the subsurface facility. In the CRCF, the TAD canisters can also be moved from the transportation cask into an aging overpack for transportation to an aging facility.

The WHF will receive non-TAD canistered commercial spent nuclear fuel (CSNF) assemblies in a transportation cask. The CSNF assemblies are transferred, under water in a pool, into the TAD canisters. The TAD canisters will be removed from the pool, dried, inerted, sealed, and then placed in an aging overpack for transportation to a CRCF or the AF. The WHF can also handle dual-purpose canisters (DPCs) that will be received in transportation casks or aging overpacks. The DPCs will then be transferred to a shielded transfer cask where the DPC is opened and the CSNF assemblies are transferred under water in the pool into the TAD canisters.

The RF will receive transportation casks containing TAD canisters or DPCs and transfer the canisters into aging overpacks. The aging overpacks are moved by a site transporter to a CRCF or AF. The horizontal DPCs can be moved by a transfer trailer for placement at the AF in horizontal aging modules.

The AF will be designed to provide support to the aging overpacks containing HLW in the TAD canisters and DPCs. The main waste handling functions of the AF will provide aging capability for the repository waste handling operations and will protect the TAD canisters and DPCs from external hazards during aging.

SAR Section 1.2.8.1 described facilities considered part of the balance of plant. SAR Table 1.2.8-1 listed the balance-of-plant facilities. SAR Sections 1.2.8.1.1.1 to 1.2.8.1.1.12 provided the descriptions and functions of the balance-of-plant facilities that the applicant classified as not-important to safety (non-ITS). The function of each non-ITS facility is briefly described next.

The EDGF will house two independent 13.8-kV ITS diesel generators and the supporting mechanical systems for those two diesel generators. The EDGF structure itself is non-ITS. The primary function of the EDGF will be to ensure that ITS power is available to the ITS loads in CRCFs and the WHF in the event of a loss of outside power.

An important function of the Administration Facility is to house the computer operations center and the emergency operations center. The computer operations center will consist of space for local network equipment and functions, while the emergency operations center will provide space for emergency management services and functions.

The CCCF will provide functional space, structures, and internal systems to support the central control center, which is the technical support center for conducting emergency management activities. This will provide centralized control and communication for plant-wide monitoring and control. The CCCF will have the capability to transfer the functions of the technical support center to the near-site emergency operations facility located in the Administration Facility.

The LLWF will store dry and liquid low-level radioactive waste (LLW). The LLWF will receive LLW from the initial handling facility, canister receipt and closure facility, wet handling facility, and the receipt facility. Unloaded dual-purpose canisters will be delivered in a shielded transfer cask or other acceptable container and will be stored in the LLWF for eventual disposal.

The Warehouse and Nonnuclear Receipt Facility will store TAD canisters; empty, new waste packages; lids; pallets; spread rings; and shield plugs. No radioactive material will be received or stored in this facility. The Aging Overpack Staging Facility will serve as an outdoor area for storing empty aging overpacks and unloaded and noncontaminated aging overpacks.

Surface runoff flooding from a probable maximum precipitation event could inundate the surface facilities (SAR Section 1.6.3.4.5). The flood control features proposed in SAR Figure 1.2.2.-7 consisted of ditches and dikes (levees) to collect and divert the surface runoff flow (potential flood) and prevent flooding of surface facilities. The applicant classified the flood control features as ITS because they are intended to prevent flooding of ITS surface facilities.

The remainder of the balance-of-plant facilities was described in SAR Sections 1.2.8.1.1.7, 1.2.8.1.1.8, 1.2.8.1.1.9, 1.2.8.1.1.10, and 1.2.8.1.1.12.

**NRC Staff's Evaluation**:

The NRC staff reviewed the applicant's description of the location of the surface facilities using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff focused its review on the description of the geologic repository at Yucca Mountain and the location and arrangement of surface facilities in the geologic repository operations area. The NRC staff compared the information on the surface facility layout and their functions contained in the SAR sections identified in SER Section 2.1.1.2.3.1, with the proposed operations of handling HLW in these facilities for ultimate disposal in subsurface emplacement drifts. The NRC staff finds that the applicant's descriptions of the functions of the surface facilities at the GROA are consistent with the proposed overall HLW handling and disposal operations at the site. The NRC staff finds that the descriptive information in the SAR about the facilities is acceptable because this information adequately described (i) the nature of operations and location and distance from the boundary, (ii) arrangement at the site, and (iii) functions of the surface facilities at the GROA site. Therefore, the descriptive information is sufficient to permit an evaluation of the applicant's PCSA and surface facilities systems design.

**NRC Staff's Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.1, the NRC staff finds, with reasonable assurance, that the applicant's description of the location of the surface facilities and their functions meets the requirements of 10 CFR 63.21(c)(2) and 10 CFR 63.112(a) because the applicant provided an adequate description of the location, functions, operations, and layout of structures of the surface facilities at the geologic repository operations area sufficient for the NRC staff to evaluate the applicant's PCSA and design.

### 2.1.1.2.3.2 Description of, and Design Details for, Structures, Systems, and Components; Equipment; and Utility Systems of Surface Facilities

This section presents the NRC staff's evaluation of the applicant's information in SAR Sections 1.2.1 through 1.2.7 on descriptions and design information for the surface facility structures, systems, and components (SSCs), equipment, and utility systems. The NRC staff evaluated surface facilities in terms of their structural features; mechanical equipment and its layout and operations; electrical power systems; heating, ventilation, and air conditioning systems; shielding and criticality control systems; fire suppression systems; piping and instrumentation diagrams; and decontamination, emergency, and radiological safety systems.

#### 2.1.1.2.3.2.1 Surface Structures

The applicant described the structural design of the building facilities in SAR Sections 1.2.3 through 1.2.7. The applicant used this information in the PCSA and in the design and performance evaluation of the building facilities. On the basis of the PCSA, the applicant designated the CRCF, IHF, WHF, RF, AF, and flood control features as important to safety (ITS). The design codes and standards used for steel and reinforced concrete structures are listed in SAR Section 1.2.2.1.8. SAR Section 1.2.2.1.7 listed the materials proposed for the construction of the ITS surface structures. SAR Section 1.2.2.1.6 described the loads and design methodologies used in ITS facilities design. SAR Section 1.2.2.1.9 described the load combinations used for ITS facilities design. SAR Table 1.2.2-1 listed the natural phenomena loading parameters used in the ITS facilities design. The NRC staff's review of the applicant's design codes and standards, materials of construction, design loads and load combinations, and design methodologies for ITS surface facilities is provided in SER Section 2.1.1.7.3.1. The GROA will also contain a number of not important to safety (non-ITS) facilities. Two of these non-ITS facilities (LLWF and EDGF) will be covered in this SER section.

***ITS Structures***

SAR Section 1.2.4 provided the general description of the canister receipt and closure facility (CRCF), and SAR Section 1.2.2.1 described the structural design of the CRCF. The applicant stated that the GROA would have three identical CRCFs constructed in phases. The CRCF building dimensions will be approximately 119-m [392-ft]-wide, 128-m [420-ft]-long, and 30-m [100-ft]-high, with the walls and floors primarily constructed of reinforced concrete. SAR Figure 1.2.2-1 showed typical reinforced concrete sections, including details of the dimensions of structural elements (e.g., foundation mat and shear walls). The general arrangement drawings for the CRCF, illustrated in SAR Figures 1.2.4-1 to 1.2.4-4, showed the ITS and non-ITS areas. SAR Figures 1.2.4-6 to 1.2.4-11 showed the cross sections of the CRCF and the location of major equipment within the facility. SAR Section 1.2.4.1.1 stated that ancillary areas of the CRCF not categorized as ITS fall outside the footprint of the main CRCF reinforced concrete structure. These non-ITS areas will be constructed using lightweight concrete and steel framing. SAR Section 1.2.4.1.1 also stated that the mat foundations associated with ancillary areas (non-ITS structures) will be reinforced concrete mats designed to adequately support the superstructures.

The initial handling facility (IHF) will be composed of two seismically independent structures isolated by a seismic joint (SAR Section 1.2.3). The main structure will consist of internal and external steel-braced frames with a concrete internal structure to provide structural support and shielding. IHF floor plans and cross-sectional views were shown in SAR Figures 1.2.3-1 to 1.2.3-14. As described in SAR Section 1.2.3.1.1, the main structure of the IHF cask handling process area will be a braced-frame steel structure approximately 52 m [170 ft] wide, 57-m [187-ft]-long, and 32-m [105-ft]-high. The interior reinforced concrete structure will consist of 1.2-m [4-ft]-thick walls and roof that comprise the waste package positioning room, the waste package loading room, the internal shielded rooms, and the cask unloading room. The IHF waste package load-out room will be a reinforced concrete structure approximately 12-m [41-ft]-wide, 43-m [140-ft]-long (excluding external north–south concrete buttresses), and 18-m [60-ft]-high. The common foundation for the IHF main structure and waste package load-out room will be a 1.8-m [6-ft]-thick mat. The applicant stated that ancillary areas are categorized as non-ITS, including the general support area, LLW sump room, and external fire water valve rooms. The non-ITS areas of the facility will be composed of slabs on grade using lightweight concrete construction and/or insulated metal panels on steel framing. These areas will be supported by reinforced concrete mat foundations independent of the ITS structures.

The wet handling facility (WHF) will be a reinforced concrete structure that consists of shear walls, roof slab diaphragms, mat foundations, and a pool (SAR Section 1.2.5). The overall footprint of the WHF will be approximately 117 × 120 m [385 × 395 ft], and the ITS portion of the structure is approximately 117 × 91 m [385 × 300 ft]. The maximum height of the building will be 30 m [100 ft] above grade, with the majority of the building approximately 24 m [80 ft] above grade. The below-grade pool substructure will be approximately 35 × 35 m [116 × 116 ft], including the rooms surrounding the pool that provide internal buttresses for the pool. The internal dimensions of the pool will be 23-m [74-ft]-wide and 19-m [61-ft]-long. The bottom of the pool will be 16 m [52 ft] below the at-grade concrete mat. The mat foundation at grade will be 1.8-m [6-ft]-thick, whereas the pool foundation mat is 2.4-m [8-ft]-thick. The foundation mats for the two structural steel vestibules will be 1.2-m [4-ft]-thick. The main WHF superstructure will be constructed of 1.2-m [4-ft]-thick exterior and interior concrete walls, and nonstructural partition walls are 0.3 m [1 ft] thick. The internal shielded rooms will be constructed of 1.2-m [4-ft]-thick concrete walls and roof slabs. Other elevated floor diaphragm slabs will be generally 0.6-m [2-ft]-thick. The below-grade portion of the pool will consist of 1.8-m [8-ft]-thick exterior earth retaining walls. Interior rooms will be separated from the pool by 1.2-m [4-ft]-thick concrete walls, and nonstructural partition walls within the pool are 0.6-m [2-ft]-thick. Ancillary areas of the facility that were categorized as non-ITS will be supported by structurally independent foundations.

SAR Section 1.2.6 discussed the receipt facility (RF), stating it will be constructed of reinforced concrete interior and exterior shear walls, concrete floor and roof slab diaphragms, and a concrete mat foundation. The RF building footprint dimensions will be approximately 96-m [315-ft]-wide by 97-m [318-ft]-long. The part of the structure that was considered ITS will have dimensions of 61-m [200-ft]-wide by 73-m [240-ft]-long. The maximum height of the building will be 30 m [100 ft] above grade with other roofs located at 22 m [72 ft] and 20 m [64 ft] above grade. The thickness of concrete walls and roof slabs will be 1.2 m [4 ft]. The RF foundation mat will be 2-m [7-ft]-thick, and elevated floor diaphragm slabs will be generally 0.5-m [1.5-ft]-thick. SAR Section 1.2.6.1.1 stated that ancillary areas of the RF that were non-ITS will be located outside the footprint of the main RF reinforced concrete structure. These non-ITS structures will be constructed on separate slabs on grade using lightweight concrete and steel framing, which will have insulated metal panels for the walls. SAR Section 1.2.6.1.1 also stated that these non-ITS ancillary areas/rooms will not compromise the integrity of the main ITS structure in a design basis seismic event.

The aging facility (AF) presented in SAR Section 1.2.7 is an ITS facility designed to provide support to the aging overpacks. The main waste handling functions of the AF will be to provide aging capability for up to $2.1 \times 10^7$ kg [21,000 metric tons] of heavy metal (MTHM) for the repository in 2,500 aging spaces and to protect TAD canisters and DPCs from external hazards. The AF will consist of the following ITS components: (i) aging pad, (ii) aging overpack, and (iii) overpack transfer systems. The location of the two aging pads the applicant proposed for aging operations was presented in SAR Figure 1.2.7-2. The aging pad 17P (SAR Figure 1.2.7-3) will consist of 7 pads for about 1,250 vertical aging overpacks. The aging pad 17R (SAR Figure 1.2.7-4) will have 8 pads with space for about 1,150 vertical aging overpacks and 2 pads with space for 100 horizontal DPCs in horizontal aging modules, with 50 modules on each pad. Vertical aging overpacks will be arrayed in groups of 16 overpacks, spaced on 4 by 4 grids with a square center-to-center pitch of approximately 5 m [18 ft]. The spacing between overpacks will be 1.8 m [6 ft] to enable access for the site transporter and to permit air circulation for cooling. Horizontal aging modules are arranged side by side.

The aging pads will consist of a 0.9-m [3-ft]-thick reinforced concrete mat foundation supported on existing soil and compacted fill where needed (SAR Section 1.2.7.1.3.1). The applicant stated (SAR Section 1.2.7.1.2) that the location of the two aging pad areas was selected to avoid faults and flooding. According to the applicant, the concrete aging pads will be designed to support the aging overpacks during credible design events and to withstand loads and load combinations imposed by natural phenomena. The applicant stated that the proposed flood drainage channels will carry away water from a probable maximum flood (PMF) surrounding the aging pads. The distance from the aging pads to upslope hillsides and the location of the drainage channel will preclude soil from the slope sliding onto the concrete aging pads and contacting the aging overpacks (see SER Section 2.1.1.7 for the NRC staff's evaluation of stability of slopes near the AF). Aging pads will provide for water runoff and will be designed to support concrete heating (heat generated by HLW) and transport equipment accessibility. SAR Section 1.2.2 further detailed the structural design of the aging pads. SAR Section 1.2.2.1 described the flood control features of the repository site areas. The aging pads will be surrounded by a security fence to control access, as shown in SAR Figure 1.2.7-2.

In SAR Figure 1.2.2-7, the applicant described ITS flood control features credited with preventing inundation of the surface facilities from a PMF at the site. The applicant stated that the ITS flood control system will include the following features to control the PMF runoff: (i) a dike and channel system west, north, and east of the AF; (ii) a dike and channel system located between the North Portal pad and AF areas; (iii) a dike and channel system east and south of the North Portal pad area; (iv) two diversion ditches in Exile Hill west of the North Portal pad area; and (v) three storm water detention ponds to the southeast of the North Portal pad.

**NRC Staff's Evaluation**

The NRC staff reviewed the information in SAR Sections 1.2.1 through 1.2.7 related to the descriptions and design information for the structural design of the ITS facilities using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of general layout; structural design information; and potential interactions among support systems and SSCs for the CRCF, IHF, WHF, RF, and AF. The NRC staff also reviewed the descriptions of the location and functional arrangement of the SSCs within each facility mentioned above and the ability of these facilities to withstand the effects of natural phenomena. The NRC staff reviewed the list of codes and standards, drawings, materials, and loads associated with each ITS facility in SER Section 2.1.1.7.3.1.1. The NRC staff concludes that these design codes and standards are acceptable because they are in conformance with standard engineering practices for similar nuclear material handling facilities. With respect to the descriptions of the SSCs, equipment, and operation activities, the NRC staff finds that the applicant's descriptions and design information are sufficient to permit an evaluation of the PCSA and design because the applicant provided (i) proposed materials for the construction of the ITS surface structures that are in conformance with standard engineering practices; (ii) an adequate description of design parameters and design methodologies; (iii) descriptions of the location and functional arrangement of the SSCs within each facility and potential interactions among support systems and SSCs; (iv) information on the reinforced concrete structural components and steel structural components; (v) general arrangement drawings of the surface facilities (DOE, 2009dm); and (vi) discussion of design information regarding the capability of the facilities to withstand the effects of natural phenomena.

For flood control features, the NRC staff finds that the information in the SAR and the applicant's response to the NRC staff's requests for additional information (RAIs) (DOE, 2009eh,fh) provided an adequate description of the layout, function, and design bases and design criteria of the flood control features and therefore can be used in evaluating the PCSA and design (see SER Section 2.1.1.7.3.1.3 for details).

***Non-ITS Structures***

SAR Section 1.2.8 provided the applicant's description of the non-ITS facilities. The location of the non-ITS facilities relative to other surface facilities was shown in SAR Figure 1.2.1-2. SAR Table 1.2.8-1 presented a list of all the non-ITS facilities and a general description of structural systems used in their design and construction. The NRC staff focused its review on the low-level radioactive waste facility (LLWF) and the emergency diesel generator facility (EDGF) because the LLWF is the surface building that will handle and store radioactive material that could pose a radiological risk to workers prior to the shipment of the LLW from the GROA to an LLW disposal facility, and the EDGF will house ITS diesel generators and their components that will serve important to safety functions.

The applicant stated that it will use the following codes and standards for the design of the non-ITS facilities: American Concrete Institute ACI 318–02/318R–02 (American Concrete Institute, 2002aa); American Institute of Steel Construction Manual of Steel Construction, Allowable Stress Design (American Institute of Steel Construction, 1997aa); American Society of Civil Engineers ASCE 7–98 (American Society of Civil Engineers, 2000ab); International Code Council International Building Code 2000 (International Code Council, 2003aa); and American Welding Society D1.1/D1.1M–04 (American Welding Society, 2006aa). Additional information related to the structural design of the non-ITS facilities was in BSC (2007av, Section 4.2.11.5). The design live loads for floor and roof, and snow loads and load combinations were also discussed in BSC (2007av). The general structural information for each specific non-ITS facility is summarized next.

SAR Section 1.2.8.1.1.1 provided the applicant's structural description of the EDGF. The EDGF will have an overall footprint of approximately 53-m [174-ft]-wide by 30-m [98-ft]-long. The applicant provided the general arrangement plans for the floor and roof in SAR Figures 1.2.8-1 through 1.2.8-3. Cross-sectional views of the facility were shown in SAR Figures 1.2.8-4 through 1.2.8-7. The applicant described the foundation of the EDGF structure as a 1.2-m [4-ft]-thick reinforced concrete mat supporting the superstructure and the ITS diesel generators. The superstructure of the EDGF, as described by the applicant, will consist of 0.9-m [3-ft]-thick concrete exterior walls and 0.6-m [2-ft]-thick interior concrete shear walls. The roof diaphragm slab will be a 0.9-m [3-ft]-thick reinforced concrete slab. There will be two non-ITS, 1,814-kg [2-T] monorail hoists in each of the two diesel generator rooms.

SAR Section 1.2.8.1.1.5 provided the structural description of the LLWF. This facility will be used for the processing, packaging, and disposal of the LLW generated from GROA operations. The general arrangement floor plans were provided in SAR Figures 1.2.8-9 to 1.2.8-11. Cross-sectional views of the facility were shown in SAR Figures 1.2.8-12 through 1.2.8-14. The LLWF will be a multistory building designed as a steel structure with a concrete floor, concrete mat foundation, concrete shield walls, steel roof truss system, and interior and exterior structural steel bracing. The facility will have four bays composed of half-height shielded walls for storage of LLW. A 45,359-kg [50-T] bridge crane will be used to move large waste containers through the facility.

**NRC Staff's Evaluation**

The NRC staff reviewed SAR Section 1.2.8 to evaluate the non-ITS facilities' description, design, and construction information of the EDGF and LLWF buildings using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of general layout, structural design information, and potential interactions among support systems and SSCs for the EDGF and LLWF. The NRC staff finds that the applicant's facility descriptions and design information are sufficient to permit an evaluation of the PCSA and design of the facilities because the applicant provided (i) descriptions of the design, location, and functional arrangement of the SSCs within each facility; (ii) discussion of potential interactions among support systems and SSCs; (iii) description of structural design information; and (iv) the codes and standards the applicant proposed to use to design the facilities, which are consistent with the standard engineering practices for structures of similar functions and can be used in the design of these facilities.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.2.1, the NRC staff finds, with reasonable assurance, that the applicant's descriptions and design information for surface facility structures meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided adequate description and design information for the surface facility structures sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed in the NRC staff evaluations in SER Section 2.1.1.2.3.2.1, the applicant provided (i) descriptions of the design, location, and functional arrangement of the SSCs within each facility; (ii) the design codes and standards and construction materials that are in conformance with standard engineering practices for structures of similar functions and are, therefore, acceptable for their intended use; (iii) information on potential interaction among support systems and SSCs; and (iv) description of design parameters and design methodologies that are sufficient to evaluate the structural capabilities of the facilities to withstand the effects of natural phenomena.

2.1.1.2.3.2.2 Layout of Mechanical Handling System

The applicant described the layout, functions, ITS components, and design information of mechanical handling systems in the surface facilities in various sections of the SAR. Major mechanical systems will be used for waste handling operations at the initial handling facility, canister receipt and closure facility, wet handling facility, and the receipt facility, where similar systems will be used in multiple facilities. Because of the system replication throughout the facilities, the NRC staff reviewed the information provided on equipment at the subsystem level while noting any significant layout and interface distinctions between facilities. The mechanical handling system consists of cask handling, canister transfer, waste package closure, waste package load-out, SNF assembly transfer, dual-purpose canister cutting, and transportation aging and disposal canister closure subsystems. This section presents the NRC staff review of whether the applicant's description of the functions, layout, and design of the aforementioned subsystems of mechanical handling systems is adequate for use in evaluating the PCSA and design. The NRC staff's evaluation of waste handling operations using these mechanical systems is in SER Section 2.1.1.2.3.6.1. The NRC staff's evaluation of ITS SSCs related to the mechanical handling systems are in SER Section 2.1.1.2.3.2.5. The codes and standards proposed by the applicant for the major ITS equipment in the mechanical handling systems, discussed in this section, are reviewed in SER Section 2.1.1.2.3.2.5. For non-ITS systems, the codes and standards, if proposed by the applicant in the SAR, are reviewed for their applicability in this section.

***Cask Handling Subsystem***

The cask handling subsystem will be used in the canister receipt and closure facility (CRCF) (SAR Section 1.2.4.2.1), initial handling facility (IHF) (SAR Section 1.2.3.2.1), wet handling facility (WHF) (SAR Section 1.2.5.2.1), and receipt facility (RF) (SAR Section 1.2.6.2.1).

The cask handling subsystem in the CRCF will consist of the cask preparation subsystem and the waste package preparation subsystem. The former will prepare (i) loaded transportation casks, loaded aging overpacks, and empty aging overpacks for canister transfer operations; (ii) unloaded transportation casks for leaving the CRCF; and (iii) unloaded aging overpacks for reuse. The latter will prepare empty waste packages for canister transfer operations (SAR Section 1.2.4.2.1.1.1). The loaded transportation casks to be handled in the CRCF will contain SNF or HLW in either TAD canisters or DPCs. SAR Figures 1.2.4-2 and 1.2.4-3 provided details of the CRCF general arrangement floor plans, illustrating locations and functional arrangements of the cask preparation and waste package preparation subsystems.

The cask handling subsystem in the IHF will consist of the cask preparation subsystem and waste package preparation subsystem, and will prepare transportation casks containing either naval SNF canisters or HLW canisters and empty waste packages for canister transfer operations (SAR Section 1.2.3.2.1.1.1). SAR Figures 1.2.3-2 and 1.2.3-3 provided details of the IHF general arrangement floor plans, illustrating locations and functional arrangements of the cask preparation and waste package preparation subsystems.

The cask handling subsystem in the WHF will provide receipt and preparation operations for loaded DPCs, loaded shielded transfer casks, empty aging overpacks, and loaded transportation casks for waste transfer, and will perform restoration activities for unloaded transportation casks, aging overpacks, and shielded transfer casks (SAR Section 1.2.5.2.1.1). The loaded transportation casks to be handled in the WHF will contain commercial SNF in vertically based DPCs or uncanistered SNF. SAR Figures 1.2.5-2 and 1.2.5-3 provided details of the WHF general arrangement floor plans, illustrating locations and functional arrangements of the cask handling subsystem.

The cask handling subsystem in the RF will prepare loaded transportation casks and empty aging overpacks for canister transfer operations and prepare unloaded transportation casks to leave the RF (SAR Section 1.2.6.2.1.1.1). The transportation casks to be handled in the RF will contain commercial SNF in TAD canisters or DPCs. SAR Figure 1.2.6-2 provided details of the

RF ground floor general arrangement plan, illustrating locations and functional arrangements of the cask handling subsystem.

The applicant provided information on the ITS SSCs for the cask preparation subsystem that will be similar among the CRCF, IHF, WHF, and RF (e.g., equipment shield and confinement doors, cask transfer trolley, cask handling yoke, cask lid-lifting grapples, cask handling crane, DPC lid adapter, horizontal lifting beam, and rail cask lid adapter that are similar to all surface facilities). The applicant provided information on the ITS SSCs that are unique to a specific facility [e.g., (i) cask preparation crane, (ii) naval cask lift bail, (iii) naval cask lift plate, and (iv) cask preparation platform in the IHF] in the above referenced SAR sections and in SAR Table 1.9-1.

The applicant identified the procedural safety controls for the cask and waste package preparation operations to be performed at each surface facility using the proposed cask handling subsystem to prevent event sequences or mitigate their effects (SAR Sections 1.2.4.2.1.4, 1.2.3.2.1.4, 1.2.5.2.4.4, and 1.2.6.2.1.4). The applicant provided the nuclear safety design bases and design criteria for the ITS SSCs in the proposed cask handling subsystems in SAR Tables 1.2.4-4 (CRCF), 1.2.3-3 (IHF), 1.2.5-3 (WHF), and 1.2.6-3 (RF).

The applicant stated that the design of the ITS SSCs (e.g., cask transfer trolley, cask handling crane, cask handling yoke, cask preparation platform, DPC lid adapter, horizontal lifting beam, rail cask lid adapter, and cask lid-lifting grapples in the CRCF) in the cask preparation subsystem similar to those used in the facilities performing similar nuclear waste handling operations uses the load combinations in accordance with the codes and standards provided in SAR Section 1.2.2.2. The applicant further stated that the design load combinations include normal conditions, event sequences, and the effects of natural phenomena (SAR Sections 1.2.4.2.1.9, 1.2.3.2.1.9, 1.2.5.2.4.9, and 1.2.6.2.1.9). For other ITS SSCs in the cask handling subsystem, the applicant uses the design load combinations in accordance with Table Q1.5.7.1 of ANSI/AISC N690-1994 [e.g., (i) equipment shield doors and equipment confinement doors in the CRCF, IHF, and RF and (ii) naval cask lift plate and naval cask lift bail in the IHF].

**NRC Staff’s Evaluation**

The NRC staff evaluated the applicant’s descriptions and design information for the cask handling subsystems using the guidance in YMRP Section 2.1.1.2. The NRC staff compared the cask handling subsystem layout information with its functions and waste handling operations in the CRCF, IHF, WHF, and RF. The NRC staff finds that the applicant’s descriptions and design information for the cask handling subsystem for the CRCF, IHF, WHF, and RF are adequate because (i) the descriptions discussed the specific functions of the subsystems that will be performed in the surface facilities (CRCF, IHF, WHF, and RF); (ii) the descriptions included discussions of the ITS SSCs in the cask handling subsystems in the surface facilities (e.g., equipment shield and confinement doors, cask transfer trolleys, cask handling yokes, cask lid-lifting grapples, cask handling cranes, DPC lid adapters, horizontal lifting beams, and rail cask lid adapters that are similar in all surface facilities with cask handling subsystems); (iii) the described functions for the cask handling subsystems of each facility are consistent with the proposed cask handling operations, procedural safety controls, and process flow in the respective surface facilities; and (iv) the descriptions provided nuclear safety design bases and design criteria, including design information regarding the codes and standards applicable to the cask handling subsystems and the capability of the cask handling subsystems to withstand the effects of natural phenomena.

The NRC staff finds that the applicant's description of the locations and functional arrangements of the cask handling subsystems is adequate because it provided floor plans for the CRCF, IHF, WHF, and RF that (i) delineated the location and arrangement of these subsystems with respect to other subsystems within each facility and (ii) illustrated the interactions between the cask handling subsystem and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant's descriptions and design information for the cask handling subsystems for the surface facilities are sufficient to permit an evaluation of the PCSA and design.

***Canister Transfer Subsystem***

The canister transfer subsystem will be used in the CRCF (SAR Section 1.2.4.2.2), IHF (SAR Sections 1.2.3.2.2), WHF (SAR Section 1.2.5.2.5), and RF (SAR Section 1.2.6.2.2).

The canister transfer subsystem in the CRCF will transfer (i) TAD, HLW, and DOE SNF canisters from transportation casks to waste packages; (ii) TAD canisters and DPCs from transportation casks to aging overpacks; or (iii) TAD canisters from aging overpacks to waste packages (SAR Section 1.2.4.2.2.1.1). SAR Figures 1.2.4-2 and 1.2.4-3 provided details of the CRCF general arrangement floor plans, illustrating locations and functional arrangements of the canister transfer subsystem.

The canister transfer subsystem in the IHF will transfer naval SNF and HLW canisters from transportation casks to waste packages (SAR Section 1.2.3.2.2.1.1). In addition, the canister transfer subsystem in the CRCF and IHF will move waste packages to the waste package positioning room after loading. SAR Figures 1.2.3-2 and 1.2.3-3 provided details of the IHF general arrangement floor plans, illustrating locations and functional arrangements of the canister transfer subsystem.

The canister transfer subsystem in the WHF will transfer (i) loaded TAD canisters from shielded transfer casks to aging overpacks; (ii) loaded dual-purpose canisters (DPCs) from transportation casks to shielded transfer casks; and (iii) loaded DPCs from aging overpacks to shielded transfer casks (SAR Section 1.2.5.2.5.1.1). SAR Figures 1.2.5-2 and 1.2.5-3 provided details of the WHF general arrangement floor plans, illustrating locations and functional arrangements of the canister transfer subsystem.

The canister transfer subsystem in the RF will transfer loaded TAD canisters and DPCs from transportation casks to aging overpacks (SAR Section 1.2.6.2.2.1.1). SAR Figures 1.2.6-2 and 1.2.6-3 provided details of the RF general arrangement floor plans, illustrating locations and functional arrangements of the canister transfer subsystems.

The applicant provided information on the ITS SSCs for the canister transfer subsystem that are similar among the surface facilities (i.e., CRCF, IHF, WHF, and RF) [e.g., canister transfer machine (CTM), CTM canister grapple, cask port slide gates that are similar to all surface facilities]. The applicant also provided information on the ITS SSCs that are unique to the canister transfer subsystem in a specific facility {e.g., slide gates for TAD and DOE canisters, Hanford multicanister overpack canister grapple, DOE SNF canister grapple {46 cm and 61 cm [18- and 24-in]}, staging racks for TAD and DOE canisters in CRCF} in the above referenced SAR sections and in SAR Table 1.9-1.

The applicant identified the procedural safety controls for the canister transfer operations to be performed at each surface facility using the proposed canister transfer subsystems to prevent event sequences or mitigate their effects (SAR Sections 1.2.4.2.2.4, 1.2.3.2.2.4, 1.2.5.2.5.4, and 1.2.6.2.2.4). The applicant provided the nuclear safety design bases and design criteria for the ITS SSCs in the proposed canister transfer subsystems in SAR Tables 1.2.4-4 (CRCF), 1.2.3-3 (IHF), 1.2.5-3 (WHF), and 1.2.6-3 (RF).

The applicant stated that the design of the ITS SSCs (e.g., canister transfer machine, grapple, and DPC lid adapter in the CRCF) in the canister transfer subsystem similar to those used in the facilities performing similar canister transfer operations uses the load combinations in accordance with the codes and standards provided in SAR Section 1.2.2.2. The applicant further stated that the design load combinations include normal conditions, event sequences, and the effects of natural phenomena (SAR Sections 1.2.4.2.2.9, 1.2.3.2.2.9, 1.2.5.2.5.9, and 1.2.6.2.2.9). For other ITS SSCs in the canister transfer subsystem, the applicant uses the design load combinations in accordance with Table Q1.5.7.1 of ANSI/AISC N690–1994 (e.g., cask and waste package ports, DOE SNF and TAD canister slide gates, and staging racks in the CRCF).

**NRC Staff's Evaluation**

The NRC staff reviewed the descriptions and design information for the canister transfer subsystem using the guidance in YMRP Section 2.1.1.2, focusing on the subsystem's relationships and interdependencies with other subsystems, equipment layout, canister transfer operations, and process flow. The NRC staff also reviewed the descriptions of the ability of the canister transfer subsystem to withstand the effects of natural phenomena. The NRC staff finds that the applicant's descriptions and design information for the canister transfer subsystem for the CRCF, IHF, WHF, and RF are adequate because (i) the descriptions discussed the specific functions of the subsystem that will be performed in the surface facilities (CRCF, IHF, WHF, and RF); (ii) the descriptions included discussions of the ITS SSCs in the canister handling subsystem in the surface facilities (e.g., HLW grapples, CTM, CTM canister grapple, cask port slide gates, and waste package port slide gates that are similar to all surface facilities); (iii) the described functions for the canister transfer subsystem are consistent with the proposed canister transfer operations, procedural safety controls, and process flow in the respective surface facilities; and (iv) the descriptions provided the nuclear safety design bases and design criteria, including design information regarding codes and standards applicable to the canister transfer subsystem and the capability of the canister transfer subsystem to withstand the effects of natural phenomena.

The NRC staff finds that the applicant's description of the locations and functional arrangements of the canister transfer subsystem is adequate because it provided floor plans for the CRCF, IHF, WHF, and RF that (i) delineated the location and arrangement of the subsystem with respect to other subsystems within each facility and (ii) illustrated the interactions between the canister transfer subsystem and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant's descriptions and design information for the canister transfer subsystem for the surface facilities are sufficient to permit an evaluation of the PCSA and design.

*Waste Package Closure Subsystem*

The waste package closure subsystem in the IHF and CRCF will consist of welding, stress mitigation, inerting, control and data management, and closure room material handling subsystems (SAR Sections 1.2.3.2.3 and 1.2.4.2.3.1.3). SAR Figures 1.2.3-3 and 1.2.4-3 provided general arrangement floor plans, illustrating locations and functional arrangements of the canister transfer subsystem for the IHF and CRCF, respectively.

The waste package closure subsystem will (i) add a seal weld between the spread ring and the inner lid, the spread ring and the inner vessel, and the spread ring ends; (ii) add a seal weld between the purge port cap and the inner lid; (iii) add a narrow groove weld between the outer lid and the outer corrosion barrier; (iv) perform nondestructive examination of the welds to verify the integrity of the welds and repair minor weld defects; (v) purge and fill the waste inner vessels with helium gas to inert the environment; (vi) perform a leak detection test of the inner-lid seals to ensure integrity of the helium environment in the inner vessel; and (vii) perform stress mitigation of the outer lid groove closure weld to induce compressive residual stress (SAR Section 1.2.4.2.3.1.1). In addition, the applicant will perform a prototype program to demonstrate the design and performance of the waste package closure subsystem (DOE, 2009dr).

This waste package closure subsystem was classified as non-ITS. The bridge of the remote handling subsystem was classified as ITS. According to the applicant, the remote handling subsystem will be used to position the closure lids and tools for inerting, leak detection, spread ring expansion, and stress mitigation (SAR Section 1.2.4.2.3.1.2). The waste package closure subsystem will be protected by preventing structural collapse of the bridge due to a spectrum of seismic events. This bridge will be designed in accordance with American Society of Mechanical Engineers (ASME, 2005aa) NOG–1–2004 for loads and accelerations associated with a DBGM-2 seismic event.

For the non-ITS waste package closure subsystem, the applicant proposed the following codes for design. The welds, weld repairs, and inspections will be performed in accordance with ASME Boiler and Pressure Vessel Code Section II, Part C; Section III, Division I, Subsection NC; Section IX; and Section V (American Society of Mechanical Engineers, 2001aa). The inerting of the waste package will be performed remotely in accordance with the applicable sections of NUREG–1536 (NRC, 1997ae). The structures, systems, and components (SSCs) of the waste package closure system will be designed using the methods and practices in American Welding Society ANSI/AWS A5.32/A5.32M–97 (American Welding Society, 1997aa), ASME B30.20–2003 (American Society of Mechanical Engineers, 2003aa), National Fire Protection Association (NFPA) 801 (National Fire Protection Association, 2003aa), and ASME NOG–1–2004 (Top Running Bridge, Multiple Girder) (American Society of Mechanical Engineers, 2005aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's descriptions and design information for the waste package closure subsystem in the CRCF and the IHF using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the waste package closure subsystem focusing on the subsystem's functions, relationships and interactions with other subsystems, equipment layout, and process flow. The NRC staff also reviewed the descriptions of the ability of the waste package closure subsystem to withstand the effects of natural phenomena. The NRC staff finds that the applicant's waste package closure subsystem descriptions and design information are adequate because (i) the descriptions discussed the specific functions of the waste package closure subsystem that will be performed in the IHF and CRCF, (ii) the descriptions included discussions of the non-ITS SSCs (e.g., welding equipment and procedures, inerting subsystem) and the ITS bridge of the remote handling SSC of the waste package closure subsystems in the IHF and CRCF, (iii) the described functions for the waste package closure subsystems are consistent with the proposed waste package closure operations and process flow in the IHF and CRCF, (iv) the descriptions included codes and standards applicable to the waste package closure subsystem, and (v) the descriptions included design information regarding the capability of the ITS bridge of the remote handling subsystem to withstand the effects of a seismic event.

The NRC staff finds that the applicant’s description of the locations and functional arrangements of the waste package closure subsystems is adequate because it provided floor plans for the IHF and CRCF that (i) delineated the location and arrangement of these subsystems with respect to other subsystems within each facility and (ii) illustrated the interactions between the waste package closure subsystem and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant’s descriptions and design information for the waste package closure subsystems for the surface facilities are sufficient to permit an evaluation of the PCSA and design.

***Waste Package Load-Out Subsystem***

SAR Sections 1.2.4.2.4 and 1.2.3.2.4 described the waste package load-out subsystem for the CRCF and the IHF. The waste package load-out subsystem will receive sealed waste packages after closure operations and prepare them for transfer to the Transport and Emplacement Vehicle (TEV). The applicant provided general arrangement floor plans, illustrating locations and functional arrangements of the waste package load-out subsystem for the CRCF in SAR Figures 1.2.4-2 and 1.2.4-3 and the IHF in SAR Figures 1.2.3-2 and 1.2.3-3.

The ITS SSCs include the waste package load-out room equipment shield doors, the waste package positioning room equipment shield doors, the waste package load-out room personnel shield doors, the waste package transfer trolley (WPTT), the waste package handling crane, and the waste package shield ring.

The applicant identified one procedural safety control for the waste package load-out operations to be performed at the CRCF and IHF using the proposed waste package load-out subsystems to limit the probability of personnel receiving direct exposure to radiation during movement of a loaded waste package into the TEV (SAR Sections 1.2.4.2.4.4 and 1.2.3.2.4.4). The applicant provided the nuclear safety design bases and design criteria for the ITS SSCs in the proposed waste package load-out subsystems in SAR Tables 1.2.4-4 (CRCF) and 1.2.3-3 (IHF).

The applicant stated that the design of the ITS SSCs (including WPTT, waste package shield ring, and waste package handling crane) in the waste package load-out subsystem uses the load combinations, in accordance with the codes and standards provided in SAR Section 1.2.2.2. The applicant further stated that the design load combinations include normal conditions, event sequences, and the effects of natural phenomena (SAR Sections 1.2.4.2.4.9 and 1.2.3.2.4.9). The applicant stated that it uses the design load combinations in accordance with Table Q1.5.7.1 of ANSI/AISC N690-1994 for design of ITS equipment and personnel shield doors.

### NRC Staff's Evaluation

The NRC staff evaluated the applicant's descriptions and design information for the waste package load-out subsystems using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the waste package load-out subsystem focusing on the subsystem's relationships and interdependencies with other subsystems, equipment layout, functions, and process flow. The NRC staff finds that the applicant's descriptions and design information for the waste package load-out subsystems for the CRCF and IHF are adequate because (i) the descriptions discussed the specific functions of the subsystems that will be performed in the CRCF and IHF; (ii) the descriptions included discussions of the ITS SSCs in the waste package load-out subsystems in the CRCF and IHF (e.g., equipment and personnel shield doors, waste package transfer trolleys, handling cranes); (iii) the described functions for the waste package load-out subsystems are consistent with the proposed canister transfer operations, procedural safety controls, and process flow in the CRCF and IHF; and (iv) the descriptions provided the nuclear safety design bases and design criteria, including design information regarding the codes and standards applicable to the waste package load-out subsystems and the capability of the waste package load-out subsystems to withstand the effects of natural phenomena.

The NRC staff finds that the applicant's description of the locations and functional arrangements of the waste package load-out is adequate because it provided floor plans for the CRCF and IHF that (i) delineated the location and arrangement of these subsystems with respect to other subsystems within each facility and (ii) illustrated the interactions between the waste package load-out subsystems and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant's descriptions and design information for the waste package load-out subsystems for the CRCF and IHF are sufficient to permit an evaluation of the PCSA and design.

#### *SNF Assembly Transfer Subsystem*

The spent nuclear fuel (SNF) assembly transfer subsystem will be in the Cask Preparation Area of the Wet Handling Facility (WHF) and its functions and components were described in SAR Section 1.2.5.2.2. This subsystem will receive SNF assemblies from a dual-purpose canister (DPC) or transportation cask and place the SNF assemblies using a spent fuel transfer machine (SFTM) in SNF staging racks or transfer the SNF assemblies into a transportation, aging, and disposal (TAD) canister. SNF assembly transfer will occur in the pool. Components of the SNF assembly transfer subsystem will be located in and above the pool.

The auxiliary pool crane and SFTM will be ITS SSCs located above the pool. The lifting grapple for boiling water reactor (BWR) SNF, pool lid-lifting grapple, long-reach grapple adapters, lifting grapple for pressurized water reactor (PWR) SNF, SNF staging rack, truck cask lid-lifting grapples, truck cask handling frame, and pool cask handling yoke will be ITS SSCs located in the pool. The functions and components of these ITS SSCs were described in SAR Section 1.2.5.2.2, with the exception of the truck cask lid-lifting grapple and pool cask handling yoke, which were described in SAR Section 1.2.5.2.1. SAR Figures 1.2.5-2, 1.2.5-3, and 1.2.5-16 provided details of the WHF general arrangement floor and pool plans, illustrating location and functional arrangement of the SNF assembly transfer subsystem.

The applicant provided the nuclear safety design bases and design criteria for the ITS SSCs in the proposed SNF assembly transfer subsystem in SAR Table 1.2.5-3. The applicant stated that the design of the ITS SSCs (except staging racks, truck cask handling frame, and SNF transfer machine) in the SNF assembly transfer subsystem uses the load combinations in accordance with the codes and standards provided in SAR Section 1.2.2.2. The applicant further stated that the design load combinations include normal conditions, event sequences, and the effects of natural phenomena (SAR Section 1.2.5.2.2.9). The applicant uses the design load combinations in accordance with Table Q1.5.7.1 of ANSI/AISC N690–1994 and ASME NOG–1–2004 for design of ITS staging racks, truck cask handling frame, and SNF transfer machine.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's descriptions and design information for the SNF assembly transfer subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the SNF assembly transfer subsystem focusing on the subsystem's relationships and interdependencies with other subsystems, functions, equipment layout, SNF assembly transfer operations, and process flow. The NRC staff finds that the applicant's descriptions and design information for the SNF assembly transfer subsystem are adequate because (i) the descriptions discussed the specific functions of the subsystem that will be performed in the WHF; (ii) the descriptions included discussions on the ITS SSCs of the SNF transfer subsystem (e.g., auxiliary pool crane, lifting grapples for BWR and PWR spent fuel, spent fuel transfer machine, and staging racks that are located in the pool); (iii) the descriptions included design features to limit event sequences and mitigate accidents (e.g., zone controls to avoid collisions, interlocks to prevent the transfer machine from raising a grapple unless a fully engaged or disengaged signal is provided, a protective wall is adjacent to the staging rack to ensure large objects cannot collide with it, and the auxiliary pool crane is equipped with seismic restraint rail clamps); (iv) the described functions for the SNF assembly transfer subsystem are consistent with the proposed SNF assembly transfer operation and process flow in the WHF; and (v) the descriptions provided discussions on the nuclear safety design bases and design criteria, including design information regarding the codes and standards applicable to the SNF assembly transfer subsystem and the capability of the SNF assembly transfer subsystem to withstand the effects of natural phenomena.

The NRC staff finds that the applicant's description of the location and functional arrangement of the SNF assembly transfer subsystem is adequate because it provided floor and pool plans for the WHF that (i) delineated the location and arrangement of this subsystem with respect to other subsystems within each facility and (ii) illustrated the interactions between the SNF assembly transfer subsystem and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant's descriptions and design information for the SNF assembly transfer subsystem in the WHF are sufficient to permit an evaluation of the PCSA and design.

***Dual Purpose Canister Cutting Subsystem***

The Dual Purpose Canister (DPC) cutting subsystem will be located in the Cask Preparation Area of the WHF, as described in SAR Section 1.2.5.2.3. SAR Figure 1.2.5-2 provided the WHF general arrangement ground floor plan, illustrating location and functional arrangement of the DPC cutting subsystem. This subsystem will receive and open various types of DPCs to access the SNF assemblies. DPC cutting will be take place outside the pool at the DPC cutting station. The DPC cutting jib crane, the DPC cutting station, and the lid-lifting grapple were categorized as ITS SSCs of the DPC cutting system. The functions and components of these

ITS SSCs and the DPC operational process were described in SAR Sections 1.2.5.2.3.1.3 and 1.2.5.2.3.2.

The applicant identified one procedural safety control for the DPC cutting operation to be performed in the WHF using the proposed DPC cutting subsystem to limit the probability of personnel receiving direct exposure to radiation. The applicant provided the nuclear safety design bases and design criteria for the ITS SSCs in the proposed DPC cutting subsystem in SAR Table 1.2.5-3.

The applicant stated that the design of the ITS SSCs (including DPC cutting jib crane and lid-lifting grapple) in the DPC cutting subsystem uses the load combinations in accordance with the codes and standards provided in SAR Section 1.2.2.2. The applicant further stated that the design load combinations include normal conditions, event sequences, and the effects of natural phenomena (SAR Section 1.2.5.2.3.9). The applicant uses the design load combinations in accordance with Table Q1.5.7.1 of ANSI/AISC N690-1994 for design of the ITS DPC cutting station.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's descriptions and design information for the DPC cutting subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the DPC cutting subsystem focusing on the system's relationships and interdependencies with other subsystems, functions, equipment layout, DPC cutting subsystem operations, and process flow. The NRC staff finds that the applicant's descriptions and design information for the DPC cutting subsystem are adequate because (i) the descriptions discussed the specific functions of the DPC cutting subsystem that will be performed in the WHF; (ii) the descriptions included discussions on the ITS SSCs (i.e., cutting jib crane, cutting station, and lid-lifting grapple); (iii) the described functions for the DPC cutting subsystem are consistent with the proposed DPC cutting operation, procedural safety control, and process flow in the WHF; and (iv) the descriptions provided discussions on the nuclear safety design bases and design criteria, including design information regarding the codes and standards applicable to the DPC cutting subsystem and the capability of the DPC cutting subsystem to withstand the effects of natural phenomena.

The NRC staff finds that the applicant's description of the location and functional arrangement of the DPC cutting subsystem is adequate because it provided floor and pool plans for the WHF that (i) delineated the location and arrangement of this subsystem with respect to other subsystems within each facility and (ii) illustrated the interactions between the DPC cutting subsystem and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant's descriptions and design information for the DPC cutting subsystem in the WHF are sufficient to permit an evaluation of the PCSA and design.

***Transportation, Aging, and Disposal Canister Closure Subsystem***

The transportation, aging and disposal (TAD) canister closure subsystem will be in the cask preparation area of the WHF, as described in SAR Section 1.2.5.2.4. The TAD canister closure will be the process that seals the loaded TAD canister by welding the shield plug and fully draining and vacuum drying the TAD canister interior, followed by backfilling the TAD canister with helium and fully welding the TAD canister lid around its circumference onto the body of the

TAD canister (SAR Section 1.2.5.2.4.1). SAR Figures 1.2.5-2 and 1.2.5-3 provided the WHF general arrangement floor plans, illustrating location and functional arrangement of the TAD canister closure subsystem.

The applicant classified the TAD canister closure subsystem and the TAD canister welding machine as non-ITS. The TAD canister closure jib crane, the lid-lifting grapple, and the shielded TAD canister closure station will be ITS SSCs of the TAD canister closure subsystem. The functions and components of these ITS SSCs and the TAD closure operational process were described in SAR Section 1.2.5.2.4.

The applicant identified one procedural safety control for the TAD canister closure operation to be performed in the WHF using the proposed TAD canister closure subsystem to limit the probability of personnel receiving direct exposure to radiation. The applicant provided the nuclear safety design bases and design criteria for the ITS TAD canister closure jib crane, lid-lifting grapple, and TAD canister closure station in the proposed TAD canister closure subsystem in SAR Table 1.2.5-3.

The applicant stated that the design the ITS TAD canister closure jib crane and lid-lifting grapple in the TAD canister closure subsystem uses the load combinations in accordance with the codes and standards provided in SAR Section 1.2.2.2. The applicant further stated that the design load combinations include normal conditions, event sequences, and the effects of natural phenomena (SAR Section 1.2.5.2.4.9). The applicant stated that it uses the design load combinations in accordance with Table Q1.5.7.1 of ANSI/AISC N690-1994 for design of the ITS TAD canister closure station.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's descriptions and design information for the TAD canister closure subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the TAD canister closure subsystem, focusing on the subsystem's relationships and interdependencies with other subsystems, functions, equipment layout, TAD canister closure subsystem operations, and process flow. The NRC staff finds that the applicant's descriptions and design information for the TAD canister closure system are adequate because (i) the description discussed the specific functions of the TAD canister closure subsystem that will be performed in the WHF; (ii) the description included discussions on the ITS SSCs (i.e., TAD canister closure jib crane, lid-lifting grapple, and the TAD canister closure station); (iii) the described functions for the DPC cutting subsystem are consistent with the proposed TAD canister closure operation, procedural safety control, and process flow in the WHF; (iv) the description identified that qualification of the TAD canister final closure welds would be in accordance with the NRC staff's ISG-18 (NRC, 2008ae); and (v) the description provided discussions on the nuclear safety design bases and design criteria, including design information regarding the codes and standards applicable to the jib crane and lid-lifting grapple and the capability of the jib crane and lid-lifting grapple to withstand the effects of natural phenomena.

The NRC staff finds that the applicant's description of the location and functional arrangement of the TAD canister closure subsystem is adequate because it provided floor and pool plans for the WHF that (i) delineated the location and arrangement of this subsystem with respect to other subsystems within each facility and (ii) illustrated the interactions between the TAD canister closure subsystem and other subsystems.

On the basis of the above evaluation, the NRC staff finds that the applicant's descriptions and design information for the TAD canister closure subsystem in the WHF are sufficient to permit an evaluation of the PCSA and design.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.2.2, the NRC staff finds, with reasonable assurance, that the applicant's descriptions and design information for the surface facilities, the facility functions, and the equipment layout of the subsystems meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided adequate descriptions and design information for the SSCs, equipment, functions, and process activities of the geologic repository operations area sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed previously in the NRC staff evaluations in SER Section 2.1.1.2.3.2.2, the applicant provided information on the major activities and subsystems of the mechanical handling system (i.e., cask handling, canister transfer, waste package closure, waste package load-out, SNF assembly transfer, dual purpose canister cutting, and transportation aging and disposal canister closure subsystems) that included descriptions of the (i) function, location, and functional arrangement of each subsystem; (ii) key subsystems and components, including identification of those that are ITS; (iii) operational processes, including any procedural safety controls to prevent event sequences or mitigate their effects; (iv) relationships and interdependencies with other subsystems; (v) the codes and standards applicable to each subsystem; and (vi) design information regarding the capability of the SSCs to perform under normal conditions and event sequences and withstand the effects of natural phenomena.

2.1.1.2.3.2.3 Geologic Repository Operations Area Electric Power Systems

The applicant described and discussed the electrical power system for the geologic repository operations area (GROA) in multiple sections of the SAR. SAR Sections 1.4.1, 1.2.8, 1.9.1, and 1.13 provided information related to (i) ITS and normal (non-ITS) alternating current (AC) electrical power system; (ii) ITS and normal (non-ITS) direct current (DC) electrical power system; (iii) ITS and normal (non-ITS) alternating current uninterruptible power supplies (UPS); and (iv) ITS and normal (non-ITS) diesel generators and associated mechanical support equipment in the GROA facilities. Additional information relevant to the subsurface normal electrical power distribution system was provided in SAR Sections 1.3.2–1.3.3, and information relevant to subsurface electrical power distribution concept of operations and functional design was provided in BSC (2008ca). Applicable codes and industry standards the applicant cited and high-level, single-line electrical drawings for representative power subsystems were also included in these SAR sections.

Facilities in the proposed GROA utilize normal electric power that the electrical power system would provide (SAR Section 1.4.1). The ITS SSCs (except the ITS HVAC SSCs in the CRCF, WHF, and EDGF) are powered by the normal electrical power system and are designed to stop in a safe condition if power is interrupted (SAR Section 1.9.1.11; DOE 2009fc). The ITS HVAC SSCs in the CRCF, WHF, and EDGF and non-ITS HVAC SSCs in the RF are powered by the ITS electrical power system (SAR Table 1.4.1-1 and SAR Sections 1.2.4, 1.2.5, 1.2.6, and 1.2.8). The applicant did not identify any ITS SSCs in the subsurface facilities (SAR Section 1.9.1.11). The applicant provided separate subsurface normal electrical power system feeds that facilitate operations simultaneously with construction and emplacement activities (SAR Section 1.4.1).

The electrical power systems include a normal electrical power system and an ITS electrical power system (SAR Section 1.4.1). Each contains respective backup diesel generators, DC, UPS, switchgear, and distribution SSCs. Independent and redundant offsite commercial 138-kV power supplies are connected to the GROA electrical power system within the normal electrical power system onsite switchyard. Switchyard facilities convert incoming 138-kV power to 13.8-kV normal power for further onsite distribution to the normal and ITS electrical power system.

The Standby Diesel Generator Facility houses four normal electrical power system standby diesel generators with mechanical support systems and two 13.8-kV switchgears, which can supply backup normal power during a loss of offsite power (LOSP) (SAR Section 1.4.1). The emergency diesel generator facility (EDGF) houses two redundant and independent ITS diesel generators with ITS mechanical support systems and ITS 13.8-kV switchgears (SAR Section 1.2.8.1.1.1). Separate normal and ITS DC and uninterruptible power supply system SSCs are located within various facilities to maintain power to designated controls and loads. The applicant described physical and electrical separation and isolation between normal and ITS electrical power system SSCs, including descriptions of cable raceways and cabling. SAR Section 1.13 also described the equipment qualification program, including seismic and environmental qualification processes, for active electrical equipment used in mild and harsh environments and plans and procedures for initial startup activities and operations, maintenance, and periodic testing of the electrical power system.

**The Normal Electrical Power System**

The normal electrical power system, described in SAR Section 1.4.1.1, provides power to non-ITS and most ITS loads through load centers and motor control centers in respective facilities. Underground distribution cables connect the 13.8-kV main switchgear to most surface facilities and to subsurface entrances. Power will be provided at 480 V and 208/120 V for most process functions and building utility loads. Codes and standards for design methods and practices for the normal electrical power system were listed in SAR Section 1.4.1.1.3. For the normal AC power supply, including subsurface facility electrical power distribution, standby diesel generators, DC and uninterruptible power supplies, the applicant provided the following codes and standards: National Fire Protection Association NFPA 70 and NFPA 110 (NFPA, 2005aa,ac); Institute of Electrical and Electronic Engineers IEEE STD 141–1993, IEEE STD 519–1992, IEEE STD 446–1995, IEEE STD 946–2004, and ANSI/IEEE STD 944–1986 (Institute of Electrical and Electronic Engineers, 1993aa, 1992ab, 1996ab, 2005ab, 1986aa); and National Electrical Manufacturers Association ANSI C84.1–2006 (National Electrical Manufacturers Association, 2006ac).

***Surface Normal Electrical Power***

The switchyard, described in SAR Section 1.4.1.1.1, connects to redundant offsite commercial power sources via high-voltage overhead transmission lines. High-voltage sources are connected to five main step-down transformers through a breaker-and-a-half scheme. The five main transformers supply 13.8-kV power to four open buses and one transfer bus. High-voltage circuit breakers, disconnect switches, surge arrestors, and other switchyard protective and distribution SSCs were presented in SAR Figures 1.4.1-1 and 1.4.1-2. SAR Section 1.4.1.1.1.1 described the normal electrical power system Switchgear Facility within the switchyard, which contains four main 13.8-kV switchgears, supported by battery-powered, 125-V DC SSCs and local distribution, control, and communications equipment.

SAR Section 1.4.1.1.1.3 described the normal electrical power system standby diesel generators. Upon detection of loss-of-offsite power (LOSP), the feed breakers providing commercial power to the two 13.8-kV switchgears in the Standby Generator Facility will be opened, and the standby diesel generators will be automatically started and connected to each switchgear. These operations will provide backup power to nonshed loads such as fire alarm panels, alarm communications and display systems, and the Emergency Operations Center. The standby diesel generators are sized such that three of the four generators are sufficient to run the nonshed loads and three of the six subsurface ventilation fans.

Redundant, normal direct current (DC) electrical power subsystems (SAR Section 1.4.1.1.1.4) provide power for switchgear medium-voltage circuit breaker control, protective relaying, and other non-ITS loads requiring continuous DC power. SAR Figure 1.4.1-7 presented a single-line diagram for the normal DC power system, including a third “swing” battery charger that the standby diesel generators can power.

Surface normal electrical power system UPSs are located in major operations facilities (SAR Section 1.4.1.1.1.5) and are sized to provide a minimum of 15 minutes of continuous alternating current power for selected processes that need time to complete ongoing operations. SAR Figures 1.4.1-8 and 1.4.1-9 presented single-line electrical diagrams for normal electrical power systems 480/277-V and 208/120-V UPS, respectively.

Two separate normal electrical power system 13.8-kV feeds, each capable of meeting full power requirements, will be converted to 480 V at each major facility. An interlock and transfer control scheme to be designed to prevent simultaneous closure of two incoming breakers and a tie breaker providing 480-V power to the CRCF, as an example, were shown in SAR Figure 1.4.1-3 (Sheet 1 of 16).

***Subsurface Normal Electrical Power***

SAR Sections 1.4.1.1.3 and 1.3.2.4.1 provided principal design codes and standards applicable to the subsurface normal electrical power system and major subsurface electrical distribution SSCs, respectively. During normal operations, electric power provided to the subsurface facilities is derived from commercial offsite power sources. The normal electrical power system standby diesel generators provide backup power for selected loads in the subsurface facilities. Separate power feeds are provided to subsurface emplacement and construction activities (SAR Section 1.3.2.4.1) to protect the subsurface electrical power system for each activity from adverse effects due to demand loads and interruptions on the alternate side.

SAR Section 1.3.3 described distribution of normal electric power within the subsurface facility. According to the applicant, the two electrical power system power feeds (13.8 kV) are converted to 480/277-V and 208/120-V power within the alcoves located inside each subsurface access main. Normal 13.8-kV power will also be supplied via overhead distribution lines to subsurface construction switchgear at the south portal facilities and the north construction portal area. SAR Figure 1.4.1-5 showed a typical single-line diagram of power distribution for a subsurface alcove, and SAR Figure 1.3.3-20 showed a typical subsurface electrical alcove physical configuration. The locations of the subsurface facility electrical stations were shown in SAR Figure 1.4.1-6.

Normal 13.8-kV electrical power for the subsurface ventilation system is provided by the normal electrical power system switchgear located in the Standby Diesel Generator Facility. The power is distributed via overhead distribution lines to the subsurface ventilation fan facilities, which are located on the surface at the openings of the exhaust shafts. There the power is converted to 4.16-kV power for the primary exhaust fan power system and to lower voltages for operational controls and supporting SSCs. In the event of a LOSP, power to the exhaust fans is supplied from two backup sources. The standby diesel generators provide up to three exhaust fans with backup power, and all exhaust shaft facility surface pads will be equipped with connections for mobile diesel backup generators (SAR Sections 1.3.3.1.8, 1.4.1.1.1.2, and 1.4.1.1.1.3).

Various types of mechanical handling equipment, such as the transport and emplacement vehicle (TEV) and drip shield emplacement gantry (DSEG), operate on 480-V, three-phase power in the subsurface facility, as described in SAR Section 1.3.2.3. The TEV and DSEG are powered via an electrified third rail that follows the rail track system planned for these vehicles. There is no provision for backup power for the emplacement side normal electrical power system SSCs, which supply power for waste package transportation and emplacement operations. The normal electrical power system power distribution and connections energizing the electrified third rail are located in the accessible areas.

The electrified third rail must extend into inaccessible areas to provide power for the TEV, DSEG, and other remotely operated vehicles (ROVs) (SAR Section 1.3.3.5.1.1). The applicant stated that the electrified third rail design will be based on applicable codes and standards and accepted industry practices (SAR Section 1.3.3.4.1) and that the materials used to construct the electrified third rail conductor will be contingent on the subsurface transportation equipment design (SAR Section 1.3.2.4.6.4). The applicant stated that commercially available materials will be used for the electrified third rail (SAR Section 1.3.4.5.7). SAR Section 1.3.3.4.1 and SAR Figure 1.4.1-5 provided a high-level conceptual description of the design of electrified third-rail SSCs to accommodate transmission of three-phase power to a vehicle (requiring at least three power contact rails for each). The applicant's description indicated that SSCs providing power to vehicles in inaccessible areas may be permanently installed within emplacement drifts, turnouts, and other inaccessible areas. Inspection and potential maintenance operations for these areas will be performed using one or more types of ROVs. The applicant also described concepts for additional specialized ROVs that may be tethered, rubber-tired, or rail-type vehicles, some of which may be battery powered (BSC, 2008ca).

**NRC Staff's Evaluation**

The NRC staff evaluated the design descriptions and design information for the proposed non-ITS normal electrical power system using the guidance in YMRP Section 2.1.1.2. For the design of the normal electrical power system providing power to surface facilities and subsurface areas, the NRC staff evaluated the following: (i) design and operational information, (ii) applicability of design codes and standards, and (iii) provisions for adequate preventive and corrective maintenance operations. The evaluation focused on whether the design could adequately perform the stated functions as defined by the applicant and whether the design would interfere with deployment of alternative designs, if needed, to support operations during the preclosure period.

The NRC staff finds that the applicant's descriptions and design information for the normal electrical power system, including the electrified third rail and power provisions for remote operated vehicles, are sufficient for an evaluation of the applicant's preclosure safety analysis and design because the applicant provided (i) descriptions of GROA normal electrical power sources, including offsite power and onsite standby diesel generators; (ii) descriptions of functions of major SSCs and operations for normal power supply and distribution to surface and subsurface distribution centers and major loads; (iii) single-line electrical schematics for the normal and ITS electrical power system; (iv) description of the electrified third rail that will provide power to the TEV, DSEG, and other planned ROVs operating in accessible and inaccessible areas; and (v) applicable codes and standards for surface and subsurface normal electrical power system and major subsurface electrical distribution SSCs that are consistent with the standard engineering practice for systems of similar functions.

The design concept description of the TEV redundant sliding collectors that interface with the electrified third rail indicated that the vehicle design could accommodate planned gaps in the third rail system, such as where roads will cross the TEV rail in accessible areas. Planned gaps in the third rail can be maintained to be functional as they can be accessed and repaired. The NRC staff finds that the conceptual descriptions of the electrified third rail and power provisions for non-ITS ROVs in inaccessible subsurface areas are acceptable. The basis for the NRC staff's acceptance is because the applicant provided (i) a conceptual description of the electrified third rail intended to provide power to the TEV, DSEG, and other planned ROVs operating in emplacement drifts and turnouts and (ii) conceptual descriptions of additional ROVs using different types of power connections for operations in inaccessible areas. For these non-ITS items, the NRC staff finds that the conceptual description is adequate to understand the intended capabilities and operational processes for the ROVs and is sufficient for use in evaluation of the PCSA and design.

**ITS Electric Power System**

The applicant provided a high-level conceptual and functional description of the ITS electrical power system design for designated surface facilities in SAR Sections 1.4.1.2 and 1.4.1.3. The proposed ITS electrical power system consists of ITS switchgear, ITS diesel generators and associated ITS diesel generator mechanical support systems, ITS 13.8-kV breaker automatic load sequencers, ITS 13.8-kV to 480-V transformers, ITS 480-V load centers and ITS motor control centers, ITS 125-V DC battery power supplies, and ITS uninterruptible power supplies (UPS) SSCs. The applicant listed and discussed applicable principal codes and standards for design methods and practices for the ITS electrical power system in SAR Section 1.4.1.2.8. The applicant provided supplemental information regarding the applicability of cited principal codes and standards (DOE, 2009dl) and discussed the proposed application of specific sections of principal codes and industry standards that the applicant would apply to the final design of the ITS electrical power system (DOE, 2009do). The applicant proposed to use IEEE 308, 379, 384, and 603 (Institute of Electrical and Electronics Engineers, 2001aa, ab; 1998aa, ab) as the principal codes and standards for the ITS electrical power system design. These standards describe the need to incorporate design criteria such as redundancy, spatial separation, independence between redundant channels, and isolation between safety and nonsafety circuits. The applicant further described how it would interpret the applicability of the specific sections of principal codes and standards to the ITS electrical power system.

The ITS electrical power system provides redundant, independent, and separate trains of power distribution to designated facilities and loads within the geologic repository operations area. Either of the ITS electrical power system trains (A or B) is capable of providing the power needed to perform defined safety functions. Major loads powered by the redundant ITS electrical power system trains, such as ITS heating, ventilation, and air conditioning SSCs, were also described as redundant, independent operating SSC trains, each powered independently by one of the two ITS electrical power system trains. The applicant stated that, during normal operations, one of the ITS operating SSC (ITS HVAC) trains will be running while the other train will be in standby (SAR Section 1.9.1.12). Upon failure of a working ITS electrical power system or respective ITS operating SSC train, the standby ITS electrical power system and respective operating SSC train automatically engage so the associated safety function can continue to perform. Upon a LOSP event, the ITS electrical power system will automatically disconnect from the offsite power grid and begin to start and load the onsite ITS diesel generators, each connected to its respective ITS switchgear and ITS electrical power system train to supply power to electrical facilities and loads that relies on the ITS electrical power system. During an LOSP, both ITS diesel generators will be started and maintained in a running condition to maintain the redundant features of the ITS electrical power system. An ITS electrical interlock for circuit breakers, which will prevent automatic connection of an ITS diesel generator to an energized or faulted bus, was described in SAR Section 1.4.1.2.1.

The ITS electrical power system also includes ITS battery-powered DC and ITS battery-powered UPS SSCs to provide power to ITS SSCs that will require continuous electrical power to perform or contribute to safety function performance. Batteries for ITS DC electrical power, as described in SAR Section 1.4.1.3.1, will be sized to have sufficient capacity to support loads for 8 hours.

**NRC Staff's Evaluation**

The NRC staff evaluated the design descriptions and design information for the proposed ITS electrical power system using the guidance in YMRP Section 2.1.1.2. The NRC staff focused on the adequacy of the design and operational information provided in the SAR and the applicability of design codes and standards cited by the applicant.

The NRC staff finds that the design descriptions and design information the applicant provided are sufficient to permit an evaluation of the PCSA and design of the ITS electrical power system because the applicant provided (i) adequate descriptions of the design and operational information for both the normal preferred source of electrical power and back-up electrical power source, including processes associated with the loss of offsite power; (ii) adequate descriptions of the components for the ITS electrical power system (e.g., diesel generators, DC batteries, automatic load sequencers); (iii) adequate descriptions of the ITS electrical power system functions; and (iv) applicable codes and standards, which the NRC staff evaluated in SER Section 2.1.1.7.3.6, and concluded are acceptable because they are in conformance with standard engineering practices.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.2.3, the NRC staff finds, with reasonable assurance, that the applicant's descriptions and design information for the electrical power system for the GROA meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the GROA electrical power system sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.2.4 Heating, Ventilation, and Air Conditioning and Filtration Systems

The applicant described the heating, ventilation, and air conditioning (HVAC) and filtration systems at the GROA surface facilities in SAR Sections 1.2.2.3, 1.2.3.4, 1.2.4.4, 1.2.5.5, and 1.2.6.4. In addition, the applicant described the HVAC systems for the balance-of-plant facilities in SAR Sections 1.2.8.3.1 and 1.2.8.3.2. The applicant proposed to use HVAC systems during normal operations to (i) control flow from areas of lesser to greater contamination potential, (ii) control temperature for the health and safety of workers and proper equipment operation, (iii) limit the release and spread of airborne contamination in and from the surface facilities through filtration, and (iv) provide a release point to the atmosphere. In addition, the applicant stated that the HVAC systems shall ensure reliable confinement and filtration of radiological releases from event sequences that involve breach of a waste container or damaged spent nuclear fuel (SNF) assembly. The applicant classified as ITS the HVAC system components required to mitigate the consequences of a radioactive release following an event sequence and provide cooling to the ITS equipment.

The applicant described the HVAC systems for the surface facilities in terms of ITS or non-ITS subsystems serving confinement zones or nonconfinement zones. Secondary confinement (i.e., an area with a potential for airborne contamination during normal operations) was identified for the pool room in the wet handling facility only. Both tertiary confinement (i.e., areas where airborne contamination will not be expected during normal operations) and nonconfinement (i.e., noncontaminated or clean areas) were identified for the RF, CRCF, WHF, IHF, and LLWF. Only nonconfinement zones were identified for the EDGF and CCCF.

The HVAC systems will consist of supply and exhaust subsystems with similar basic features but varying capacities for different surface facilities. In particular, the confinement areas of the surface facilities will be equipped with a recirculation supply subsystem and an exhaust subsystem. In SAR Section 1.2.2.3.2, the applicant stated that each facility will be equipped with a discharge duct capable of a minimum discharge velocity of 15.24 m/s [3,000 ft/min]. According to the HVAC description in SAR Section 1.2.2.3.1, the HVAC system components will include dampers (e.g., isolation, volume, back draft, tornado, and fire/smoke dampers), ductwork, fans, HEPA filters, moisture separators, and prefilters. In addition, the HVAC system will have the necessary instrumentation and control (I&C) listed in SAR Table 1.2.2-14.

The applicant addressed the location and arrangement of the HVAC supply and exhaust equipment in SAR Sections 1.2.3.4, 1.2.4.4, 1.2.5.5, 1.2.6.4, and 1.2.8.3 for individual surface facilities. According to the applicant, the location and arrangement of the HVAC systems within the surface facilities will ensure no interference with the safety functions of adjacent equipment and/or other systems. In SAR Section 1.2.4.4.2, the applicant described the operational processes for the HVAC systems and potential interaction between the HVAC systems and other SSCs or support systems (e.g., electrical power, fire protection, radiation monitoring, and alarm systems).

The applicant stated that, for structural design, ITS components of the HVAC systems will be designed for dead weight; dynamic weight; constraint of free displacement; system operational transient; fluid momentum; and external loads, pressure differentials, and seismic events. The applicant stated that the load combinations used in the design analysis of ITS HVAC systems, with the exception of seismic loads, will be in accordance with ASME AG–1–2003, including 2004 addenda (AG–1a–2004) Articles SA–4212 and SA–4216 (Table SA–4216), Articles BA–4131 and AA–4212 (Table AA–4212), and Article HA–4212 (Table HA–4212) (American Society of Mechanical Engineers, 2004ac). In addition, the applicant applied the seismic loads in accordance with the International Building Code 2000 (International Code Council, 2003aa). The NRC staff's evaluation of ITS HVAC systems' structural and thermal design is provided in SER Section 2.1.1.7.3.3. In addition, according to the applicant, the non-ITS components of the HVAC systems will be designed to seismic loads (International Code Council, 2003aa), and their design ensures that failures of a non-ITS component will not prevent an ITS SSC from performing its intended safety function.

The applicant defined the materials of construction for the HVAC systems as follows: minimum 18–gauge, 304L stainless steel for ductwork and 14–gauge, 304L stainless steel castings for the glass fiber HEPA filters and HEPA filter housing (ASTM International, 2006aa), and fans in accordance with ASME AG–1–2003, including 2004 addenda (AG–1a–2004), Article BA 3000, and Table BA-3100 (American Society of Mechanical Engineers, 2004ac).

The applicant identified the codes and standards applicable to the design and fabrication of the ITS (HVAC) systems in SAR Section 1.2.2.3.8 and provided the codes and standards for specific HVAC components in SAR Table 1.2.2-12. In addition, the regulatory guidance used for the design and analysis of the HVAC systems was summarized in SAR Table 1.2.2-9. In response to an NRC staff request for additional information (RAI), the applicant also provided some examples that specify certain sections of the codes and standards intended to be used for HVAC component design (DOE, 2009dw).

**NRC Staff's Evaluation**

The NRC staff evaluated the ITS and non-ITS surface heating, ventilation and air-conditioning (HVAC) systems information using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the HVAC layouts, design information, and location and arrangement of the HVAC systems to prevent interference with the safety functions of adjacent systems, potential interactions between the HVAC and other systems, and the ability of the HVAC systems to withstand the effects of natural phenomena. The NRC staff finds that the applicant provided adequate descriptions and design information of the HVAC systems because the applicant (i) discussed the functions of the HVAC systems during the normal operations for the surface facilities, including the design of the HVAC systems to maintain flow from low to higher potential for radioactive contamination; (ii) described the safety functions of the portions of the HVAC systems related to mitigating the consequences of a radioactive release following an event sequence and providing cooling to the other ITS equipment; (iii) discussed the components of the HVAC systems, such as dampers, ductwork, fans, HEPA filters, moisture separators, and prefilters; (iv) included information on materials of construction, fabrication, and design codes and standards for the ITS HVAC systems; and (v) discussed the design loads and load combinations to be used to design ITS HVAC systems.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the HVAC systems because the applicant included floor plans for the CRCF, IHF, WHF, and RF that delineated the location and arrangement of the HVAC systems with respect to other systems within each facility.

The NRC staff finds that the applicant provided an adequate discussion of potential interactions between the HVAC systems and other systems because (i) the general arrangement floor plans gave a basic understanding of the interactions between the HVAC systems and other systems (e.g., electrical power, fire protection, radiation monitoring, and alarm systems) and (ii) the location and arrangement of the HVAC systems within the surface facilities will help ensure no interference with the safety functions of adjacent equipment and/or other systems.

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the capability of the surface facilities to withstand the effects of natural phenomena because (i) the applicant will install tornado dampers in the ITS outdoor air intake and exhaust ductworks to prevent duct collapse during a tornado event, to mitigate the damage to the ITS systems and components (SAR Section 1.2.2.3.1) and (ii) the ITS HVAC equipment will be designed for loads resulting from seismic events.

On the basis of the above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of the HVAC systems.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.2.4, the NRC staff finds, with reasonable assurance, that the applicant's descriptions and design information for the HVAC systems meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the HVAC systems sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.2.5 Mechanical Handling Equipment

The applicant provided a description of the design of specialized and one-of-a-kind ITS mechanical handling equipment to be used in the IHF, CRCF, WHF, and RF operations. This specialized equipment included the following five types of ITS mechanical equipment: (i) canister transfer machine (CTM), (ii) cask handling crane (CHC), (iii) spent fuel transfer machine (SFTM), (iv) canister transfer trolley (CTT), and (v) waste package transfer trolley (WPTT). In addition, the applicant provided information on other ITS mechanical handling equipment that is not specialized or one-of-a-kind as the aforementioned major ITS mechanical handling equipment. These ITS mechanical handling systems are more prevalent in the nuclear industry.

***Canister Transfer Machine***

The CTM is classified by the applicant as an ITS SSC in the canister transfer subsystem. The CTM is a special-purpose overhead bridge crane with two trolleys. The first trolley will hoist the canisters with a grapple attachment. The second trolley will support a shield bell assembly that will be used to provide radiation shielding during canister transfer operations. The bottom end of the shield bell will support a motorized slide gate, which, when closed, will provide bottom shielding of the canister. The CTM bridge is similar to a typical crane bridge with end trucks riding rails supported by wall corbels. Each bridge girder supports two sets of trolley rails. The two inner rails will be used for the canister hoist trolley, and the two outer rails will be used for the shield bell trolley. The design and operation of the CTM is the same for all facilities.

The applicant described the CTM in SAR Section 1.2.4.2.2.1.3. The canister transfer machine will be used to transfer canisters from transportation casks or aging overpacks to waste packages, gaining overpacks, shielded transfer casks, or staging areas for temporary staging. In SAR Section 1.2.4.2.2.2, the applicant described the operational processes for the CTM and potential interaction between the CTM and other SSCs or support systems in the canister transfer subsystem. The general arrangement floor plans in SAR Figures 1.2.3-4 (IHF), 1.2.4-3 (CRCF), 1.2.5-4 (WHF), and 1.2.6-3 (RF) provided locations and functional arrangements of the CTM.

SAR Figure 1.2.4-50 showed the plan and elevation view of the CTM with a defined clearance envelope. The applicant stated that the CTM will be designed in accordance with ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The applicant stated that it includes design features such as load path redundancy, overload protection, redundant braking systems, and over-travel limit switches to limit the likelihood of a load drop for the CTM (SAR Section 1.2.2.2.1). For the overhead cranes, the CTM, and the SFTM, the applicant considered the following load cases: normal operation load combinations (including testing and operating events) as in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), site-specific ground motions (DBGM-2), extreme wind, and collision.

The applicant provided approximate dimensions of the CTM in BSC (2008bg, Section 6.2.2.12). The shield bell is approximately 8-m [25-ft]-tall with an inside diameter of 1.8 m [6 ft]. The bottom end of the shield bell is attached to a larger chamber to accommodate cask lids with a diameter of 2 m [7 ft]. The CTM bottom plate supports a 0.3-m [1-ft] motorized slide gate. Further, the applicant relies on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods.

**NRC Staff's Evaluation**

The NRC staff evaluated the CTM system description and design information using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the CTM layout; design information including drawings, functions, and potential interactions of the CTM system with other ITS SSCs; and the ability of the CTM to withstand the effects of natural phenomena. The NRC staff finds that the description and design information of the ITS CTM is adequate because the applicant (i) discussed the functions of the CTM for the IHF, CRCF, WHF, and RF; (ii) discussed the components of the CTM, including bridge crane and trolleys; (iii) provided CTM mechanical drawings, illustrating basic mechanical design details; (iv) described CTM dimensions and geometry; (v) will design the CTM to include design features such as load path redundancy, overload protection, redundant braking systems, and over-travel limit switches to limit the probability of a load drop; (vi) included information on materials of construction and design code and a standard for the CTM, which the NRC staff evaluates in SER Section 2.1.1.7.2.1, and concludes is acceptable because it is consistent with the design codes and standards in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa); and (vii) discussed the design loads and load combinations to be used to design CTM.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the CTM because the applicant included floor plans for the CRCF, IHF, WHF, and RF that delineated the location and arrangement of the CTM with respect to other systems within each facility.

The NRC staff finds that the applicant provided an adequate discussion on potential interactions between the CTM and other systems because the general arrangement floor plans and the applicant's description of the CTM operational processes gave a basic understanding of potential interactions of the CTM with other SSCs, including the support systems in the canister transfer subsystem.

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the CTM's ability to withstand the effects of natural phenomena because the CTM will be designed for loads resulting from site-specific seismic events and extreme wind.

On the basis of above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of the ITS CTM.

### *Cask Handling Crane*

The cask handling crane (CHC) is an ITS SSC in the cask preparation subsystem with the main function of moving a transportation cask into a cask transfer trolley (CTT). The CHC will also be used to upend the transportation cask. The CHC is a large gantry crane with a rated payload capacity ranging from 181,437 to 272,155 kg [200 to 300 T] (SAR Table 1.2.2-10). The CHC is a top running, double-girder-type bridge crane with a top running trolley. The CHC will be used in all four surface facilities (IHF, CRCF, RF, and WHF). The applicant (i) detailed the CHC equipment in SAR Section 1.2.4.2.1.1.3; (ii) provided mechanical design details of the CHC specific to the IHF in SAR Figure 1.2.3-19, to the CRCF in SAR Figures 1.2.4-34 and 1.2.4-35, and to the receipt facility (RF) in SAR Figure 1.2.6-15; and (iii) presented logic diagrams in SAR Figures 1.2.4-36 and 1.2.4-37 for the CRCF.

In SAR Section 1.2.4.2.1.2.1, the applicant described the operational processes for the cask handling subsystem involving using the CHC and potential interaction between the CHC and other SSCs or support systems (e.g., cask handling yoke, cask tilting frame, and cask transfer trolley). The general arrangement floor plans in SAR Figures 1.2.3-2 (IHF), 1.2.4-2 (CRCF), 1.2.5-4 (WHF), and 1.2.6-2 (RF) and the cross-section views in SAR Figures 1.2.3-8 (IHF), 1.2.4-7 (CRCF), 1.2.5-7 (WHF), and 1.2.6-7 (RF) provided locations and functional arrangements of the CHC.

The applicant will use ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as the design code and standard for the CHC. The CHC will be equipped with seismic-restraint clamps to prevent derailment and load drop (SAR Section 1.2.4.2.1.1.3.1). The applicant stated that the design has features such as load path redundancy, overload protection, redundant braking systems, and over-travel limit switches to limit the likelihood of a load drop for the CHC (SAR Section 1.2.2.2.1). As stated in SAR Section 1.2.2.2.9.2.1, for overhead cranes such as the CHC, the applicant considered the following load cases: normal operation load combinations (including testing and operating events) as in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), site-specific ground motions (DBGM-2), extreme wind, and collision. In the IHF, the CHC will be rated at 272,155 kg [300 T]. For the CRCF, the main hoist will be rated at 181,437 kg [200 T] with an auxiliary hoist rated at 18,144 kg [20 T].

### NRC Staff's Evaluation

The NRC staff evaluated the description and design information for the CHC system using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the CHC layout; design information including drawings, functions, and potential interactions of the CHC system with other ITS SSCs; and the ability of the CHC to withstand the effects of natural phenomena. The NRC staff finds that the description and design information of the ITS CHC is adequate because the applicant (i) discussed the functions of the CHC for the IHF, CRCF, WHF, and RF; (ii) described the components of the CHC including crane bridge, trolley, and crane hoist; (iii) provided CHC mechanical drawings including geometry; (iv) will design the CHC to include the features such as load path redundancy, overload protection, redundant braking systems, and over-travel limit switches to limit the probability of a load drop; (v) included information on design codes and standards for the CHC, which the NRC staff evaluates in SER Section 2.1.1.7.2.3.4, and concludes is acceptable because it is consistent with design codes and standards in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa); and (vi) discussed the design loads and load combinations to be used to design CHC.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the CTM because the applicant included floor plans and cross-sectional views for the CRCF, IHF, WHF, and RF that delineated the location and arrangement of the CHC with respect to other systems within each facility.

The NRC staff finds that the applicant provided an adequate discussion on potential interactions between the CHC and other systems because the general arrangement floor plans and cross-sectional views and the applicant's description of the CHC operational processes gave a basic understanding of potential interactions of the CHC with other SSCs including the support systems in the cask preparation subsystem (e.g., cask handling yoke, cask tilting frame, and cask transfer trolley).

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the CHC's ability to withstand the effects of natural phenomena because the CHC will be designed for loads resulting from site-specific seismic events and extreme wind.
On the basis of above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of the ITS CHC.

***Spent Fuel Transfer Machine***

The spent fuel transfer machine (SFTM) is an ITS SSC in the spent nuclear fuel assembly transfer subsystem in the WHF. The SFTM will transfer spent nuclear fuel (SNF) arriving in transportation casks and dual-purpose canisters (DPCs) into spent fuel racks and into transportation aging and disposal (TAD) canisters or alternatively to a staging rack in the pool. The SFTM is a bridge-type crane that will span the pool of the WHF and runs on rails on the edge of the pool. The trolley will run on a set of rails on the bridge. The applicant described the SFTM in SAR Section 1.2.5.2.2.1.3. The applicant provided mechanical design drawings of the SFTM in SAR Figure 1.2.5-47, the process and instrumentation diagram in SAR Figure 1.2.5-48, the logic diagram for the SFTM mast hoist in SAR Figure 1.2.5-49, and the logic diagram for the SFTM grapple in SAR Figure 1.2.5-50.

In SAR Section 1.2.5.2.2.2, the applicant described the operational processes for the SNF assembly transfer subsystem involving the use of the SFTM and potential interactions between the SFTM and other SSCs or support systems (e.g., PWR and PWR lifting grapples). The general arrangement floor plan in SAR Figure 1.2.5-2 for the WHF provided location and functional arrangement of the SFTM.

The applicant stated that the SFTM will be designed in accordance with ASME NOG–1–2004 Sections 4200 and 5200 (American Society of Mechanical Engineers, 2005aa) for a Type I crane and to meet the site-specific ground motions (DBGM-2). In SAR Section 1.2.2.2.9.2.1, the applicant described the load combinations used in the design: normal operation (including testing and operating events) load combinations as in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), site-specific ground motions (DBGM-2), extreme wind, and collision. The applicant stated that the SFTM's design will prevent the device from overturning, derailing, losing any main structural components, or dropping an SNF assembly or a load (SAR Section 1.2.5.2.2.1.3). The applicant stated that the design includes design features such as load path redundancy, overload protection, redundant braking systems, and end-of-travel limit switches to limit the likelihood of a load drop for the SFTM (SAR Section 1.2.2.2.1). The applicant provided approximate dimensions for the SFTM in BSC (2008bg, Section 6.2.2.14). The minimum clearance between the top of the SFTM and the ceiling of the WHF will be 0.6 m [2 ft]. A retractable camera will be at the end of a 4.6-m [15-ft] pole that will be attached to the SFTM. Finally, the applicant will apply the ASME NOG–1–2004 standard (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods.

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the SFTM system using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the SFTM layout and design information, including drawings, functions, and potential interactions of the SFTM with other ITS SSCs. The NRC staff finds that the description and design information of the ITS SFTM is adequate because the applicant (i) discussed the functions of the SFTM for the WHF; (ii) provided SFTM mechanical drawings, illustrating basic mechanical design details; (iii) described SFTM dimensions and geometry; (iv) will design the SFTM to include features such as load path redundancy, overload protection, redundant braking systems, and end-of-travel limit switches to limit the probability of a load drop; (v) the SFTM is designed to prevent overturning, derailment, or a load drop; (vi) included information on materials of construction and design codes and standards for the SFTM, which the NRC staff evaluates in SER Section 2.1.1.7.3.2.3, and concludes is acceptable because it is consistent with design codes and standards in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa); and (vii) discussed the design loads and load combinations to be used to design SFTM.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the SFTM because the applicant included floor plans for the WHF that delineated the location and arrangement of the SFTM with respect to other systems in the WHF.

The NRC staff finds that the applicant provided an adequate discussion on potential interactions between the SFTM and other systems because the general arrangement floor plans and the applicant's description of the operational processes relevant to the SFTM gave a basic understanding of potential interactions of the SFTM with other SSCs (e.g., BWR and PWR lifting grapples) and the support systems in the SNF assembly transfer subsystem.

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the SFTM's ability to withstand the effects of natural phenomena because the SFTM will be designed for loads resulting from site-specific seismic events and extreme wind.

On the basis of above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of the ITS SFTM.

***Cask Transfer Trolley***

The cask transfer trolley (CTT) is an ITS component in the cask preparation subsystem. The CTT is a unique air-powered transport machine to be used in the CRCF, WHF, RF, and IHF to transfer the transportation casks between the cask preparation area and the cask unloading room to the canister transfer room. The trolley consists of a platform, a cask support assembly, a pedestal assembly, a seismic restraint system, and an air system that levitates the CTT between 1.27 and 2.22 cm [0.5 and 0.875 in] above the floor. The CTT is propelled and steered using two pneumatically powered traction drive units. To handle the different sizes of casks, pedestals will be used in the bottom of the CTT. The pedestal is loaded into the CTT using the cask handling crane. The applicant described the CTT in SAR Section 1.2.4.2.1.1.3.1 (CRCF), provided mechanical drawings in SAR Figure 1.2.4-26, and provided a process and instrumentation diagram in SAR Figure 1.2.4-27. The applicant described the CTT of the IHF in SAR Section 1.2.3.2.1.1.3.1, as illustrated in SAR Figure 1.2.3-20.

In SAR Section 1.2.4.2.1.2.1, the applicant described the operational processes involving the CTT and potential interaction between the CTT and other SSCs or support systems in the cask preparation subsystem (e.g., cask handling crane). The general arrangement floor plans in SAR Figures 1.2.3-2 (IHF), 1.2.4-2 (CRCF), 1.2.5-2 (WHF), and 1.2.6-2 (RF) provided locations and functional arrangements of the CTT.

The CTT will be designed in accordance with the requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) applicable to a Type I crane trolley, except for the unique features associated with the pneumatic components. In addition to ASME NOG–1–2004, the applicant will use specific design codes and standards for the pneumatic valves, pressure relief valves, air cylinders, air bearings/casters, air motors, and piping of the CTT (DOE, 2009dq). These design codes include ASME B16.34–2004 (American Society of Mechanical Engineers, 2005ab) for ball, gate, and throttle valves; ASME Boiler and Pressure Vessel Code, Section VIII, Paragraph UG–131 (American Society of Mechanical Engineers, 2004aa) for safety relief valves; API 526 (American Petroleum Institute, 2002aa) for pressure relief valves; API 527 (American Petroleum Institute, 1991aa) for seat tightness of pressure relief valves; and ASME B31.3–2004 (American Society of Mechanical Engineers, 2004ab) for process piping. DOE also stated that design of commercially available air cylinders, air motors, and air bearings/casters will follow manufacturer's standards (DOE, 2009dq).

The applicant provided approximate dimensions for the CTT in BSC (2008bg, Section 6.2.2.7). The applicant will follow ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods. The applicant stated that it will include design features such as redundant systems and speed limitations to limit the likelihood of a tip over, collision, or uncontrolled movement for the CTT (SAR Section 1.2.2.2.1). The CTT will be designed to meet the site-specific seismic ground motions such that the trolley does not tip over but may slide. For beyond design basis seismic events that would produce greater movements, energy-absorbing features will be used to minimize the effect of impact forces on the cask and to prevent tip over. In SAR Section 1.2.4.2.1.9, the applicant stated that the load combination analysis for the CTT will be in accordance with ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The CTT to be used in the IHF will be part of the cask preparation subsystem and will be rated at 240,404 kg [265 T]. The CTT to be used in the CRCF, WHF, and RF will be rated at 181,437 kg [200 T].

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the CTT system using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the cask transfer trolley system layout and design information including drawings, functions, and potential interactions of the CTT system with other ITS SSCs. The NRC staff finds that the description and design information of the ITS CTT is adequate because the applicant (i) discussed the functions of the CTT for the surface facilities (IHF, CRCF, WHF, and RF); (ii) described the components of the CTT including a platform, a cask support assembly, a pedestal assembly, a seismic restraint system, and an air system; (iii) provided CTT mechanical drawings, illustrating basic mechanical design details; (iv) described CTT's dimensions, geometry, and capacity in each surface facility; (v) will design the CTT to include features such as redundant systems and speed limitations to reduce the likelihood of a tip over, collision, or uncontrolled movement for the CTT; (vi) included information on materials of construction and design codes and standards for the CTT, which the NRC staff evaluates in SER Section 2.1.1.7.3.2.4, and concludes is acceptable because it provides applicable guidance for designing SSCs for their intended use; and (vii) discussed the design loads and load combinations to be used to design CTT.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the CTT because the applicant included floor plans for the IHF, CRCF, WHF, and RF that delineated the locations and arrangements of the CTT with respect to other systems within each facility.

The NRC staff finds that the applicant provided an adequate discussion on potential interactions between the CTT and other systems because the general arrangement floor plans and the applicant's description of the operational processes relevant to the CTT gave a basic understanding of potential interactions of the CTT with other SSCs (e.g., cask handling crane) and the support systems in the cask preparation subsystem.

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the CTT's ability to withstand the effects of natural phenomena because the CTT will be designed for loads resulting from the site-specific design basis seismic ground motion and the CTT will include energy-absorbing features to minimize the effect of impact forces on the cask and to prevent tip over for beyond design basis seismic events.

On the basis of above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of the ITS CTT.

***Waste Package Transfer Trolley***

The ITS waste package transfer trolley (WPTT) consists of two main components: the shielded enclosure and the trolley. The WPTT is a trolley that operates on rails and is part of the waste package load-out subsystem of both the IHF (SAR Section 1.2.3.2.4.1.3) and the CRCF (SAR Section 1.2.4.2.4.1.3). The capacity of the WPTT is 90,718 kg [100 T]. The applicant provided basic mechanical design details of the WPTT in SAR Figure 1.2.4-88, the process and instrumentation diagram in SAR Figure 1.2.4-89, and the WPTT logic diagram in SAR Figure 1.2.4-90.

The WPTT will be used to orient and transport a waste package for placement of canisters, lid installation, and delivery to the transport and emplacement vehicle (TEV). The WPTT has a shielded enclosure that allows access to the top of the loaded waste package for closure activities and includes a pedestal that positions the top of the loaded waste package at the required elevation for closure. The applicant provided approximate dimensions for the WPTT in BSC (2008bg, Section 6.2.2.17). The WPTT will be remotely controlled.

In SAR Section 1.2.4.2.4.1.3, the applicant described the operational processes for the WPTT and potential interactions between the WPTT and other SSCs or support systems (e.g., waste package transfer carriage docking station, waste package port slide gate, waste package handling crane, waste package retrieval assembly, CTM, and TEV) in the waste package load-out subsystem. The general arrangement floor plans in SAR Figures 1.2.3-2 (IHF) and 1.2.4-2 (CRCF) provided locations and functional arrangements of the WPTT.

The applicant selected ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as the main design and materials construction code and standard for the WPTT. The load combinations to be used for the WPTT are based on normal operations, earthquakes, extreme winds, and collisions, as stated in SAR Section 1.2.2.2.9.2.4. The applicant stated that the WPTT includes features to reduce the likelihood of a tip over, collision, or uncontrolled movement. For example, the WPTT will be equipped with seismic restraints to help prevent derailment leading to tip over.

The process and instrumentation diagrams provided in the SAR contained pictorial descriptions of the safety features and their interactions with various components; number of programmable logic controllers (PLCs) within the WPTT system; flow of drive commands to the WPTT; and interaction of electrical signals from the slide gate, seismic sensors, and disconnect switch to the WPTT.

The applicant also provided information on potential interactions of the WPTT with other ITS systems. The WPTT will interact primarily with two ITS SSCs: the CTM in the waste package positioning room and the TEV in the waste package load-out room. Interlocks between these systems will ensure safe interaction of the WPTT with the aforementioned ITS SSCs. In SAR Sections 1.2.4.2.4.1.3 and 1.2.4.2.4.2, the applicant provided additional details on the interactions of the WPTT with other SSCs.

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the WPTT system using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the WPTT layout and design information, including drawings, functions, and potential interactions of the WPTT with other ITS SSCs. The NRC staff finds that the description and design information of the ITS WPTT are adequate because the applicant (i) discussed the functions of the WPTT for the IHF and CRCF; (ii) described the components of the WPTT, including a shielded enclosure and a trolley; (iii) provided WPTT mechanical drawings, illustrating basic mechanical design details, and process and instrumentation diagrams showing safety features and their interactions with various components; (iv) described the WPTT's dimensions, geometry, and capacity in the IHF and CRCF; (v) will design the WPTT to include features to reduce the WPTT's likelihood of a tip over, collision, or uncontrolled movement; (vi) included information on materials of construction and design code and standard for the WPTT, which the NRC staff evaluates in SER Section 2.1.1.7.3.2.2, and concludes is acceptable because it is consistent with design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa); and (vii) discussed the design loads and load combinations to be used to design the WPTT.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the WPTT because the applicant included floor plans for the IHF and CRCF that delineated the locations and arrangements of the WPTT, with respect to other systems in the IHF and CRCF.

The NRC staff finds that the applicant provided an adequate discussion on potential interactions between the WPTT and other systems because the general arrangement floor plans and the applicant's description of the operational processes relevant to the WPTT gave a basic understanding of potential interactions between the WPTT with other SSCs and support systems in the waste package load-out subsystem (e.g., waste package transfer carriage docking station, waste package port slide gate, waste package handling crane, waste package retrieval assembly, CTM, and TEV).

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the WPTT's ability to withstand the effects of natural phenomena because the WPTT will be (i) designed for loads resulting from site-specific seismic events and (ii) equipped with seismic restraints to prevent derailment during a seismic event.

On the basis of above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of the ITS WPTT.

***Other ITS Mechanical Handling Equipment***

The applicant categorized other ITS mechanical handling equipment that would not be specialized or one-of-a-kind, as follows: (i) crane systems, (ii) special lifting components, (iii) shield and confinement doors and sliding gates, (iv) rails, (v) platforms, and (vi) racks.

The overhead crane systems consist of standard cranes with varying load ratings and are designed following ASME NOG–1–2004 requirements, including load combinations for Type 1 cranes (American Society of Mechanical Engineers, 2005aa). The jib crane systems design follows ASME NUM–1–2004 standard requirements, including load combinations (SAR Section 1.2.2.2.9.2). In addition to the load combinations in the codes and standards, the applicant specified load combinations specific to earthquake, extreme wind, and collision (SAR Sections 1.2.2.2.9.2.1 and 1.2.2.2.9.2.2). Both types of cranes will have features such as load path redundancy, conservative design factors, overload protection, redundant braking systems, and over-travel limit switches to limit the possibility of a load drop (SAR Section 1.2.2.2.1).

Special lifting components, such as cask yokes, canister grapples, and lifting beams, will be attached to the end of standard mechanical handling equipment to lift and transport casks, overpacks, or canisters containing waste. The applicant stated in SAR Section 1.2.2.2.1 that these components design is in accordance with ANSI N14.6–1993 (American National Standard Institute,1993aa), as modified by NUREG–0612, Section 5.1.1(4) (NRC, 1980aa]. The applicant stated that it will apply the load combinations in ASME NOG–1–2004 and the load combinations the applicant developed specifically for earthquake and collision (SAR Section 1.2.2.2.9.2.3) to design these components. The applicant stated that the design of the special lifting components will account for load-drop prevention as well as load drop onto a cask or canister and incorporates design features to prevent unintentional load disengagement. The applicant further stated that these special lifting components are commonly used in nuclear facilities. The applicant provided mechanical drawings for canister grapples and yokes in the SAR (e.g., SAR Figure 1.2.4-66 for the SNF canister grapples in the CRCF, SAR Figure 1.2.5-43 for the WHF pool lid-lifting grapple, and SAR Figure 1.2.5-17 for WHF pool cask-handling yoke).

Shield and confinement doors and sliding gates are intended to protect facility personnel from direct exposure. The applicant stated that the shield and confinement doors design includes consideration of load combinations, in accordance with ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). The applicant described shield and confinement doors in SAR Section 1.2.4.2.1.1.3.1, with design drawings presented in SAR Figures 1.2.4-19 and 1.2.4-22 and process and instrumentation diagrams illustrated in

SAR Figures 1.2.4-20 and 1.2.4-23. The applicant stated that the shield doors are commonly used in nuclear facilities to prevent personnel exposure to radiation.

Rails are intended to support the WPTT, as well as the TEV, to transport casks from one location to another. The rails for the WPTT and the portion of the rails in the waste package load-out room for the TEV were classified by the applicant as ITS (SER Section 1.2.2.2.1). In SAR Sections 1.2.2.2.1 and 1.2.4.1.6, the applicant stated that the rails design will include load combinations, to be used in accordance with ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa).

Platforms will include multilevel steel structures that provide access to personnel and tools to the top of aging overpacks or transportation casks. The applicant will use the methods and practices provided in American Institute of Steel Construction (1997aa) to design the platforms and include seismic considerations. The applicant described platform categories in SAR Sections 1.2.3.2.1.1.3.1 (IHF), 1.2.4.2.1.1.3.1 (CRCF), 1.2.5.2.1.1.3 (WHF), and 1.2.6.2.1.1.3 (RF). The applicant provided relevant mechanical design drawings in SAR Figures 1.2.3-26 (IHF), 1.2.4-41 (CRCF), 1.2.5-33 (WHF), and 1.2.6-17 (RF).

Racks are intended to stage spent nuclear fuel (SNF) canisters and TAD canisters. The applicant stated that racks design is in accordance with the applicable provisions of ANSI/AISC N690–1994 (American Institute of Steel Construction (1994aa). The applicant addressed thermal safety of the SNF canister staging racks by designing them with fixed neutron absorbers, in accordance with ANSI/ANS–8.21–1995 (American Nuclear Society, 1995aa) and ANSI/ANS–8.14–2004 (American Nuclear Society, 2004aa) to maintain criticality control. The applicant will also include a thermal barrier that will enclose the bottom and sides of the canisters so that the canister temperatures will not rise to unsafe levels in the event of a fire. The applicant described DOE and TAD canister staging racks in SAR Section 1.2.4.2.2.1.3 and provided the design drawings in SAR Figures 1.2.4-68 and 1.2.4-69. The applicant provided design drawings for the SNF staging racks in SAR Figure 1.2.5-51.

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for other ITS mechanical handling equipment (crane systems, special lifting components, shield and confinement doors and sliding gates, rails, platforms, and racks) using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the layouts of other ITS mechanical handling equipment, design information including drawings, and functions. The NRC staff finds that the applicant's description and design information of other ITS mechanical handling equipment are adequate because the applicant (i) discussed the functions of this ITS mechanical handling equipment; (ii) provided mechanical drawings, illustrating basic mechanical design details and equipment geometry; (iii) will design this ITS mechanical handling equipment to include safety features to prevent event sequences from occurring, as appropriate (e.g., inclusion of fixed neutron absorbers in designing staging racks to maintain criticality control and inclusion of design features for the special lifting components to prevent unintentional load disengagement causing load drops that could result in radiological consequences); (iv) included codes and standards for this ITS mechanical handling equipment, which the NRC staff evaluates in SER Section 2.1.1.7.3.4 and Table 7-1, and concludes are acceptable for use at the GROA as; and (v) discussed the design loads and load combinations to be used to design this ITS mechanical handling equipment.

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the ability of this ITS mechanical handling equipment to withstand the effects of natural phenomena because (i) this ITS mechanical equipment will be designed for loads resulting from site-specific seismic events and (ii) the ITS overhead cranes in the IHF will be designed for load combinations.

On the basis of above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient for the NRC staff to evaluate the PCSA and design of this ITS mechanical handling equipment.

**NRC Staff's Conclusion**

On the basis of the evaluation described in SER Section 2.1.1.2.3.2.5, the NRC staff finds, with reasonable assurance, that the applicant's descriptions and design information for the mechanical handling equipment meet the regulatory requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided adequate description and design information for the mechanical handling equipment sufficient for the NRC staff to evaluate the PCSA and design.

2.1.1.2.3.2.6 Shielding and Criticality Control Systems

The applicant described SSCs to be used for shielding and criticality control for the proposed GROA in SAR Sections 1.10.3 and 1.14.

**Shielding**

The applicant described the shielding design of the surface facilities in SAR Section 1.10.3. The facility shielding will be designed to reduce dose rates from radiation sources such that worker doses will be within the standards of 10 CFR Part 20 and will be as low as is reasonably achievable (ALARA) when combined with the program to control personnel access and occupancy of restricted areas. Facility shielding will include concrete walls, floors, shield doors, ceilings, and shielded viewing windows. Design of concrete used for shielding will be in accordance with ACI–349–01/349R–01 (American Concrete Institute, 2001aa) and ANSI/ANS–6.4–2006 (American Nuclear Society, 2006aa).

The applicant described the shielding design by providing the shielding design objectives (SAR Section 1.10.3.1) and shielding design considerations (SAR Section 1.10.3.1.1). The applicant's shielding design objectives were taken from NRC Regulatory Guide 8.8 (NRC, 1978ab). The shielding design considerations the applicant provided defined the bases for the shielding evaluation of the various facility areas and the radiation zones established for each area. The individual radiation zoning characteristics were presented in SAR Table 1.10-1, and specific area dose rate criteria used in the shielding evaluation were presented in SAR Table 1.10-2. The primary material to be used for the shielding evaluation will be Type 04 concrete, based on ANSI/ANS–6.4–2006 (American Nuclear Society, 2006aa, Table 5.2). Other materials used in the shielding evaluation were described in SAR Sections 1.2.3 to 1.2.8.

The applicant used radiation sources (SAR Figure 1.10-18) and bounding terms (SAR Section 1.10.3.4) to approximate the geometry and physical condition of sources in various repository facilities. In addition, the applicant used flux-to-dose-rate conversion factors taken from ANSI/ANS–6.1.1–1977 (American Nuclear Society, 1977aa) to develop dose rates. The applicant assessed the basic design regarding protection of workers and the public using commonly accepted industry computer codes, such as MCNP and SCALE. The applicant's shielding assessment was summarized for various areas and components in SAR Tables 1.10-35 through 1.10-46.

**NRC Staff's Evaluation**

The NRC staff evaluated the descriptions and design information for the shielding systems presented in SAR Section 1.10.3 using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the locations of the shielding systems such as concrete walls, floors, shield doors, ceilings, and shielded-viewing windows. The NRC staff also reviewed design information and potential interactions of the shielding systems with other SSCs within each facility. The NRC staff finds that the applicant's descriptions and design information for shielding systems are sufficient to permit an evaluation of the PCSA and design of these systems because the applicant provided (i) adequate descriptions of the locations and functional arrangement of the shielding systems; (ii) adequate descriptions of their interactions with other SSCs within each facility; and (iii) information that showed the applicant's shielding assessment used codes, standards, and methods, such as American Nuclear Society (1991aa) and International Commission on Radiological Protection (1996aa), that are consistent with NRC guidance.

**Criticality Control**

In SAR Table 1.14-2, the applicant identified several parameters that may need to be controlled to prevent criticality, as discussed next. The applicant described how the other parameters (e.g., geometry and reflection) are bounded in the analysis in SAR Section 1.14.2.3.

In SAR Section 1.14.2.3.2.4, the applicant stated that criticality of high-level radioactive waste (HLW) in glass waste form will be controlled by relying on the low concentrations of fissile isotopes in the HLW glass. SAR Table 1.14-1 presents estimated concentrations of fissile isotopes in HLW glass canisters. This concentration of fissile isotopes in HLW glass is significantly less than the minimum concentration limit for fissile isotopes, identified in ANSI/ANS-8.1, Table 1, Single Parameter Concentration Limits (American Nuclear Society, 1998ab). Therefore, the applicant stated that the HLW glass canisters will be safely subcritical.

According to the applicant, moderation will be the most important control parameter for commercial SNF and DOE SNF. With certain exceptions discussed next, all canisters used in the GROA will remain subcritical when unmoderated. The applicant described and discussed the control of neutron moderation in SAR Section 1.14.2.3 and stated that the guidance of ANS/ANSI–8.22–1997 (American Nuclear Society, 1997ac), which addresses the control of moderators, will be followed.

Neutron interaction between canisters will be controlled in the CRCF by physical barriers (such as the canister staging racks), where neutron interaction among several canisters containing DOE SNF Criticality Groups 2, 3, and 6 (SAR Section 1.5.1.3.1.1.3) can result in criticality. Criticality resulting from interaction among canisters in other facilities will be prevented by physical barriers that will make it impossible to have enough canisters in the same location that would result in criticality.

The applicant plans to use fixed neutron absorbers as part of the canister internals and as part of the SNF staging racks in the WHF pool. During wet handling operations, the presence of 2,500 mg/L [0.02 lb/gal] of soluble boron enriched to 90 wt% B-10 is credited as the primary criticality control parameter. The soluble neutron absorber will be in the form of orthoboric acid ($H_3BO_3$), which will be injected into the water in the WHF pool or in the transportation cask and DPC fill water. To ensure the presence of sufficient concentrations of enriched boron, the applicant developed procedural safety control (PSC)-9 and will sample and analyze the pool water on a regular basis (SAR Sections 1.2.5.3.2.1.3.3 and 1.2.5.3.2.2). Additionally, SAR Table 5.10-1 specifically lists a limiting condition for operations for soluble boron concentration control (corresponding to PSC-9 of SAR Table 1.9-10) that must be established and maintained within a specific range of values (DOE, 2009dh).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description and design information for the criticality control systems using guidance in YMRP Section 2.1.1.2. The NRC staff's review focused on the applicant's descriptions of its criticality controls for maintaining all canisters subcritical when unmoderated, preventing potential interactions among canisters for dry handling, and controlling boron concentration for wet handling operations. The applicant described the isotope characteristics and range of parameters for other radioactive wastes, such as DOE and commercial SNF.

The NRC staff finds that the fissile isotope concentration in HLW glass is subcritical because the concentration is less than the limit for fissile isotopes, as identified in ANSI/ANS–8.1, Table 1 Single Parameter Concentration Limits. Thus, further criticality controls for HLW glass canisters are not necessary and were not discussed by the applicant.

The NRC staff reviewed the applicant's analysis of neutron interaction and the design of DOE SNF staging racks and canister handling equipment and finds that potential criticality conditions would be controlled by physical barriers that passively prevent criticality of DOE SNF canisters. Furthermore, the NRC staff finds that only the most reactive DOE SNF, in Criticality Groups 2, 3, and 6, poses a concern for criticality due to neutron interaction. To ensure that criticality will not occur, the applicant's design and operation controls will physically limit the number of DOE SNF canisters containing DOE SNF Groups 2, 3, and 6, that can be placed in one location next to each other.

The NRC staff finds the applicant's description and design information for the design and operation controls in SAR Section 1.14.2.3.2.2.4 are adequate because the applicant (i) explained that there were no configurations of SNF in the WHF pool that needed more than 30 percent of the minimum required soluble boron concentration in the WHF pool to remain subcritical, (ii) described PSC-9 that establishes the soluble boron concentration at 90 percent, and (iii) described the design and procedures that will provide a margin to protect against potential dilution and uncertainty in the amount of soluble boron in the pool.

The applicant stated that it will follow the guidance in ANSI/ANS–8.22–1997 (American Nuclear Society (1997ac), which is acceptable to the NRC staff because the NRC staff endorsed this standard in Regulatory Guide 3.71 (NRC, 2010ai). Further evaluation of criticality controls is provided in Section 2.1.1.7.

On the basis of the above evaluation, the NRC staff finds that the applicant's description, design, and operations information regarding the criticality control of SNF, DOE SNF, and HLW glass are sufficient for use in PCSA, and review of the design for ensuring subcriticality is maintained. The staff, however, notes that this evaluation of the applicant's criticality controls does not assess whether the applicant could actually receive and possess certain fuel types at this time. For additional discussion and evaluation of the types and quantities of fuel the applicant proposes to handle at the GROA, see SER Section 2.1.1.2.3.6.1.

**Criticality Monitoring and Alarms**

The applicant will not use a criticality accident alarm system to control the consequences of a potential criticality event. Instead, the applicant relied on its screening of criticality as beyond Category 2 to justify that a criticality accident alarm system will not be needed. The applicant also stated that because of the risk of false alarms and potential injury due to unnecessary evacuation, a criticality accident alarm system is considered to have a net adverse effect on worker safety (DOE, 2009di; BSC, 2008ba). This statement was supported by DOE with accounts of false alarms. In addition, radiation monitoring systems in the facility will function as an alarm to indicate any radiation as a result of a criticality event.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information regarding criticality monitoring and alarms using guidance in YMRP Section 2.1.1.2. The NRC staff focused its review on the applicant's design approach that a criticality accident alarm system is not needed to control the consequences of a potential criticality event. The NRC staff reviewed the descriptions of false alarm accounts described by the applicant (NRC, 2008ad, 2004aa, 2002ab) and notes that these incidents did not result in any injury to workers but did occur at a frequency much higher than criticality accidents involving SNF. The NRC staff's evaluation of the applicant's information on design and operations resulting in screening of initiating events that would lead to criticality is in SER Section 2.1.1.3.3.2.6. In addition, in SER Section 2.1.1.4, the NRC staff reviews the applicant's calculation of the probability of event sequences that may lead to an end state important to criticality. In these SER sections, the NRC staff accepted the applicant's screening of criticality as beyond Category 2 and the applicant's conclusion that there is a negligible probability of a criticality event. Because of the negligible probability of a criticality event, the NRC staff finds that the applicant's proposal not to install a criticality accident alarm system is acceptable. The NRC staff also evaluated the applicant's proposal to rely on the facility radiation monitoring system, in lieu of the criticality alarm, to function as an alarm to indicate any radiation from a criticality event. The NRC staff considers this to be a defense-in-depth measure for a potential criticality event because the alarm would sound when radiation levels were high enough to be of immediate concern to personnel safety.

**NRC Staff's Conclusion**

On the basis of the evaluation described in SER Section 2.1.1.2.3.6, the NRC staff finds, with reasonable assurance, that the applicant's descriptions and design information for the shielding and criticality control systems meet the requirements of 10 CFR 63.21(c)(3)(i) and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the shielding and criticality control systems sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed above in the NRC staff's evaluations, the applicant provided (i) adequate descriptions of the surface and subsurface shielding, the criticality control parameters, and justification for not using a criticality monitoring system and (ii) adequate information on the surface and subsurface shielding and the criticality controls used to ensure subcriticality.

2.1.1.2.3.2.7 Fire Safety Systems

The applicant described the fire safety systems in SAR Sections 1.4.3.2 and 1.4.5.1.1.2. These systems will include the site water supply and distribution systems and other active and passive fire protection systems.

The applicant stated in SAR Section 1.4.3.2.1.2 that the site will have a dedicated fire protection water supply and two water distribution systems (Loops 1 and 2). The distribution systems will be composed of site water tanks and fire pumps designed and installed per NFPA 20, NFPA 22, and NFPA 24 (National Fire Protection Association, 2007ad, ae; 2003ab). Four 1,136-$m^3$ [300,000-gal] storage tanks supply Loop 1, and one 1,136-$m^3$ [300,000-gal] storage tank supplies Loop 2. Loop 1 will be configured as a redundant system (e.g., two tanks with associated pumps will be provided in two separate locations to feed Loop 1) because Loop 1 will supply the more critical operational facilities. Loop 2 will supply the geologic repository operations area (GROA) support facilities such as administration, security, and general warehousing areas. The SAR indicated that the building fire-suppression systems, including the site water supply tanks and stationary pumps, were designated as not important to safety (non-ITS).

The applicant described the fire water effluent collection in SAR Section 1.4.5.1.1.2. The fire water effluent may be contaminated resulting in liquid low-level waste. The Liquid LLW Management system will include a series of local containment tanks at the individual buildings. The tanks will be sized in accordance with NFPA 801 (National Fire Protection Association, 2008aa) to contain 30 minutes of overhead sprinkler effluent, along with sufficient freeboard. The applicant will also include containment volume for the largest anticipated vessel spill within each handling facility. Supplemental design data were provided in the applicant’s response to the NRC staff’s request for additional information (RAI) (DOE, 2009dm). The applicant stated that the design capacity of the effluent tanks will include sprinkler effluent, liquid waste from normal facility operation, the contained spill from the largest credible vessel, and appropriate freeboard volumes. The calculations did not include the added volume from manual suppression efforts, which is consistent with NFPA 801.

As described in SAR Section 1.4.3.2.1, the applicant stated that standard water-filled pipe systems will be provided over the majority of the facilities. These systems will deliver water through a pressurized piping network and discharge the water through specific sprinkler heads (nozzles) in the vicinity of a fire. The system will be driven by water pressure provided by the site fire pumps. These systems will be designed per national standards, such as NFPA 13 (National Fire Protection Association, 2007ab). NFPA 13 governs the location of sprinkler heads, installation and support of the overhead sprinkler piping (including seismic bracing requirements), the design methodology used to size the system, and minimum requirements for system maintenance and testing. The applicant indicated that an Ordinary Hazard Group 2 density will be used. This designation will establish the water delivery rate and head-spacing requirements for the systems. Automatic suppression systems are not planned for the subsurface facilities.

The applicant described the double-interlock preaction system in SAR Section 1.4.3.2.1. These sprinkler systems will be a variation of a traditional wet pipe system and will be used in areas where moderator control will be required (e.g., IHF, WHF, RF, and CRCF). The piping network, pipe supports, and overall system components will be identical to a wet-pipe system and will be all designed using the same national standard. The preaction system will be charged with air, rather than water. A sequence of events that includes two independent forms of fire detection will be necessary before water will be discharged through the system. When the interlocks are made, the system will rely on water pressure to deliver suppression water to the source. The fire detection component of the preaction system will include interfaces with local fire detectors (e.g., heat or smoke detectors) and control logic provided by the building fire alarm system. The fire alarm control function of the preaction system will be on standard power, with integral battery backup per NFPA 72 (National Fire Protection Association, 2007af). These systems were identified as ITS from the standpoint of reliability against spurious operation (DOE, 2009dm). The fire control and suppression aspects of these systems were not credited in the PCSA.

As described in SAR Section 1.4.3.2.1.2, the applicant provided standpipes and manual hose stations for local, manual fire suppression. The applicant stated that these stations are designed and installed per NFPA 14 (National Fire Protection Association, 2007ac). The hose stations will be constantly pressurized, and water will be available for use at all times. As described in SAR Section 1.4.3.2.1.2, these outlets will be designed for use with 3.8-cm [1.5-in] and 6.4-cm [2.5-in] hoses and will be designed to flow 0.9 $m^3$/min [250 gpm] per outlet. The overall design and installation of standpipes and manual hose stations will be in accordance with NFPA 14. NFPA 14 provides the specific design details of the standpipes and manual hose stations, such as specific location, hose travel distance, and interconnections. The standpipe system will be a completely manual system and was classified as non-ITS.

Portable extinguishers will be provided throughout the surface and subsurface facilities, as described in SAR Sections 1.4.3.2 and 1.4.3.2.1. These extinguishers will be manual systems, similar to the standpipe and manual hose systems described previously. The applicant stated that the portable extinguishers will be sized and installed per NFPA 10 (National Fire Protection Association, 2007aa). NFPA 10 provides the specific design details of the portable extinguishers, such as specific locations, travel distance, and individual unit sizes. In addition, the applicant designated the extinguishers as non-ITS.

As described in SAR Section 1.4.3.2.2.1, the transport and emplacement vehicle (TEV) will be provided with a dedicated, pre-engineered suppression system to protect the unit from electrical fires (BSC, 2007bf). The applicant stated that the consequence of a fire will be limited, because the TEV would have limited combustible material onboard (BSC, 2007bf). The applicant stated that the suppression system onboard the TEV was not credited to mitigate any fire event that could result in a waste package breach (SAR Section 1.3.2.4.7).

As the applicant described in SAR Sections 1.4.3.2.1.3 and 1.4.3.2.2.2, site fire alarm systems will be primarily notification systems for building occupants and onsite/offsite fire and security personnel. The building fire alarm systems will also play a role in heating, ventilation, and air conditioning (HVAC) control and will provide key input to double-interlock preaction suppression systems. These systems will be installed in accordance with national standard NFPA 72 (National Fire Protection Association, 2007af). Only portions of the fire alarm system responsible for double-interlock sprinkler system operation (as shown in SAR Figure 1.4.3.21) were designated as ITS. Non-ITS fire alarm systems were detailed in SAR Sections 1.2 and 1.3. These systems will be traditional installations that are well described by NFPA 72. As non-ITS systems, no special design bases outside of the national standard will be applied.

Passive fire protection will be provided in each of the main handling facilities and subsurface facilities to compartmentalize the facilities and prevent fire from spreading. The barriers will be noncombustible and will provide a fire resistance rating of up to 3 hours. Openings in fire barrier subsystems will be protected with fire-rated closures (e.g., rated doors, fire dampers, and penetration seals). These systems were not credited in the preclosure safety analysis (PCSA) to reduce the spread of fire throughout the surface facilities (SAR Section 1.7.1.2.2).

Fire barrier subsystems will be used to delineate between emplacement areas and construction areas in the subsurface portions of the GROA. Although these barriers will have a demonstrated fire resistance rating, no credit for a reduction in fire spread was taken in the PCSA for these barriers in the subsurface areas (SAR Section 1.7.1.2.2).

Passive fire barrier designs will follow recognized construction practices to achieve the intended fire resistance ratings. SAR Section 1.4.3 provides national codes and standards to be used in the design of surface and subsurface fire protection. Furthermore, the applicant designated these passive fire-resistance-rated assemblies as non-ITS, and they were not credited in the PCSA (SAR Section 1.7.1.2.2).

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's description and design information for the fire safety systems using the guidance in YMRP Section 2.1.1.2. The NRC staff's review focused on design information for the fire safety systems that were used (i) in areas where moderator control will be required (i.e., IHF, WHF, RF, and CRCF), (ii) for the TEV, and (iii) for the liquid LLW management system. The NRC staff also reviewed the applicant's approach of deviating from NFPA 90A (National Fire Protection Association, 2002aa) by not requiring automatic shutdown of the air handler on duct smoke detection. The NRC staff finds that the applicant's description and design information for these fire safety systems (the traditional wet-sprinkler systems, double-interlock preaction systems, standpipes, fire alarm systems, fire-barrier system, portable fire extinguishers, and manual hose stations) are acceptable because their design and installation will be consistent with national standards, which will ensure the desired level of fire fighting capabilities. The NRC staff also finds that design of passive fire barriers using national design standards is acceptable because GROA operations do not pose any unusual fire loads or thermal challenges to fire barriers (e.g., no high-density flammable liquids or other combustible materials will be stored). Therefore, the NRC staff concludes that the description and design information provided for the fire safety systems are sufficient for the NRC staff to review the PCSA and design of the GROA.

The NRC staff determined during its review that a supplemental role of the fire alarm system will be to control fire and smoke dampers in the facility to maintain the integrity of fire barriers. This role would require the fire alarm system to close dampers and shutdown air handlers under certain fire conditions. The applicant's response to an RAI on how the interlock between smoke detectors and fire and smoke dampers of air handling units impacts the reliability of the cooling function (DOE, 2009dm) stated that the HVAC control function of the fire alarm system would conflict with the confinement requirements of the HVAC system. In its response to this RAI (DOE, 2009dm), the applicant stated that the PCSA identified that having an interlock between the smoke detector and air handling units results in an unacceptable reduction in reliability of the cooling function where the cooling function is designed as ITS (e.g., battery rooms in the CRCF and WHF). Therefore, air handler automatic shutdown on duct smoke detection is not warranted, and uninterrupted HVAC confinement would take priority in all cases. Duct smoke detection will remain and will provide notification functions only. The NRC staff finds that the applicant's deviation from NFPA 90A (National Fire Protection Association, 2002aa) by not requiring automatic shutdown of an air handler on duct smoke detection of the ITS HVAC systems is acceptable because this deviation is supported by the PCSA results, to ensure that confinement takes precedence and that this variance from NFPA 90A will provide operators greater control over the HVAC confinement features of the facilities, while still providing code-required alarm and manual control functions. The applicant stated that the SAR will be updated to reflect this hierarchy in HVAC control (DOE, 2009dm). Non-ITS HVAC systems will have fire alarm and HVAC controls in accordance with NFPA 90A.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.2.7, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for fire safety systems meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i) and 10 CFR 63.112(a) because the applicant provided adequate description and design information for the fire safety systems sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed in the NRC staff evaluation, the applicant provided adequate information regarding the fire safety systems that included information on the (i) site water supply and distribution systems, (ii) active and passive fire protection systems and barriers for surface facilities and the subsurface, (iii) codes and standards applicable to the design; and (iv) identification of components and systems of the fire safety systems that are important to safety.

2.1.1.2.3.2.8 Piping and Instrumentation Diagrams

The applicant described and discussed Piping and Instrumentation Diagrams (P&IDs) of surface facility process subsystems in SAR Sections 1.2.1 through 1.2.8. These sections provided the description and the design details of 13 process subsystem P&IDs.

The NRC staff's review of the design description of the P&ID of a process subsystem focused on understanding the function, operational sequence, logic of the layout of the subsystem components, and safety significance of ITS components. The design description review also included assessing the applicability of codes and standards the applicant proposed in the subsystem design. A representative sample of P&ID component descriptions was reviewed, comprising design codes and standards, equipment layout and arrangement, process flow, piping connections, potential interactions among support systems, and pressure-relief systems. The NRC staff reviewed subsystem design descriptions and subsystem requirements to ensure that all the major components shown on P&IDs are in general agreement with the operations information for the subsystem.

***Cask Cavity Gas Sampling Subsystem***

The cask cavity gas sampling subsystem will be used in the IHF, CRCF, RF, and WHF, and this non-ITS subsystem will be similar in all facilities. Only the IHF cask cavity gas-sampling subsystem is reviewed in this section, as this subsystem is expected to be similar in the CRCF, WHF, and RF. The applicant stated in SAR Section 1.2.3.3.1 that the subsystem will sample gaseous contents of a loaded transportation cask before it is opened. The gas sample will be analyzed to detect the presence of gaseous fission products, such as xenon and krypton. The P&ID for the cask cavity gas sampling subsystem was shown in SAR Figure 1.2.3-40 and described in SAR Section 1.2.3.3.1, which illustrated the flow sequence and operational aspects of the subsystem. The subsystem mainly comprises primary and secondary piping, temperature and pressure indicators, vacuum pumps, gas-sampling portals, particulate samplers, valves, and sample acquisition and analysis ports. A portable sample vacuum flask will collect gas samples for analysis of the gaseous fission products. The applicant also provided a simplified table of valve positions (SAR Figure 1.2.3-40) showing the valve layout of the cask cavity gas sampling subsystem and orientation for different operation modes. In addition, in SAR Section 1.2.4.3.1.3, the applicant described the design methods and applicable codes and standards. The applicant stated that it will use methods and practices in ANSI/ANS–57.7–1988 (American Nuclear Society, 1988aa) and ANSI/ANS–57.9–1992 (American Nuclear Society, 1992aa), as appropriate.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's descriptions and design information for the P&ID for the IHF cask cavity gas-sampling subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the subsystem description, subsystem functions, location, functional arrangement of major components, and operational processes. The NRC finds that the descriptions and design information for the P&ID for the IHF cask cavity gas-sampling subsystem are adequate because the applicant (i) explained that the cask cavity gas-sampling subsystem will sample gaseous contents of a loaded transportation cask before it is opened to detect the presence of gaseous fission products, such as xenon and krypton; (ii) described and illustrated the flow sequence and operational aspects of the subsystem, including the primary and secondary piping, temperature and pressure indicators, vacuum pumps, gas-sampling portals, particulate samplers, valves, and sample acquisition and analysis ports; (iii) provided a simplified table of valve positions for different operation modes; and (iv) proposed codes and standards for design of the non-ITS cask cavity gas sampling subsystem that are consistent with the standard engineering practices for systems of similar functions.

On the basis of the above evaluation, the NRC staff also finds that the applicant's descriptions and design information for the P&ID of the cask cavity gas-sampling subsystem for the IHF are sufficient to permit an evaluation of the PCSA and design of this subsystem. Because the cask cavity gas-sampling subsystem is similar in the IHF, CRCF, WHF, and RF, the NRC staff's findings for the cask cavity gas-sampling subsystem in IHF are applicable to the subsystem in the CRCF, WHF, and RF.

***Liquid LLW Sampling and Sump Collection Subsystem***

The non-ITS liquid LLW sampling and sump collection subsystem (SAR Section 1.2.3) will be similar in the IHF, CRCF, and RF. The NRC staff reviewed only the liquid LLW sampling and sump-collection subsystem in the IHF, as the subsystem is expected to be similar in the CRCF and RF. This subsystem contains floor drains designed to collect small amounts of potentially contaminated water from IHF operations. The P&ID in SAR Figure 1.2.3-41 provided graphical representation of the mechanical flow of the IHF liquid LLW sampling and sump-collection subsystem. The applicant classified this subsystem as non-ITS. The subsystem contains primary piping that transfers waste water effluents to the liquid LLW sampling tank. The effluents will be pumped from the LLW sampling tank through a system of pipes, valves, and pumps to trucks that transfer the effluents to the LLW collection tank. Sample lines on secondary piping has an access port to collect samples for analysis. This subsystem is also designed to collect water from the fire-suppression system. The applicant provided the design basis requirements for this subsystem in SAR Section 1.2.4.3.2. In addition, the applicant proposed to follow Regulatory Guide 1.143, Table 1, excluding footnotes intended for a liquid HLW subsystem, which are not applicable for this non-ITS liquid LLW subsystem (NRC, 2001ab).

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's description and design information for the P&ID for the non-ITS liquid LLW subsystem using guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the liquid LLW sampling and sump-collection subsystem layout, functional arrangement of the major components, and design information, including drawings and P&ID. The NRC staff finds that the applicant's description and design information for the P&ID for the liquid LLW sampling and sump collection subsystem is adequate because the applicant (i) explained that floor drains will be designed to collect small amounts of potentially contaminated water from IHF operations; (ii) described and illustrated representation of the mechanical flow **of** the IHF liquid LLW sampling and sump-collection subsystem and the primary piping that will transfer waste water effluents to the liquid LLW sampling tank; (iii) explained that the effluents will be pumped from the LLW sampling tank through a system of pipes, valves, and pumps to trucks that transfer the effluents to the LLW collection tank, and sample lines on secondary piping will contain an access port to collect samples for analysis; (iv) explained that the subsystem will also be designed to collect water from the fire-suppression system; and (v) will design the non-ITS liquid LLW subsystem using the methods and practices prescribed in Regulatory Guide 1.143, Table 1. The NRC staff evaluated the applicant's proposed exclusion of footnotes from Regulatory Guide 1.143 Table 1 (NRC 2001ab) and concludes that the exclusions are acceptable because the excluded provisions apply to high-level radioactive waste systems and are not applicable to this non-ITS liquid LLW subsystem.

Based on the above evaluation, the NRC staff finds the applicant's description and design information for the P&ID for the non-ITS liquid LLW sampling and sump-collection subsystem are sufficient to permit an evaluation of the PCSA and design. Because the liquid LLW sampling and sump-collection subsystem is similar in IHF, CRCF, and RF, the NRC staff's findings for the liquid LLW sampling and sump-collection subsystem in IHF are applicable to the subsystem in CRCF and RF.

***Waste Package Inerting Subsystem of Waste Package Closure System***

The waste package inerting subsystem is part of the waste package closure system, and the system is similar in the CRCF and IHF buildings. The applicant classified this subsystem as non-ITS. Waste package closure involve engaging and welding the inner lid spread ring, inerting the waste package with helium, setting and welding the outer lid to the outer corrosion barrier, performing leak testing on the inner vessel closure, performing nondestructive examination of welds, and conducting post-weld stress mitigation on the outer lid closure weld. The P&ID of only the waste package inerting subsystem of the waste package closure system is reviewed here. SAR Section 1.2.4.2.3.1.3 and SAR Figure 1.2.4-76 presented P&ID of the waste package inerting subsystem of the waste package closure system for the CRCF. The inerting subsystem vacuum will dry the waste package and then pressurize the container with helium. The subsystem contains sensors and instruments, which monitor and measure the waste package inerting operations. SAR Figure 1.2.4-76 showed instruments and controls (e.g., pumps, dial indicators, helium leak detectors, and position sensors) related to the waste package inerting process. According to the applicant, the inerting process will follow the procedures in accordance with the applicable sections of NUREG–1536 (NRC, 1997ae) except the operations will be done remotely. After the waste package inner vessel is backfilled with helium, both the spread ring welds and purge port plug will be seal tested remotely in accordance with 2001 ASME Boiler and Pressure Vessel Code (ASME 2001, Section V, Article 10, Appendix IX) to verify that no leakage can be detected that exceeds the rate of

$10^{-6}$ std $cm^3/sec$ [$6.1 \times 10^{-8}$ $in^3/sec$]. The applicant proposed the following codes and standards for welding of the waste package closure subsystem: (i) welds, weld repairs, and inspections, in accordance with ASME Boiler and Pressure Vessel Code, Section II, Part C; Section III, Division I, Subsection NC; Section IX; and Section V (American Society of Mechanical Engineers, 2001aa); and (ii) ANSI/AWS A5.32/A5.32M–97 (American Welding Society, 1997aa). For design of the waste package closure subsystem the applicant proposed using (i) ASME B30.20–2003 (American Society of Mechanical Engineers, 2003aa); (ii) NFPA 801 (National Fire Protection Association, 2008aa); and (iii) ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). For inerting of the waste package the applicant proposed using applicable sections of NUREG–1536 (NRC, 1997ae).

In its response to the NRC staff's RAI on the details of the closure system, the applicant stated that the waste package and its closure system were selected for prototyping programs prior to their use in the repository and performed a prototype test of the waste package inerting subsystem in the laboratory. The applicant stated that its prototype testing of the waste package inerting subsystem conducted in the laboratory demonstrated the overall feasibility for the following operations: (i) seal welding of the inner lid spread ring, seal welding of the purge port cap, and narrow-groove welding of the outer lid; (ii) nondestructive examination of the welds; (iii) evacuation and helium backfill of the inner vessel; (iv) leak detection of the inner-lid seals; and (v) stress mitigation of the outer-lid groove weld (DOE, 2009dr). The applicant stated that the test encountered problems with seating the purge port plug during the demonstration of the inerting subsystem. Also, the temporary heating system (electrical) located inside the inner vessel of the waste package mockup did not function properly, resulting in insufficient simulation of the expected temperature range. The applicant recorded recommendations and lessons learned from the laboratory prototype test and planned a retest as part of the prototype testing in connection with the final design activities. In addition, the applicant stated that to clarify the performance capability of the waste package closure system, as part of the prototype testing, it will perform capability demonstrations (full system qualification testing) to ensure conformance with waste package safety criteria (DOE, 2009dr).

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.2.4.2.3.1.3 and SAR Figure 1.2.4-76 and the applicant's response to the NRC staff's request for additional information (RAI) (DOE, 2009dr) for the waste package inerting subsystem's description and P&ID, using the guidance in YMRP Section 2.1.1.2. The NRC staff also reviewed the descriptions of the location and functional arrangement of the waste package inerting subsystem. The NRC staff finds that the applicant's description and design information for the P&ID of this subsystem are adequate because the applicant (i) explained that the inerting subsystem vacuum will dry the waste package and then pressurize the container with helium, and the subsystem contains sensors and instruments, which monitor and measure the waste package inerting operations; (ii) illustrated instruments and controls (e.g., pumps, dial indicators, helium leak detectors, and position sensors) related to the waste package inerting process; (iii) described how the inerting process will follow the procedures in accordance with the applicable sections of NUREG–1536 (NRC, 2010ah); (iv) explained that after the waste package inner vessel is backfilled with helium, both the spread ring welds and purge port plug will be seal tested remotely; and (v) provided codes and standards for the subsystem that are consistent with the standard engineering practices for equipment performing similar functions. The applicant stated that it will follow the guidance in NUREG–1536, except it will do the welding remotely to mitigate exposure to workers. The staff finds that using NUREG–1536 but doing the welding remotely for the inerting subsystem, an operation that will be done numerous times at the GROA, is acceptable because remote welding will minimize exposure to workers. Based on the above evaluation, the applicant's description and design information for the waste package inerting subsystem's P&ID are sufficient to permit an evaluation of the PCSA and design of the waste package inerting subsystem.

The NRC staff notes that the problem encountered in the laboratory prototype testing is a test set-up detail problem and has no bearing on the previous design description and P&ID information. The applicant has recorded recommendations and lessons learned from the laboratory prototype test, and the applicant plans to conduct a prototype testing demonstration of the waste package closure system as part of the final design activities for this system.

***Wet Handling Facility Pool Water Treatment and Cooling System***

The applicant presented the P&ID for the pool water treatment and cooling system in SAR Figures 1.2.5-58 to 1.2.5-63. The pool water treatment and cooling system and its subsystems were classified as non-ITS. In SAR Section 1.2.5.3.2, the applicant provided information on functions, location, and components for the pool water treatment and cooling system that consist of (i) the pool water treatment subsystem (Trains A, B, and C), (ii) the pool water cooling subsystem, (iii) the pool water makeup subsystem, (iv) the boric acid makeup subsystem, and (v) the leak detection subsystem.

Using filters, the pool water treatment subsystem removes crud and particulates, radionuclides, and other ionic species; maintains optical clarity of pool water to allow identification of spent nuclear fuel (SNF) assembly identifiers; and facilitates SNF handling. The pool water cooling subsystem removes decay heat from the pool water caused by the heat load of fuel in the pool. The pool water makeup subsystem controls the level of deionized water in the pool. The boric acid makeup subsystem maintains the required concentration of boron in the WHF pool to prevent criticality. The leak detection subsystem is designed to monitor and detect leaks between the pool liner and the concrete wall of the pool. In addition, it will include cameras and sumps to monitor any leak.

The applicant stated that the design will conform to the following codes, standards, and general guidance commonly used in nuclear industry: (i) ANSI/ANS–57.7–1988 (American Nuclear Society, 1988aa) and (ii) Regulatory Guide 1.143 (NRC, 2001ab).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID for the pool water treatment and cooling system using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the pool water treatment and cooling subsystem and P&ID. The NRC staff finds the applicant's description and design information for the pool water treatment and cooling subsystem are adequate because the applicant (i) provided information on the system functions, location and functional arrangements, components, maintenance considerations, operational processes, and design codes and standards for each of the five subsystems comprising the pool water treatment and cooling system (i.e., the pool water treatment subsystem, the pool water cooling subsystem, the pool water makeup subsystem, the boric acid makeup subsystem, and the leak detection subsystem); (ii) described the use of filters in the pool water treatment subsystem to remove crud and particulates, radionuclides, and other ionic species to maintain optical clarity of the pool water; (iii) described the boric acid makeup subsystem for maintaining the required concentration of boron in the WHF pool to prevent criticality; (iv) described the leak detection subsystem design for monitoring and detecting leaks between the pool liner and the concrete wall of the pool, including cameras and sumps to monitor any leak; and (v) stated that the design will conform to codes, standards, and general guidance commonly used in the nuclear industry (i.e., ANSI/ANS–57.7–1988 and Regulatory Guide 1.143). Therefore, the NRC staff finds that the applicant's description and design information for the pool water treatment and cooling system P&ID are sufficient to permit an evaluation of the PCSA and design.

***Cask Decontamination Subsystem***

The applicant presented the P&ID for the cask decontamination subsystem in SAR Figure 1.2.5-67. The cask decontamination subsystem uses deionized water to rinse casks when removed from the WHF pool. The cask decontamination subsystem was classified as non-ITS, but the decontamination pit and seismic restraints were classified as ITS. The decontamination pit will include seismic restraints to ensure that the transportation cask or shielded transfer cask inside the decontamination pit is restrained to prevent tip over. The ITS SSCs in the cask decontamination subsystem (decontamination pit and seismic restraints) will be designed in accordance with ANSI/AISC N690–1994, Sections Q1.2 (design methodologies), Section Q1.4 (selection of appropriate material), and Table Q1.5.7.1 (meeting load combinations) (American Institute of Steel Construction, 1994aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID in SAR Figure 1.2.5-67 for the cask decontamination subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff finds the applicant's description and design information for the cask decontamination subsystem P&ID adequate because the applicant (i) provided information on the subsystem functions (i.e., rinse casks when removed from the WHF pool); (ii) described the operational processes and the components identified as ITS (e.g., decontamination pit will include seismic restraints to ensure that the transportation cask or shielded transfer cask inside the decontamination pit is restrained to prevent tip over); and (iii) provided the design codes and standards used for design of the ITS components, which are consistent with standard engineering practices for systems performing similar functions. Therefore, on the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the cask decontamination subsystem P&ID are sufficient to permit an evaluation of PCSA and design.

***Cask Cooling and Filling Subsystem***

The applicant described the cask cooling and filling subsystem in the cask preparation area of the WHF in SAR Section 1.2.5.3.4. SAR Figures 1.2.5-69 through 1.2.5-72 presented P&IDs of this subsystem. The function of this subsystem is to cool the inside of dual-purpose canisters (DPCs) and casks and to fill the casks and transportation, aging, and disposal (TAD) canisters with borated water prior to placement in the pool. The primary function of the WHF TAD canister closure station cask cooling subsystem is to fill TAD canisters and annulus spaces with pool water and cool the inside of the TAD canister prior to opening or placement in the pool. An alternate cooling method is to cool the casks with a forced helium dehydrator, which will be located in the canister transfer machine (CTM) maintenance room of the WHF. The forced helium dehydrator consists of a refrigeration unit, condensing module, demoisturizer module, helium circulation module, and preheater module (not used for cooling). The cask cooling and filling subsystem was classified as a non-ITS system but will have both ITS and non-ITS components. The pressure relief valves, which will be used to implement the overpressure protection function, were classified as ITS components. The applicant stated that the design of the SSCs in the cask cooling and filling system uses the methods and practices in ASME B31.3–2004 and 2004 ASME Boiler and Pressure Vessel Code, Section VIII, Division I (American Society of Mechanical Engineers, 2004ab,aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID for the cask cooling and filling subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the cask cooling and filling subsystem layout and the P&ID. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes to evaluate potential interactions with other subsystems. Though the cask cooling and filling subsystem P&ID (SAR Figure 1.2.5-69) identified ITS function for cask overpressure protection, the title block of the piping and instrumentation diagram figures did not identify the subsystem as ITS. Additionally, the applicant stated in SAR Section 1.2.5.3.4 that a subsystem will be designated as non-ITS, while portions of the structure or components in the subsystem are ITS. The NRC staff determined that this information is inconsistent with the practice of labeling a subsystem as ITS, even if only one component of the subsystem is ITS, as discussed in BSC (2008bx). The applicant stated in its response to an RAI (DOE, 2009du) that it will address this labeling inconsistency by revising the text in SAR Section 1.2.5.3.4 as follows:

"The cask cooling subsystem has an ITS classification. However, as shown in SAR Figures 1.2.5-69 through 1.2.5-72, it is only the ITS overpressure protection components of the cask cooling subsystem that are relied upon to satisfy the overpressurization prevention safety function. All other components of the cask cooling subsystem are classified as non-ITS." (DOE, 2009du).

The applicant further identified similar labeling inconsistencies (DOE, 2009ec) in SAR Chapter 1 and provided revised ITS designations for components and subsystems (see table in response for detailed list, DOE, 2009ec). The applicant stated that it will update the SAR to ensure a consistent statement of the system and subsystem safety classification among the SAR text, tables, figures, and SAR Table 1.9-1. The NRC staff finds that the applicant's statement in response to RAIs (DOE, 2009ec,du) proposing revision to the relevant sections of the SAR is acceptable because it would result in ITS identification in the SAR text, tables, and figures being consistent.

The NRC staff finds that the applicant's description and design information for the cask cooling and filling subsystem P&ID is adequate because the applicant (i) described that the function of the subsystem is to cool the inside of DPCs and casks, and to fill the casks and TAD canisters with borated water prior to placement in the pool; (ii) illustrated the components and functions of the subsystem via P&IDs; (iii) described an alternate cooling method that would cool the casks with a forced helium dehydrator; (iv) described the ITS pressure relief valves, which will be used to implement the overpressure protection function; (v) provided codes and standards for the subsystem that are consistent with the standard engineering practices with similar cask handling operations; and (vi) addressed the labeling inconsistency regarding ITS classification.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the cask cooling and filling and the P&ID are sufficient to permit an evaluation of the PCSA and design.

### *Wet Handling Facility TAD Canister and Shielded Transfer Cask Drying and TAD Canister Inerting Subsystem*

In SAR Section 1.2.5.3.5, the applicant described the subsystem and P&ID of the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem. The applicant also presented the P&ID for this drying and inerting subsystem in SAR Figure 1.2.5-73. The applicant classified this subsystem as non-ITS. The system consists of a forced helium dehydrator or vacuum dryer to drain and dry the TAD or shielded transfer cask when it is taken out of the WHF pool. The forced helium dehydrator will also be used to inert the TAD canister containing spent fuel after loading it in the WHF pool. If a vacuum system is used, the system will consist of a vacuum pump, filter, and condenser, which will dry the TAD/shielded transfer cask. In SAR Section 1.2.5.3.5.2, the applicant described the operational process for the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem. In SAR Figure 1.2.5-73, the applicant presented a simplified P&ID that showed components and their arrangement in this non-ITS subsystem.

The codes, standards, and regulatory guidance that the applicant proposed to use are (i) ANSI/ANS–57.7–1988 (American Nuclear Society, 1988aa), (ii) ANSI/ANS–57.9–1992 (American Nuclear Society, 1992aa), (iii) ANSI N14.5–1997 (American National Standards Institute,1998aa), and (iv) NUREG–1536 (NRC 1997ae).

### NRC Staff’s Evaluation

The NRC staff evaluated the description and design information for the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem and associated piping and instrumentation diagrams information using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the layout descriptions of the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem and the P&ID. The NRC staff finds that the applicant’s description and design information for the P&ID for the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem are adequate because the applicant (i) described the system and its principal components (e.g., a forced helium dehydrator or vacuum dryer to drain and dry the TAD or shielded transfer cask when it is taken out of the WHF pool; the forced helium dehydrator will also be used to inert the TAD canister containing spent fuel after loading it in the WHF pool; and the vacuum system will consist of a vacuum pump, filter, and condenser, which will dry the TAD/shielded transfer cask); (ii) described the operational process; (iii) provided a P&ID showing the components and their arrangement in this non-ITS subsystem; and (iv) provided the codes and standards for the subsystem that are consistent with the standard engineering practices with similar handling operations.

On the basis of the above evaluation, the NRC staff finds that the applicant’s description and design information for the P&ID for the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem are sufficient to permit an evaluation of the PCSA and design.

### *Wet Handling Facility Water Collection Subsystem*

In SAR Section 1.2.5.3.6, the applicant presented the description, function, and design information of the water collection subsystem and P&ID for the WHF. The subsystem consists of floor drains, collection tanks, and pumps to collect (i) small amounts of water that will be discharged or leaked from process SSCs, (ii) decontamination and wash water, and (iii) fire-suppression water. There are two water collecting tanks—C2, which collects normally noncontaminated water, and C3, which collects water that will have the potential to be contaminated. The contaminated water will be transferred to the low-level radioactive waste facility (LLWF) for treatment. SAR Figures 1.2.5.-74 and 1.2.5-75 showed P&IDs for the C2 and C3 tanks, respectively. The subsystem was classified as non-ITS. In addition, the applicant proposed to follow Regulatory Guide 1.143, Table 1, excluding footnotes NRC (2001ab).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID of the Wet Handling Facility water collection subsystem using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the water collection subsystems layout, design information, and the P&ID. The NRC staff finds the description and design information for the WHF water collection subsystem P&ID are adequate because the applicant (i) described the components of the water collection subsystem (e.g., floor drains, collection tanks, and pumps); (ii) described the origination of the water collection sources (e.g., small amounts of water that will be discharged or leaked from process SSCs, decontamination and wash water, and fire-suppression water); (iii) described the two water-collecting tanks (i.e., one tank to collect normally noncontaminated water and a second tank to collect water that will have the potential to be contaminated); (iv) explained that contaminated water will be transferred offsite to an LLWF; (v) included both collection tanks in the P&IDs; and (vi) stated the design would be consistent with Regulatory Guide 1.143, Table 1, excluding footnotes. The NRC staff finds that the aforementioned exclusion of footnotes is acceptable because the provisions to be excluded apply to liquid high level waste and do not apply to liquid LLW, as is the case in this facility.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the WHF water collection subsystem P&ID are sufficient to permit an evaluation of the PCSA and design.

**Emergency Diesel Generator Facility**

The Emergency Diesel Generator Facility (EDGF) is designed to house the two independent 13.8-kV ITS diesel generators (Trains A and B) and supporting mechanical systems in separate areas of the EDGF. In SAR Section 1.2.8.2, the applicant provided P&ID information for the following ITS subsystems of each train: (i) ITS diesel generator fuel oil subsystem, (ii) ITS diesel generator air start subsystem, (iii) ITS diesel generator jacket water cooling subsystem, (iv) ITS diesel generator lubrication oil subsystem, and (v) ITS diesel generator air intake and exhaust subsystem. The NRC staff focused its review on Train A P&ID, and because Trains A and B will have similar designs; the NRC staff evaluation findings for Train A are applicable to Train B also.

***ITS Diesel Generator Fuel Oil Subsystem***

In SAR Section 1.2.8.2.1 and Figure 1.2.8-18, the applicant provided a description of the design and P&ID information for the ITS diesel generator fuel oil subsystem. The ITS diesel generator fuel oil subsystem consists of an underground diesel fuel oil storage tank, from which fuel will be drawn through duplex basket filters by diesel fuel oil transfer pumps to the diesel fuel oil day-tank. A diesel-engine-driven fuel oil pump will draw fuel from the day-tank through another set of duplex basket strainers to the ITS diesel generator. There is one underground diesel fuel oil storage tank per ITS diesel generator, providing diesel fuel to the dedicated day-tank that supports each ITS diesel generator. Two diesel fuel oil transfer pumps transfer fuel oil from the diesel fuel oil storage tank to the associated diesel fuel oil day-tank. The design methodologies proposed for the design of ITS SSCs in the ITS diesel generator fuel oil system are in accordance with codes and standards provided in SAR Section 1.2.8.2.1.8, such as ANSI/ANS–59.51–1997 (ANS, 1997ae); ASME 2004 Section VIII, Boiler and Pressure Vessel Code (ASME, 2004aa); ASME B31.3–2004, Process Piping (ASME, 2004ab); and Regulatory Guide 1.137, Fuel-oil System for Standby Diesel Generators (NRC, 1979ab). SAR Figure 1.2.8-17 showed the interface between the ITS diesel generator Train A and the mechanical system that will support it. SAR Figure 1.2.8-18 presented the ITS diesel generator fuel oil system P&ID. SAR Figure 1.2.8-19 presented the ITS diesel generator fuel oil transfer pump logic diagram.

**NRC Staff's Evaluation**

The NRC staff evaluated information in SAR Section 1.2.8.2.1 and SAR Figure 1.2.8-18 on the ITS diesel generator fuel oil subsystem's purpose, function, operation, and design descriptions, and P&ID, using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the ITS diesel generator fuel oil system and the P&ID. The NRC staff finds the description and design information for the ITS diesel generator fuel oil system P&ID adequate because the applicant (i) described the diesel generator fuel oil subsystem, including the underground diesel fuel oil storage tank, filters, and fuel oil pumps; (ii) illustrated the interface between ITS diesel generator Train A and the mechanical system that will support it; (iii) provided the ITS diesel generator fuel oil system P&ID and the ITS diesel generator fuel oil transfer pump logic diagram; and (iv) provided the codes, standards, and guidance, including Regulatory Guide 1.137, to be used in the design that are consistent with the standard engineering practices for emergency diesel generator systems in nuclear facilities.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the ITS diesel generator fuel oil subsystem P&ID are sufficient to permit an evaluation of the PCSA and design.

***ITS Diesel Generator Air Start Subsystem Train A***

In SAR Section 1.2.8.2.2 and Figure 1.2.8-20, the applicant described the diesel generator air start system and its P&ID. This system provides air to the ITS diesel generator during startup. The air start system consists of one air compressor, after cooler, air dryer, air receiver, compressor air intake filter, piping, valves, associated instrumentation, and an air distribution system on the diesel engine. The air start system components that are downstream of the ITS isolation gate valve were classified as ITS, and components that are upstream of the ITS isolation gate valve (the compressor, after cooler, and air dryer) were classified as non-ITS (SAR Section 1.2.8.2.2). The air receiver will be maintained at operating pressure. The applicant stated that the system would alarm when pressure drops below its set point, and the compressor automatically starts. SAR Figure 1.2.8-20 showed the ITS diesel generator air start system Train A. SAR Figure 1.2.8-21 showed the logic diagram for the ITS diesel generator air compressor.

The ITS diesel generator air start system will be designed in accordance with (i) 2004 ASME Boiler and Pressure Vessel Code, Section VIII (American Society of Mechanical Engineers, 2004aa); (ii) ASME B31.3–2004, Process Piping (American Society of Mechanical Engineers, 2004ab); and (iii) CGA G–7.1–2004 (Compressed Gas Association, 2004aa). The cited codes are routinely used in the design of emergency diesel generators in nuclear facilities.

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID for the diesel generator air start subsystem Train A using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the diesel generator air start subsystem Train A layout and the P&ID. The NRC staff finds that the applicant's description and design information for the diesel generator air start subsystem Train A P&ID are adequate because the applicant (i) described the major components of the subsystem (e.g., air compressor, after cooler, air dryer, air receiver, compressor air intake filter, and an air distribution system on the diesel engine); (ii) explained that air start system components that are downstream of the ITS isolation gate valve were classified as ITS and components that are upstream of the ITS isolation gate valve (the compressor, after cooler, and air dryer) were classified as non-ITS; (iii) described that the air receiver system would be alarmed to automatically start the compressor when pressure drops below its set point; (iv) illustrated the ITS diesel generator air start system Train A, including a logic diagram for the ITS diesel generator air compressor; and (v) provided the applicable codes and standards that are routinely used in the design of emergency diesel generators in nuclear facilities.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the diesel generator air start subsystem Train A P&ID are sufficient to permit an evaluation of the PCSA and design of the diesel generator air start subsystem.

***Diesel Generator Jacket Water Cooling Subsystem***

In SAR Section 1.2.8.2.3 and Figure 1.2.8-22, the applicant described the ITS diesel generator jacket water cooling system and its P&ID. The jacket water cooling system provides sufficient heat sink to permit the diesel engine to start and operate without the need for external cooling water. Major components include after coolers (engine-mounted combustion air coolers), a lube oil cooler, a jacket water air cooler, jacket water pumps, a jacket water expansion tank, an electric immersion heater, and a keep-warm circulating pump (SAR Figures 1.2.8-17 and 1.2.8-22). The system will be designed such that the cooling water chemistry criteria will be compatible with the materials of the system's various components. The ITS diesel generator jacket water cooling system Train A P&ID was presented in SAR Figure 1.2.8-22. The codes and standards to be used in the design were listed in SAR Section 1.2.8.2.3.8: (i) 2004 ASME Boiler and Pressure Vessel Code, Section VIII (American Society of Mechanical Engineers, 2004aa); (ii) ASME B31.3–2004 (American Society of Mechanical Engineers, 2004ab); (iii) Pump Standards (Hydraulic Institute, 2005aa); and (iv) Standards of the Tubular Exchanger Manufacturers Association (Tubular Exchanger Manufacturers Association, 2007aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID for the diesel generator jacket water cooling system using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed the descriptions of the diesel generator jacket water cooling system and the P&ID. The NRC staff finds that the applicant's description and design information for the diesel generator jacket water cooling system P&ID are adequate because the applicant (i) described the major components of the system (e.g., engine-mounted combustion air coolers, a lube oil cooler, a jacket water air cooler, jacket water pumps, a jacket water expansion tank, an electric immersion heater, and a keep-warm circulating pump); (ii) described the system design for the cooling water chemistry criteria of compatibility with the materials of the system's various components; (iii) included the ITS diesel generator jacket water cooling system Train A in the P&ID; (iv) explained that, as depicted in SAR Figure 1.2.8-22, the jacket water air cooler and associated piping that deliver cooling water to the lubricating oil heat exchanger are ITS, which clarifies the interface and boundary between ITS and non-ITS portions of the diesel generator jacket water cooling system (DOE, 2009du); and (v) provided codes and standards that are consistent with standard engineering practice for equipment of similar functions in nuclear facilities.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the ITS diesel generator jacket water cooling subsystem P&ID are sufficient to permit an evaluation of the PCSA and design.

***Diesel Generator Lubricating Oil Subsystem Train A***

In SAR Section 1.2.8.2.4 and in SAR Figure 1.2.8-23, the applicant described the ITS diesel generator lubricating oil system and its P&ID for Train A.  Major components of the system include one engine-driven pump; an engine-mounted lube oil collection sump; a full-flow filter; a full-flow strainer; a lube oil cooler; an electric keep-warm heater; an electric motor-driven keep-warm circulating pump; an electric motor-driven prelubricating pump; and associated valves, piping, and instrumentation.  SAR Figure 1.2.8-17 showed the engine-mounted lubricating oil pump and the lubricating oil sump connections to the diesel generator engine. The design bases, materials, and design methodologies to be incorporated and applied are based on ANSI/ANS–59.52–1998 (American Nuclear Society, 1998aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the P&ID for the diesel generator lubricating oil system using the guidance in YMRP Section 2.1.1.2.  The NRC staff reviewed the descriptions of the diesel generator lubricating oil system and P&ID.  The NRC staff finds that the applicant's description and design information for the diesel generator lubricating oil system P&ID are adequate because the applicant (i) described the major components of the lubricating oil system (e.g., one engine-driven pump, an engine-mounted lube oil collection sump, a full-flow filter, a full-flow strainer, a lube oil cooler, an electric keep-warm heater, an electric motor-driven keep-warm circulating pump, and an electric motor-driven prelubricating pump); (ii) illustrated the engine-mounted lubricating oil pump and the lubricating oil sump connections to the diesel generator engine; and (iii) provided the applicable codes and standards that are consistent with standard engineering practice for equipment of similar functions in nuclear facilities.

The applicant stated that it will follow ANSI/ANS–59.52–1998, Section 4 (American Nuclear Society, 1998aa), which provides for the use of a gravity drain to collect oil leaks from diesel generators.  The standard states that a gravity drain system is acceptable; however, consideration shall be given to potential system leakage and its consequences, such as potential hazards and collection and ultimate disposal of leaked oil.  Additionally, the scope of ANSI/ANS–59.52–1998 (American Nuclear Society, 1998aa) excluded engine-mounted components, except to define interface requirements.  The NRC staff requested that the applicant clarify its description for the use of this guidance in the design (DOE, 2009dt).  Based on its review of the applicant's response, the NRC staff finds the applicant's design description and design information for the ITS diesel generator lubricating oil system P&ID are adequate because the applicant (i) clarified that the detailed design process for the diesel generators includes consideration of potential system leakage when assessing and implementing design features that provide leak prevention, detection, containment, and isolation of the lubricating oil system [these potential design features include double-lined sumps, appropriately sized catch basins to collect lubricating oil leaks, catch basins with oil-detection capabilities, automatic oil level regulation (make-up) systems, and remote lubricating oil pump shutdown capabilities]; (ii) clarified that potential design features include standard diesel generator sensory components that monitor lubricating oil pressure and temperature; (iii) stated that fire prevention and fire risk control and storage (including containment, drainage, and spill control) will be implemented, in accordance with NFPA 30, Flammable and Combustible Liquids Code (NFPA, 2006ac); (iv) stated that isolation of the lubricating oil system will be in accordance with Chapter 7 of NFPA 37, Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines (NFPA, 2006ab); (v) explained that a remote shutdown capability for the lubricating oil pump will be incorporated into the diesel generator lubricating oil system design to prevent any significant oil leakage by precluding lubricating oil flow from the lubricating oil storage tank or the diesel engine sump; (vi) stated that a dedicated inspection, testing, and reliability-centered maintenance program will be employed for the important to safety diesel generators (including their supporting subsystems) to monitor and consequently prevent or remedy potential issues that could adversely affect their safe and reliable operation; (vi) stated that the inspection, testing, and maintenance of the ITS diesel generators will be in accordance with manufacturer's recommendations and IEEE Standard 387-1995 (IEEE, 1996aa), Standard Criteria for Diesel-Generator Units Applied as Standby Power Generating Stations; and (vii) explained that although there are no event sequences, including a seismic event, that require the operation of the ITS diesel generators during or after a seismic event, the ITS diesel generators and their support systems are designed to perform their intended safety function for the ground motions identified in the International Building Code 2000 (International Code Council, 2003aa) for the location of the repository (DOE, 2009ec). The NRC staff also notes that a diesel generator lubricating oil system is common in emergency diesel generators in nuclear facilities.

On the basis of the above evaluation, the NRC staff finds that the applicant’s description and design information for the ITS diesel generator lubricating oil system P&ID for Train A are sufficient to permit an evaluation of the PCSA and design.

***ITS Air Intake and Exhaust Subsystem Train A***

In SAR Section 1.2.8.2.5, the applicant described that the ITS air intake and exhaust system Train A and its P&ID were presented in SAR Figure 1.2.8-24. The major components of the system are the air-intake filter, intake and exhaust silencers, and piping and expansion joints (features that supply air to the ITS diesel generator without an excessive pressure drop). The size, layout, and arrangement of the ITS air intake and exhaust system Train A will allow air to be routed through intake piping, an intake filter, an in-line silencer, and a turbocharger. The system design reduces the potential exhaust gas from entering through the air intake system. For this reason, the exhaust piping will be monitored for pressure and temperature. A high-temperature or back-pressure alarm will trip the diesel engine. In SAR Section 1.2.8.2.5.8, the applicant listed codes and standards for the design of the ITS air intake and exhaust system: ASME B31.3–2004, Process Piping (ASME 2004ab), and NUREG/CR–0660, Enhancement of On-site Emergency Diesel Generator Reliability (NRC, 1979ac).

**NRC Staff’s Evaluation**

The NRC staff evaluated the description and design information in the P&ID for the air intake and exhaust system using the guidance in YMRP Section 2.1.1.2. The NRC staff finds that the applicant’s description and design information for the air intake and exhaust system P&ID are adequate because the applicant (i) described the major components of the system (e.g., the air-intake filter, intake and exhaust silencers, and piping and expansion joints); (ii) described the size, layout, and arrangement of the ITS air intake and exhaust system for Train A; (iii) explained that the system will be designed to reduce the potential exhaust gas from entering through the air intake, and the exhaust piping will be monitored for pressure and temperature (e.g., a high-temperature or back-pressure alarm will trip the diesel engine); and (iv) provided applicable codes and standards that are consistent with standard engineering practice for equipment of similar functions in nuclear facilities.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the ITS air intake and exhaust subsystem P&ID for Train A are sufficient to permit an evaluation of the PCSA and design.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.2.8, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the P&IDs of various surface facilities' subsystems meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the P&IDs of various surface facilities' subsystems sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.2.9 Decontamination, Emergency, and Radiological Safety Systems

The applicant described and discussed the design of the decontamination, emergency, and radiological safety systems in the SAR.

In SAR Section 1.2.1.3, the applicant provided an overview of the decontamination systems. The applicant noted that each of the handling facilities will be capable of performing activities, including (i) decontaminating exterior surfaces of casks, waste packages, and canisters; (ii) decontaminating the interior surfaces of casks in a dry environment; and (iii) in the case of the WHF, decontaminating underwater using the cask decontamination subsystem. Other than minor decontamination in the CRCF and RF, if surface contamination levels exceed acceptable limits, canisters will be sent to the WHF for decontamination in the cask decontamination subsystem. In SAR Section 1.2.5.3.3, the applicant described the cask decontamination subsystem of the WHF. The cask decontamination system will rinse (i) unloaded transportation casks, (ii) unloaded shielded transfer casks, and (iii) the shielded transfer casks containing DPC or TAD canisters to prepare for export or removal from the WHF (SAR Section 1.2.5.3.3.1). Further decontamination, if necessary, will be performed in the decontamination pit. The only ITS components of the cask decontamination subsystem will be the decontamination pit and the seismic restraints discussed in SAR Section 1.2.5.3.3.1.3. SAR Figure 1.2.5-2 provided the WHF general arrangement ground floor plan of the decontamination pit. SAR Figure 1.2.5-66 provided a mechanical equipment envelope drawing of decontamination pit mechanical equipment, and SAR Figure 1.2.5-67 provided decontamination pit process and instrumentation diagram. The decontamination pit includes (i) stainless steel walls, (ii) retractile spray heads with high pressure nozzles, and (iii) a pump module. ITS decontamination pit and seismic restraints will be designed in accordance with the design methodologies, construction of material, and load combinations specified in ANSI/AISC N690–1994 (SAR Section 1.2.5.3.3.3).

In SAR Section 5.7, the applicant described the emergency plan that will be used to mitigate the consequences of a potential radiological accident. The description of the emergency plan identified the safety systems to be put in place. Specifically, these will include equipment and design features relied upon to mitigate emergency events; facilities to be available to support mitigation efforts; response equipment to be available; and provisions to periodically inventory, test, and maintain these systems and equipment. The description of the emergency plan provided in SAR Section 5.7 is evaluated in detail in SER Section 2.5.7. The applicant's emergency and radiological safety systems were described in SAR Section 5.11 as part of the operational radiation protection program description. The applicant stated that it will set aside an area for the operational radiation protection program to support monitoring of radiological work and facility conditions, access control, and the generation of radiation work controls and permits to provide for radiological safety. The process and area radiation monitoring equipment and instruments that will be part of the GROA were described in SAR Section 1.4.2.2. The radiation/radiological monitoring systems (RMS) were designated as non-ITS. The systems will be used to monitor the surface and subsurface areas and effluents from the GROA release points. Monitoring equipment will alert operators through a central monitoring station of any radiological release and potential Category 1 or Category 2 event sequences or conditions.

The applicant proposed to follow periodic testing and calibration of the RMS using the practices in ANSI/ANS–HPSSC–6.8.1–1981 (American Nuclear Society, 1981aa) and ANSI N42.18–2004 (American National Standards Institute, 2004aa). The applicant stated that it will also design and use area radiation monitors using the methods and practices of ANSI/ANS–HPSSC–6.8.1–1981, continuous air monitors using the methods and practices of ANSI N42.17B–1989 (American National Standards Institute, 1989aa), and airborne radioactivity effluent monitors using the methods and practices of ANSI N42.18–2004 and ANSI/HPS N13.1–1999 (American National Standards Institute/Health Physics Society, 1999aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the decontamination and emergency and radiological safety systems using the guidance in YMRP Section 2.1.1.2. Specifically, the NRC staff reviewed description, functions, location, and functional arrangement of major components of the decontamination systems, emergency and radiological safety systems, and the potential interactions of the decontamination systems with other subsystems. The NRC staff finds that the applicant's description and design information of the cask decontamination subsystem are adequate because (i) the description discussed the specific functions of the subsystem that will be performed in the WHF; (ii) the description discussed the ITS SSCs in the cask decontamination subsystem in the WHF (decontamination pit and seismic restraints); (iii) the described functions for the decontamination subsystem are consistent with the proposed decontamination operation and process flow in the WHF; (iv) the description provided components of the decontamination pit, including stainless steel walls, retractile spray heads, and pump modules; (v) the applicant provided mechanical drawings illustrating mechanical equipment envelope and process instrumentation information; and (vi) the design information included design codes and standards, design methodologies, materials of construction, and load combinations to be used to design the decontamination pit and seismic restraints.

The NRC staff finds that the applicant provided adequate descriptions of the location and arrangement of the decontamination pit because the applicant included a floor plan for the WHF that delineated the locations and arrangements of the decontamination pit with respect to other systems within each facility.

The NRC staff finds that the applicant provided an adequate discussion on potential interactions between the decontamination pit and the seismic restraints in the cask decontamination subsystem because (i) the seismic restraints will be tied to the decontamination pit structure with the ability for adjustment to accommodate casks of various sizes and (ii) the decontamination pit process and instrumentation diagram showed that the seismic restraints are an integral part of the cask decontamination subsystem.

The NRC staff finds that the applicant provided an adequate discussion of design information regarding the ability of the cask decontamination subsystem to withstand the effects of natural phenomena because this subsystem will be designed to include seismic restraints to prevent tip over of the casks inside the decontamination pit.

On the basis of the above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient to permit an evaluation of the PCSA and design of the cask decontamination subsystem.

The NRC staff finds that the applicant's descriptions and design information of the emergency and radiological safety systems are adequate because (i) the applicant described the use of equipment and design features to mitigate emergency events; (ii) the applicant described a set-aside area for the operational radiation protection program to support radiological monitoring; (iii) the applicant described the systems to be used to monitor the surface and subsurface areas and effluents from the GROA release points and alert operators of any radiological releases resulting from Category 1 or Category 2 event sequences or conditions; and (iv) the descriptions included design codes and standards for the radiation/radiological monitoring systems. On the basis discussed above, the NRC staff finds that the description and design information the applicant provided are sufficient to permit an evaluation of the PCSA and design of the emergency and radiological safety systems.

**NRC Staff's Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.2.9, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the WHF, CRCF, and RF decontamination, emergency, and radiological safety systems meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the WHF, CRCF, and RF decontamination, emergency, and radiological safety systems sufficient for the NRC staff to evaluate the applicant's PCSA and design.

#### 2.1.1.2.3.3 Descriptions of, and Design Details for, Structures, Systems, and Components; Equipment; and Utility Systems of the Subsurface Facility

In this SER section, the NRC staff evaluates the applicant's description of the subsurface facility SSCs and operational process activities on the basis of information in SAR Section 1.3. The NRC staff's evaluation of the subsurface description focused on the geometrical and other physical characteristics of the SSCs, their functions, and the design features the applicant used to accomplish the functions. Functions of the subsurface facility openings and structures identified in the NRC staff's review are summarized in SER Table 2-1 and show how functions are linked to design features of the openings. Each function for the structures presented in Table 2-1 has its own separate number that coincides with the controlling design features with the same number. However, if a specific function does not have any controlling design features, a corresponding number is not provided in the controlling design features column of the table.

2.1.1.2.3.3.1 Subsurface Facility Layout and Development Plan

***Subsurface Facility Layout***

The applicant described the layout of subsurface facility structures in SAR Sections 1.3.1.1 and 1.3.2.2.1. According to the applicant, the subsurface facility will consist of nonemplacement and emplacement areas. The nonemplacement area will consist of the North Ramp, access mains, turnouts (curved openings that connect the access mains to the emplacement area), intake shafts, openings used for performance confirmation (e.g., an observation drift and alcove), ventilation raises, exhaust mains, shaft-access drifts, and exhaust shafts (SAR Figure 1.3.1-1). The emplacement area will consist of a horizontal array of emplacement drifts divided into four panels (SAR Figure 1.3.1-1). Panel 1, the smallest panel, will consist of six emplacement drifts in the central area of the subsurface facility and will be developed first, as the applicant stated. Each emplacement drift will be connected to an access main through the turnout drift at one end of the emplacement drift (SAR Figure 1.3.1-4). The other end of the emplacement drift will be connected to a ventilation exhaust main, which, in turn, will be connected to an exhaust shaft (SAR Figure 1.3.1-1).

**Table 2-1. Functions of the Subsurface Facility Structures Based on NRC Staff Evaluation of the Applicant's Description of the Subsurface Facility Design**

| Structure | Functions | Controlling Design Features |
|---|---|---|
| North Portal | (1) Access control to the subsurface facility<br>(2) Waste package transportation to subsurface facility<br>(3) Fresh air intake opening for the emplacement ventilation system<br>(4) Supports closure operations<br>(5) Protects the subsurface facility against storm water | (3) Stability of roof and walls<br>(5) Invert elevation; water diversion and control structures; slope to North Ramp entrance |
| North Ramp | (1) Supports crane rails for the TEV and DSEG<br>(2) Supports a third rail for power supply<br>(3) Fresh air intake conduit for the emplacement ventilation system | (1) Stability of invert<br>(2) Stability of invert<br>(3) Stability of roof and walls |
| Access Main | (1) Provides infrastructure for transportation, power supply, and communications and control systems<br>(2) Supports crane rails for the TEV and DSEG<br>(3) Supports a third rail for power supply<br>(4) Provides access to waste emplacement areas<br>(5) Fresh air conduit for the emplacement ventilation system | (1) Overall stability of opening<br>(2) Stability of invert<br>(3) Stability of invert<br>(4) Overall stability of opening<br>(5) Stability of roof and walls |

**Table 2-1. Functions of the Subsurface Facility Structures Based on NRC Staff Evaluation of the Applicant's Description of the Subsurface Facility Design (continued)**

| Structure | Functions | Controlling Design Features |
|---|---|---|
| Turnout | (1) Limits radiation dose rate in the access main<br>(2) Controls access to emplacement drift<br>(3) Regulates air flow into the emplacement drift<br>(4) Provides smooth elevation transition from access main to emplacement drift<br>(5) Supports crane rails for the TEV and DSEG<br>(6) Supports a third rail for power supply | (1) Curvature and length<br>(2) Emplacement access doors<br>(3) Air flow regulator; stability of roof and walls<br>(4) Invert slope and elevation<br>(5) Stability of invert<br>(6) Stability of invert |
| Exhaust Main | (1) Exhaust conduit for heated air from emplacement drifts<br>(2) Provide remote access for inspection and maintenance | (1) Stability of walls and roof<br>(2) Overall stability and invert stability |
| Intake Shaft | Fresh air conduit for the emplacement ventilation system | Stability of shaft walls |
| Exhaust Shaft | Exhaust conduit for heated air from emplacement drifts via the exhaust main | Stability of shaft walls |
| Ventilation Raise | Exhaust conduit for heated air from emplacement drifts via the exhaust main | Stability of walls |
| Observation Drift for Performance Confirmation | Used for installation of test equipment and infrastructure needed for performance confirmation monitoring of the rock mass around the thermally accelerated drift | Overall stability of observation drift and of the rock pillar shared with the thermally accelerated drift |
| Observation Alcove Under Emplacement Drift Panel 1 | Used for installation of test equipment and infrastructure needed for performance confirmation monitoring of the rock mass around the thermally accelerated drift | Overall stability of opening and of the rock pillar shared with the thermally accelerated drift |
| Seepage Alcoves | Measure seepage in the unsaturated zone | Stability of opening |
| Emplacement Drift Opening | (1) Waste package emplacement and inspection<br>(2) Drip shield installation<br>(3) Fresh air conduit for waste package ventilation | (1) Stability of roof and walls<br>(2) Stability of shape and dimension of drift opening<br>(3) Stability of drift opening |
| Emplacement Drift Invert Structure | (1) Foundation of crane rail and power supply third rail<br>(2) Drip shield alignment and interlocking | (1) Stability of invert structure; serviceability of crane rail<br>(2) Serviceability of crane rail |

The applicant in SAR Section 1.3.2.2 identified geometrical constraints for the subsurface facility layout to satisfy design features that the applicant used in assessing operational safety and postclosure performance.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to evaluate the applicant's description and design information for the subsurface facility layout. The NRC staff reviewed layouts of the nonemplacement and emplacement areas in the subsurface facility. The NRC staff also reviewed the geometrical constraints of the subsurface facility. The NRC staff finds that the applicant's layout description provided the geometrical relationships among underground openings and identified geometrical constraints for the opening. Therefore, the NRC staff finds that the layout description and design information are sufficient for the NRC staff to evaluate the PCSA and the subsurface facility design.

***Subsurface Facility Development Plan***

The applicant described the subsurface facility development plan in SAR Section 1.3.1. The applicant stated that operations in the subsurface facility will be preceded by a period of initial construction, during which three emplacement drifts will be built and commissioned to receive waste. The start of waste emplacement will mark the end of the period of initial construction and the beginning of repository operations in the subsurface facility. The applicant plans for a period of operations, also referred to as the preclosure period, of approximately 100 years (SAR Section 1.3.1). The preclosure period would consist of 50 years of waste emplacement, including an initial period of 24 years of concurrent repository development, and 50 years of postemplacement monitoring. The subsurface facility will include a ventilation system that will use forced air flow to cool waste packages through the preclosure period. The first set of emplaced waste packages would be subjected to approximately 100 years of forced ventilation, and the last set would be subjected to 50 years of forced ventilation, on the basis of information in SAR Section 1.3.1.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's description of the subsurface facility development plan. The NRC staff reviewed the sequence of construction of the underground facility, initial waste emplacement, plan for concurrent repository development and waste emplacement, and ventilation. The NRC staff finds that the applicant's information on the sequence and time estimate for drift development and waste emplacement and duration of ventilation are sufficient to permit an evaluation of the PCSA and design because this information enables the staff to evaluate the development schedule for the subsurface facility structures.

***Thermal Load Design***

The applicant described its approach to thermal management in SAR Section 1.3.1.2.5. The applicant stated that it will manage the repository thermal load by controlling the arrangement of waste packages in emplacement drifts and providing forced ventilation to remove waste-generated heat. The applicant specified thermal load control parameters, including (i) maximum waste package thermal power at emplacement of 18 kW (17 Btu/sec.) for a CSNF waste package or 11.8 kW (11.2 Btu/sec.) for a naval spent nuclear fuel (SNF) waste package, (ii) maximum line load limit for a drift of 2.0 kW/m [0.61 kW/ft] or 1.45 kW/m [0.44 kW/ft] for any seven-waste-package segment that includes a naval SNF waste package, and (iii) end-to-end spacing of 10 cm [4 in] between adjacent waste packages. The applicant stated that the actual waste stream that will be emplaced in the drifts will depend on a number of variable and unspecified factors. The applicant indicated that a customized loading plan will be developed for each emplacement drift to meet the overall repository thermal goals after definitive shipping schedules for SNF from utilities and other sources become available.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's description and design information for the approach to managing the repository thermal load. The NRC staff finds that the applicant's thermal load description and design information are sufficient to permit an evaluation of the thermal load design and to use in the PCSA and postclosure performance assessment because the applicant provided (i) adequate information regarding the thermal load design and (ii) the approach the applicant will use to determine whether a waste package arrangement satisfies the thermal load design.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.3.1, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the subsurface facility layout, development plan, and thermal load meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the subsurface facility layout, development plan, and thermal load sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.3.2 Nonemplacement Areas of the Subsurface Facility

The applicant described the nonemplacement areas of the subsurface facility in SAR Section 1.3.3. The applicant stated that the nonemplacement areas of the subsurface facility will consist of all underground openings and their SSCs, except the emplacement drifts. The NRC staff's understanding of the functions of underground openings and their inverts in the nonemplacement areas of the subsurface facility during the preclosure period is summarized in SER Table 2-1. The functions support operation of ITS TEV or operations and activities, such as thermal management, that control parameter values that the applicant used for postclosure performance assessment.

***North Portal and North Ramp***

The applicant described the North Portal and North Ramp in SAR Section 1.3.3.1.1. The North Portal will connect the surface facilities to the subsurface facility through the North Ramp (SAR Figures 1.3.3-4 and 1.3.3-5). The North Ramp will be sloped at 2.15 percent to connect the surface facilities to the emplacement horizon (SAR Section 1.3.3.1.1). Ground support for the North Ramp will consist of fully grouted rock bolts, steel-fiber-reinforced shotcrete, and occasional lattice girders (SAR Section 1.3.3.3.1). The invert of the North Ramp will consist of a reinforced concrete slab with embedded anchor bolts to support crane rails for the transport and emplacement vehicle and a third rail for the power supply (SAR Section 1.3.3.4.1).

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's descriptions and design information for the North Portal and North Ramp designs. Specifically, the NRC staff reviewed the descriptions of the geometry and layout of the North Portal and North Ramp, ground supports to be used, and the intended functions. On the basis of the applicant's description, the NRC staff finds that the North Portal, North Ramp, and invert need to be sufficiently stable during the preclosure period to support functions listed in SER Table 2-1. The invert elevation at the North Portal needs to be high enough to protect against storm water flow to the subsurface facility. The NRC staff finds that the applicant's North Portal and North Ramp descriptions and design information are sufficient to permit an evaluation of the PCSA and design of the North Portal and North Ramp because the applicant (i) provided the basic geometry and layout, (ii) identified the construction materials, and (iii) defined the intended functions. The NRC staff's evaluation of the North Portal and North Ramp designs' capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7.2, where the NRC staff finds that the applicant's design of the North Portal and North Ramp, along with the proposed monitoring and maintenance activities, would ensure the stability of the access mains during the preclosure period. The NRC staff's evaluation of protection of the invert at the North Portal against storm water is discussed in SER Section 2.1.1.1.3.4, where the NRC staff finds that the applicant demonstrated the adequacy of the proposed engineered flood barriers by verification of available freeboard (vertical distance between the top of an engineered barrier and the maximum flood depth) for the areas subject to inundation along the entire length of each engineered barrier.

***Access Mains***

The applicant described the access mains in SAR Section 1.3.3.1.2. The applicant stated that the subsurface facility will include three access mains that will connect the North Ramp to the emplacement drifts (SAR Figure 1.3.3-8): the access main for Panels 1 and 2, the access main for Panels 3E and 3W, and the access main for Panel 4. The access mains will be excavated with a tunnel-boring machine to a circular cross section of a 7.62-m [25-ft]-diameter, except for the access main cross drift to Panel 4, which will have a diameter of 5.5 m [18 ft] (SAR Section 1.3.3.1.2). The access mains will be connected to the emplacement drifts via turnouts that will accommodate the turning radius of the TEV. Ground support for the access mains will consist of fully grouted rock bolts and wire mesh, except at the intersections with the turnouts, where the ground support will include fully grouted rock bolts with steel-fiber-reinforced shotcrete and occasional lattice girders (SAR Section 1.3.3.3.1). The access main invert will consist of a reinforced concrete slab with embedded anchor bolts to support crane rails for the TEV and a third rail for the power supply (SAR Section 1.3.3.4.1).

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's descriptions and design information for the access mains. Specifically, the NRC staff reviewed the descriptions of the geometry and layout of the access mains, ground supports to be used, and the intended functions. On the basis of the applicant's description, the NRC staff finds that the access mains and invert need to be sufficiently stable during the preclosure period to support the functions listed in SER Table 2-1. The NRC staff finds that the applicant's access mains descriptions and design information are sufficient to permit an evaluation of the PCSA and design of the access mains because the applicant (i) provided the basic geometry and layout, (ii) identified the construction materials, and (iii) defined the intended functions. The NRC staff's evaluation of the capability of the access mains' design to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7.2, where the NRC staff finds that the applicant's design of the access mains, along with the proposed monitoring and maintenance activities, would ensure the stability of the access mains during the preclosure period.

***Turnouts***

The applicant described the turnouts' design in SAR Section 1.3.3.1.4. The applicant stated that the turnouts will connect the emplacement drifts to the access mains and will contain facilities and equipment to control access and ventilation to the emplacement drifts. The turnout cross sections will vary in shape and dimensions, from a rectangular section at the intersection of the access main, to a circular section with a 5.5-m [18-ft]-diameter at the intersection of the emplacement drift (SAR Figure 1.3.3-13). The invert of the turnout will slope up toward the emplacement drift. As described in SAR Section 1.3.3.1.4, the invert slope will increase from 1.35 percent at the access main intersection to a maximum of 1.75 percent and decrease thereafter to a 0 percent slope at the emplacement drift intersection. The applicant stated in SAR Section 1.3.3.1.4 that the curvature and length of the turnout is designed to prevent direct-line radiation from any emplaced waste package to the access main. SAR Figure 1.3.3-13 indicated a radiation dose rate at the access main intersection approximately six orders of magnitude smaller than the dose rate at the emplacement drift entrance because of the length and curvature of the turnout. Ground support for the turnouts will vary along the turnout axis, as described in SAR Section 1.3.3.3.1. For the turnout segment closest to the access main, the ground support will consist of fully grouted rock bolts with steel-fiber-reinforced shotcrete and occasional lattice girders. The rock bolts will have a nominal length of 5 m [16.4 ft] and will be spaced in a square-grid pattern at 1.25-m [4.1-ft]-centers. The shotcrete will be 0.1-m [0.3-ft]-thick. Ground support for the other turnout segments will consist of stainless steel, friction-type rock bolts with stainless steel welded wire fabric. The rock bolts will have a nominal length of 3 m [9.8 ft] and will be spaced in a square-grid pattern at 1.25-m [4.1-ft] centers. The fabric will be W4 × W4 with 75-mm [3-in] center-to-center wire spacing. The turnout invert will consist of reinforced concrete in the segments closest to the access main and a carbon steel invert structure with ballast in the segments closest to the emplacement drift entrance (SAR Section 1.3.3.4.1).

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's descriptions and design information for the turnouts. Specifically, the NRC staff reviewed the descriptions of the turnout geometry and layout, geometrical constraints, ground supports to be used, and the intended functions. The NRC staff determines that, based on the applicant's information in the SAR, which was summarized by the NRC staff in SER Table 2-1, the applicant will rely on the following functions and design features for the turnouts: (i) protecting the access main from radiation by controlling the curvature and length, (ii) providing a smooth transition from the invert of the access main to the invert of the connected emplacement drift by controlling the turnout invert slope, (iii) supporting crane rails for transportation of waste packages or drip shields and a third rail for a power supply by assuring invert and cross section stability, (iv) providing the operating envelope for the drip shield emplacement gantry (DSEG) by assuring invert and cross section stability, (v) providing access to the connected emplacement drift by assuring invert and cross section stability, and (vi) functioning as a fresh air conduit for ventilation of disposed waste packages by assuring invert and cross section stability. The NRC staff finds that the applicant's descriptions and design information for the turnouts are sufficient to permit an evaluation of the PCSA and design of the turnouts because the applicant (i) provided the basic geometry and layout, (ii) identified the construction materials, and (iii) defined the intended functions. The NRC staff's evaluation of the turnouts design capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7.2, where the NRC staff finds that the applicant's design of the turnouts, along with the proposed monitoring and maintenance activities, would ensure the stability of the turnouts during the preclosure period.

### *Exhaust Mains*

The applicant described the exhaust mains' design in SAR Section 1.3.3.1.3. The applicant stated that each exhaust main will connect the exhaust end of several emplacement drifts to an exhaust shaft via a shaft access drift (SAR Figure 1.3.3-8). The other end of the emplacement drifts will connect to an access main via a turnout. According to the applicant, the exhaust mains will have the same diameter as the access mains {i.e., 7.62 m [25 ft]}, except the exhaust main for Panel 1 will have a diameter of 5.5 m [18 ft]. The applicant will use an isolation barrier, where an exhaust main and access main will intersect to separate the intake air in the access main from the exhaust air (SAR Section 1.3.3.1.3). SAR Section 1.3.3.1.3 and Figure 1.3.3-8 indicated that emplacement drift Panels 4, 3W, and 1 will have separate but closely spaced exhaust mains. The applicant explained that separate exhaust mains will be needed to allow concurrent development in Panel 4 and waste emplacement in Panel 3W or development in Panel 4 adjacent to a waste-loaded Panel 1. According to the applicant (DOE, 2009dm), the majority of the exhaust main lengths for Panels 4, 3, and 1 will be parallel and will have a centerline-to-centerline spacing of approximately 22.9 m [75 ft]. The primary function of an exhaust main will be to remove hot ventilation air from the repository during the preclosure period (SAR Section 1.3.3.1.3). Thus, the exhaust main will play a key role in the applicant's thermal management strategies to satisfy the applicant's thermal performance requirements. According to the applicant, the exhaust mains also will be used for remote access for inspection and maintenance. Ground support for the exhaust mains will consist of fully grouted rock bolts with welded wire fabric (SAR Section 1.3.3.3.1). The rock bolts will have a nominal length of 3 m [9.8 ft] in a square-grid pattern at 1.25 m [4.1 ft] center to center. Ground support where an exhaust main and an emplacement drift intersect will consist of fully grouted rock bolts with steel-fiber-reinforced shotcrete and occasional lattice girders. The rock bolts will be approximately 5 m [16.4 ft] long and placed in the same square-grid pattern. The exhaust mains may have an invert to facilitate mobile equipment access, as the applicant stated. The applicant explained in SAR Section 1.3.1.2.1.6 that the exhaust mains, like the exhaust shafts, will be inaccessible because of high temperature and potential high radiation levels.

### NRC Staff's Evaluation

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's descriptions and design information for the exhaust mains. The NRC staff reviewed the descriptions of the exhaust main geometry and layout, ground supports to be used, and the intended functions. The exhaust mains need to be sufficiently stable during the preclosure period to (i) function as return air conduits for ventilation of disposed waste packages and (ii) provide access to remote-controlled equipment for inspection and maintenance. The NRC staff's understanding of the functions of the exhaust mains is summarized in SER Table 2-1. The NRC staff finds that the applicant's descriptions and design information for the exhaust mains in the SAR and subsequent information provided to respond to an NRC staff request for additional information (RAI) (DOE, 2009dm) are sufficient to permit an evaluation of the PCSA and design of the exhaust mains because the applicant (i) provided the basic geometry and layout, (ii) identified the construction materials, and (iii) defined the intended functions. The NRC staff's evaluation of the exhaust mains' design capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7.3, where the NRC staff finds that the applicant's design of the exhaust mains, along with the proposed monitoring and maintenance activities, would ensure the stability of the exhaust mains during the preclosure period.

### *Shafts and Ventilation Raises*

The applicant described the design of shafts and ventilation raises in SAR Section 1.3.3.1.5. The applicant stated that the subsurface facility will include three intake shafts and six exhaust shafts (SAR Figure 1.3.3-8). The shafts will connect the emplacement areas to the ground surface and will be used primarily for ventilation intake and exhaust. SAR Table 1.3.3-1 summarized the dimensions of the shafts and indicated a finished diameter of approximately 7.3 m [24 ft] for seven shafts and approximately 4.4 m [14.4 ft] for two exhaust shafts. SAR Section 1.3.3.3.1 stated that the larger diameter shafts will be lined with 0.3 m [12 in] of plain concrete (i.e., concrete without reinforcement) and the smaller diameter shafts will be lined with 0.25 m [10 in] of plain concrete. The applicant stated that plain concrete will provide adequate support for the shaft walls because the liner will be applied after full relaxation of the walls following excavation; the applicant stated the concrete liner will be installed stress free. As the applicant described in SAR Section 1.3.3.3.1, ground support for the shaft base where the shaft will intersect the shaft access drift will consist of fully grouted rock bolts with a nominal length of 3 m [9.8 ft] in a square-grid pattern at 1.25 m [4.1 ft] center to center. The applicant stated in SAR Section 1.3.1.2.1.6 that the exhaust shafts will be inaccessible because of high temperature and potential high radiation levels and, therefore, will be monitored remotely using observation vehicles equipped with video cameras to determine concrete liner conditions. According to the applicant, the subsurface facility also will include two short vertical openings, referred to as raises, as described in SAR Section 1.3.3.1.5. The applicant stated that one raise will connect the exhaust main of emplacement drift Panel 1 to the enhanced characterization of the repository block (ECRB) cross drift exhaust shaft, and the other raise will connect the exhaust main of Panel 4 to the ECRB cross drift exhaust shaft (SAR Figures 1.3.3-8 and 1.3.5-5).

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's descriptions and design information for shafts and ventilation raises. The NRC staff reviewed the descriptions of the ventilation raise geometry and layout, ground supports to be used, and the intended functions. The shafts and ventilation raises need to be sufficiently stable during the preclosure period for (i) the exhaust shafts and ventilation raises to function as return air conduits and (ii) the intake shafts to function as fresh air intake for the ventilation of disposed waste packages. The NRC staff's understanding of the shafts' and raises' functions is summarized in SER Table 2-1. The NRC staff finds that the applicant's descriptions and design information for the shafts and ventilation raises are sufficient to permit an evaluation of the PCSA and design of the shafts and ventilation raises because the applicant (i) provided the basic geometry and layout, (ii) identified the construction materials, and (iii) defined the intended functions. The NRC staff's evaluation of the shafts' and raises' design capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7, where the NRC staff finds that the applicant's design of the shafts and ventilation raises, along with the proposed monitoring and maintenance activities, would ensure the stability of the shafts and ventilation raises during the preclosure period.

### *Subsurface Facility Openings Dedicated to Performance Confirmation*

The applicant described the design of underground openings dedicated to performance confirmation in SAR Section 1.3.3.1.6. The applicant stated that the subsurface facility will include an observation drift and three alcoves dedicated to performance confirmation. The observation drift and one alcove will be located under Panel 1 of the emplacement drift layout.

As shown in SAR Figure 1.3.3-18, the east end of the observation drift will be connected to an existing thermal test alcove off the access main of Panel 1. The drift will extend under Panel 1 and ramp up to connect to the Panel 1 exhaust main. The observation drift will be approximately 20 m [66 ft] north of emplacement drift number 3 of Panel 1 and a minimum of 10 m [33 ft] below the emplacement drift. An alcove attached to the observation drift will extend southward under the emplacement drift, as shown in SAR Figure 1.3.3-18. The applicant stated that the observation drift and alcove will be used to install the instrumentation and equipment needed to monitor the rock mass under emplacement drift number 3 of Panel 1 for performance confirmation. According to SAR Section 1.3.3.3, the ground support for the observation drift and alcove will consist of fully grouted, approximately 3-m [9.8-ft]-long rock bolts spaced in a square-grid pattern at 1.25 m [4.1 ft] center to center, and welded wire fabric. According to the applicant, the subsurface facility will also include two alcoves for monitoring unsaturated zone seepage: one in the nonlithophysal rock zone and another in the lithophysal zone. The applicant stated that the alcoves will be located using fracture mapping data from early stages of repository development and will be excavated off the access mains or the ECRB cross drift.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's descriptions and design information for the underground openings dedicated to performance confirmation. The NRC staff reviewed the descriptions of the geometry and layout of the underground openings dedicated to performance confirmation, ground supports to be used, and the intended functions. To provide space and a platform for instrumentation and equipment to monitor the performance of the rock mass (rock pillar) as part of the emplacement drift performance confirmation program, the rock mass (rock pillar) under emplacement drift 3 of Panel 1 and overlying the observation drift and alcove needs to be stable during the preclosure period. The applicant discussed the functions of the observation drift and alcove, and these functions are summarized by the NRC staff in SER Table 2-1. The NRC staff finds that the applicant's descriptions and design information for the observation drift and alcoves are sufficient to permit an evaluation of the PCSA and design of the observation drift and alcoves because the applicant (i) provided the basic geometry and layout, (ii) identified the construction materials, and (iii) defined the intended functions. The NRC staff's evaluation of the capability of the observation drift and alcoves' design to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7.2, where the NRC staff finds that the underground openings dedicated to performance confirmation along with the proposed monitoring and maintenance activities would ensure the stability of the openings during the preclosure period.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.3.2, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for underground openings in nonemplacement areas of the subsurface facility meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for underground openings in nonemplacement areas of the subsurface facility sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.3.3 Emplacement Areas of the Subsurface Facility

The applicant described the emplacement areas of the subsurface facility in SAR Section 1.3.4. The applicant stated that the emplacement areas of the subsurface facility will consist of a series of emplacement drifts (horizontal underground openings) organized into four panels, as illustrated in SAR Figure 1.3.4-2. One end of each drift will be connected to an access main via a turnout, and the opposite end will be connected to an exhaust main. Each emplacement drift will consist of the drift opening, ground support for stabilizing the immediately surrounding rock, and an invert that will carry the waste emplacement and disposal infrastructure. The emplacement drift is designed to contain the engineered barrier components (i.e., waste package supported on pallet and drip shield). According to SAR Section 1.3.4.1, the emplacement drift will function as (i) a space for disposed waste packages, (ii) a foundation for the waste emplacement infrastructure, (iii) an air flow conduit for ventilation of disposed waste packages, (iv) an operating space for the remote-controlled vehicle used to monitor waste packages as part of a performance confirmation program, and (v) an operating space for the installation of drip shields prior to closure. In addition, emplacement drift Number 3 of Panel 1 will be operated as a thermally accelerated drift through special ventilation controls to develop in-drift environmental conditions for the performance confirmation program (SAR Section 1.3.4.2.3).

As described in SAR Section 1.3.4.2.3, the emplacement drifts will be aligned at an azimuth of 72° (measured eastward from north). The applicant stated that this drift orientation would favor drift stability, considering the prevalent orientation of rock joints. According to the applicant, the drifts will be laid out in a parallel pattern and spaced 81 m [266 ft] horizontally from centerline to centerline. The applicant stated that this drift spacing was chosen to prevent thermal interaction between adjacent drifts and to allow natural and thermally mobilized water percolation to drain between the drifts (SAR Section 1.3.4.2.3). The drift opening will have a circular cross section with a nominal excavated diameter of 5.5 m [18 ft] (SAR Figure 1.3.4-4). The applicant stated that the total length of disposed waste packages in a drift, including an end-to-end spacing of 10 cm [3.9 in] between adjacent waste packages, will be limited to 800 m [875 yd] to maintain the applicant-specified ventilation efficiency. Other features of the emplacement drift design described in SAR Section 1.3.4.2.3 will include the emplacement drift invert, with a horizontal grade at the same elevation as the invert of the connected exhaust main; emplacement drifts excavated using a tunnel-boring machine; and drift mapping after installation of the initial ground support and before installation of final ground support. According to the applicant, geologic mapping of drifts will include documentation of fractures, fault zone characteristics, stratigraphic contacts, and lithophysal content.

The applicant described the emplacement drift ground support in SAR Section 1.3.4.4.1. According to the applicant, an initial ground support and a final ground support will be installed in each emplacement drift. The initial ground support will consist of carbon steel frictional rock bolts and wire mesh installed in the drift crown only, immediately after excavation. The wire mesh will be removed before installation of the final ground support, but the rock bolts will be left in place. The final ground support will consist of a 3-mm [0.12-in]-thick Bernold-type perforated stainless steel (Type 316) liner and a pattern of Super Swellex-type stainless steel (Type 316) rock bolts. The rock bolts will be 3 m [9.8 ft] long and set in a square-grid pattern at a center-to-center spacing of 1.25 m [4.1 ft]. The steel liner and rock bolts will be installed in a 240° arc around the drift periphery above the invert structure, as illustrated in SAR Figure 1.3.4-4. The applicant stated that the emplacement drift ground support will be designed to last at least 100 years without planned maintenance, and any maintenance needs will be evaluated using information from inspection and monitoring (SAR Section 1.3.4.4).

The emplacement drift invert will consist of a steel invert structure and crushed tuff ballast fill (SAR Section 1.3.4.5). The steel invert structure will consist of transverse beams interconnected to four longitudinal beams as illustrated in SAR Figures 1.3.4-5 and 1.3.4-8–10. The transverse beams will be spaced 1.5 m [5 ft] center to center and bolted to the longitudinal beams. The two outermost longitudinal beams will be attached to stub columns that transfer loads to the drift floor. The stub columns will be anchored to the underlying rock. In addition, the ends of the transverse beams will be attached to plates that will be anchored to the drift wall rock. As shown in SAR Figures 1.3.4-5, 1.3.4-8, and 1.3.4-9, crane rails will be mounted on the two outer longitudinal beams, also referred to in the SAR as rail runway beams. The applicant stated in SAR Section 1.3.4.5.1 that the steel invert structure and crane rail will be fabricated from corrosion-resistant steel. The applicant also mentioned a third rail that will be used for the power supply, but the third rail was not shown in the illustrations provided in the SAR. The crushed tuff ballast will fill the void space formed by the steel invert structure and surrounding rock. The applicant stated that the top of the ballast will coincide with the top of the steel structure.

The steel invert structure will provide a platform that supports the emplacement pallets, waste packages, and drip shields during the preclosure period and will gradually transfer the support to the ballast as the steel structure corrodes after emplacement drift closure (SAR Section 1.3.4.5). The steel invert structure will also function as the foundation for the crane rail system for operation of the transport and waste emplacement vehicle (TEV), drip shield emplacement gantry (DSEG), and remote-controlled inspection vehicle.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.2 to review the applicant's description and design information for the emplacement areas of the subsurface facility. The NRC staff reviewed the descriptions of the geometry and layout of the emplacement areas, ground supports to be used, emplacement drift inverts, and the intended functions. The NRC staff determines that the drift opening and invert structure need to be sufficiently stable during the preclosure period to (i) support the crane rails used to operate the TEV, DSEG, and remote-controlled inspection vehicle; (ii) provide the operating envelope for drip shield emplacement; (iii) support the third rail used for power supply; and (iv) function as an air conduit for ventilation of disposed waste packages. The NRC staff summarized the functions of the emplacement drifts, invert structure, and ground support in SER Table 2-1. The NRC staff finds that the description and design information for the emplacement drift, invert structure, and ground support designs in the SAR provided the basic geometry and layout (e.g., drift dimensions, number of packages in a drift, and spacing between waste packages), identified the construction materials (e.g., steel invert structure, crushed tuff ballast fill, stainless steel rockbolts), and defined the intended functions of the design [e.g., longitudinal and transverse supports will function as the foundation for the crane rail system for operation of the TEV, DSEG, and remote-controlled inspection vehicle]. Therefore, the NRC staff finds the description and design information are sufficient to support the evaluation of (i) the design of underground openings in the emplacement areas of the subsurface facility, and (ii) the preclosure safety analysis (PCSA). The NRC staff's evaluation of the capability of the emplacement drifts, invert structure, and ground support designs to perform the functions through the preclosure period is documented in SER Section 2.1.1.2.3.7.3, where the NRC staff finds the design of the emplacement areas of the subsurface facility, along with the proposed monitoring and maintenance activities, would ensure the stability of the emplacement openings during the preclosure period.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.3.3, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the emplacement areas of the subsurface facility meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the emplacement areas of the subsurface facility sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.3.4 Waste Package Transportation and Emplacement System

The applicant described and discussed the transport and waste emplacement vehicle (TEV) design in SAR Sections 1.2 and 1.3. The applicant used this information in the PCSA and iterative design of the TEV (SAR Section 1.3.2.7). The applicant designated the TEV as ITS.

**Description of the Transport and Waste Emplacement Vehicle and Functions**

The applicant described the TEV as a rail-based, self-propelled, multiwheeled vehicle designed for transporting waste packages from the surface facilities (CRCFs and IHF) to the subsurface emplacement areas of the repository. The applicant categorized five main TEV functions: (i) handling the waste packages on associated pallets in the surface facilities by performing docking, lifting, and lowering maneuvers; (ii) providing waste package radiation shielding to personnel in unrestricted areas; (iii) transporting waste packages from the surface facilities to the subsurface facility in a controlled and safe manner; (iv) lifting, lowering, and positioning the waste package during the emplacement process in the drift; and (v) safely returning the TEV to the surface facility. The applicant also proposed to use the TEV for retrieval operations, if needed, by reversing the emplacement operations. The applicant emphasized that even though the TEV is a one-of-a-kind transportation system, its construction, material, and functions are considered similar to those of mining equipment and gantry cranes in the nuclear industry.

For the surface facilities, the applicant provided the layout of the surface rails that illustrated the specific routes of the TEV at the surface (SAR Figures 1.2.1-2). The applicant also provided general descriptions related to the role of the TEV in these surface facility areas, such as the CRCF and IHF. The applicant provided specific functions and interactions between the TEV and other SSCs in the individual rooms within these facilities. It also briefly discussed contamination surveying and interlocking system requirements prior to the TEV exiting from both the surface and subsurface facilities.

Similarly, the applicant provided the routes the TEV will follow in the subsurface facility. This information was described in the form of layouts of the facility's rail system, shown in SAR Figures 1.3.3-9 to 1.3.3-11. The applicant discussed the subsurface crane rail, which will be an integrated rail system that will connect the IHF and CRCF buildings to the subsurface emplacement areas. The applicant provided detailed location, length, direction, and magnitude of expected slopes that the TEV will be designed to travel. It also included the specification for a turning radius of 61 m [200 ft] or larger within the subsurface facilities to allow TEV travel without binding the wheels. The applicant also provided estimates of the TEV travel distances and travel times, such as the minimum one-way travel distance of the TEV {2,760 m [9,055 ft]} in 60 minutes and maximum one-way distance {7,200 m [23,622 ft]} corresponding to a travel time of 160 minutes, excluding stops and delays.

**NRC Staff's Evaluation**

The NRC staff reviewed the description and design information for the TEV using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the intended functions of the TEV and design specifications to prevent occurrence of event sequences. The NRC staff also reviewed the applicant's description of the locations, both in the surface and subsurface facilities, in which the TEV will operate. The purpose of the review was to determine whether sufficient detail was provided to (i) understand TEV operations, (ii) determine TEV propulsion and braking requirements for the worst case elevation changes (grade) and other possible environmental conditions, (iii) determine design requirements for potential locations of runaway initiating events, (iv) determine TEV turn-negotiation capabilities and potential for tip over initiating events, (v) compute throughput, and (vi) determine bounding values for TEV component reliability calculations. On the basis of this review, the NRC staff finds that the applicant's description and design information for the TEV are sufficient to permit an evaluation of the PCSA and design of the TEV because the applicant provided (i) diagrams showing surface and subsurface routes; (ii) adequate description of TEV activities in the surface facilities (e.g., docking, lifting, and lowering maneuvers in surface facilities and lifting, lowering, and positioning the waste package during the emplacement process in the drift); and (iii) adequate information on the route lengths, grades, curvature, activities, minimum and maximum travel times for the TEV routes, and interactions with other SSCs necessary to support a TEV design review.

**Transport and Waste Emplacement Vehicle Design Information**

The applicant described the TEV conceptual design in SAR Section 1.3.3.5.1, SAR Figures 1.3.1-4 and 1.3.3-39 to 1.3.3-41. Information provided included the length of the TEV considering the longest waste package ("South Texas") of 630 cm [248 in]. It also included (i) clearances of at least 5 cm [2 in] circumferentially between the waste package and the TEV; (ii) a factor of safety of 10 percent added to the weight of the TEV; (iii) a lifting mechanism spaced 203 cm [80 in] on each side with 90,718 kg [100 T] and 136,078 kg [150 T] lifting motors (4 of the former and 2 of the latter); (iv) wheel block dimensions such as height, width, and length with pivots fabricated from 5-cm [2-in]-thick steel plates; (v) shape and construction of the TEV steel chassis and shielded enclosure that can withstand a 2,500-kg [2.5-metric ton] rockfall; (vi) shielding material layers consisting of a 3.8-m [1.5-in] inner layer of stainless steel, a 3.8-cm [1.5-in]-layer of depleted uranium (for gamma shielding), a 1.3-cm [0.5-in]-layer of SS316L stainless steel (for structural strength), a 15.2-cm [6.0-in]-layer of NS-4-FR (for neutron shielding), and a 1.3-cm [0.5-in] outer layer of stainless steel; and (vii) description of drive motors, lifting motors, shielding enclosures, shield doors, ITS mechanical switch, extendable base plate, third rail power, sensors (speed, range, and temperature), fire-suppression system, communication devices, programmable logic controllers (PLCs), interlock switches, and video cameras.

The applicant stated that the electric drive motors and the lifting jacks will be selected with a type of gearing unit that will prevent the load from back-driving the units under runaway or loss-of-power conditions. The applicant specified commercially available "thruster" brakes because of their ability to utilize the TEV's own weight and motion to exert a vertical force (directly proportional to a braking force) to the top of the rail to prevent TEV movement. The applicant specified that the TEV rail wheel material will be of a lower hardness than the subsurface crane rail. This will result in more wear of TEV wheels rather than the subsurface rails, which will be more difficult to repair inside the radiation environment of the drifts.

In addition to design information in the SAR Section 1.3.3.5, the applicant provided other supplemental references; in particular, TEV drawings, dimensions, weight, materials of construction, and subsystem descriptions (BSC, 2008ad,cb and DOE, 2009gv). The applicant also discussed applicable industry codes and standards for the wheel and rail design. An example of cited codes for crane rail specifications were ASTM A 759–00 (ASTM International, 2001aa,), as specified in ASME–NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), and American Railway Engineering and Maintenance of Way Association (2007aa).

**NRC Staff's Evaluation**

The NRC staff reviewed the TEV design information using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the TEV dimensions, TEV subsystems and components, design information, and subsystem descriptions. On the basis of the information provided in the SAR and the information included in the supplemental documents (BSC, 2008,cb, 2006aj, DOE,2009gv), the NRC staff finds that the TEV description and design information the applicant provided are sufficient to permit an evaluation of the PCSA and design because the applicant provided (i) adequate descriptions of TEV drawings, dimensions, weight, materials of construction, and subsystem descriptions and (ii) codes and standards that are consistent with the standard engineering practices for equipment performing similar functions.

**Transport and Waste Emplacement Vehicle Structural and Thermal Analysis**

In SAR Section 1.3.3.5.1, the applicant discussed calculations to size TEV components. The applicant also discussed the methodology employed, the key TEV performance computations, and the specifications resulting from the analyses. The applicant used standard guidelines and references from ASME–NOG–1 (American Society of Mechanical Engineers, 2005aa); Given (1992aa); Avallone, et al. (2006aa); and American Institute of Steel Construction (1997aa) to perform sizing calculations.

The applicant evaluated the impact of a collision between the TEV and an emplaced waste package in BSC (2007cd). The applicant performed this analysis using ANSYS®, an industry-accepted simulation software. The results demonstrated that a TEV travelling at 3 km/hour [2.0 mph] (17 percent higher than the nuclear safety design bases speed limit target for the TEV of 1.7mph), with a driving force of 50 percent more than the total propelling force of the TEV, would not cause waste package outer corrosion barrier failure. The applicant used methods in its analyses that are commonly used in the engineering community to define boundary conditions and to derive simulation results that could be validated during the "live load" confirmation program.

The applicant provided analyses predicting the temperature within the TEV during its emplacement operations. The analyses (BSC, 2007ce) considered the geometry of the TEV, a range of heat generation of the waste packages (11.8 kW to 30 kW TAD), heating from solar energy incidents {200 cal/cm$^2$ [1,290 cal/in$^2$]} on the TEV surface, and thermal parameters (i.e., density, conductivity, and specific heat) of the different constituents of the shielding enclosure materials. The applicant used this information to define inputs to the design of the TEV regarding onboard cooling system requirements, as well as drift ventilation and emergency backup power requirements, in the event of a power failure in the subsurface facility.

**NRC Staff's Evaluation**

The NRC staff reviewed the description of structural and thermal analyses using the guidance in YMRP Section 2.1.1.2. The NRC staff verified that the applicant included computation of expected TEV chassis frame deflections under waste package and shielding weight. In addition, the NRC staff reviewed the information the applicant provided regarding specifications of the minimum power requirement and maximum speed limits for lifting, restraining, and propelling; redundant braking of the TEV in the presence of elevation changes; reduced traction coefficient in steel-on-steel wheel/rail interfaces; allowable frame deflection from heaviest loads; and drag from rail curvature, 145-km/hour [90-mph] winds, and seismic loading. The NRC staff finds that the applicant's descriptions of its thermal propulsion and structural analyses for the TEV are sufficient to permit an evaluation of the PCSA and design because they (i) included the methodology applied using either standards or simulation software (ANSYS®), both of which are consistent with the standard engineering practices; (ii) utilized appropriate boundary conditions; (iii) included bounding parameters and specifications needed for proper TEV design; and (iv) provided sufficient basis for the applicant's quantification of temperature harshness within the emplacement drifts.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.3.4, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the TEV meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for the TEV sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed in the NRC staff evaluations, the applicant provided (i) adequate descriptions of operations at both surface and subsurface locations, (ii) adequate design information that included codes and standards that are consistent with the standard engineering practices for equipment performing similar functions, and (iii) structural and thermal analyses.

2.1.1.2.3.3.5 Waste Package Emplacement Pallet System

The applicant described and discussed the waste package emplacement pallet design in SAR Section 1.3.4.6. The applicant proposed to use the waste package emplacement pallet to support the waste package for handling, transportation, and emplacement during the preclosure and postclosure periods. The applicant classified the waste package emplacement pallet as non-ITS because it will not be relied on to prevent or mitigate the effects of Category 1 and Category 2 event sequences related to the waste package. In addition, the applicant classified the waste package emplacement pallet as not important to waste isolation (non-ITWI) because it will not have a barrier function or a potential to reduce damage to waste packages during a seismic event.

The applicant described the design and design drawings for two waste package emplacement pallet configurations that will be used at the repository: (i) the standard waste package emplacement pallet, which is designed to accommodate all waste package configurations, except the 5-DHLW/DOE short waste package, and (ii) the short waste package emplacement pallet, which is specifically designed to accommodate the 5-DHLW/DOE short waste package. Both waste package emplacement pallet configurations will have a single design (SAR Figure 1.3.4-13) that will consist of two waste package supports containing V-shaped cradles to accommodate all waste package diameters with the supports connected by square tubes. According to SAR Figures 1.3.4-11 and 1.3.4-12, the waste package emplacement pallet length will be varied between 2,501 and 4,148 mm [98.5 and 163.3 in] for short and standard configurations, and the height and the width will be 726.3 and 1,845 mm [28.59 and 72.65 in] for both configurations, respectively. In addition, the SAR indicated that the maximum weight of the waste package emplacement pallet will be 1,970 kg [4,340 lb] as reported in SNL (2007ap).

In BSC (2007ca), the applicant stated that the waste package emplacement pallet will be considered a Class 2 vessel plate-type support, and its design will be governed by 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF–3000 (American Society of Mechanical Engineers, 2001aa), which governs design requirements for the support-type systems and components. In the design analyses, the applicant will consider two normal loading conditions for the waste package emplacement pallet: (i) the horizontal lifting of the emplacement pallet loaded with the waste package and (ii) the emplacement pallet under waste package static load, as emplaced in the drift.

The material used for the waste package supports will be Alloy 22, as outlined by the American Society of Mechanical Engineers SB–575, UNS N06002 (American Society of Mechanical Engineers, 2001aa), which will be chosen to provide only Alloy 22-to-Alloy 22 contact surfaces for the waste packages. According to the applicant, using the same material on the waste package supports and the waste package outer corrosion barrier would minimize the potential for galvanic corrosion at the areas of contact between the two. The material used for the square tubes will be Stainless Steel Type 316, as described by the American Society of Mechanical Engineers SA–240, UNS S31600 (American Society of Mechanical Engineers, 2001aa). According to the applicant, this material was selected because the general corrosion rate of the stainless steel tubes in the repository-relevant environment is low.

The applicant stated that the waste package emplacement pallet will be fabricated using appropriate sections of the following ASME codes and standards: 2001 ASME Boiler and Pressure Vessel Code Section II; Section III, Division 1, Subsections NF/NCA; Section V; Section IX (American Society of Mechanical Engineers, 2001aa); Y14.5–M–1994 (American Society of Mechanical Engineers, 1994aa); B46.1–1995 (American Society of Mechanical Engineers, 1995aa); NQA–1–2000, Subparts 2.1 and 2.2 (American Society of Mechanical Engineers, 2000aa); ANSI/AWS A2.4–98 (American Welding Society, 1998aa); and ANSI/AWS A5.32/A5.32M–97 (American Welding Society, 1997aa).

**NRC Staff's Evaluation**

The NRC staff evaluated the description and design information for the waste package emplacement pallet system as a part of the subsurface facility using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the design descriptions, including specifications and design drawings of the waste package emplacement pallet configurations and the proposed construction materials. The NRC staff finds that the applicant's description and design information of the waste package emplacement pallet system are adequate because the applicant (i) discussed the function of the system as it will provide support to the waste package during handling, transport, and emplacement; (ii) described the design configurations of the waste package emplacement pallet system including standard and short versions {with 4,148 mm [163.3 in] log for the standard version and 2,501 mm [98.5 in] for the short version}; (iii) provided dimensions and weight of the two configurations; (iv) described the components of the system, including two V-shaped cradles connected by square tubes; (v) provided the codes and standards for designing and fabricating the system, including design loads that are consistent with the standard engineering practices for equipment performing similar function; (vi) specified construction materials for the V-shaped cradles and square tubes, respectively; and (vii) described the interactions of the waste package emplacement pallet system with the TEV and waste package.

On the basis of the above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient to permit an evaluation of the PCSA and design of the waste package emplacement pallet system.

**NRC Staff's Conclusion**

On the basis of the evaluation in SER section 2.1.1.2.3.3.5, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the waste package emplacement pallet meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided adequate description and design information for the waste package emplacement pallet sufficient for the NRC staff to evaluate the applicant's PCSA and the design.

2.1.1.2.3.3.6 Drip Shield Emplacement System

The applicant described the drip shield emplacement gantry (DSEG) design in SAR Section 1.3.4.7.2. The applicant designated the DSEG as non-ITS because it will not be relied upon during the preclosure period to prevent or mitigate Category 1 and Category 2 event sequences.

In SAR Section 1.3.4.7.2, the applicant described the functions of the DSEG that will include the following operations: (i) moving into the drip shield staging area (Heavy Equipment Maintenance Facility) and straddling a drip shield; (ii) lifting a drip shield from its specially designed brackets; (iii) transporting a drip shield to a predetermined location at the turnout of a designated drift at a speed of 46 m/min [150 ft/min]; (iv) waiting for confirmation of precise location and directions from the Central Control Center Facility (CCCF) to proceed; (v) installing the drip shield by straddling and moving over emplaced waste packages in the 600- and 800-m [1,969- and 2,625-ft]-long emplacement drifts, as commanded, at an initial crawl speed of 4.6 m/min [15 ft/min] and a subsequent slow crawl speed of 0.5 m/min [1.5 ft/min]; and (vi) returning to the surface facility to repeat the process.

The applicant described the DSEG design as a self-propelled, rail-based crane, which will be similar to the TEV based on nuclear and industrial crane technology. The main components of the DSEG will include (i) a steel frame structure capable of supporting the weight of a drip shield; (ii) a lifting system composed of four lifting brackets, screw jacks, shot bolts, and gantry motors that can vertically lift the drip shield for transportation; (iii) a self-propulsion system containing electric drive motors with integrated disk brakes and fail-safe capabilities; (iv) an onboard programmable logic controllers (PLCs) network that communicates with the CCCF and with thermal and radiological sensing instrumentation onboard the DSEG; (v) an electrified third rail supplied by a dual-power-pickup mechanism to provide power to onboard electrical systems; (vi) air-conditioned cooled electronic cabinets to protect temperature-sensitive equipment; (vii) a fire-suppression system that detects and automatically operates when needed; and (viii) instrumentation and control system (I&C) containing articulated cameras, ultrasonic sensors, and high-intensity lights. The applicant provided a more extensive discussion on the drip shield emplacement operations and its conceptual design, including drive system, electrical and control systems, braking controls, cooling system, vision system, thermal and radiation monitoring system, fire protection, and communication systems in a supplemental document

(BSC, 2007cf). The applicant also provided the specific routes for the DSEG from the surface to the subsurface facility. Furthermore, the applicant described the motion of the DSEG, including stops, inspection, and calibration of its precise position, which will closely resemble the TEV operational sequence toward the subsurface.

The applicant provided a schematic of the system and its envelope (SAR Figures 1.3.4-17 and 1.3.4-18) as well as a supplemental drawing (BSC, 2007ca). The diagrams included overall DSEG dimensions {923.9-cm [363.75-in]-wide × 467.4-cm [184-in]-long × 321.9-cm [126.75-in]-high}; maximum DSEG weight {90,718 kg [100 T]}; conceptual locations of major DSEG components; diametrical size {4.9 m [16 ft]} of the DSEG envelope relative to the drift's 5.5-m [18-ft]-diameter envelope; and the lifting features of the DSEG, illustrating the drip shield in its maximum nuclear safety design bases design lift height of 102 cm [40 in] inside the 15.5-m [18 ft]-diameter drift. The applicant also specified that additional clearance will be required at different locations of DSEG operations with appropriate justification.

The applicant further described the DSEG in BSC (2008bk, Section B.4.2). BSC (2008bk, Table B4.3-1) listed dependencies and interactions with other SSCs. The applicant's description provided design similarities between the DSEG and the TEV. The DSEG will rely on a number of infrastructure components, such as the subsurface crane rail {85 kg/m [171 lb/yd]}, several rail switches, radiation monitoring equipment, redundant third rail power, a communication system with the CCCF, and subsurface ventilation. The applicant will use ASME–NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as guidance for designing the DSEG; specifically, for defining structural construction, material selection, and operational limits for gantry cranes operating in nuclear facilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the description and design information for the DSEG as a part of the subsurface facility and its operations described in SAR Section 1.3.4.7.2 and supporting documents (BSC, 2007ca,cg–ci) using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the DSEG geometry and layout, design features, design specifications, and the intended functions, including emplacement activities. On the basis of this information, the NRC staff finds that the applicant's description and design information of the drip shield emplacement system are adequate because the applicant (i) discussed the functions of the system as it will lift, transport, and install drip shields in the emplacement drafts at the predetermined speed limits; (ii) described the major components of the DSEG system, including a steel frame structure, a lifting system, a self-propulsion system, a programmable logic controllers (PLCs) network communicating with the CCCF, an electrified third rail, air-conditioned cooled electronic cabinets to protect temperature-sensitive equipment; (iii) identified the dimensions for comparison of operating envelopes among the DSEG, the drip shield, and the drift openings; (iv) described the relationship between the DSEG motion and the interlocking requirements of the drip shields; (v) provided mechanical drawings showing basic design details, including equipment geometry and maximum weight; (vi) identified codes and standards for design, fabrication, material selection, and operational limits; and (vii) discussed dependencies and interactions with other SSCs, including subsurface crane rail, redundant third rail power, communication system with the CCCF, and subsurface ventilation. The NRC staff also finds that the applicant's information on the DSEG operations is adequate because it included key design specifications to (i) prevent contacting the outer surface of the drip shield, except at the lift points; (ii) provide only vertical lifting motion (no lateral motion); (iii) rely on human interaction/confirmation to control its operation; and (iv) operate at different design speeds during distinct modes of operation.

On the basis of the above evaluation, the NRC staff finds that the description and design information the applicant provided are sufficient to permit an evaluation of the PCSA and design of the drip shield emplacement system.

**NRC Staff' Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.3.6, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the drip shield emplacement system meet the requirements of 10 CFR 63.112(a), 10 CFR 63.21(c)(2), and 10 CFR 63.23(c)(3)(i) because the applicant provided an adequate description and design information for the drip shield emplacement system sufficient for the NRC staff to evaluate the applicant's PCSA and design.

### 2.1.1.2.3.4 Description of Waste Form Characteristics

The applicant provided information on the kind, amount, and specification of the radioactive material proposed to be received and possessed at the GROA. LLW produced as a result of GROA operations will be temporarily stored onsite but will not be disposed of at the GROA.

2.1.1.2.3.4.1 High-Level Radioactive Waste Characteristics

The applicant described commercial spent nuclear fuel (CSNF) in SAR Section 1.5.1.1, summarizing CSNF physical characteristics (SAR Section 1.5.1.1.1.1), thermal characteristics (SAR Section 1.5.1.1.2), nuclear characteristics (SAR Section 1.5.1.1.3), and source term characteristics (SAR Section 1.5.1.1.4). The applicant discussed how it used the CSNF characteristics in the PCSA (SAR Section 1.8.1.3.1) and created representative CSNF characteristics for the applicant's ALARA analysis (SAR Section 1.10.3.4.1).

The applicant described high-level radioactive waste (HLW) glass in SAR Section 1.5.1.2 with the physical, thermal, nuclear, and source term characteristics in SAR Sections 1.5.1.2.1.1, 1.5.1.2.2, 1.5.1.2.3, and 1.5.1.2.4, respectively. DOE spent nuclear fuel (SNF) will consist of SNF from numerous test and research reactors {$2.3 \times 10^6$ kg [2,265 MTHM]} and naval SNF {65,000 kg [65 MTHM]}. The applicant summarized the physical, thermal, nuclear, and source term characteristics in SAR Sections 1.5.1.3.1.1, 1.5.1.3.2, 1.5.1.3.3, and 1.5.1.3.4, respectively. The physical, thermal, nuclear, and source term characteristics of naval SNF were discussed in SAR Sections 1.5.1.4.1.1, 1.5.1.4.2, 1.5.1.4.3, and 1.5.1.4.4.

***Physical Characteristics of Radioactive Waste***

The CSNF inventory of the repository will be $63 \times 10^6$ kg [63,000 MTHM]. SAR Tables 1.5.1-2 and 1.5.1-3 summarized physical characteristics of pressurized water reactor (PWR) and boiling water reactor (BWR) assemblies. SAR Tables 1.5.1-4 and 1.5.1-5 presented the initial uranium load, enrichment, and burnup of CSNF assembly types and a summary of the initial uranium load, initial enrichment, and discharge burnup of the CSNF inventory.

The average PWR assembly is a Babcock & Wilcox 15 × 15 Mark B, and the average BWR assembly is a General Electric 2/3 8 × 8. SAR Section 2.3.7.4 discussed the CSNF radionuclide inventory used in the total system performance assessment (TSPA). The distribution of radionuclides in the $UO_2$ matrix was summarized in SAR Section 2.3.7.7.1. Most radionuclides would be retained in the $UO_2$ matrix, but some of the more mobile fission products and activation products would accumulate in gap regions and grain boundaries. In

SAR Section 2.3.7.7.3.1, the applicant discussed these isotopes, which are available for instantaneous release when the cladding is breached. The applicant used the rod breakage fraction of CSNF in evaluating radionuclide isotopes used in normal and accident conditions.

HLW glass is highly radioactive waste that was mixed with silica and/or other glass-forming chemicals that were melted and poured into canisters where they solidified into glass. The chemical composition of the glass was listed in SAR Table 1.5.1-14.

SAR Table 1.5.1-15 listed the approximate mass of HLW per canister for each site (Hanford, Savannah River, Idaho National Laboratory, and West Valley). The applicant used 500 kg [0.5 MTHM] per canister equivalence for DOE HLW to determine how many canisters can be disposed of at the repository. The applicant expects to receive approximately 9,300 DOE HLW canisters containing a total of $4.7 \times 10^6$ kg [4,667 MTHM]. A 2,300 kg [2.3 MTHM] per canister equivalence was used for the 275 commercial HLW glass canisters from West Valley with a total of approximately 640,000 kg [640 MTHM] of HLW.

The DOE SNF waste form comes from a range of backgrounds with a variety of fuel types, moderators, enrichments, shapes, and chemistries. The approximately $2.3 \times 10^6$ kg [2,265 MTHM] of DOE SNF proposed for disposal at the Yucca Mountain site may be stored in 2,500 to 5,000 DOE canisters. In SAR Section 1.5.1.3.1.1.1, the applicant developed 34 groups of DOE SNF, including naval SNF, that are based on the characteristics the applicant believes have the greatest impact on release and criticality. SAR Table 1.5.1-23 listed these 34 fuel groups and described how they were analyzed. SAR Table 1.5.1-24 described the ranges of the properties of the 34 groups. In SAR Section 1.5.1.3.1, the applicant stated that approximately 98 percent of DOE SNF will be shipped in sealed canisters and the remaining amount of DOE SNF (approximately 2 percent) has intact cladding and requires no special handling considerations compared to CSNF.

Naval SNF has been allocated 65,000 kg [65 MTHM] for proposed disposal at the repository. Naval fuel is uranium metal highly enriched in U-235 and, as a result, contains very small amounts of transuranics compared to commercial spent nuclear fuel (CSNF). The applicant stated in SAR Section 1.5.1.4.1.1, “In a few cases, destructive evaluations of disassembled components result in nonintact cladding and exposed fission products and actinides; some test specimens have nonintact cladding because they were intentionally tested until the cladding failed.” The applicant modeled Naval SNF as CSNF, which did not take credit for cladding. Structural components made of Alloys 600, 625, X-60, or SS304 provide support to the assemblies in the canister.

**NRC Staff’s Evaluation**

The NRC staff reviewed the applicant’s description of physical characteristics of HLW using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the inventory and waste form compositions for the commercial spent nuclear fuel, HLW, DOE SNF, and naval SNF. The NRC staff also reviewed the applicant’s description of grouping the DOE SNF waste forms. The NRC staff finds that the applicant adequately described the waste form composition and amount because (i) the amount allocated for disposal will not exceed the Nuclear Waste Policy Act limit of $7 \times 10^7$ kg [70,000 MTHM] (note that this limit is addressed further in SER Volume 5) and (ii) the applicant provided detailed descriptions of the type of SNF and HLW, including physical dimensions, initial uranium loading, enrichment, burnup, crud deposits on cladding, thermal input, radionuclide inventory, chemical characteristics, and post-irradiation cooling time. In addition, the NRC staff finds that the applicant’s use of rod-breakage fractions is acceptable because use of rod-breakage fractions is consistent with Interim Staff Guidance (ISG) 5 (NRC, 2000af, p. 7). Although the applicant did not discuss burnable poison absorbers or integral burnable poison absorbers that may remain in the fuel, it did not take credit for the neutron-absorbing properties of these absorbers. The NRC staff finds not taking credit for burnable poison absorbers to be conservative because taking credit for neutron absorption by poison absorbers would reduce the margin in the criticality assessment. Evaluation of the means to prevent and control criticality is discussed in SER Section 2.1.1.6. The applicant bounded the cooling time with PSC-20, which provides for at least 5 years' cooling time, as per 10 CFR Part 961. The NRC staff finds this specified cooling time acceptable and conservative because the applicant will not accept any commercial SNF at the GROA with a cooling time shorter than 5 years, and a large amount of the commercial SNF currently stored at the reactor sites have cooling times longer than 5 years.

The NRC staff reviewed the applicant's description of grouping DOE SNF waste forms in the SAR and supporting documents (DOE–Idaho, 2000aa) and finds the applicant's description of the DOE SNF groupings acceptable because these groupings were based on the fuel properties that were most important to the design and safety analyses in terms of their influence on design basis events and radionuclide releases, nuclear criticality scenarios, and the total system performance assessment. The NRC staff also finds the applicant's description of the range of waste form characteristics adequate because the range of the waste form characteristics includes parameters important to safety and important to repository performance. The NRC staff also finds that creating groups of similar types of DOE SNF and using representatives of the groups for analysis purposes is acceptable because the process used to define group representatives will not underestimate risk.

The NRC staff reviewed the applicant's description of naval SNF and characterization of naval SNF with non-intact cladding. Because the cladding of naval SNF will not be relied upon for safety and the applicant-modeled naval SNF as CSNF, which did not credit cladding, the NRC staff finds the small amount of nonintact cladding is bounded by the CSNF assumption. Therefore, the applicant's description of damaged fuel cladding is acceptable.

***Thermal Characteristics***

SAR Table 1.5.1-11 provided the average (25 years' cooling time) and maximum thermal power (5 years' cooling time) for PWR and BWR assemblies. SAR Figure 1.5.1-6 showed thermal power per assembly as a function of time. The applicant chose to analyze limiting values so that the uncertainties were bounded by the maximum cases.

The applicant calculated the heat generation rate from the high-level radioactive waste (HLW) radionuclide inventory and displayed the results in SAR Table 1.5.1-19. The applicant stated that it will impose a limit that the maximum allowable canister temperature and maximum allowable heat generation rate will be 400 °C [752 °F] and 1.5 kW/canister (1.4 Btu/sec./canister), respectively. HLW canisters that do not meet this limit will not be disposed of in the repository (SAR Table 5.10-3).

SAR Table 1.5.1-28 provided the nominal and bounding estimated decay heat of all DOE SNF canisters in 2010 and 2030. The applicant stated that it will impose a limit on the heat generation rate of DOE SNF canisters to less than 1,970 watts/canister. The DOE SNF canisters that do not meet this limit will not be disposed of in the repository (SAR Table 5.10-3).

According to the applicant, naval SNF canisters will not be shipped until the heat output, when received at the GROA, is less than or equal to 11.8 kW. Those that do not meet this limit will not be disposed of (SAR Table 5.10-3). For preclosure, the applicant stated that the canister surface in the emplacement drift should be not greater than 160 and 204 °C [320 and 400 °F] for normal and loss-of-ventilation conditions, respectively. SAR Figure 1.3.1-8 showed the surface temperature of a naval SNF canister as a function of time after emplacement.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of thermal characteristics and heat generation rate using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the thermal power for PWR and BWR assemblies, calculated the heat generation rate from the HLW radionuclide inventory, and estimated decay heat of all DOE SNF canisters and allowable heat output for the naval SNF canisters. The NRC staff finds the applicant's description of thermal characteristics is sufficient to permit an evaluation of the PCSA and design because (i) the description included adequate information of thermal power for the PWR and BWR assemblies, heat generation rate from HLW radioactive inventory, and decay heat of DOE SNF canisters; (ii) the description contained the commercial spent nuclear fuel (CSNF) heat generation rate; (iii) the thermal characteristics included conservatism to bound uncertainties to permit an NRC staff evaluation of the applicant's thermal calculations; and (iv) the contents of all canisters for disposal will have thermal characteristics less than the aforementioned applicant-imposed limits, and actual cases will be bounded by the applicant's thermal calculation results because the applicant chose to analyze limiting values.

***Nuclear Characteristics***

The applicant used the SCALE computer software to calculate the nuclear characteristics of the CSNF (SAR Section 1.5.1.1.3). SAR Table 1.5.1-12 recorded the amount each radionuclide in the assembly contributes to the radioactivity for the average and bounding PWR and BWR assemblies. The radionuclides in the table included those from the fuel section, top and bottom end fittings, fuel plenum, and crud (SAR Section 1.5.1.1.3). SAR Figure 1.5.1-7 showed the activity per assembly as a function of time for the average and bounding assemblies. SAR Table 1.5.1-20 provided the radionuclide inventories for HLW from each site in 2017. SAR Table 1.5.1-21 provided the maximum radionuclide inventories for each canister type. These values were used as inputs to source term and thermal calculations. The maximum allowed fissile isotope concentrations were shown in SAR Table 1.14-1. SAR Table 1.8-5 listed the values the applicant used for its HLW glass consequence analysis.

The applicant used ORIGEN, part of the SCALE software package, to develop a template of radionuclide inventories, at 10-decay intervals between 5 and 100 years, for typical SNF, which was scaled based on burnup and fuel mass to get approximate radionuclide inventories for similar fuels. The inventories of the template contain 145 radionuclides that account for 99.9 percent of the total curie inventory of the DOE SNF. DOE-Idaho (2004aa) described how the radionuclide inventory was calculated. The projected total inventories were listed in SAR Table 1.5.1-29 for nominal and bounding cases in 2010.

For purposes of criticality evaluations, the applicant sorted the DOE SNF into nine groups with a representative DOE SNF for each group. The groups and their representatives were listed in SAR Section 1.5.1.3.1.1.3 and analyzed in SAR Sections 1.14.2.3.2.3 and 2.2.1.4.1 for preclosure and postclosure, respectively. The applicant listed the postclosure critical limits for the nine groups of DOE SNF analyzed for criticality purposes in SAR Table 2.2-11.

SAR Table 1.5.1-32 presented an initial radionuclide inventory developed for a representative naval SNF canister, with an assumed cooling time of 5 years.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the radionuclide inventories using the guidance in YMRP Section 2.1.1.2. The NRC staff focused its review on the methods the applicant used to generate the radionuclide inventories and conservatisms in the models and calculations. The NRC staff finds the applicant's use of SCALE acceptable because use of this software package is consistent with the standard engineering practice for analyzing radionuclide inventories and is consistent with NRC guidance for source term, criticality, and shielding analyses (refer to SER Section 2.1.1.5.3.2 for additional discussion on the acceptability of the SCALE software).

***Source Term Characteristics***

The applicant considered the PWR fuel assembly to be bounding and used it in shielding design (e.g., worker dose assessments, process facility design, ALARA) and repository consequence analysis for preclosure. In SAR Table 1.10-18, the applicant provided the radiation sources from the maximum PWR assembly {5 wt% initial enrichment, 0.08 GWd/kg [80 GWd/MTU] burnup, and 5-year cooling}, which will be used in shielding calculations for permanent structural components because it represents the bounding fuel assembly. SAR Table 1.10-19 provided the radiation sources of the design basis PWR assembly {4 wt% initial enrichment, 0.06 GWd/kg [60 GWd/MTU] burnup, 10-year cooling}, which the applicant claims will bound at least 95 percent of the fuel inventory, and it will be used in shielding calculations for some transfer shield designs to limit shield weight. For normal operation airborne releases, the applicant used representative PWR {4.2 wt% initial enrichment, 0.05 GWd/kg [50 GWd/MTU] burnup, and 10-year cooling} and BWR {4 wt% initial enrichment, 0.05 GWd/kg [50 GWd/MTU] burnup, and 10-year cooling} assemblies to generate the radionuclide inventories in SAR Table 1.8-3. For airborne releases from Category 1 and Category 2 event sequences, radionuclide inventories (see SAR Table 1.8-3) from the maximum assemblies were used.

The applicant stated that no Category 1 event sequences for HLW glass were identified by its PCSA. Potential doses to the public were discussed in SAR Section 1.8.3.2 and doses to workers in SAR Section 1.8.4. The applicant stated that the maximum per canister inventories were provided in SAR Table 1.5.1-21 and used in the PCSA. The applicant also stated that the Savannah River Site HLW canister would represent the bounding glass compositions from a dose perspective (SAR Section 1.5.1.2.4). By limiting the radionuclide inventory to the values in SAR Table 1.8-5, PSC-21 will ensure that the dose limits in SAR Tables 1.8-30 and 31 are met.

The applicant discussed DOE SNF shielding source term characteristics in SAR Section 1.10.3. Neutron and gamma energy spectra and source intensity and fuel composition the applicant used in shielding calculations for a homogenized TRIGA-FLIP fuel were presented in SAR Sections 1.10.3.3.2.3 and 1.10.3.4.3 and Tables 1.10-14 and 1.10-23. The TRIGA-FLIP fuel was used because it bounds other DOE SNF waste forms from a shielding and dose perspective.

In SAR Section 1.5.1.4.1.2.6.4, the applicant discussed the gamma and neutron source terms for the naval SNF canisters, which the applicant considered in the initial handling facility (IHF) design. The source term assumed a cooling time of 5 years, and it was increased by 30 percent to provide extra margin. The gamma and neutron source terms were listed in

SAR Tables 1.10-21 and 1.10-22, respectively. The applicant determined that breaches of DOE SNF and naval SNF canisters were beyond Category 2 event sequences; therefore, the applicant did not develop a source term to analyze doses from these canisters.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the radiation source term using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the radiation sources of CSNF, radionuclide inventories for the HLW canisters, DOE SNF shielding source term characteristics, and gamma and neutron source terms for naval SNF canisters. The NRC staff finds the description of the source term is acceptable because it included conservatisms and sufficient detail for the NRC staff to evaluate the applicant's shielding and dose calculations, as documented in SER Section 2.1.1.5.3.2. The NRC staff confirmed that the Savannah River Site HLW is bounding, with respect to dose, by calculating the radioactivity per unit mass using the information in SAR Tables 1.5.1-15 and 1.5.1-21. The NRC staff also finds that dividing the DOE SNF into groups with only a representative sample of the group being analyzed is adequate given that the other members of the DOE SNF groups will still be subjected to the applicant's waste form and waste package qualification program, as specified in SAR Table 5.10-3.

The waste form characteristics evaluated in this SER section formed the bases for the applicant's PCSA. The NRC staff finds that the description of source term characteristics used in the PCSA is adequate because the applicant provided (i) its basis for the limiting values and/or bounding values of waste form characteristics used in its thermal designs (i.e., conservative assumption of at least 5 years of cooling for commercial SNF) and (ii) assumptions regarding heat generation rate in SAR Table 1.5.1-11 and thermal design limits on cladding and canister temperature relevant to the SNF waste form (SAR Section 1.5.1.1.1.2.5.3) that are based on relevant data.

**NRC Staff's Conclusion**

On the basis of the evaluations discussed in SER Section 2.1.1.2.3.4.1, the NRC staff finds, with reasonable assurance, that the applicant's description of HLW characteristics meets the requirements of 10 CFR 63.21(c)(4) and 10 CFR 63.112(a) because the applicant provided an adequate description of HLW characteristics sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.4.2 Description of Low-Level Radioactive Waste

The applicant described how it intends to handle and process low-level radioactive waste (LLW) that will be produced at the GROA in SAR Section 1.4.5.1. More specifically, the applicant discussed solid LLW in SAR Section 1.4.5.1.1.1, liquid LLW in SAR Section 1.4.5.1.1.2, and gaseous LLW in SAR Section 1.4.5.1.1.3. SAR Table 1.4.5-1 listed the expected annual LLW volumes. SAR Table 1.4.5-2 listed the expected LLW radionuclide concentration. SAR Section 1.10.3.4.5 provided the LLW source terms. The applicant considered LLW when performing the PCSA and assumed that containers used to transport LLW would lose containment after a structural challenge (SAR Section 1.7.2.3.1) or a fire (SAR Section 1.7.2.3.3.1). The resulting event sequences involving LLW were provided in SAR Table 1.7-19. The applicant assessed the consequences of these event sequences assuming unfiltered radionuclide release and identified no significant worker exposures (SAR Section 1.7.5).

**NRC Staff's Evaluation**

The NRC staff reviewed the LLW description using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the characteristics for the solid, liquid, and gaseous LLW and LLW handling operations. The NRC staff finds that the applicant's description of LLW is adequate because the applicant provided (i) a discussion of the activities that would generate solid, liquid, and gaseous LLW; (ii) the anticipated annual volume of LLW for each of the relevant repository facilities (e.g., wet handling facility, receipt facility, aging facility); (iii) a description of how LLW would be collected, handled, and, as applicable, transferred to a LLW facility; and (iv) information on how LLW handling facilities will be designed to withstand event sequences, including natural phenomena. Thus, the NRC staff finds that the description of LLW the applicant provided is sufficient to support an evaluation of the PCSA and design.

**NRC Staff's Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.4.2, the NRC staff finds, with reasonable assurance, that the applicant's description of LLW meets the requirements of 10 CFR 63.21(c)(4) and 10 CFR 63.112(a) because the applicant provided an adequate description of LLW sufficient for the NRC staff to evaluate the applicant's PCSA and design.

**2.1.1.2.3.5 Waste Package, Canisters, Casks, and Engineered Barrier System Components**

This section provides the NRC staff's review and evaluation of the applicant's overview description of canisters, casks, and the engineered barrier system (EBS). The EBS will be composed of the waste package, waste package emplacement pallet, drip shield, and invert structure. The following four sections detail the NRC staff's evaluation of the (i) waste package, (ii) waste canisters, (iii) aging overpack and shielded transfer cask, and (iv) drip shield descriptions.

2.1.1.2.3.5.1 Waste Packages

The applicant described and discussed the waste package design in SAR Section 1.5.2 and other applicable sections of the SAR (e.g., 1.2.1.4.1, 1.2.4.2.3.1.3, 1.3.1.2.5, and 2.3.6.7.4). The applicant proposed to use the waste package as an engineered barrier for disposal of SNF and HLW. Waste packages would be loaded with TAD, HLW, DOE, and naval SNF canisters at the surface facilities. The applicant classified waste packages as ITS because they will be relied upon to protect against the release of radioactive gases or particulates during normal operations and Category 1 and Category 2 event sequences. Moreover, the applicant classified waste packages as important to waste isolation (ITWI) because, after repository closure, the waste package will be relied upon to meet the postclosure performance objectives.

The applicant described six waste package configurations: (i) waste package loaded with one 21-PWR/44-BWR TAD canister, (ii) waste package loaded with five short HLW canisters and one short DOE SNF canister in the center location (5-DHLW/DOE short codisposal), (iii) waste package loaded with five long HLW canisters and one long DOE SNF canister in the center location (5-DHLW/DOE long codisposal), (iv) waste package loaded with two DOE multicanister overpacks (MCO) and two long HLW canisters (2-MCO/2-DHLW), (v) waste package loaded with one short naval SNF canister, and (vi) waste package loaded with one long naval SNF canister.

The approximate percentage, by waste package configuration, was provided in SAR Table 1.5.2-2. According to this table, the configuration of the waste package with one 21-PWR or one 44-BWR TAD canister will be the most commonly used configuration, which will account for approximately 71 percent of all waste packages. Also, the waste package configurations with five long HLW and one long DOE SNF canisters or five short HLW and one short DOE SNF canister will account for approximately 23 percent of the waste packages.

All waste package configurations will have a single design that will consist of two concentric cylinders (i.e., inner vessel and outer corrosion barrier) with the upper and lower sleeves on the end of the outer corrosion barrier for additional structural support. The inner vessel will include an inner cylinder, bottom inner lid, and top closure inner lid. The outer corrosion barrier will include an outer cylinder, bottom outer lid, and top closure outer lid. In addition, a purge port will be added to the top closure inner lid, and the inner vessel will be helium filled (SAR Figures 1.5.2-3 through 1.5.2-8).

The applicant used codes and standards typically used in the industry for the waste package design. As the applicant specified, the inner vessel will be designed for internal pressure and deadweight loads, in accordance with the provisions of the 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC for Class 2 components (American Society of Mechanical Engineers, 2001aa). The applicant stated that the inner vessel will be stamped with an N symbol and, therefore, will be identified as a pressure vessel that will undergo stringent quality assurance controls. Furthermore, the outer corrosion barrier will be designed with applicable technical requirements of the 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC for Class 2 components. However, according to the applicant, the outer corrosion barrier will not be stamped with an N symbol and, therefore, will not be identified as a pressure vessel.

Although all waste packages will have a single design, different waste package configurations will have multiple internal structures and different external dimensions to accommodate various waste forms. According to SAR Tables 1.5.2-3 and 1.5.2-5, the waste package nominal length will range from 369.7 to 585.0 cm [145.57 to 230.32 in], the nominal diameter will range from 183.1 to 212.60 cm [72.07 to 83.70 in], and the maximum loaded weight will range from 40,800 to 73,500 kg [90,000 to 162,000 lb].

The applicant stated that the materials used for the waste package will meet the requirements of the 2001 ASME Boiler and Pressure Vessel Code, Section II (American Society of Mechanical Engineers, 2001aa). The material of construction for the inner vessel will be identified as ASME SA-240 (UNS S31600) with additional controls on nitrogen and carbon, referred to as Stainless Steel 316. The material of construction for the outer corrosion barrier and the upper and lower sleeves will be identified as ASME SB-575 (UNS N06022) with limited constituents of 20.0 to 21.4 percent chromium, 12.5 to 13.5 percent molybdenum, 2.5 to 3.0 percent tungsten, and 2.0 to 4.5 percent iron, referred to as Alloy 22 (SAR Section 2.3.6.7.4). The material used for divider plates and support tubes for 5-DHLW/DOE short codisposal, 5-DHLW/DOE long codisposal, and 2-MCO/2-DHLW waste package configurations will be carbon steel SA 516 (UNS K02700).

According to the applicant, fabrication materials and processes will conform to the requirements of the 2001 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa), as follows: (i) the welding processes used on the inner vessel and the outer corrosion barrier (identified as gas tungsten arc and gas metal arc methods) are in accordance with NC-4000 Sections IX and Section III, Division 1; (ii) the welding filler materials are in accordance with NC-2400 Section III, Division 1; (iii) the heat treatment procedure is in accordance with NC-4600 Section III, Division 1; (iv) the examination of welds for the inner vessel and the outer corrosion barrier is in accordance with NC-5000 Section III, Division 1; (v) the hydrostatic and pneumatic testing of the inner vessel is in accordance with NC-6220 Section III, Division 1 and NC-6324; and (vi) the helium leakage test of the inner vessel is in accordance with Section V, Article 10, Appendix IX.

**NRC Staff's Evaluation**

The NRC staff reviewed the description of the waste package and its components using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the principal characteristics, functional features, and design information of the waste packages. The NRC staff finds that not stamping the outer corrosion barrier with an N symbol is acceptable because the applicant will not take credit for the outer corrosion barrier to function as a structural component and therefore, an N symbol certification is not needed. The NRC staff finds that the applicant defined the principal characteristics of the waste package because the applicant (i) discussed the principal functions of waste packages as they will be relied upon to prevent the release of radioactive gases or particulates during normal operations and Category 1 and Category 2 event sequences and will be an engineered barrier for disposal of SNF and HLW; (ii) described the six configurations for housing nuclear waste canisters of various types and numbers of canisters; (iii) discussed the waste package configuration that will be most commonly used (i.e., containing one 21-PWR or one 44-BWR TAD canister accounting for approximately 71 percent of all waste packages); (iv) provided dimensions and maximum loaded weights of the waste package {the nominal length ranging from 369.7 to 585.0 cm [145.57 to 230.32 in] and the nominal diameter ranging from 183.1 to 212.60 cm [72.07 to 83.70 in], and the maximum loaded weight ranging from 40,800 to 73,500 kg [90,000 to 162,000 lb]}; (v) discussed internal structures (e.g., divider plates and support tubes) of the waste package for various waste forms; and (vi) identified codes and standards for design, fabrication, material selection, welding process, heat treatment, helium leakage tests of the waste package, and the inner vessel design for internal pressure and deadweight loads that are consistent with the standard engineering practices.

The NRC staff finds that the applicant's characterization of functional features of the waste package is adequate because (i) the applicant defined the waste package inner vessel as a pressure vessel, which will be made of Stainless Steel 316; (ii) the applicant defined the waste package outer layer as a corrosion-resistance barrier that will be made of Alloy 22; (iii) the waste package will be designed to accommodate internal pressurization of the waste package, including effects of a high temperature of 350°C (662$^0$F)and fuel rod gas release (SAR Table 1.5.2-7); and (iv) a loaded waste package will maintain structural integrity during various design basis events.

On the basis of the above evaluation, the NRC staff finds that the principal characteristics and functional features the applicant provided are sufficient to permit an evaluation of the PCSA and design of the waste package.

**NRC Staff's Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.5.1, the NRC staff finds, with reasonable assurance, that the applicant's description of the waste package meets the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description of the waste package sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.5.2 Waste Canisters

The applicant described the design of waste forms and waste packages in SAR Section 1.5. The applicant used this information in its PCSA and design of the waste canisters (SAR Sections 1.6 through 1.9). The SNF and vitrified HLW will be shipped to Yucca Mountain in TAD canisters, DOE standardized canisters, HLW canisters, and dual-purpose canisters (DPCs). In addition, the naval SNF will be shipped to Yucca Mountain in naval SNF canisters. On the basis of the PCSA, the applicant designated these waste canisters as ITS.

***TAD Canisters***

The applicant provided performance specifications in SAR Section 1.5.1.1.1.2.1.3 for the transportation, aging, and disposal (TAD) canisters. These specifications were generally based upon nuclear safety design bases developed from the PCSA and/or transportation and storage requirements. In SAR Figure 1.5.1-5, the applicant also provided a conceptual drawing of the TAD canister. The information the applicant provided included key dimensions, weights, fabrication specifications and materials of construction, sealing (welding) specifications, and descriptions of drying processes. In SAR Section 1.5.1.1.1.2.1.4, the applicant presented the principal physical characteristics of the proposed TAD canister. The TAD canister will have a diameter of 1,689 mm [66.5 in], a minimum height of 4,724 mm [186.0 in], and a maximum height of 5,385 mm [212.0 in]. For a TAD canister with a height less than the maximum height, a TAD waste package spacer will be used to restrict the axial movement of the canister while in the waste package. The maximum loaded weight of the TAD canister, including the TAD spacer, will not exceed 49,215 kg [54.25 T]. To facilitate underwater handling, the applicant specified that the TAD lid will be designed for underwater handling, and the canister and lid can be centered while submerged. The TAD canister will have lifting features to allow overhead lifting when the canister is open, empty, and vertical, while a closed, loaded TAD canister should be capable of being lifted by its lid.

The applicant specified that the fabrication of the TAD canister shell and lid will follow 2004 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (American Society of Mechanical Engineers, 2004aa). The TAD canister will be constructed of Type 300-series stainless steel, as per ASTM A 276–06 (ASTM International, 2006ab), for the canister shell and structural internals. The applicant chose this material because of its resistance to degradation. The applicant stated that the TAD canister and its basket materials are required to be compatible with either borated or unborated pool water because the canister will be submerged during fuel loading at the repository and/or reactor sites. With respect to the canister internals, the applicant specified that the neutron absorbers necessary for criticality safety control will be fabricated from borated stainless steel with a boron content of 1.1 wt% to 1.2 wt% and will meet ASTM A 887-89, Grade "A" alloys (ASTM International, 2004ab). The neutron absorber plates will have a minimum thickness of 11 mm [0.4375 in], while the nominal thickness will be based on structural requirements to maintain the stored geometry of the spent nuclear fuel (SNF) inside the canister. The length of the neutron absorber plates will cover the full length of the active fuel region, to account for any axial shifting of the SNF assemblies within the TAD canister.

In SAR Section 1.5.1.1.1.2.6.1.2, the applicant described the TAD canister containment characteristics pertaining to welding of the TAD lid. The applicant stated that the TAD canister design will meet either of two requirements: (i) welding specifications will be in accordance with Interim Staff Guidance–18 (NRC, 2003af) or (ii) the TAD closure welds will be helium leak tested using procedures that conform to the requirements in ANSI N14.5–97 (American National Standards Institute, 1998aa).

The applicant specified it will use helium to inert the TAD canister to prevent SNF cladding oxidation and limit the cladding temperature to be less than 570 °C [1,058 °F] during draining, drying, and backfill operations. SAR Section 1.2.5.3.5 described the TAD canister drying and inerting systems, which will consist of a generic, forced helium dehydrator system package, and a traditional vacuum drying system.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description for TAD canisters using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the physical dimensions (internal and external) and functional features of the TAD canister and compared them with waste package dimensions. The NRC staff also reviewed consistency between the dimensions of the proposed SNF packages to be placed inside TAD canisters with the internal dimensions of the canisters. In addition, the NRC staff reviewed the consistency of information on material, specifications, and codes proposed for TAD canister design. The NRC staff finds that the principal characteristics of the TAD canister are defined because the applicant (i) provided a conceptual drawing of the TAD canister; (ii) provided TAD dimensions {1,689 mm [66.5 in] in diameter, a minimum height of 4,724 mm [186.0 in], and a maximum height of 5,385 mm [212.0 in]}; (iii) specified the maximum loaded weight of the TAD canister, including the TAD spacer, to not exceed 49,215 kg [54.25 T]; (iv) specified applicable codes and standards for material selection and fabrication of TAD shell, lid, and canister internals; and (v) specified applicable codes and standards for welding of the TAD canister lid. The NRC staff also finds the drying and backfilling information adequate because drying and backfilling with helium are consistent with standard engineering practices, and the applicant indicated that it will follow the guidance given in NUREG–1536 (NRC, 1997ae) for draining and drying the TAD canisters. The NRC staff finds that the applicant's statement that the TAD canister design will meet the standards of either ANSI N14.5–97 (American National Standards Institute, 1998aa) or NRC Interim Staff Guidance 18 (NRC, 2003af) acceptable because (i) ANSI N14.5–97 is an industry-accepted standard for leakage tests on packages for shipment and (ii) NRC ISG–18 provides an alternate method to meet ANSI N14.5–97 standards.

The NRC staff finds that the applicant provided adequate characterization of the functional features of the TAD canisters because (i) the TAD canisters will be designed for underwater handling; (ii) the TAD canisters will be resistant to degradation by using Type 300-series stainless steel, as per ASTM A 276–06 (ASTM International, 2006ab), as its material of construction; (iii) the TAD canister and its basket materials will be compatible with either borated or unborated pool water because the canister will be submerged during fuel loading; (iv) the canister internals will include neutron-absorber plates for criticality safety control; and (v) the TAD canisters will provide a containment function by welding the lid to the TAD canister.

On the basis of the above evaluation, the NRC staff finds that the principal characteristics and functional features are sufficient to permit an evaluation of the PCSA and design of the TAD canisters.

***DOE Standardized Canister***

SAR Section 1.5.1.3.1.2.1.1 presented the design description of the DOE standardized canister. There will be four different DOE standardized canisters, but the applicant stated that the functions and requirements will be the same. The applicant specified that the standardized canister will have two different diameters with differing wall thicknesses: (i) a large-diameter standardized canister will have an outer diameter of 610 mm [24 in] and a wall thickness of 12.7 mm [0.5 in] and a (ii) small-diameter standardized canister will have an outer diameter of 457 mm [18 in] and wall thickness of 9.525 mm [0.375 in]. The applicant stated that these two standardized canisters will have two lengths: 3.1 and 4.6 m [10 and 15 ft]. The maximum allowable weight of the standardized canister including its contents will be approximately 4,536 kg [10,000 lb] for the 610-mm [24-in]-diameter, 4.6-m [15-ft] canister; 4,082 kg [9,000 lb] for the 610-mm [24-in]-diameter, 3.1-m [10-ft] canister; 2,722 kg [6,000 lb] for the 457-mm [18-in]-diameter, 4.6-m [15-ft] canister; and 2,268 kg [5,000 lb] for the 457-mm [18-in]-diameter, 3.1-m [10-ft] canister. SAR Figure 1.5.1-9 showed a representative small-diameter DOE standardized canister. The standardized canister will include an integral, energy-absorbing skirt and will have a lifting ring integral within the skirt (Section 1.5.1.3.1.2.1.1 of SAR). During a drop event, the energy-absorbing skirt will deform, which reduces drop-induced damage to the standardized canister containment barrier. The standardized canisters will be fabricated from Stainless Steel Type 316L SA-312 welded or seamless pipe for the shell, while Stainless Steel Type 316L SA-240 plate will be used for the heads and lift rings. For the canisters with the optional plugs, the plugs will be fabricated from Stainless Steel Type 316L SA-479 (bar).

In SAR Section 1.5.1.3.1.2.2, the applicant described the operational processes that it will use for drying, sealing, inerting, and leak testing the canisters. The applicant specified that the inerting process will utilize an inert gas such as helium. In terms of sealing, the standardized canister boundary components will be joined with full-penetration welds. The applicant stated that these welds will meet the requirements of the 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 3, Subsections WA and WB (American Society of Mechanical Engineers, 2001aa). The applicant stated that the type of weld inspection will be a volumetric inspection using ultrasonic testing. The final closure weld will be performed using an ASME-acceptable welding procedure. Prior to transportation to the repository, any required threaded plugs will be installed and seal welded in place to establish an ASME-acceptable containment boundary. The applicant will demonstrate leak tightness by utilizing a helium leak test in accordance with 2001 ASME Boiler and Pressure Vessel Code, Section V, Article 10, Appendix IV (American Society of Mechanical Engineers, 2001aa).

In SAR Section 1.5.1.3.1.2.6.1, the applicant presented information on the structural design methodology used for the standardized canister. The applicant stated that standardized canisters will be designed to the 1998 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998aa). The applicant also described the finite element analyses performed on the canisters by which the design can be evaluated.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of DOE standardized canisters using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the principal characteristics of DOE standardized canisters; structural design methodology to be used; numerical analyses for the DOE standardized canisters; and the operational processes to be used for drying, sealing, inerting, and leak testing the DOE standardized canisters. Based on the NRC staff's evaluation of the information presented, the NRC staff finds that the principal characteristics of the DOE standardized canisters are defined because the applicant (i) provided a conceptual drawing of the DOE standardized canister, including components; (ii) specified two standardized canister sizes {one with an outer diameter of 610 mm [24 in] and a wall thickness of 12.7 mm [0.5 in] and the other with an outer diameter of 457 mm [18 in] and wall thickness of 9.525 mm [0.375 in]}; (iii) specified the maximum allowable weight of the DOE standardized canister, including its contents ranging from 2,268 kg [5,000 lb] to 4,536 kg [10,000 lb]; (iv) specified applicable codes and standards for material selection and fabrication of the DOE standardized canister shell, head, lift ring and optional plugs; (v) specified applicable codes and standards for design and welding of the DOE standardized canisters; and (vi) described the operational processes that it will use for drying, sealing, inerting, and leak testing the DOE standardized canisters.

The NRC staff finds that the applicant provided adequate characterization of the functional features of the DOE standardized canisters because (i) the applicant discussed an integral, energy-absorbing skirt of the DOE standardized canisters that will reduce drop-induced damage to the standardized canister containment and (ii) the DOE standardized canister basket assemblies with neutron absorber materials added to the design will provide criticality control.

On the basis of the above evaluation, the NRC staff finds that the principal characteristics and functional features are sufficient to permit an evaluation of the PCSA and design of the DOE standardized canisters.

***HLW Canister***

SAR Section 1.5.1.2.1.2 described four high-level radioactive waste (HLW) canisters: (i) Hanford canisters with a nominal outside diameter of 0.7 m [2 ft] and a nominal height of 4.6 m [15 ft], (ii) Savannah River Site canisters with a nominal outside diameter of 0.7 m [2 ft] and a nominal height of 3.1 m [10 ft], (iii) Idaho National Laboratory canisters with a nominal outside diameter of 0.7 m [2 ft] and a nominal height of 3.1 m [10 ft], and (iv) West Valley Demonstration Project canisters with a nominal outside diameter of 0.7 m [2 ft] and a nominal height of 3.1 m [10 ft]. SAR Figure 1.5.1-8 showed the types of HLW canisters mentioned. In SAR Table 1.5.1-16, the applicant showed the physical characteristics of each HLW canister, including length, outside diameter, wall thickness, and material type. The applicant stated that it is not necessary for HLW canister criticality controls during normal repository operations and waste emplacement, because the fissile radionuclides in each HLW canister have low concentrations (SAR Section 1.5.1.2.1.2.4).

SAR Section 1.5.1.2.1.7 provided the design codes and standards for the HLW canisters. The applicant provided the materials of construction, welding, weld testing, and leak testing in SAR Table 1.5.1-18 for the HLW canisters. Specifically, the canisters will be fabricated from austenitic stainless steel and welded in accordance with the 2001 ASME Boiler and Pressure Vessel Code, Section IX (American Society of Mechanical Engineers, 2001aa). The nondestructive evaluation of the canister welds for the Hanford, Idaho, National Laboratory and Savannah River Site canisters is specified to be radiographic examination of all full penetration butt welds, in accordance with 2001 ASME Boiler and Pressure Vessel Code, Section V (American Society of Mechanical Engineers, 2001aa). The nondestructive evaluation of the canister welds for the West Valley Demonstration Project is a dye penetration examination of all fabrication welds, in accordance with 2001 ASME Boiler and Pressure Vessel Code, Section V (American Society of Mechanical Engineers, 2001aa). Further, DOE stated that the canister welds will be required to pass pressure and helium leak tests. Full-scale testing of HLW canisters is evaluated in SER Section 2.1.1.7.3.9.3.2.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of high-level radioactive waste (HLW) canisters using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the description of the principal characteristics of the HLW canisters and the description of design information. Based on the NRC staff's evaluation of the information presented, the NRC staff finds the principal characteristics of the HLW canisters are defined because the applicant (i) provided a conceptual drawing of two types of the HLW canisters; (ii) provided outer shell diameter, length, wall thickness, and empty and loaded weights of the Hanford, Savannah River Site, Idaho National Laboratory, and West Valley Demonstration Project HLW canisters (SAR Table 1.5.1-16); and (iii) provided the materials of construction, welding, weld testing, and leak testing for the HLW canisters (SAR Table 1.5.1-18).

The NRC staff finds that the applicant provided adequate characterization of the functional features of the HLW canisters because (i) the HLW canisters will provide containment and (ii) with the low concentrations of fissile radionuclides in an HLW canister, criticality controls will not be necessary.

On the basis of the above evaluation, the NRC staff finds that the principal characteristics and functional features are sufficient to permit an evaluation of the PCSA and design of the HLW canisters.

***Dual-Purpose Canister***

The applicant discussed dual-purpose canisters (DPCs) in SAR Section 1.5.1.1.1.2.1.2. Currently, the applicant plans to accept DPCs at the repository. In terms of storage, the applicant stated that DPCs would be placed in an appropriate aging overpack used for CSNF aging. The applicant stated that current DPC designs are not appropriate for disposal. For disposal, SNF in DPCs would be repackaged into a TAD canister and this operation would be performed in the wet handling facility (WHF). The applicant stated that it performed structural analyses on generic canisters (BSC, 2008cp), and it will perform additional structural, thermal, and criticality analyses once receipt of a specific DPC type is planned.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of DPCs using the guidance in YMRP Section 2.1.1.2.2. Specifically, the NRC reviewed the intended functions of the DPCs. The NRC staff finds that the applicant provided an adequate description of DPCs because the applicant (i) described the various functions of the DPC (e.g., shipment of spent fuel to the repository, placement of a DPC in aging overpack for CSNF aging); (ii) described how SNF in DPCs would be repackaged into TAD canisters in a wet handling facility prior to disposal; (iii) provided structural analysis on generic canisters; and (iv) explained that the local conditions at Yucca Mountain (e.g., temperature, rainfall, and tornado winds) are within those values specified in the Certificate of Compliance for many certified DPC systems.

On the basis of the above evaluation, the NRC staff finds that the description of the intended functions of the DPCs is sufficient to permit an evaluation of the PCSA and design. The NRC staff notes that additional analysis would be necessary should a specific DPC type have geometry or properties outside those considered for the generic canister.

### *Naval SNF Canister*

SAR Section 1.5.1.4.1.2.1 described the naval short or naval long spent nuclear fuel (SNF) canisters, which accommodate different naval fuel assembly designs. SAR Figure 1.5.1-29 showed a typical naval SNF canister. The applicant specified that the naval SNF canister will be fabricated from stainless steel that is similar to Stainless Steel Types 316 and 316L (Stainless Steel Type 316/316L). The naval SNF canister can be described as a cylinder with 2.5-cm [1-in]-thick shell walls, an 8.9-cm [3.5-in]-thick bottom plate, and a 38-cm [15-in]-thick top shield plug. The top shield plug will have six, 7.6-cm [3-in]-diameter threaded holes for lifting purposes. The shield plug will be welded to the canister shell; details of the redundant canister closure system were shown in SAR Figure 1.5.1-30. The naval short SNF canister will have a 471-cm [185.5-in]-nominal length {maximum length is 475 cm [187 in]}, and the naval long SNF canister will have a 535-cm [210.5-in] nominal length {maximum length will be 538 cm [212 in]}. The maximum outer diameter of the naval SNF canister will be 168.9 cm [66.5 in]. The canisters will be sized to fit within a waste package. The maximum design weight of the loaded long or short naval SNF canister will be 44,452 kg [98,000 lb]. However, the applicant noted that for establishing a margin in crane capability, the canister will be assigned a maximum weight of 49,215 kg [108,500 lb].

Design codes and standards for the naval SNF canister were given in SAR Section 1.5.1.4.1.2.8. For normal and accident conditions of storage and transportation, a naval SNF canister will be designed to the specifications of the 1998 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (American Society of Mechanical Engineers, 1998aa). The lifting features of the naval SNF canister will follow ANSI N14.6–1993 (American National Standard Institute, 1993aa) to define the structural limits for normal handling operations at the repository surface facilities. Leak testing of the naval SNF canister will follow the guidelines of ANSI N14.5–1997 (American National Standard Institute, 1998aa).

### NRC Staff's Evaluation

The NRC staff reviewed the applicant's description of the naval spent nuclear fuel (SNF) canisters using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the principal characteristics of the naval SNF canisters and design information. The NRC staff finds the principal characteristics of the naval SNF canisters are defined because the applicant (i) provided a cutaway drawing and a schematic on closure design of a typical naval SNF canister, including components; (ii) described the naval short or naval long SNF canisters with dimensions and weights; (iii) described the components to be used to package the naval SNF (e.g., naval SNF baskets, basket spacers, hafnium control rods, control rod retention hardware, and installed neutron poison assemblies); and (iv) provided the materials of construction to be used for the naval SNF canisters and associated components.

The NRC staff finds that the applicant provided adequate characterization of the functional features of the naval SNF canisters because (i) the naval SNF canisters will provide containment; (ii) the applicant plans to use hafnium control rods or install neutron poison assemblies, when necessary, to reduce the reactivity of the naval SNF (SAR Section 1.5.1.4.1.2.1); and (iii) the applicant will control criticality by designing the naval SNF canisters with a reliability such that the breach of a naval SNF canister is beyond Category 2 and introduction of a moderator into naval SNF canisters is beyond Category 2 (SAR Section 1.5.1.4.1.2.2.2).

On the basis of the above evaluation, the NRC staff finds that the principal characteristics and functional features are sufficient to permit an evaluation of the PCSA and design of the naval SNF canisters.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed in SER Section 2.1.1.2.3.5.2, the NRC staff finds, with reasonable assurance, that the applicant's description of the waste canisters meets the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description of the waste canisters (e.g., TAD canisters, DOE standardized canisters, HLW canisters, dual-purpose canisters, and naval SNF canisters) sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.5.3 Aging Overpack and Shielded Transfer Casks

The applicant provided information on aging overpacks and transfer casks in SAR Sections 1.2.7 and 1.2.5.4, respectively. Vertical aging overpacks will be used to age CSNF received in TAD canisters or vertical DPCs. Horizontal aging modules will be used for aging SNF in horizontal DPCs. Shielded transfer casks will be used during TAD canister and DPC handling operations.

CSNF will be aged at the repository in TAD canisters and DPCs. TAD canisters could be loaded at utility sites or loaded at the repository in the wet handling facility. Commercial DPCs will be loaded at utility sites. Therefore, it will be necessary to have an overpack designed specifically for the TAD canister and a set of overpacks designed for the DPC; both of these systems will be required to satisfy the applicant's aging facility (AF) design criteria. The TAD canister will use a vertical aging overpack, while the commercial DPCs will use either concrete vertical aging overpacks or concrete horizontal aging modules.

There were three different types of shielded transfer casks proposed: (i) vertical shielded transfer casks for use in the WHF for handling TAD canisters during loading and canister closure operations (e.g., drying and sealing), (ii) vertical shielded transfer casks for handling DPCs during opening and unloading operations in the WHF, and (iii) horizontal shielded transfer casks for moving horizontal DPCs from the AF to the WHF. Horizontal shielded transfer casks will also be used for handling horizontal DPCs during opening and unloading operations in the WHF.

***Vertical Aging Overpacks and Horizontal Aging Modules***

The applicant described aging overpacks in SAR Section 1.2.7. The applicant stated that the aging overpacks will include both a vertical aging overpack and a horizontal aging module. The vertical aging overpack will be used with TAD canisters and some DPCs, and the horizontal aging modules will be used only with horizontal DPCs. The aging overpack's function is to serve as a missile barrier and a radiation shield. In addition, the aging overpacks provide containment when subjected to natural hazards (SAR Table 1.2.2-1), such as lightning, a tornado-generated missile, snow, or volcanic ash. In SAR Table 1.9-1, the applicant classified aging overpacks (i.e., vertical aging overpacks and horizontal aging modules) as ITS. The nuclear safety design bases and design criteria were given in SAR Table 1.2.7-1.

In SAR Section 1.2.7.1.3.2.1, the applicant provided a general description of vertical aging overpacks. The applicant described the vertical aging overpack as a cylinder with a metal liner surrounded by steel-reinforced concrete that will be surrounded by an outer steel shell. A vertical aging overpack will have a maximum fully loaded weight of 226,796 kg [250 T], a maximum diameter of 3.7 m [12 ft], and a maximum height of 6.7 m [22 ft]. These dimensions are specified such that a vertical aging overpack will be able to support the inserted canister during the aging process. A vertical aging overpack will be fitted with a bolted lid, which also will provide shielding and protection, and will be designed to protect the internal canister against impact/collision and drop loads. The overpack will also be designed to provide passive cooling through convective movement of the air surrounding the canisters. Bottom inlets and top outlets will allow ventilation air to be passively drawn into the annular area between the TAD or vertical DPC canister and the metal liner. The inlets and outlets will be designed to prevent radiation streaming. SAR Figure 1.2.7-6 showed most of these design features.

The applicant described the horizontal aging module to be a boxlike, thick-walled, reinforced concrete structure having a minimum concrete shielding thickness of 0.9 m [3 ft], a maximum height of 6.4 m [21 ft], a maximum width of 2.6 m [8.5 ft], and a minimum length {with the minimum of 0.9 m [3 ft] of shielding} of 7.1 m [23 ft 4 in]. A shield wall will be used behind each horizontal aging module and at each end of a row of modules to supplement shielding and reduce the radiation dose emanating from the horizontal aging modules. Similar to the vertical aging overpack, a horizontal aging module was described as being configured with vents and flow paths to permit natural circulation airflow to transfer the heat from the canister to the atmosphere and will be equipped with temperature sensors to measure outlet air temperature.

The applicant stated that the aging overpack systems will be evaluated for normal handling loads, dead loads, thermal loads, and event sequence loads (SAR Section 1.2.7.9). The applicant further stated that the aging overpack systems will withstand the natural phenomena of loading parameters at Yucca Mountain, as shown in Table 1.2.2-1.

SAR Section 1.2.7.8 listed the design codes and standards for the vertical aging overpack and horizontal aging modules. The applicant stated that the concrete used to construct the aging overpacks will follow ACI 349–01/349R–01 (American Concrete Institute, 2001aa) and the reinforcing steel should comply with ASTM A 706/A 706M–06a or ASTM A 615/A 615M–06a (ASTM International, 2006ac,ad). In addition, the aging overpack design will follow ASCE/SEI 43–05 (American Society of Civil Engineers, 2005aa); ACI 349–01/349R–01 (American Concrete Institute, 2001aa); and ANSI/ANS–6.4–1997, as described in American Nuclear Society (2006aa, Appendix A).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's aging overpack (i.e., vertical aging overpack and horizontal aging module) description using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of (i) functional requirements of the aging overpacks; (ii) design information including the basic drawings, geometry, materials, and protection against natural phenomena; and (iii) functional arrangement of the aging overpacks in the aging facilities. The NRC staff finds that the applicant's description of the aging overpacks (i.e., vertical aging overpack and horizontal aging module) is adequate because (i) the description discussed the specific functions of aging overpacks, including acting as a missile barrier, providing radiation shielding, providing containment when subjected to natural hazards, and providing passive cooling; (ii) the described functions for the aging overpacks are consistent with the proposed aging operation and process flow in the aging pads; (iii) the description discussed the two aging overpack configurations (vertical aging overpacks with a cylindrical shape and horizontal aging modules with a box shape); (iv) the description provided the physical characteristics of the aging overpacks, including dimensions and materials of construction for both configurations (reinforced concrete and metal liner for vertical aging overpacks and reinforced concrete for horizontal aging modules); (v) the applicant provided schematic drawings showing the conceptual design for the aging overpacks; (vi) the description discussed radiation protection features to be implemented for the aging overpacks; and (vii) the design information included design codes and standards that are consistent with standard engineering practices, design methodologies, materials of construction, and load combinations to be used to design aging overpacks.

The NRC staff finds that the applicant provided adequate descriptions of the arrangement of the aging overpacks (i.e., vertical aging overpack and horizontal aging module) on the aging pads because the applicant provided a schematic drawing showing the placement scheme.

The NRC staff finds that the applicant provided an adequate discussion of the ability of the aging overpacks (i.e., vertical aging overpack and horizontal aging module) to withstand the effects of natural phenomena because (i) aging overpacks will be designed for natural phenomena and (ii) the applicant, through simple analyses, showed that the impact of missiles generated by tornadoes, consistent with the site-specific information, will not be able to penetrate the reinforced concrete structure and lightning strike will not be able to breach the casks inside aging overpacks (SAR Section 1.6.3.4; BSC, 2008ai).

On the basis of the above evaluation, the NRC staff finds that the description the applicant provided is sufficient to permit an evaluation of the PCSA and design of the aging overpacks (i.e., vertical aging overpack and horizontal aging module).

***Shielded Transfer Cask***

The applicant described shielded transfer casks in SAR Section 1.2.5.4. Shielded transfer casks will be used for processing TAD canisters and DPCs in the WHF. They will be also used for moving horizontal DPCs from the AF to the WHF. Shielded transfer casks provide integral shielding, structural strength, and passive cooling functions. The applicant classified shielded transfer casks as ITS, and the nuclear safety design bases and design criteria were given in SAR Table 1.2.5-3.

Shielded transfer casks will be required to maintain their structural integrity, retain the canister, and continue to provide shielding when subjected to drops, tip over, collisions, fires, seismic events, and natural phenomena such as wind loading, missiles, or precipitation (SAR Table 1.2.2-1). Because the shielded transfer cask will be used in the WHF, the applicant stated that it needs to be compatible with pool water. The applicant specified that the materials of construction for the shielded transfer cask design will be in accordance with 2004 ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (American Society of Mechanical Engineers, 2004aa).

Shielded transfer casks will perform different functions when handling TAD canisters and DPCs; however, some common design features will be used to standardize operations and maintenance. SAR Figures 1.2.5-76 to 1.2.5-78 showed the general design features of shielded transfer casks. A vertical DPC shielded transfer cask contains a DPC, which will be moved into the WHF by a bottom-lift site transporter and moved within the WHF by an overhead crane and cask transfer trolley. Inside the WHF, the vertical DPC shielded transfer cask will be lifted by trunnions using an overhead crane and will be required to stand and remain upright when set down upon a flat horizontal surface. SAR Figure 1.2.5-76 showed a representative vertical DPC shielded transfer cask. Similarly, a TAD shielded transfer cask will contain a TAD canister (in a vertical orientation) and will be able to be moved and lifted in a manner similar to the vertical DPC shielded transfer cask. The TAD shielded transfer cask will also be required to stand upright on a flat horizontal surface. SAR Figure 1.2.5-77 showed a representative drawing of a TAD shielded transfer cask. A horizontal shielded transfer cask will be designed for a single horizontal DPC. After loading, the cask will be rotated to a vertical position and lifted by its trunnions using an overhead crane. This cask will also be required to remain in a vertical orientation when set on a flat horizontal surface. A drawing of a horizontal shielded transfer cask was shown in SAR Figure 1.2.5-78.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on the shielded transfer casks presented in SAR Section 1.2.5.4 using the guidance in YMRP Section 2.1.1.2. The NRC staff finds that the applicant's description of the shielded transfer casks is adequate because the applicant (i) described specific functions of shielded transfer casks (including processing TAD canisters and DPCs in the WHF; moving horizontal DPCs from the AF to the WHF; and providing integral shielding, structural strength, and passive cooling functions); (ii) discussed the three types of shielded transfer cask (two vertical ones for DPCs and TADs and one horizontal one for DPCs); (iii) specified that a shielded cask will be designed to accommodate one canister; (iv) provided mechanical drawings, illustrating the general design features of shielded transfer casks; and (v) provided design codes and standards that are consistent with standard engineering practices, design methodologies, materials of construction, and design load combinations to be used to design shielded transfer casks.

The NRC staff finds that the applicant provided an adequate description regarding the ability of the shielded transfer casks to withstand the effects of natural phenomena because shielded transfer casks will be designed to maintain structural integrity when subject to natural phenomena (e.g., seismic events, lightning strikes, tornado winds, and tornado-generated missiles, snow, and volcanic ash).

On the basis of the above evaluation, the NRC staff finds that the description the applicant provided is sufficient to permit an evaluation of the PCSA and design of the shielded transfer casks.

**NRC Staff's Conclusion**

On the basis of the evaluations discussed in SER Section 2.1.1.2.3.5.3, the NRC staff finds, with reasonable assurance, that the applicant's description of the aging overpacks (i.e., vertical aging overpack and horizontal aging module) and shielded transfer casks meets the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description of the aging overpacks (i.e., vertical aging overpack and horizontal aging module) and shielded transfer casks sufficient for the NRC staff to evaluate the PCSA and design.

2.1.1.2.3.5.4 Drip Shield

The applicant described and discussed the drip shield design in SAR Section 1.3.4.7. The applicant proposed to use the drip shield as an engineered barrier system (EBS) during the postclosure period to divert the liquid moisture that drips from the drift wall around the waste package and down to the drift invert structure and protect the waste packages from rockfall.

The applicant classified the drip shield as non-ITS because it will not be relied on to prevent or mitigate Category 1 and Category 2 event sequences. The applicant classified the drip shield as Important to Waste Isolation (ITWI) because it will be relied upon to prevent or substantially reduce the rate of movement of water and radionuclides during the postclosure period.

The applicant stated that the drip shield will have a single design, will be uniformly sized to enclose all waste package configurations, and will be designed for both corrosion resistance and structural strength. The drip shield will consist of Titanium Grade 7 (UNS R52400) plates for water diversion, Titanium Grade 29 (UNS R56404) structural members for structural support, and Alloy 22 (UNS N06022) base plates to prevent direct contact between the titanium drip shield components and the invert structure steel members. The codes and standards that govern Titanium Grades 7 and 29 and Alloy 22 properties (e.g., density, elongation, yield, and ultimate tensile stresses) were listed in SAR Table 1.3.2-5.

According to SAR Table 1.3.4-3, the drip shield height will vary between 2,821 and 2,886 mm [111 and 113.6 in], the width will vary between 2,526 and 2,535 mm [99 and 99.8 in], and the length will be 5,805 mm [228.5 in]. The drip shield weight will be 4,897 kg [10,796 lb]. The standard nomenclature used and construction material for the drip shield components were provided in SAR Table 1.3.4-4. SAR Figure 1.3.4-15 provided dimensions for an assembled (welded) drip shield, and in response to an NRC staff request for additional information (RAI), the applicant provided drip shield main assembly, subassemblies, and components drawings (DOE, 2009dr).

According to the applicant, drip shields will form a continuous barrier throughout the entire length of the emplacement drift by interlocking the drip shield segments. The applicant stated that the drip shield will accommodate an interlocking feature to prevent the separation between contiguous drip shield segments and a minimum lift height of 1,016 mm [40 in] will be required to interlock the drip shield segments. Furthermore, the drip shield interlocking feature will include water diversion rings and connector plates that will divert the liquid moisture at the seams between the drip shield segments. SAR Figure 1.3.4-15 detailed the drip shield interlock feature, and in response to an NRC staff RAI, the applicant provided a sequence of isometric sketches that illustrated the drip shield interlocking process and figures that demonstrated the height clearance required to interlock two drip shields (DOE, 2009dr).

The applicant stated that, except for the attachment to the Alloy 22 base, drip shield components will be connected to each other by welding. According to the applicant, the Alloy 22 base plates will be mechanically attached to the titanium components by Alloy 22 pins because titanium and Alloy 22 cannot be reliably welded together. The applicant included codes and standards governing physical and mechanical properties (e.g., density, elongation, yield, and ultimate tensile stresses) of Titanium Grades 7 and 29 in SAR Table 1.3.2-5. In response to the NRC staff RAI (DOE, 2009dr) regarding the codes and standards for the drip shield design and fabrication, the applicant extracted codes and standards as applicable for materials, welding, postweld heat treatment, and postweld nondestructive examination of the drip shield from Yucca Mountain Project Engineering Specification Prototype Drip Shield (BSC, 2007bu). The applicant stated that the codes and standards cited in the prototype specification were adopted from the ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa) and American Welding Society standards for welding. In addition, the applicant stated that the prototype program will be used to demonstrate and confirm the design suitability and progressively develop and refine the production fabrication process.

According to the applicant, for similar welds, including Titanium Grade 7 to Titanium Grade 7 and Titanium Grade 29 to Titanium Grade 29, the filler metal matching the base metal will be used. However, for Titanium Grade 7 to Titanium Grade 29 welds, Titanium Grade 28 filler material will be used. In its response to the NRC staff's RAIs (DOE, 2009ab, Enclosure 8; 2009dr, Enclosure 4) for justification that welding Titanium Grade 7 to Grade 29 using Grade 28 as filler metal is appropriate, the applicant reviewed industry experience of welding dissimilar alpha-phase and (alpha+beta)-phase alloys and stated that this dissimilar welding joint will mitigate hydrogen embrittlement because of the presence of palladium (Pd) and ruthenium (Ru) noble elements in the material and the use of Titanium Grade 28 intermediate layer. In addition, the applicant cited two examples from the literature (American Welding Society, 2007aa; Boyer, et al., 1994aa) to indicate that welding joints similar to Titanium Grade 7/Grade 28/Grade 29 have been used in the industry.

**NRC Staff's Evaluation**

The NRC staff reviewed the drip shield information using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the characteristics, functional features, the intended functions, and design information of the drip shield. The NRC staff finds that the applicant has characterized the drip shield sufficient to support evaluation of the PCSA and design of the drip shield because the applicant (i) described the function of the drip shield during the postclosure period (the drip shield will divert the liquid moisture that drips from the drift wall around the waste package and down to the drift invert structure and will protect the waste packages from rockfall); (ii) described the design of the drip shield for both corrosion resistance and structural strength (i.e., titanium plates for water diversion and titanium structural members for strength); (iii) provided the weight of the drip shield and dimensions for an assembled (welded) drip shield; (iv) provided the drip shield main assembly, subassemblies, and components drawings; (v) described the shield interlocking feature that will include water diversion rings and connector plates that will divert the liquid moisture at the seams between the drip shield segments; (vi) provided codes and standards used in the design of the drip shield; and (vii) described how the drip shields would be welded.

The NRC staff finds the applicant's proposal for welding similar metals, Titanium Grade 7 to Titanium Grade 7 and Titanium Grade 29 to Titanium Grade 29, using filler metal matching the base metal is acceptable because, as per the applicant, the welding will be in accordance with the American Welding Society standards for welding and industrial practices. The NRC staff also finds that the proposed codes and standards pertaining to drip shield design and fabrication in the prototype program are acceptable because they are consistent with the standard engineering practices for equipment of similar functions.

With respect to welding Titanium Grade 7 to Grade 29 using Grade 28 as filler metal, the NRC staff had a question regarding the adequacy of this dissimilar materials welding because the two examples the applicant cited are not directly comparable to the dissimilar welding joint proposed for the drip shield. In one example (American Welding Society, 2007aa), Titanium Grade 7 was used as a filler metal in welding the ruthenium-containing Titanium Grade 26 to Titanium Grade 26 (UNS R52404). The NRC staff notes that this example is not similar to the welding joint the applicant proposed to use for the drip shields. This is because both Titanium Grades 26 and 7 are single alpha-phase materials having similar microstructure and mechanical properties, whereas Grade 29, which the applicant proposed to use for the drip shields, is an (alpha+beta)-phase alloy with chemical composition, microstructure, and mechanical properties different from Titanium Grade 7. The other example (Boyer, et al., 1994aa) indicated that the industry occasionally uses unalloyed or low-alloyed titanium as filler metal to weld titanium alloy grades with higher strength for improved joint ductility [e.g., using unalloyed filler metal to weld Titanium Grade 5 (Ti-6Al-4V) to Titanium Grade 6 (Ti-5Al-2.5Sn)]. However, the NRC staff also notes that this example is not similar to the welding joint the applicant proposed to use for the drip shields, because the alloy content and type of alloys are different.

Nevertheless, the NRC staff notes that Titanium Grade 28, the filler material proposed by the applicant, is a near-alpha alloy. A near-alpha alloy contains more of alpha crystalline phase in the alloy than the alpha crystalline phase in an [alpha+beta]-phase alloy, such as Titanium Grade 29. Titanium Grade 28 microstructure is compatible with both Titanium Grade 7 and Grade 29 because the amount of beta crystalline phase in Titanium Grade 28 is in between those in Titanium Grades 7 and 29. In addition, the concentration of aluminum and vanadium elements in Titanium Grade 28 is lower than that in Titanium Grade 29, whereas Grade 7 does not have aluminum and vanadium elements. The lower concentration of these elements in Grade 28 results in a mechanical strength lower than that of Grade 29 but closer to Grade 7. The microstructure and mechanical properties of Grade 28 are in between Grade 7 and Grade 29. Also, the chemical composition of Grade 28 is compatible with the chemical composition of both Grades 7 and 29 because the concentration of aluminum and vanadium elements in Grade 28 is in between Grade 7 and Grade 29. Overall, Titanium Grade 28 is compatible in chemical composition, microstructure, and mechanical properties with both Titanium Grades 7 and 29. As stated in the applicant's response to the NRC staff's RAI (DOE, 2009ab,dr), similar composition (containing noble elements such as ruthenium and palladium) of Grade 28 and Grade 29 will help in preventing hydrogen embrittlement of Grade 7 alloy near-weld joints by hydrogen uphill diffusion in dissimilar metals. Therefore, the NRC staff determines that Titanium Grade 28 is an appropriate choice as the filler material in welding Titanium Grade 7 to Grade 29.

On the basis of the evaluation discussed above, the NRC staff finds that the applicant provided sufficient information about the drip shield design because it provided descriptive information on dimensions, structural features, functions, design codes and standards, materials, fabrication, and welding.

The NRC staff notes that, as part of the final design, the applicant proposed a drip shield prototype program in which the compatibility and performance of the materials in terms of, welding joints, capability to assemble within nominal dimensions, structural strength of fabricated drip shield, and weld designs and procedures, will be confirmed and finalized (DOE, 2009dr; BSC, 2007bu).

**NRC Staff's Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.5.4, the NRC staff finds, with reasonable assurance, that the applicant's description of the drip shield meets the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description of the drip shield sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed above in the NRC staff evaluation, the applicant (i) defined the principal characteristics of the drip shield and its components that included drip shield dimensions, weight, materials, fabrication, and welding; (ii) characterized the functional features of the drip shield and its components (i.e., to divert the liquid moisture around the waste package and down to the drift invert structure and protect the waste packages from rockfall); and (iii) described the design, along with the applicable codes and standards used, for the drip shield and its components.

### 2.1.1.2.3.6 Description of Geologic Repository Operations Area Processes, Activities, and Procedures, Including Interfaces and Interactions Between Structures, Systems, and Components

In this section, the NRC staff evaluates the applicant's description of operational processes in the surface and subsurface facilities of the geologic repository operations area (GROA) and onsite transportation to determine compliance with 10 CFR 63.21(c)(3)(i) and 10 CFR 63.112(a). The applicant's description of operational processes addresses (i) the operational sequences and material flow, (ii) the major waste processing functions performed, and (iii) the waste form inventory present within the facility.

In SER Section 2.1.1.2.3.6.1, the NRC staff reviewed each facility in terms of its descriptive information pertaining to how a particular waste form is handled in the GROA operations. The NRC staff evaluated (i) waste form handling operations, including the process flow diagram; (ii) planned waste throughput in each facility; (iii) subsystem/equipment and interactions and interfaces among the subsystems; (iv) human interactions; and (v) the proposed operation plan, in terms of permitting permanent disposal of the mandated quantity of high-level radioactive waste (HLW) within the period the applicant stipulates.

In SER Section 2.1.1.2.3.6.2, the NRC staff reviewed the communication, instrumentation, and control systems for the surface and subsurface facilities. The review covered ITS and non-ITS control systems to provide an overall description of the general control philosophy of the GROA operations.

#### 2.1.1.2.3.6.1 Operational Processes

The applicant described GROA process activities in SAR Section 1.2 to identify hazards and event sequences in the PCSA. The applicant's description included operations in the initial handling facility (IHF), canister receipt and closure facility (CRCF), wet handling facility (WHF), receipt facility (RF), aging facility (AF), and subsurface facility, as well as onsite transportation. The NRC staff's evaluation of the description of the layout of mechanical handling systems and mechanical handling equipment at the GROA is provided in SER Sections 2.1.1.2.3.2.2 and 2.1.1.2.3.2.5, respectively.

***Surface Facility Operations***

The applicant presented its overview of the CRCF operational processes in SAR Section 1.2.4.1.2 and the detailed description of the operations for the cask handling, canister transfer, waste package closure, and waste package load-out subsystems in SAR Sections 1.2.4.2.1.2, 1.2.4.2.2.2, 1.2.4.2.3.2, and 1.2.4.2.4.2, respectively. The overview of the operational processes for the IHF was discussed in SAR Section 1.2.3.1.2, and the detailed description of the operations for the cask handling, canister transfer, waste package closure, and waste package load-out subsystems was provided in SAR Sections 1.2.3.2.1.2, 1.2.3.2.2.2, 1.2.3.2.3, and 1.2.3.2.4.2, respectively. For the RF, the overview of the operational processes was discussed in SAR Section 1.2.6.1.2, and the detailed description of the operations for the cask handling and canister transfer subsystems was discussed in SAR Sections 1.2.6.2.1.2 and 1.2.6.2.2.2, respectively.

The applicant proposed to construct three identical CRCFs. The main operations in the CRCF involve handling canisters containing different waste forms and handling transportation casks, aging overpacks, and waste packages. The CRCF will handle TAD canisters, HLW canisters, DPC, and DOE SNF canisters. The overall operations in the CRCF will be performed using four mechanical handling subsystems: cask handling, canister transfer, waste package closure, and waste package load out. The process subsystems will include cask cavity gas sampling and water collection subsystems. The facility will be divided into several major areas of operation, consisting of the transportation cask and site transporter vestibule area; cask unloading and preparation areas; gas-sampling area; canister transfer area; and waste package positioning, closure, and load-out areas. The major rooms to support waste handling operations will include the heating, ventilation, and air conditioning (HVAC) equipment room, electrical rooms, maintenance areas, and waste package closure support rooms. The major mechanical equipment used in the facility will be overhead bridge cranes, canister transfer trolleys (CTTs), canister transfer machines (CTMs), waste package transfer trolleys (WPTTs), and associated lifting fixtures and devices.

Transportation casks on rail- or truck-based trailers will be received in the cask handling area. The casks will be moved onto the cask transfer trolley after the impact limiters are removed from them. In the cask preparation area, the cask cavity will be sampled and depressurized and lid bolts will be removed. The aging overpacks will be received on a site transporter, and the lid bolts will be removed. In the canister transfer subsystem, the canisters will be transferred from the transportation casks into a waste package or aging overpack, or they will be placed in the staging area. A staging area will be provided for TAD, HLW, and DOE SNF canisters. However, in response to the NRC staff RAI (DOE, 2009dx), the applicant stated that TAD canister staging will not be part of normal operations. The CTM will be operated remotely to remove cask lids, transfer canisters to waste packages positioned on the WPTT, and place inner lids on waste packages. Waste package closure subsystems will be used for welding waste package lids to waste packages, stress mitigation, nondestructive tests, and inerting of the waste package inner vessel. Waste package load-out operations will include transfer of sealed waste packages to the load-out area using the WPTT and loading of waste packages onto the transport and emplacement vehicle (TEV).

The main operations in the IHF will involve handling of naval SNF canisters or HLW canisters, transportation casks, and waste packages. Similar to the CRCF, the overall operations in the IHF will be performed using four mechanical handling subsystems: cask handling, canister transfer, waste package closure, and waste package load out. The process subsystems will include cask cavity gas sampling and water collection. The major operational areas will consist of the cask preparation area, canister transfer area, waste package closure area, and waste package load-out area. The major mechanical equipment used in the facility will be overhead bridge cranes, cask transfer trolleys, CTMs, WPTTs, and associated lifting fixtures and devices.

The main operations in the receipt facility (RF) will involve handling of TAD canisters or DPCs, transportation casks, and aging overpacks. The overall operations in the RF will be performed using two mechanical handling subsystems: cask handling and canister transfer. The process subsystems include cask cavity gas sampling and water collection subsystems. The facility will be divided into major areas of operation consisting of cask preparation, cask unloading and loading, canister transfer, lid bolting, and transportation cask and site transporter vestibule areas. The major mechanical equipment used in the facility will be overhead bridge cranes, CTTs, CTMs, and associated lifting fixtures and devices.

The CRCF, IHF, and RF will include several personnel and equipment shield doors of different configurations. The floor plans and cross-sectional views were shown in SAR Figures 1.2.4-1 to 1.2.4-11 for CRCF, Figures 1.2.3-2 to 1.2.3-14 for IHF, and Figures 1.2.6-2 to 1.2.6-11 for RF. The major waste processing functions were shown in SAR Figure 1.2.4-12 for CRCF,

Figure 1.2.3-17 for IHF, and Figure 1.2.6-14 for RF. The operational sequence, material flow, and waste form inventory locations were illustrated in SAR Figures 1.2.4-12 to 1.2.4-14 for CRCF; Figures 1.2.3-15, 1.2.3-16, and 1.2.3-18 for IHF; and Figures 1.2.6-12 and 1.2.6-13 for RF. The process flow diagrams were shown and discussed in BSC (2008ab,ao,bd, Section 6, Figure 15, and Attachments A and B). Human interactions during operations were discussed in BSC (2008ac,as,be, Section E6) as a part of the applicant's analysis of human failures.

SAR Sections 1.2.5.1.2 and 1.2.5.2.1.2 provided an overview of the operational processes for the WHF. The main operations will include handling of transportation casks and aging overpacks, SNF assemblies, and DPC and TAD canisters. Overall operations will be performed using five mechanical handling subsystems: cask handling, SNF assembly transfer, DPC cutting, TAD canister closure, and canister transfer. The facility will be divided into several major areas of operation, consisting of the transportation cask and site transporter vestibule area, cask preparation areas (including SNF transfer area in the pool), DPC cutting area, TAD closure area, and canister transfer area. The process subsystems will include cask cavity gas sampling, pool water treatment, and cooling subsystems. The major rooms to support waste handling operations will include the heating, ventilation, and air conditioning (HVAC) equipment room, electrical rooms, maintenance areas, and waste package closure support rooms. The major mechanical equipment used in the facility will be overhead bridge cranes, cask transfer trolleys, CTMs, spent fuel transfer machine (SFTM), and associated lifting fixtures and devices. The facility will include several personnel and equipment shield doors of different configurations. The floor plans and cross-sectional views of the WHF were shown in SAR Figures 1.2.5-1 to 1.2.5-16. The major waste processing subsystems and functions were shown in SAR Figure 1.2.5-19, while the operational sequence, material flow, and waste form inventory locations were in SAR Figures 1.2.5-17 and 1.2.5-18. The process flow diagrams and operations description were shown in BSC (2008bq, Section 6, Figures 16–17, and Attachments A and B). Human interactions during operations were discussed in BSC (2008bq, Appendix E, Section E6) as a part of the applicant's analysis of human failures.

SAR Section 1.2.5.1.2 provided an overview of the operational processes for the WHF. SAR Sections 1.2.5.2.1.2, 1.2.5.2.2.2, 1.2.5.2.3.2, 1.2.5.2.4.2, and 1.2.5.2.5.2 described the operations for cask handling, SNF assembly transfer, DPC cutting, TAD canister closure, and canister transfer, respectively. The cask handling system will receive transportation casks containing uncanistered CSNF, rail casks with DPCs, shielded transfer casks with DPCs, and aging overpacks with DPCs. For transportation casks containing uncanistered SNF, cask handling cranes (CHCs) will be used to remove impact limiters and to upend the cask and move it to the preparation station. After sampling, venting, and cooling, the cask interior will be filled with borated water, the cask lid will be unbolted, and the cask will be moved to the pool. For rail casks containing DPCs, impact limiters will be removed and the cask will be upended and then placed on a cask transfer trolley. Shielded transfer casks containing DPCs will be upended and then placed on a cask transfer trolley. DPCs in an aging overpack will be transferred to a shielded transfer cask at the WHF using the CTM. All casks containing DPCs will be transferred to the DPC cutting station. The cask handling system will also transfer TAD canisters from shielded transfer casks to aging overpacks using the CTM and send aging overpacks to the AF or the CRCF. In the SNF assembly transfer system, the auxiliary pool crane removes the transportation cask or shielded transfer cask lids and DPC shield plugs. The SFTM will move SNF assemblies from transportation casks or DPCs to TAD canisters or a staging rack and from the staging rack to TAD canisters. After loading TAD canisters, the TAD canister shield lid and shielded transfer cask lid will be replaced prior to lifting the TAD canister out of the pool.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on surface facility operations using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of waste handling operation processes for both dry and wet handling and related process flow diagrams. The NRC staff also reviewed the discussions on the interfaces and interactions of the operational processes during waste handling. The NRC staff finds that the description of the surface facility operations is adequate because the applicant (i) described the waste handling operations and provided process flow diagrams for the relevant facilities (e.g., IHF, CRCF, WHF) and each of the cask and canister types to be handled (e.g., TAD canister, DPC, transportation cask, aging overpack); (ii) discussed the subsystem/equipment interactions, operations, and interfaces between the subsystems (e.g., the WHF will include cask cavity gas sampling, pool water treatment, and cooling subsystems; for rail casks containing DPCs, impact limiters will be removed, the cask will be upended, and then placed on a cask transfer trolley; the SFTM will move SNF assemblies from transportation casks or DPCs to TAD canisters or a staging rack and from the staging rack to TAD canisters); and (iii) described human interactions for the CRCF, WHF, IHF, and RF that provide a general understanding and overview of the operational processes at the surface facilities. The detailed review of the operations, as it pertains to the identification and quantification of initiating events, is described in SER Sections 2.1.1.3.3.2.1 and 2.1.1.3.3.2.3.4.

The NRC staff finds that the applicant discussed the probability of failure from drop for various containers (casks, canisters, and waste packages) from normal operating and two-block lift heights for inclusion in the event sequence analysis and presented in various BSC documents (e.g., BSC, 2008ac, Table 6.3-7). The NRC staff notes that the potential drop from the lift height during operations would affect the container failure probability that is needed for event sequence identification and categorization. The NRC staff finds that the drop height information provided in various BSC documents (e.g., BSC, 2008ac) is acceptable because (i) the drop height information is consistent with the descriptions and designs provided for container handling operations at the surface facilities and (ii) the applicant used these heights consistently in the event sequence analysis in the PCSA. The NRC staff's evaluation of event sequences related to container drops is discussed in SER Section 2.1.1.4.3.3.1.1.

On the basis of the above evaluation, the NRC staff finds that the description the applicant provided for the surface facility operations is sufficient to permit an evaluation of the PCSA and design.

***Intrasite Surface Operational Processes***

The applicant described the intrasite operations in SAR Section 1.2.8.4 and in supplemental documents (BSC, 2008at,au). These documents described activities related to the aging facility (AF), low level waste handling facility (LWF), emergency diesel generator facility (EDGF), and the intrasite transportation system. For example, site transportation activities will include movement of the transportation casks, security/radiological inspections, and transfer of aging overpacks and horizontal casks from one surface facility to another. The AF activities considered as part of intrasite activities, as described in BSC (2008at, Section 4.3.4), will include positioning of aging overpacks, loading of horizontal canisters in horizontal aging modules, canister aging and monitoring, and retrieval of aged canisters. The AF's operational process was described in SAR Section 1.2.7.2. LLW management activities will include onsite loading, onsite transfer to the LLWF, unloading at the LLWF, storage at the LLWF, and the offsite disposal process. BSC (2008at, Attachments B and C) provided additional details on the operations. Balance-of-plant activities will include support systems, such as site roadways and railways for GROA operations, and nonnuclear facilities, such as the craft shop, equipment yard, and maintenance facility. The EDGF activity would provide emergency power in the event of a loss of offsite power (LOSP).

The applicant provided an intrasite operational process flow diagram in BSC (2008at, Figure 14). BSC Attachment C (2008at) included onsite transportation routes and the relative location of the aging pads, LLWF, and buffer areas within the GROA. The applicant provided a flow diagram showing the flow path of each type of waste container and the transportation equipment used to move the waste container from one surface facility to another. The surface facilities included in the flow diagram were the Cask Receipt Security Station, truck buffer area, railcar buffer area, IHF, CRCF, WHF, RF, and AF.

The applicant identified and described that several types of site transportation SSCs would be utilized: the site transporter (SAR Section 1.2.8.4.1), the cask tractor and cask transfer trailer (SAR Section 1.2.8.4.2), and the site prime mover (SAR Section 1.2.8.4.3). All three were categorized as ITS and will be further described and evaluated in SER Section 2.1.1.7.3.5. In BSC (2008au, Appendices B1.4, B2.3, and B3.3), the applicant also listed the dependencies and interactions associated with each of the three transportation systems mentioned previously. It included functional, environmental, spatial, human, and external events interactions.

For human-related operations, the applicant listed human interactions and described human-induced failures during intrasite operations. For the site transporter, in particular, the applicant provided additional detail on human interfaces. This detail included a list of 15 remote control activation devices (push buttons and selector switches) that human operators manipulate or activate.

**NRC Staff's Evaluation**

The NRC staff reviewed intrasite operations using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the waste handling operational processes of the intrasite surface activities, SSCs to be used for intrasite transportation, and interfaces and interactions among subsystems. The NRC staff finds that the description of intrasite surface operational activities and procedures is adequate because the applicant (i) described site transportation activities (e.g., the movement of the transportation casks, security/radiological inspections, and transfer of aging overpacks and horizontal casks from one surface facility to another); (ii) described aging facility activities (e.g., positioning of aging overpacks, loading of horizontal canisters in horizontal aging modules, canister aging and monitoring, and retrieval of aged canisters); (iii) described LLW management activities (e.g., onsite loading, onsite transfer to the LLWF, unloading at the LLWF, storage at the LLWF, and the offsite disposal process); (iv) described the site roadways, railways, and transportation routes for GROA operations; (v) described how EDGF activity would provide emergency power in the event of a LOSP; (vi) provided an intrasite operational process flow diagram; (vii) provided a flow diagram showing the flow path of each type of waste container and the transportation equipment used to move the waste container from one surface facility to another; and (viii) listed human interactions and described human-induced failures during intrasite operations.

On the basis of the above evaluation, the NRC staff finds that the description the applicant provided for intrasite surface operational processes is sufficient to permit an evaluation of the PCSA and design.

***Subsurface Operational Processes***

The applicant described the subsurface operations in SAR Section 1.3.1 and summarized these operations in BSC (2008bj, Appendix B). The subsurface operations will include activities such as WP load out, WP emplacement, drip shield loadout, drip shield transport, and drip shield emplacement. The applicant provided a process flow diagram in a supplemental document (BSC, 2008bj) that detailed waste package transportation from the surface facility to the subsurface facility for emplacement. This document outlined the operation of the TEV as it exits the Heavy Equipment Maintenance Facility until it returns from the subsurface facility. The applicant also provided a process flow diagram for the drip shield emplacement operations in BSC (2008bj) and indicated that the only ITS SSC associated with the subsurface operations will be the TEV. The non-ITS SSCs mentioned were the invert system, crane rail switches, ventilation system, access door, the drip shield emplacement gantry (DSEG), fire protection system, the electric power system of the third rail, and the communication and control system of the control center.

There will be two distinct normal operation sequences in the subsurface: waste package emplacement and drip shield emplacement. The normal TEV emplacement operation will consist of several steps, such as opening TEV front shield doors, driving forward, lifting rear shield doors, extending the baseplate, lowering the shielded enclosure, and lifting the waste package. These operations were described in SAR Section 1.3.3.5.2.1.

In BSC Appendix B (2008bj), the applicant also briefly described the construction operations that will occur at the same time as emplacement operations during a portion of the preclosure period. These operations will include excavation using common drill and blast techniques, as well as mechanical excavators, which could potentially affect waste handling operations.

The applicant indicated that the primary human interactions during subsurface operations will involve communication and control of the TEV by operators in the control center. The applicant described the use of high-intensity lights and a camera onboard the TEV to provide feedback to operators in the control center. The applicant also described the use of programmable logic controllers (PLCs) that will accept initiating commands from operators to execute predefined, preprogrammed instructions and maneuvers. The applicant discussed subsurface operations as they relate to human interactions in BSC (2008bk, Section 6.4).

**NRC Staff's Evaluation**

The NRC staff reviewed the information on subsurface operational processes using the guidance in YMRP Section 2.1.1.2. The NRC staff finds that the applicant's descriptions of the subsurface operational processes and procedures involving waste package transport and emplacement using the TEV are adequate because the applicant (i) described the subsurface operations for the waste package and drip shield transport and emplacement (i.e., WP load out, WP emplacement, drip shield load out, drip shield transport, and drip shield emplacement); (ii) provided a process flow diagram detailing the waste package transportation using the TEV from the surface facility to the subsurface facility for emplacement; (iii) identified non-ITS SSCs for subsurface operations (e.g., crane rail switches, ventilation system, access door, the drip shield emplacement gantry, fire protection system, the electric power system of the third rail, and the communication and control system of the control center); (iv) described the human interactions during subsurface operations involving communication and control of the TEV by operators in the control center; and (v) described the construction operations that will occur during emplacement operations.

On the basis of the above evaluation, the NRC staff finds that the description the applicant provided for subsurface operational processes is sufficient to permit an evaluation of the PCSA and design.

***Waste Form Throughput***

According to the applicant, the repository will be designed for $7 \times 10^7$ kg [70,000 MTHM] of radioactive waste, as shown in SAR Table 1.5.1-1 and BSC (2007bh, Section 6). More specifically, the waste to be disposed of in the repository will include $6.3 \times 10^7$ kg [63,000 MTHM] of CSNF and HLW of commercial origin, $4.7 \times 10^6$ kg [4,667 MTHM] of defense HLW, $2.3 \times 10^6$ kg [2,268 MTHM] of DOE SNF, and 65,000 kg [65 MTHM] of naval SNF. Waste will be shipped to Yucca Mountain in transportation casks. The applicant estimated an annual rate of waste handling, including $3 \times 10^6$ kg [3,000 MTHM] of CSNF; 763 defense HLW canisters; 179 DOE standardized canisters; and 24 naval SNF canisters.

Most waste shipped to the repository site will be canistered. The applicant indicated in SAR Section 1.5.1 and described in BSC (2007bh) that about 90 percent of the CSNF will be loaded in TAD canisters prior to being placed in transportation casks for shipping. The remaining 10 percent will be either in the form of bare fuel assemblies (uncanistered) or loaded in DPCs before being shipped to the repository in transportation casks. The HLW glass will be in HLW canisters, DOE SNF will be either in DOE standardized canisters or multicanister overpack (MCO) canisters, and naval SNF will be loaded in U.S. Navy-designed canisters.

According to the applicant (BSC, 2007bh), most canistered waste will be either (i) transferred into aging overpacks for aging and transferred into waste packages for disposal after the waste is aged or (ii) directly transferred into waste packages for emplacement when it is received at the surface facility. The bare fuel received at the site will be repackaged into TAD canisters in the WHF before being loaded into waste packages or aging overpacks. The received DPCs may be transferred into overpacks or horizontal shielded transfer casks in the CRCF or RF for aging. After aging, these DPCs will then be transferred into TAD canisters in the WHF. Alternatively, the SNF assemblies in DPCs can first be transferred into TAD canisters in the WHF. Then, the TAD canisters will be either loaded into aging overpacks for aging or into waste packages in the CRCF for disposal.

The waste receipt and handling throughputs for the surface facilities during the preclosure period and the number of canister and SNF assembly transfers in each surface facility were provided in SAR Tables 1.2.1-1 and 1.7-5. The expected number of occurrences of the event sequences and categorization of event sequences performed in the applicant's PCSA was based on these throughput numbers, which were used as point estimates in the event tree analysis for quantification of event sequences. These throughput numbers were the same as the mean values listed in BSC (2007bh). SAR Section 1.2.1.1.2 provided the annual rate estimates of waste handling, as described previously. The annual rate information was necessary for the applicant to assess normal operational exposures in demonstrating compliance with 10 CFR 63.111(a)(1).

The applicant indicated that the quantification of event sequences in PCSA involving MCO canisters was not addressed in the SAR Section 1.5.1.3.1.2.9. The applicant stated that the basis for MCO acceptance and disposal will be included in an update of the license application and that it will follow the processes prescribed in 10 CFR 63.22 and 10 CFR 63.46, as appropriate, to obtain authorization to receive DOE SNF in MCOs (DOE, 2009av,gw). In addition, SAR Section 1.5.1 stated that the applicant has not completed the necessary safety analyses for commercial mixed oxide fuel. As in the case of aforementioned MCOs, the applicant stated that it plans to include MOX fuel waste in future licensed operations and will follow the aforementioned processes, as appropriate, to obtain authorization to receive this waste when the safety analyses have been completed.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's waste throughput information using the guidance in YMRP Section 2.1.1.2. The NRC staff finds that the applicant's throughput numbers listed in SAR Tables 1.2.1-1 and 1.7-5, except information for MCO canisters and MOX fuel, which are discussed next, are sufficient to permit an evaluation of the applicant's event sequence quantification and categorization in the PCSA because these numbers represent the maximum capacity and rate of receipt during the preclosure period for the various waste forms and canisters to be handled in various facilities (DOE, 2009dz).

The NRC staff finds that the description of MCO canisters is not adequate for review of event sequences associated with MCO canisters in the PCSA, because DOE has not provided sufficient design information and reliability analyses necessary to determine nuclear safety design bases for the MCO canisters (SAR Section 1.5.1.3.1.2.9). Therefore, the NRC staff determines that the applicant cannot receive DOE SNF in MCO canisters. The NRC staff further determines that, if the applicant wishes to receive DOE SNF using MCO canisters, it will need to update the SAR and obtain NRC approval to receive DOE SNF in MCO canisters. Thus, the NRC staff finds that the throughput information for all forms of canisters, except for MCO canisters, is adequate for the NRC staff to review the applicant's PCSA.

The applicant has also not presented safety analyses for handling commercial MOX fuel and has stated that it will follow the same procedure as for MCOs in obtaining authorization to receive this waste. Therefore, the NRC staff finds that DOE has also not provided necessary analyses regarding MOX fuel and, if DOE wishes to receive MOX fuel, it will need to seek NRC authorization. For the foregoing reasons, the following condition for construction authorization should be included:

**Proposed Condition of Construction Authorization**:

> DOE shall not, without prior NRC review and approval, accept DOE spent nuclear fuel (SNF) in multicanister overpacks (MCOs) or commercial mixed oxide (MOX) fuel.
>
> Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these MCOs and MOX fuel at the GROA or (ii) demonstrates, through the PCSA, that the MCOs and MOX fuel can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

***Operational Period***

The applicant indicated in the SAR and in DOE (2009av, Section 2.2) that the total preclosure period will be 100 years, while the receipt and emplacement operations period is projected to span 50 years. In addition, as stated in DOE (2009av, Section 1.1.2.1), the surface facilities will have a design operating life of 50 years. The applicant also used screening criteria of $2 \times 10^{-6}$ for an aircraft crash, on the basis of a 50-year preclosure operating period, and used a 50-year exposure time for surface facility structures to screen tornado missiles, as discussed in SER Section 2.1.1.3.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on the operational period using the guidance in YMRP Section 2.1.1.2. The NRC staff finds that the applicant's projected operational period is acceptable because the 50-year period is (i) consistent with the rates and capacities for the SSCs involved with the receipt and emplacement operations reviewed in SER Section 2.1.1.2 and (ii) used for initiating event screening and event sequence categorization in the PCSA. Although the preclosure period is 100 years in duration, the DOE has stated it will emplace the waste during the initial 50 years. Thus, operational activities for emplacing waste will occur over a 50-year period.

**NRC Staff's Conclusion**

On the basis of the evaluations discussed in SER Section 2.1.1.2.6.1 and the proposed condition of construction authorization, the NRC staff finds, with reasonable assurance, that the applicant's description of the surface and subsurface facility operations, waste form throughput, and operational period meets the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided an adequate description of the surface and subsurface facility operations, including waste form throughput and an operational period sufficient for the NRC staff to evaluate the applicant's PCSA and design.

2.1.1.2.3.6.2 Instrumentation and Control Systems

The applicant provided information on instrumentation and control (I&C) and related communications systems in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.1–1.3.6, 1.4.2, 1.9, 1.13, 5.5, and 5.6 for the surface and subsurface facilities. This information also included conceptual process diagrams, equipment outline drawings, and digital control logic diagrams for various ITS and non-ITS controls. The NRC staff evaluated this information to determine whether the applicant acceptably described (i) control philosophy, conceptual process diagrams, and digital control logic diagrams for ITS and non-ITS controls; (ii) design codes, standards, and acceptable industry practices used for ITS and non-ITS controls; and (iii) plans and procedures for initial startup, operation, maintenance, and periodic testing of ITS and non-ITS controls. For I&C and related communication systems to be used in the underground inaccessible areas, the NRC staff also evaluated the adequacy of the I&C and related communication systems design descriptions to support potential operations and/or waste retrieval subject to 10 CFR 63.111(e).

***Surface and Subsurface Facility Instrumentation and Control***

Most normal facility operations will be based on repetitive cask unloading, transfer, repackaging, and reloading steps; hence, the normal facility production/throughput functions implement automation where practical. Such automation will typically use non-ITS programmable logic controllers (PLCs) or other non-ITS digital devices to control machines that will be specially designed to handle the shipping overpacks, waste packages, and storage canisters. The control philosophy for non-ITS I&C SSCs was provided in SAR Section 1.4.2.1.1. Codes and standards for the design and application of non-ITS I&C SSCs were provided as general references in SAR Section 1.4.2.6. Specific descriptions of non-ITS I&C SSCs, functions, and operations were in the SAR in descriptions and figures provided for specific facilities, systems, and other SSCs.

The ITS control philosophy was described generally in SAR Section 1.4.2, and more specifically in SAR Sections 1.2.4, 1.2.5, and 1.2.8, for relevant surface operating facilities. ITS heating, ventilation, and air conditioning (HVAC) and ITS electrical power system SSCs, including ITS diesel generators, also will include ITS I&C SSCs, as generally described in SAR Sections 1.2.2.3 and 1.4.1.2, respectively. The applicant described no ITS I&C SSCs for subsurface facilities.

In general, ITS controls will be made up of individual instruments, sensors, or devices that will be hardwired to control devices to perform safety-related control functions, interlock functions, and other protective functions. The applicant stated that all ITS controls and interlocks that implement safety functions needed for preventing event sequences and mitigating consequences will be hardwired and cannot be overridden by non-ITS, automation-based controls.

The SAR contained conceptual process diagrams and logic diagrams for SSCs containing ITS and non-ITS controls. In its response to an NRC staff RAI (DOE, 2009dk), the applicant identified the ITS controls and the related safety functions that will be implemented for the CRCF. These ITS controls were considered representative of designs for ITS controls for other surface facilities. The applicant provided supplemental information regarding the applicability of cited principal codes and standards (DOE, 2009dl) and discussed the proposed application of specific sections of principal codes and industry standards that the applicant will apply to the final design of the ITS controls (DOE, 2009do). The applicant proposed to use IEEE 308, 379, 384, and 603 (Institute of Electrical and Electronics Engineers, 2001aa,ab; 1998aa,ab) and ASME NOG–1–2004 Section 6000 for Type I cranes (American Society of Mechanical Engineers, 2005aa) as the principal codes and standards for ITS control design. These standards describe the need to incorporate design criteria such as redundancy, spatial separation, independence between redundant channels, and isolation between safety and nonsafety circuits. The applicant further described how it would interpret the applicability of specific sections of principal codes and standards to ITS controls.

A high-level description of the proposed environmental qualification process, maintenance, and functional testing procedures for ITS and non-ITS controls was provided in SAR Section 1.13.2. The applicant proposed that ITS I&C equipment associated with ITS cranes will be environmentally qualified in accordance with IEEE 323–2003 (Institute of Electrical and Electronics Engineers, 2004aa) and the equipment qualification program will be developed consistent with Regulatory Guide 1.89 (NRC, 1984aa). SAR Sections 5.5 and 5.6 stated that channel functional tests and channel calibrations for control systems will be performed (specifically, tests for cranes, trolleys, heating, ventilation, and air conditioning (HVAC) systems, and TEV were outlined in SAR Table 5.5-1). Test procedures for control systems will be developed consistent with Regulatory Guide 1.30 (NRC, 1972aa).

Specific plans and procedures for preventive and corrective maintenance of ITS controls and ITS SSCs have not been completed and will be developed during detailed design (DOE, 2009dk,dm). The applicant also stated that preventive maintenance of hardwired ITS interlocks will be based upon manufacturer's recommendations, industry codes and standards, and equipment qualification and reliability requirements from the PCSA, as identified in SAR Section 1.9. The applicant further stated that potential future upgrade of the ITS interlocks will be based upon a reliability-centered maintenance program and that a reliability-centered maintenance process will be used to develop plans and procedures by analyzing the inspection, testing, and maintenance needs for each component. The applicant stated that safety controls will be designed in accordance with applicable criteria in IEEE 603–1998, Paragraphs 5.7–5.12

(Institute of Electrical and Electronics Engineers, 1998ab) to ensure testability and maintainability (DOE, 2009do).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description and design information for I&C SSCs using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions and functions of the I&C SSCs for ITS HVAC and ITS electrical power system SSCs and for the systems that will be used to handle the overpacks, waste packages, and storage canisters. The NRC staff finds that the applicant's description and design information for ITS I&C SSCs (ITS controls) are adequate because the applicant (i) described the ITS control philosophy (e.g., all ITS controls and interlocks that implement safety functions needed for preventing event sequences and mitigating consequences will be hardwired and cannot be overridden by non-ITS automation-based controls); (ii) described the ITS I&C SSCs and their functions for the (a) ITS HVAC system, (b) ITS electrical power system, and (c) ITS diesel generators; (iii) described the hardwired ITS I&C SSCs that will provide the reliability to ensure the associated ITS systems and components will perform the intended safety functions; (iv) described the proposed environmental qualification process, maintenance, and functional testing procedures for ITS and non-ITS controls (e.g., ITS cranes will be environmentally qualified in accordance with IEEE 323–2003, the equipment qualification program will be developed consistent with Regulatory Guide 1.89, and test procedures for control systems will be developed consistent with Regulatory Guide 1.30); and (v) identified applicable sections of the codes and standards for design of ITS controls.

The NRC staff notes that the applicant did not identify any ITS I&C SSCs for the subsurface facility.

The NRC staff finds that the applicant's description and design information for non-ITS I&C SSCs in the GROA is acceptable to permit an evaluation of the PCSA and design of the non-ITS I&C SSCs because the applicant provided (i) adequate information regarding the function and use of particular non-ITS I&C SSCs within descriptions of other GROA facilities and operating systems, (ii) adequate descriptions of controls and monitoring for the GROA and a high-level description of distributed control philosophy, and (iii) codes and standards that are consistent with standard engineering practices for non-ITS I&C SSCs performing similar functions.

On the basis of the above evaluation, the NRC staff finds that the description and design information the applicant provided for surface and subsurface facility instrumentation and control are sufficient to permit an evaluation of the PCSA and the instrumentation and control design.

***Special Vehicle Instrumentation and Control***

The ITS TEV was described in SAR Sections 1.3.2.1, 1.3.3.5.1.1, and 1.3.4.8. The ITS electrical components of the TEV will include the mechanical location switch and shield door motors (DOE, 2009dp). The applicant confirmed that design criteria in IEEE 384–1992 and IEEE 603–1998 (Institute of Electrical and Electronics Engineers, 1998aa,ab) will be used for the TEV ITS electrical components and interlocks (DOE, 2009dp).

Programmable logic controllers (PLCs) will be used for remote-controlled, non-ITS operations of the TEV. In response to an NRC staff RAI (DOE, 2009dm), the applicant confirmed the use of ASME NOG–1–2004 Sections 6410 to 6419 (American Society of Mechanical Engineers, 2005aa), NUREG/CR–6090 (Wyman, 1993aa), and portions of International Electrotechnical Commission (IEC) 61131-3 (Part 3) (International Electrotechnical Commission, 2003aa) for the PLC design. The applicant stated that the onboard PLC will be non-ITS because it will not perform ITS safety functions and will not be relied upon to prevent or mitigate an event sequence. The TEV ITS mechanical location switch (activated when the TEV is in an emplacement drift) and the onboard PLC will be functionally independent of each other. When the ITS location switch is deactivated, a remote operator will not be able to inadvertently open the TEV shielded enclosure doors using the non-ITS PLC on the TEV. The TEV will have an air conditioning unit and a fire protection system for onboard temperature-sensitive electronic components.

The non-ITS drip shield emplacement gantry (DSEG) will be a custom vehicle designed specifically to install drip shields over waste packages in the emplacement drifts before repository closure (SAR Section 1.3.4.7.2 and Figure 1.3.4-17). The DSEG will be controlled by an onboard PLC network, which controls cameras, high-intensity lights, and thermal and radiological sensing instruments. Like the TEV, the DSEG will have an air conditioning unit for temperature-sensitive electronic components and a fire protection system. The applicant stated that the DSEG control system will be similar to the system used on the TEV (SAR Section 1.3.4.7.2). The applicant indicated in SAR Table 1.3.2-4 that DSEG design generally will conform to ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa,).

According to SAR Section 1.3.1.2.1.6 and BSC (2008ca), the applicant proposed a number of planned non-ITS remotely operated vehicles (ROVs) for performing visual inspections, material sampling, and potential maintenance and repair tasks within emplacement drifts after the emplacement of waste packages. ROVs would also be used in other inaccessible subsurface areas. Remote observations will be conducted using onboard video cameras to monitor the drift condition, drift ground support system, and emplaced waste packages. Inspections will involve monitoring drift stability and the status of the rail or other systems. In addition, the emplacement drift ROV will utilize sensing devices to measure temperature and other environmental conditions, and make remote observations of potential seepage in the drift. Additional versions of ROVs will be used in nonemplacement areas that are inaccessible for human entry, such as exhaust mains and shafts. The SAR also described a potential need for additional, unplanned special purpose ROVs designed specifically for performing off-normal or unplanned inspection, observation, maintenance, or repair activities in inaccessible areas.

In SAR Section 4.2.1.8, the applicant indicated that while the technology to remotely inspect emplacement drifts is available, the high-temperature and high-radiation environments representative of postemplacement conditions within the drifts will require developing a first-of-a-kind application of existing technologies to build ROVs able to perform the intended operations and inspections. There are similarities between the design of the emplacement drift ROV and the design of the TEV and the DSEG pertaining to how the ROV will be expected to move among the drifts and use I&C and supporting communications. The emplacement drift ROV will have enhanced monitoring capabilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's descriptions and design information for special vehicle I&C using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the I&C SSCs for the TEV, DSEG, and ROVs, including the intended functions and design information of the ITS I&C SSCs for the TEV. The NRC staff finds that the applicant's description and design information for TEV I&C is adequate because the applicant (i) described the ITS electrical components of the TEV (e.g., mechanical location switch, shield door motors) and non-ITS electrical components of the TEV (e.g., programmable logic controllers used for remote-controlled, non-ITS operations); (ii) discussed that the TEV will have an air conditioning unit and a fire protection system for onboard temperature-sensitive electronic components; (iii) described the design and associated safety functions of the TEV ITS I&C; and (iv) provided the applicable codes and standards used for design. A safety evaluation of the design of TEV ITS I&C SSCs is included in SER Sections 2.1.1.6.3.2.8.3 and 2.1.1.7.3.5.

The NRC staff determines that the applicant's description and design information for non-ITS I&C for the DSEG is adequate because the applicant (i) described the specialized function of the DSEG (e.g., custom vehicle to be designed specifically to install drip shields over waste packages in the emplacement drifts); (ii) discussed how the DSEG will be controlled by an onboard PLC network, which controls cameras, high-intensity lights, and thermal and radiological sensing instruments; (iii) explained that the DSEG will have similar control systems as the TEV, an air conditioning unit for temperature-sensitive electronic components, and a fire protection system like the TEV; and (iv) provided the applicable codes and standards used for design.

The NRC staff finds that the applicant's descriptions and design information for the planned ROVs provided in the SAR and BSC (2008ca) are adequate because the applicant (i) described the planned functions of the ROVs (e.g., performing visual inspections, material sampling, and potential maintenance and repair tasks within emplacement drifts after the emplacement of waste packages); (ii) provided discussions of the intended concepts for design and operations, including outline drawings and other diagrams at the current stage of design; and (iii) explained that the ROVs represent a first-of-a-kind application of existing technologies that would have similarities with the design of the TEV and the DSEG, pertaining to how the ROV will be expected to move among the drifts, and would use I&C and supporting communications. The capability to reliably perform observations and inspections and potential maintenance of inaccessible underground openings and SSCs within them throughout the preclosure period is discussed in SER Section 2.1.1.2.3.7.

On the basis of the above evaluation, the NRC staff finds that the descriptions and design information the applicant provided for special vehicles (TEV, DSEG, and ROVs) instrumentation and control are sufficient to permit an evaluation of the PCSA and the instrumentation and control design.

***Subsurface Ventilation, Instrumentation, and Control***

Drift temperature, pressure, and relative humidity are important parameters indicating the effectiveness of the ventilation system and will be closely monitored. The applicant indicated that the proposed sensors/monitors needed to determine the effectiveness of the subsurface ventilation system will not be required to operate under extreme environmental conditions (DOE, 2009dm). This is because the primary use of the subsurface facility sensors will be to monitor temperature, barometric pressure, relative humidity, and dose rate in subsurface facility places, which are accessible to repository personnel, to ensure that ventilation to the emplacement drifts is maintained at design values.

The sensors were identified as non-ITS; therefore, commercial-grade sensors, which are environmentally qualified for the expected environment, will be used. However, the applicant indicated that selection of these industrial-grade components will be based on the guidance provided within Regulatory Guide 1.23 (NRC, 2007aa) and in accordance with applicable sections of ANSI/ANS–3.1 1–2005 and EPA–454/R–99–005 (American Nuclear Society, 2005ab; EPA, 2000). In accordance with Regulatory Guide 8.8 (NRC, 1978ab), the monitoring sensors will be located and/or shielded so that they can function to help maintain occupational radiation exposures ALARA.

**NRC Staff's Evaluation**

The NRC staff reviewed the description and design information for controls for the subsurface ventilation system provided in the SAR using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the description and design information of the sensors for monitoring the performance of the subsurface ventilation system. The NRC staff finds that the applicant's description and design information for the controls for the subsurface ventilation system are adequate because the applicant (i) described the controls and instrumentation for monitoring performance of the subsurface ventilation system; (ii) stated that the selection of industrial-grade components will be based on the guidance provided in Regulatory Guide 1.23 (NRC, 2007aa) and in accordance with applicable sections of ANSI/ANS–3.1 1–2005 and EPA–454/R–99–005 (American Nuclear Society, 2005ab; EPA, 2000); and (iii) explained that monitoring sensors will be located and/or shielded so that they can function to help maintain occupational radiation exposures ALARA, consistent with Regulatory Guide 8.8.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the I&C for the subsurface ventilation system are sufficient to permit an evaluation of the PCSA and the instrumentation and control design.

***Digital Control Management Information Systems***

The Digital Control Management Information Systems (DCMIS) will be part of the proposed non-ITS I&C system and will provide control and monitoring for the GROA facilities during the preclosure period (SAR Section 1.4.2.1). The DCMIS will rely on the proposed communications system (described next) to "connect" operators in the central control center facility (CCCF) with non-ITS monitoring and controlling SSCs distributed throughout the GROA and nearby support facilities. In other words, the DCMIS will provide the CCCF operators with the capability to control and monitor the operations of the TEV, DSEG, and ROVs. The major components of the DCMIS will be controllers, human–machine interface consoles, input and output modules, engineering workstations, data historians, networks and network interface devices, and foreign-device interfaces.

The applicant provided applicable codes and standards for the DCMIS in SAR Section 1.4.2.1.3. In its response to an NRC staff's RAI (DOE, 2009du), the applicant identified additional applicable industry codes and standards for network interface design to protect the DCMIS from undesired interactions and intrusions.

The DCMIS architecture will use a redundant control network operating under a nonproprietary protocol to which a distributed set of local controllers, cameras, digital video multiplexers, and other devices can provide data-to-data historians (SAR Figures 1.4.2-1 and 1.4.2-2). The historians will make data available to a redundant supervisory network, which facility operators and managers may access to monitor the status of operations. The DCMIS will also be capable of transmitting data offsite (SAR Section 1.4.2). Firewall devices will be used to protect repository operations networks and offsite locations (DOE, 2009du). The applicant stated that it would incorporate criteria contained within NIST 800–53 (National Institute of Standards and Technology, 2007aa) and other standards, which will provide for the incorporation of security controls to help guard against intrusion. The applicant also stated that the repository cyber security program will be risk-based and will provide for continuous improvements to the protection of information and information systems through ongoing threat analysis and vulnerability assessments.

SAR Section 1.4.2 indicated that the DCMIS controllers will process remote operators' commands and execute the logic to control virtually all operations within the proposed GROA and adjacent facilities, with the exception of some mechanical handling equipment such as locally controlled jib cranes. Although there was no specific information in the SAR regarding the design characteristics for the local controllers, in its response to an NRC staff RAI (DOE, 2009dn), the applicant stated that local controllers, in general, will be PLCs. Operational process information provided in the SAR for applicable SSCs (e.g., CRCF waste package load-out subsystem, SAR Section 1.2.4.2.4) described operations whereby local controllers would be used to regulate GROA operations so that the combination of hardware and software will be configured to wait for permissive signals from either a local or remote operator before the automation functions will be able to proceed. The applicant provided the design characteristics for the local controllers to include separation of power supply feeds and digital input/output modules, diagnostics, and shielding (DOE, 2009dn). The applicant also defined high-level plans for periodic calibration and surveillance requirements for analog signals.

Controllers and input/output modules (non-ITS, non-Class 1E equipment) will be distributed throughout the GROA and will be close to the signal source. To ensure that the DCMIS will be able to perform monitoring functions during a loss of normal power, portions of the DCMIS will be powered by ITS uninterruptible power supplies (UPSs) (SAR Sections 1.4.2.1 and 1.4.1.1.1.5).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description and design information for DCMIS using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the design information and locations and functional arrangements of the DCMIS. The NRC staff finds that the applicant's description and design information for DCMIS are adequate because the applicant (i) described the functions of the DCMIS to provide the CCCF operators with the capability to control and monitor the operations of the TEV, DSEG, and ROVs; (ii) identified the major components of the DCMIS (e.g., controllers, human–machine interface consoles, input and output modules, engineering workstations, data historians, networks and network interface devices, and foreign-device interfaces); (iii) explained that the DCMIS architecture will utilize a redundant control network operating under a nonproprietary protocol to which a distributed set of local controllers, cameras, digital video multiplexers, and other devices can provide data to data historians; (iv) described that the DCMIS will be capable of transmitting data offsite and firewall devices will be used to protect repository operations networks and offsite locations; (v) explained that security controls to help guard against intrusion will be incorporated consistent with criteria contained within NIST 800–53; and (vi) explained that portions of the DCMIS will be powered by ITS uninterruptible power supplies.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the digital control management information systems (DCMIS) are sufficient to permit an evaluation of the PCSA and the DCMIS design.

*Communications System*

The communications system, which the applicant categorized as non-ITS, will facilitate interchange of video, voice, and data communications for surface and subsurface facilities during the preclosure period. Communications will be provided for the geologic repository operations area (GROA) facilities using both wired and wireless media.

Two-way radio communications will be used to facilitate voice operations during emergencies, and hardwire telephone lines will be used to facilitate offsite voice and data communications in the event of a site emergency.

The DCMIS will be supported by a dual-ring network topology that provides the physical transport media within a Synchronous Optical NETwork (SONET) communications backbone, which will be a permanently installed (wired) network. The proposed SONET architecture will consist of a redundant, fiber optic ring connecting all network nodes. One ring will be typically the active ring and will be referred to as the working facility, while the other ring will be the standby ring, referred to as the protection facility (Black and Walters, 2001aa). The applicant identified codes and standards applicable to the communications systems in SAR Section 1.4.2.4.3.

Radio-frequency wireless transmission communications systems will be provided to interconnect the DCMIS and the mobile TEV, DSEG, and ROVs. The applicant stated that the wireless communication system will meet Federal Communications Commission standard 47 CFR Part 15, to prevent interference with operations within and external to the communications system. The communications system's functional organization, network architecture, organization, and site topology were presented in SAR Figures 1.4.2-5 to 1.4.2-8.

To protect the communications system from possible compromise due to deliberate attacks or naturally occurring phenomena, the applicant (DOE, 2009eg) stated that it will incorporate the methods and practices of NIST 800–53 and NIST 800–53A (National Institute of Standards and Technology, 2008aa; 2007aa). In the event of a loss of offsite power (LOSP), the selected portions of the DCMIS and communications system will be designed to be powered by uninterruptable power supply (UPS), with a capacity to provide a minimum of 15 minutes of sustaining power (SAR Section 1.4.1.1.1.5).

The applicant described the dual-ring wired network for the subsurface facilities as physically separate and independent from the dual-ring SONET wired network that will serve the surface facilities; however, the surface and subsurface networks will be interconnected by firewall SSCs, as shown in SAR Figure 1.4.2-8. SAR Figure 1.4.2-8 illustrated how SONET nodes will be installed in several electrical equipment alcoves positioned along the access mains and how they will interface with various radio frequency transceivers collocated in the same alcoves. SAR Figure 1.4.2-9 depicted the subsurface wireless configuration for the access main and emplacement drifts.

The communication system will make extensive use of both wired and wireless technology. The wired component will consist of both fiber optic and copper cabling. A radio frequency radiating coaxial cable antenna will wirelessly connect mobile transceivers within the wired (SONET) communications system and the various vehicles to transmit video and data between the vehicle-mounted operation/sensor SSCs and the CCCF. The TEV will carry a battery backup power system that maintains onboard communications with the CCCF for an undefined period of time if normal power becomes unavailable. The applicant confirmed that the radiating cables, antennas, and transceivers will be installed in the access mains and alcoves of the subsurface facility and, in some cases, they will be located in the air intake shafts.

The applicant proposed to use antennas located in the access main and just inside drift entrances (SAR Section 1.4.2.4.1.7 and SAR Figure 1.4.2-9) to provide reliable wireless mobile data communications between DCMIS and the TEV when located in inaccessible turnouts and emplacement drifts. In a response to an NRC staff's RAI, the applicant also suggested use of a slotted microwave guide system (DOE, 2009ee) as an alternate communications approach. The system, according to the applicant, would provide payload data rates of up to 54 Mbps and should provide adequate capacity for communications with the TEV. The applicant also described an additional alternative communication system using power line carrier transmission techniques operating through vehicle sliding electrified third rail contacts. Bandwidth of up to 100 Mbps through static power lines is based on proposed IEEE P1901 (Institute of Electrical and Electronics Engineers, 2010aa). The applicant stated that it will further develop these and other potential in-drift communications alternatives during the detailed design phase (DOE, 2009ee).

The SAR contained no description of wired communications provisions to inaccessible enhanced characterization of the repository block (ECRB) cross drift and exhaust mains and shafts where maintenance activities will be conducted, according to BSC (2008ca). The applicant stated in BSC (2008ca) that inspection and potential maintenance operations for these areas will be performed by the applicant using one or more types of ROVs and mobile communications.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description and design information for the non-ITS communications systems in the geologic repository operations area (GROA) facilities using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the design information and locations and functional arrangements of the non-ITS wired communications systems. The NRC staff also reviewed the description of the interactions of the non-ITS wired communications systems with other systems. The NRC staff finds the applicant's description and design information for the communications system are adequate because the applicant (i) described non-ITS wired and mobile communications between operators and the TEV, DSEG, and other ROVs through antennas, slotted microwave guides, electrified third power rails, or other alternative means using commercially available products; (ii) explained that two-way radio communications will be used to facilitate voice operations during emergencies, and hardwire telephone lines will be used to facilitate offsite voice and data communications in the event of a site emergency; (iii) described the radio-frequency wireless transmission communications systems to be used to interconnect the DCMIS and the mobile TEV, DSEG, and ROVs and stated that the wireless communication system will meet Federal Communications Commission standard 47 CFR Part 15; (iv) described use and location of antennas in the access main and just inside drift entrances; and (v) provided the standards and codes pertaining to the protocols and interfaces used for wired communications systems, which are consistent with the standard engineering practices for these systems.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the communications system are sufficient to permit an evaluation of the PCSA and design.

### *Radiation/Radiological Monitoring System*

The function of the radiation/radiological monitoring system (RMS) will be to monitor both surface and subsurface facilities during the preclosure period. The applicant categorized the RMS as non-ITS. SAR Figure 1.4.2-3 provided the RMS functional block diagram for the GROA. The major components will be area radiation monitors, continuous air monitors, and airborne radioactivity monitors. The facility RMS will be powered by a set of uninterruptible power supplies (UPSs). Although the monitoring equipment will be able to alert operators to the occurrence of Categories 1 or 2 event sequences, or potential off-normal radiological releases, the RMS will not alert operators to take manual action to mitigate an analyzed event.

According to the applicant, area radiation monitors will not be required for the subsurface facilities, because administrative controls will be used to prevent personnel from entering areas that potentially will contain high levels of radiation. However, continuous air monitors will be located at strategic places within the subsurface nonemplacement areas to sample airborne radioactivity effluent particulate and gases leaving the exhaust shafts and to continuously monitor (particulate only) air in the access main, the alcoves, and other personnel work areas.

The applicant identified the standards and codes for the RMS in SAR Section 1.4.2.2.2. The applicant stated (DOE, 2009dm) that evaluation and selection of area radiation monitors and continuous air monitors would follow the guidance of ANSI/ANS–HPSSC–6.8.1–1981, ANSI N42.17B–1989, and ANSI N42.18–2004 (American Nuclear Society, 2004aa, 1988aa, 1981aa). The applicant confirmed that only the sensor probes for the instruments located at the ventilation shaft collars will be exposed to greater than ambient temperatures, as the exhaust air will be heated by decay heat from the emplaced waste. The exhaust shaft sensors will monitor for airborne radioactivity.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description and design information for RMS using the guidance in YMRP Section 2.1.1.2. The NRC staff reviewed the descriptions of the design information of the RMS. The NRC staff finds the applicant's description and design information for the RMS are adequate because the applicant (i) described the major components of the RMS (i.e., area radiation monitors, continuous air monitors, and airborne radioactivity monitors); (ii) provided the RMS functional block diagram for the geologic repository operations area; (iii) explained that the facility RMS will be powered by a set of uninterruptible power supplies; (iv) described the continuous air monitors located at strategic places within the subsurface nonemplacement areas to sample airborne radioactivity effluent particulate and gases leaving the exhaust shafts; (v) described the plans for continuously monitoring (particulate only) air in the access main, the alcoves, and other personnel work areas; and (vi) identified the standards and codes used in the design of the RMS, which are consistent with the standard engineering practices for radiation monitoring—these codes and standards contain recommendations for obtaining valid samples of airborne radioactive material in effluents and guidelines for sampling from ducts and stacks. In particular, the applicant cited ANSI N42.18–2004 (American National Standards Institute, 2004aa) that specifies guidelines for protection of instrument systems from environmental conditions (e.g., high temperatures).

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the RMS are sufficient to permit an evaluation of the PCSA and design of the RMS.

*Environmental/Meteorological Monitoring System*

The environmental/meteorological monitoring system, which the applicant classified as a non-ITS system, will monitor seismic and meteorological parameters for the GROA through the preclosure period and transmit the collected data through the Digital Control Management Information System (DCMIS) so that the data will be available in the central control center facility (CCCF). The system will perform only monitoring functions, and there will be no control functions associated with it. Remotely located environmental/meteorological equipment will be powered by solar panels with battery backup, and by UPS for other equipment. The applicant confirmed (DOE, 2009dq) that the sensors/monitors to be used will not be Class 1E equipment, and there will be no plans to qualify these environmental sensors/monitors (SAR Section 1.13) for harsh environments.

Meteorological instruments will monitor wind speed, wind direction, temperature, humidity, barometric pressure, solar radiation, and precipitation. SAR Figure 1.4.2-4 provided a functional block diagram of the system. The general design requirements for this system will be similar to those used for other types of nuclear facilities ANSI/ANS 3.11–2005 and Regulatory Guide 1.23 (American Nuclear Society, 2005ab; NRC, 2007aa), except that the elevation locations for key sensors will be commensurate with the operations requirements for the GROA.

The seismic monitoring subsystem will consist of triaxial accelerometers, which will be hardwired to seismic motion analysis equipment, and a postevent monitoring console in the CCCF. The system design will be similar to that described in Regulatory Guide 1.12 (NRC, 1997af).

**NRC Staff's Evaluation**

The NRC staff reviewed the description and design information for the environmental/meteorological monitoring system provided in the SAR using the guidance in YMRP Section 2.1.1.2. The NRC staff finds that the applicant's description and design information for the environmental/meteorological monitoring system are adequate because the applicant (i) described the meteorological instruments to be used to monitor wind speed, wind direction, temperature, humidity, barometric pressure, solar radiation, and precipitation, and provided a functional block diagram of the system; (ii) described how the seismic monitoring subsystem will consist of triaxial accelerometers, which will be hardwired to seismic motion analysis equipment, and a postevent monitoring console in the CCCF; and (iii) explained that system designs for the monitoring systems will be similar to the system design described in Regulatory Guide 1.12.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the environmental/meteorological monitoring system are sufficient to permit an evaluation of the PCSA and design.

**NRC Staff's Conclusion**

On the basis of evaluations discussed in SER Section 2.1.1.2.3.6.2, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the I&C for the GROA meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), and 10 CFR 63.112(a) because the applicant provided adequate description and design information for the I&C for the GROA sufficient for the NRC staff to evaluate the PCSA and design.

#### 2.1.1.2.3.7 Design of Subsurface Facility Structures, Systems, and Components

This SER section evaluates the design of subsurface structures and systems to determine the capability of these structures and systems to perform the functions the applicant defined. The NRC staff's evaluation of the design of structures, systems and components (SSCs) that are important to safety (ITS) is presented in SER Section 2.1.1.7. The evaluation in this section focuses on the design of structures and systems that are not evaluated in SER Section 2.1.1.7 but which the applicant will rely on to perform functions important to subsurface facility operations relevant to the applicant's demonstration of compliance with NRC regulations.

The applicant described the subsurface facility operations in SAR Section 1.3.1.2. The applicant stated that the operations will include (i) waste package transportation and emplacement, (ii) waste package ventilation to support thermal management, (iii) repository performance monitoring, (iv) waste retrieval if necessary, and (v) repository closure. The applicant explained in SAR Sections 1.3 and 1.4 and in its responses to the NRC staff requests for additional information (RAIs) (DOE, 2009bb,ed) that the subsurface facility structures and systems will be designed to provide several functions to support the subsurface facility operations. These functions will include (i) base support for crane rails and the operating envelope for the TEV, DSEG, and remote-controlled equipment for postemplacement inspection and monitoring of emplacement drifts; (ii) alignment support for crane rails, third rail for power supply, and communications for remote vehicle control and inspection (DOE, 2009ee); and (iii) fresh air or exhaust air conduits for waste package ventilation, designed to provide a continuous air flux of 15 $m^3/s$ [32,000 cfm] through the emplacement drifts during the preclosure period.

In order for subsurface facility structures and systems to function through the preclosure period, the applicant indicated that underground openings and the inverts will remain stable and retain their as-designed alignments and grades through the preclosure period. The applicant (DOE, 2009ed) stated that it established the appropriate design criteria and bases to ensure stability of the structures and systems and will implement a monitoring, inspection, and maintenance program to ensure the structures and systems will perform their functions. As explained in SAR Section 1.11 and evaluated by the NRC staff in SER Section 2.1.2, the applicant will rely on subsurface facility structures and systems to be available and perform their functions so that waste packages would be accessible through the preclosure period, and any necessary retrieval could be performed by reversing the operational procedures used for waste emplacement.

Therefore, the NRC staff's evaluation in this SER section focuses on determining whether the non-ITS subsurface facility structures and systems are acceptably designed to perform their functions through the preclosure period.

##### 2.1.1.2.3.7.1 Thermal Load and Ventilation Design

The applicant described and discussed the thermal management and loading strategy in SAR Section 1.3.1.2.5 and the subsurface facility ventilation design in SAR Section 1.3.5. The applicant categorized the subsurface ventilation system as non-ITS, because it will not prevent or mitigate an event sequence, and as non-ITWI, because the subsurface ventilation system will not function as a barrier to potential release during the postclosure performance period.

### *Thermal Management Analysis*

The applicant performed a three-step thermal management analysis to ensure compliance with the repository thermal limits described in SAR Section 1.3.1.2.5. The first step of this analysis involved development of a total system model that determined a range of possible waste streams and a representative limiting waste stream on the basis of several inputs, such as waste inventories at utilities and queuing priorities established through agreement between, for example, the applicant and utilities (BSC, 2007cb). The applicant's analysis also assumed that TAD canisters having a heat load as high as 22.0 kW [20.9 Btu/sec.] will be aged at the repository aging pads until the emplacement thermal load limit {18.0 kW [17 Btu/sec.]} is met. For the second step of analysis, the applicant used the estimated representative limiting waste stream to determine the waste package emplacement sequence that would result in meeting the local thermal loading condition, such as the midpillar index temperature (BSC, 2007cc).

The third and final step of the applicant's analysis involved evaluating the thermal-hydrologic, geomechanical, and geochemical response to the loading arrangement determined in the previous step (SNL, 2008ai). The applicant applied a number of criteria (i.e., waste package heat load at receipt and emplacement, waste package canister types, and line load limit) that were described in SAR Section 1.3.1.2.5. In its design analysis (SNL, 2008ai) and in response to the NRC staff's RAI (DOE, 2009ea), the applicant stated that emplacement will take place with three constraints: (i) the seven-waste-package running average midpillar temperature will be a maximum of 96 °C [205 °F], (ii) the maximum thermal load per waste package will be 18 kW [17 Btu/sec], and (iii) the maximum average line load will be 2.0 kW/m [0.61 kW/ft]. According to the applicant, the proposed thermal emplacement loading plan will result in satisfying the temperature limits specified in SAR Table 1.3.1-2.

The 18.0 kW [17 Btu/sec] maximum waste package heat load and 2kW/m [0.61 kW/ft] linear heat load, considered at emplacement in the final step of the analysis, are substantially higher than those assumed in reference thermal loading in total system performance assessment (TSPA) {11.45 kW (10.85 Btu/sec) maximum waste package and 1.45kW/m [0.44 kW/ft] linear heat loads}. The thermal load reference case the applicant used to assess postclosure performance assumed instantaneous emplacement of all the waste followed by 50 years of forced ventilation at 15 $m^3$/s [32,000 cfm] with an efficiency of 86 percent heat removal. The 86 percent efficiency was obtained by integrating the local efficiency values over the drift length in space and the duration of preclosure period in time (BSC, 2004bg). In response to the NRC staff's RAI (DOE, 2009eb), the applicant asserted that ventilation efficiencies were calculated for a higher heat load of 2kW/m [0.61 kW/ft] that justified using an integrated efficiency value of 86 percent (BSC, 2008by). The proposed waste package arrangement analysis assumed phased, time-dependent emplacement, with ventilation lasting up to 100 years. A waste package will be subjected to a minimum of 50 years to a maximum of 100 years of cooling by ventilation, depending on the emplacement time of the waste package. The applicant performed a thermal analysis to show that the thermal load reference case bounds any thermal loading scenario on the basis of the proposed emplacement thermal load strategy. Results of a sample analysis were shown in SAR Figure 1.3.1-6.

### NRC Staff's Evaluation

The NRC staff reviewed the information on thermal loading strategy and thermal management using the guidance in YMRP Section 2.1.1.7.3.3(II). The NRC staff evaluated the applicant-provided information on design assumptions, constraints, design technical basis, uncertainty, and analytical or modeling techniques.

The NRC staff reviewed the applicant's analytical thermal loading calculations and finds that the applicant's information on thermal characteristics of the waste in the waste package emplacement plan is acceptable because the emplacement plan is consistent with the expected waste receipt and operations at the surface facilities. The NRC staff also evaluated the applicant's thermal-hydrologic, geomechanical, and geochemical studies of the repository for a given waste package emplacement sequence that, according to the applicant, will satisfy the preclosure and postclosure temperature limits. The NRC staff finds that the analytical methods used to assess the repository performance are acceptable because they are standard engineering techniques for thermal analyses. The NRC staff finds that the applicant's description of the thermal analysis technique is acceptable because the applicant used a process flow diagram (highlighted in SAR Figure 1.3.1-9) to illustrate steps involved. On the basis of these evaluations, the NRC staff finds that the applicant provided an adequate description and technical basis information for the thermal loading strategy.

The NRC staff also reviewed the applicant's description of how the proposed thermal loading will result in meeting the temperature limits of different in-drift components, as listed in SAR Table 1.3.1-2. These temperatures, which are relevant to ITS component integrity, include SNF cladding temperature on emplacement and maximum cladding temperature, waste package surface temperature, emplacement drift wall temperature, HLW waste form temperature, and naval SNF canister temperature. The applicant performed a thermal analysis to show that the thermal load reference case bounds any thermal loading scenario based on the proposed emplacement thermal load strategy (SAR Figure 1.3.1-6). However, the applicant stated that there is uncertainty in the heat load for the waste as received. DOE stated [SAR Section 1.3.1.2.5; DOE (2009ct, Enclosure 1); DOE (2009eb)] that it will develop a comprehensive emplacement plan prior to actual waste emplacement with specific information on waste characteristics, waste package emplacement location, and ventilation duration, and will use this emplacement strategy to demonstrate that the preclosure and postclosure temperature limits will be achieved (SAR Table 1.3.1-2). Therefore, the NRC staff finds the uncertainty in heat load is adequately addressed because the description addresses how the applicant will develop a comprehensive waste emplacement plan following the thermal load criteria described in the SAR prior to waste emplacement when specific information on waste characteristics, waste package emplacement location, and ventilation duration is available.

***Subsurface Facility Ventilation System***

The applicant will design a forced air subsurface ventilation system to remove heat from the emplaced waste and maintain temperature limits in the drift, as listed in SAR Tables 1.3.1-2 and 1.3.5-2, and to provide fresh air to personnel and equipment. The subsurface ventilation system components will include fans, isolation barriers, airflow regulators, access doors, and instrumentation for controlling and monitoring the system. An interconnected system of subsurface openings that will consist of intake ramps, access and exhaust mains, access turnouts, emplacement drifts, intake and exhaust shafts, and shaft access drifts will be utilized to circulate ventilation air. The ventilation system location and functional arrangement were described in SAR Section 1.3.5.1.2. The function of specific system components and their design was described in SAR Section 1.3.5.1.3. In SAR Section 1.3.5.1.3.2, the applicant described the operation of the ventilation system during simultaneous emplacement and development in which isolation barriers will be used to direct airflow in the desired direction. SAR Figure 1.3.5-5 showed the ventilation system layout after full emplacement, and SAR Figures 1.3.5-6 and 1.3.5-7 highlighted ventilation system operation during concurrent emplacement and development. The description of the airlock system and isolation barriers that will isolate (i) inlet airflow from exhaust airflow and (ii) the emplacement area from the development area was provided in SAR Section 1.3.5.1.3.2. The applicant plans to provide a nominal airflow rate of 15 $m^3/s$ [32,000 cfm] in each emplacement drift with thermal loading of up to 2.0 kW/m [0.61 kW/ft] and, if required, will be able to vary the drift airflow rate between 0 and 47 $m^3/s$ [0 and 100,000 cfm]. The applicant stated that the total power required for ventilation fans at the exhaust shaft will be approximately 1,343 kW [1,800 hp].

The applicant provided information on the operability of ventilation system components under normal and off-normal conditions. According to the applicant, large-diameter exhaust shafts will normally have two fans operating simultaneously, and each of the fans individually is capable of producing approximately 70 percent of the required airflow rate. The applicant also stated that small-diameter exhaust shafts will normally operate with only one fan delivering 100 percent of the required airflow, with another fan in standby. As described in SAR Sections 1.3.5 and 1.4.1.1.1.3, three of the exhaust fans will be connected to diesel standby generators, and all exhaust shaft fan pads will have connections for backup mobile diesel power generators. Therefore, the exhaust fans will continue to function during a loss of power because backup power is available.

SAR Section 1.3.5.4 identified the relevant codes and standards applied for designing the subsurface ventilation system. The steel structures will be designed in accordance with the methodology in American Institute of Steel Construction (1997aa). The subsurface ventilation system components that are located on the surface, such as the exhaust fan foundation, pad, and footings, will be designed according to International Building Code Seismic Use Group I and II (International Code Council, 2003aa). The applicant stated that it will use NFPA 801, NFPA 70 (National Fire Protection Association, 2005aa; 2003aa) to design the cables and other electrical components to minimize fire hazards. Other codes and standards related to features such as diesel use, air pollutant level, operational safety, and hazards were also provided in SAR Section 1.3.5.4. The applicant stated that these components will not be relied on to prevent or mitigate any event sequence, and use of specialized codes and standards that deal with nuclear air and gas, such as ASME–AG–1–2003 (American Society of Mechanical Engineers, 2004ac), will not be necessary.

**NRC Staff's Evaluation**

The NRC staff reviewed the description and design information for the subsurface facility ventilation system provided in SAR Section 1.3.5 using the guidance in YMRP Sections 2.1.1.2 and 2.1.1.7.3.3(II). The NRC staff reviewed the descriptions of the components for the subsurface facility ventilation system, including the intended functions and design information. The NRC staff finds that the applicant's description and design information for the subsurface ventilation system design are adequate because the applicant (i) described the ventilation system location and functional arrangement, including the function of specific system components and their design; (ii) described the operation of the ventilation system during simultaneous emplacement waste and development of drifts in which isolation barriers will be used to direct airflow in the desired direction; (iii) described the ventilation system layout after full emplacement; (iv) described the airlock system and isolation barriers that will isolate inlet airflow from exhaust airflow and the emplacement area from the development area; (v) provided information on the operability of ventilation system components under normal and off-normal conditions; (vi) explained that exhaust fans will be connected to diesel standby generators and all exhaust shaft fan pads will have connections for backup mobile diesel power generators in the event of a loss of electrical power; and (vii) provided codes and standards used for the design that are accepted by the industry and used in industrial installations.

The design and function of emplacement access doors and airflow regulators, under normal operating conditions, were described in SAR Section 1.3.5.1.3.3, which did not provide details on the operation and function of these components in the event of power failure. However, the applicant provided an analysis in SAR Section 1.3.5.3.2.1 concluding that in the absence of any ventilation for 30 days, in-drift components will not exceed their limiting temperatures. During a power failure, the emplacement access door will temporarily stop operation as the motorized actuator needs electrical power to function. The NRC staff notes that an immobile emplacement access door would not cause any safety hazard, because the access door has an emergency escape and maintenance access hatch for personnel to exit during off-normal operations, if needed. Hence, during a power failure, nonfunctional airflow regulators and louvers will not pose a safety hazard during the 30-day period, because the maximum allowable temperature limits of in-drift components will not be reached in the absence of ventilation, and the 30-day period allows sufficient time for power to be restored. The NRC staff finds that, in the event of a power failure, the components of the subsurface ventilation system will continue to operate normally because the applicant's fan installation design will have multiple sources of backup power, and temporarily nonfunctioning equipment would not pose a safety hazard.

On the basis of the above evaluation, the NRC staff finds that the applicant's description and design information for the subsurface ventilation system design are sufficient for evaluating the PCSA and design.

***Subsurface Ventilation System Maintenance***

The applicant described the subsurface ventilation system maintenance considerations in SAR Section 1.3.5.1.5. It asserted that ventilation fans will be monitored and maintained according to manufacturer guidelines, and the fans will be located on the surface, providing easy access for maintenance. According to the applicant, emplacement access doors will require regular periodic inspection with the bulkhead and frame requiring minimal maintenance. The applicant stated that emplacement door components will have a modular design that facilitates easy replacement. The applicant does not plan any routine maintenance activities for door actuators, which will be remotely monitored and replaced, if necessary. The applicant also anticipates that emplacement door actuators will operate only a few hundred times, as approximately 100 waste packages will be emplaced per drift.

SAR Section 1.3.5.3.2 presented an analysis of thermal effects under off-normal conditions, such as ventilation shutdown. The applicant considered three different cases: (i) analysis of complete ventilation shutdown in the absence of natural convection, (ii) naval SNF behavior under ventilation shutdown with natural convection, and (iii) thermal effect of drift obstruction. In the first analysis, the applicant demonstrated that waste package components will not reach their temperature limit within 30 days after loss of ventilation, as shown in SAR Figures 1.3.5-17 and 1.3.5-18. The thermal analysis of naval SNF, considering only natural convection, showed that the waste package temperature will be below values mentioned in SAR Table 1.3.1-2. The applicant also stated that the probability of an emplacement sequence within the drift, where a naval SNF waste package [(12.9 kW) (12.2 Btu/sec)] will be placed beside one with commercial spent nuclear fuel (CSNF) having the limiting thermal load 18.0 kW (17 Btu/sec), is extremely small. In the third analysis, the applicant showed that the ventilation system will be capable of maintaining normal airflow with 94 percent localized blockage of a single emplacement drift. The applicant also stated that any potential rockfall during the preclosure period in the lithophysal and nonlithophysal rock will be prevented by the perforated stainless steel sheet and rock bolts of the ground support system.

**NRC Staff's Evaluation**

The NRC staff reviewed the subsurface ventilation system maintenance considerations using the guidance in YMRP Sections 2.1.1.2 and 2.1.1.7.3.3(II). The NRC staff reviewed the descriptions of the maintenance activities for the ventilation fans and emplacement access doors. The NRC staff also reviewed the design features that will facilitate maintaining the subsurface facility ventilation system. Additionally, the NRC staff reviewed the applicant's analysis of thermal effects, in the event of ventilation shutdown. The NRC staff finds that the applicant's description of the subsurface ventilation system maintenance considerations are adequate because the applicant (i) explained that ventilation fans will be monitored and maintained according to manufacturer guidelines, and the fans will be located on the surface, providing easy access for maintenance; (ii) explained that emplacement access doors will be regularly inspected, while the bulkhead and frame will receive minimal maintenance; (iii) explained that emplacement door components will have a modular design that facilitates easy replacement and anticipates that emplacement door actuators will operate only a few hundred times, as approximately 100 waste packages will be emplaced per drift; and (iv) used standard techniques and tools to perform the thermal analyses that determined that the ventilation system will be capable of maintaining normal airflow with 94 percent localized blockage of a single emplacement drift.

On the basis of the above evaluation, the NRC staff finds that the applicant's description of the subsurface ventilation system maintenance is sufficient for evaluating the PCSA and design.

**NRC Staff's Conclusion**

On the basis of the evaluations discussed in SER Section 2.1.1.2.3.7.1, the NRC staff finds, with reasonable assurance, that the applicant's description, discussion, and design information for thermal management strategy meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), 10 CFR 63.112(a), and 10 CFR 63.112(f) because the applicant provided adequate description, discussion, and design information for the thermal management strategy sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed in NRC staff evaluations, the applicant (i) provided justification for the waste package emplacement sequence through an analysis that accounts for site-specific thermal properties, uncertainties, and engineering input parameters such as ventilation efficiency; (ii) provided design information and a design basis for subsurface ventilation system components; (iii) provided plans for inspection, maintenance, and replacement of critical components; and (iv) identified applicable codes and standards for ventilation system design.

2.1.1.2.3.7.2 Underground Openings in Accessible Areas

The applicant provided the design of underground openings in accessible areas of the subsurface facility in SAR Section 1.3.3. The applicant identified the underground openings as non-ITS. The applicant classified the subsurface facility into nonemplacement areas (SAR Section 1.3.3) and emplacement areas (SAR Section 1.3.4). In addition, in response to an NRC staff RAI on the applicant's approach to assure adequate functionality of the openings and their SSCs during the preclosure period, the applicant classified the openings as accessible or inaccessible on the basis of personnel accessibility because of thermal and radiation conditions (DOE, 2009bb). According to the applicant (DOE, 2009bb), the accessible openings will consist of the North Portal, North Ramp, access mains, entrance to the turnouts, intake shafts, and the performance confirmation observation drift. The accessible openings will be occupied frequently enough such that approaches used in underground mines and in the tunneling industry are applicable and will be used by the applicant to assure adequate functionality of the openings. The applicant stated in SAR Section 1.3.3.2 that the horizontal openings will be excavated using tunnel-boring machines and vertical openings with raise-boring machines. The applicant also stated that it will monitor the performance of the accessible openings through regular visual inspection by qualified personnel and will implement a geotechnical instrumentation program to measure drift convergence, ground support loads, and potential overstressed zones. The monitoring and maintenance program will be performed using methods similar to those used in underground openings in civil and mining industries (DOE, 2009bb).

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.7.3.3(II) to review the applicant's description and design information for underground openings in the accessible areas of the subsurface facility. The NRC staff reviewed the methods the applicant proposed to excavate the underground openings in the accessible areas and selection of the ground support system. The NRC staff also reviewed the construction materials the applicant proposed to use for steel ground support, grout for fully grouted rock bolts, and shotcrete. The NRC staff finds that the applicant's description and design information for the underground openings in the accessible areas are adequate because (i) the excavation methods the applicant selected will minimize construction damage to the surrounding rock and thereby enhance stability of the openings; (ii) the applicant will use well-established empirical methods to select the ground support system (SAR Section 1.3.3.3); and (iii) the applicant will select materials for steel ground support, grout for fully grouted rock bolts, and shotcrete, in conformance with established industry standards (SAR Section 1.3.3.3.3).

The NRC staff finds that the applicant's descriptions of the design, monitoring, and maintenance plans are adequate because the applicant (i) described that the accessible openings of the subsurface facility (North Portal, North Ramp, access mains, entrance to the turnouts, intake shafts, and the performance confirmation observation drift) will be designed consistent with the applicant's assumptions in the PCSA regarding the geometry and serviceability of the openings during the preclosure period; (ii) will use design approaches that are used in underground mines and in the tunneling industry; (iii) described how the excavations would be performed (i.e., horizontal openings will be excavated using tunnel-boring machines and vertical openings with raise-boring machines); (iv) explained that it will monitor the performance of the accessible openings through regular visual inspection by qualified personnel and will implement a geotechnical instrumentation program to measure drift convergence, ground support loads, and potential overstressed zones; and (v) stated the monitoring and maintenance program will be performed using methods similar to those used in underground openings in civil and mining industries.

**NRC Staff's Conclusion**

On the basis of the evaluation in SER Section 2.1.1.2.3.7.2, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for the design of underground openings in accessible areas of the subsurface facility meet the requirements of 10 CFR 63.111(d), 10 CFR 63.111(e), and 10 CFR 63.112(a) because the applicant provided an adequate description and design information for underground openings in accessible areas of the subsurface facility sufficient for the NRC staff to evaluate the PCSA and design.

### 2.1.1.2.3.7.3 Underground Openings in Inaccessible Areas

The applicant provided the design of underground openings in inaccessible areas of the subsurface facility in SAR Sections 1.3.3 and 1.3.4. According to the applicant (DOE, 2009bb), the inaccessible openings will consist of emplacement drifts, turnouts, exhaust mains, exhaust shafts, and shaft access drifts. The applicant expects high radiation levels in the emplacement drifts and turnouts (SAR Figure 1.3.3-13) and thermal and radiological conditions in the openings on the exhaust-air side of the emplacement drifts (exhaust mains, exhaust shafts, and shaft access drifts) that will be high enough to make these openings inaccessible to personnel.

SAR Sections 1.3.3.3 and 1.3.4.4 described the applicant's approach to the subsurface facility opening design. The applicant selected the ground-support system using empirical methods, as described in BSC (2007an, Section 6.3; 2007ao, Section 6.4), and site-specific rock mechanical properties. The applicant then assessed the stability of the resulting design using numerical modeling, as outlined in BSC (2007an, Section 6.7; 2007ao, Section 6.5). In the numerical model analyses, the applicant considered the effects of *in-situ* stress, thermal loads, and seismic ground motions and performed analyses to examine the stability of the openings with and without ground support. The applicant concluded, on the basis of the analyses, that the openings will be stable without ground support but the surrounding rock will sustain stress-induced damage within a zone approximately 0.3–1.0 m [1–3.28 ft] from the circumference, around the entire opening in the lower quality rock categories but only in the roof areas for higher quality rock categories, as outlined in BSC (2007an, Section 6.4.3; 2007ao, Section 7.2). The applicant also concluded that the repository thermal loading and potential seismic ground motion will not have a significant effect on the damaged zone. According to the applicant in BSC (2007an, Section 7), repository thermal loading will not have significant effect on emplacement drift stability, because subsurface ventilation will be used to ensure the drift wall temperature will not increase more than approximately 50 °C [122 °F] during the preclosure period, as described in BSC (2007an, Section 7). In SER Section 2.1.1.2.3.3.3, the NRC staff evaluated the applicant's description of the design of the ground support system for underground inaccessible drifts. The applicant also assessed the effect of ground support on stability of the openings. The applicant's analysis indicated that rock bolts in the exhaust mains and intersections between exhaust mains and emplacement drifts may experience a load of up to approximately 75 percent of the bolt-yield strength, whereas rock bolts in the emplacement drifts will have a factor of safety of approximately 2.9 (i.e., loading of up to 35 percent of the bolt capacity). In response to an NRC staff's RAI, the applicant stated that the repository emplacement drift environment will not be conducive to corrosion of stainless steel rock bolts, because of ventilation and temperature in the drift (DOE, 2009gu). Relative humidity in an emplacement drift will be very low and will prevent formation of an aqueous environment conducive to stress corrosion cracking (SCC) of rock bolts. However, the relative humidity inside the rock bolt boreholes is expected to be higher, especially near the end of the borehole. Void zones in the lithophysal rock, in the rock bolt boreholes, will create local zones of stress concentration in a rock bolt that may result in localized SCC of the rock bolt. The applicant stated that the localized SCC would have a limited impact on the effectiveness of the ground support system.

The applicant stated in SAR Section 1.3.2.4.4.3 that the inaccessible underground openings are designed to function without planned maintenance during the preclosure period, but the applicant proposed a plan for monitoring and maintenance, which is a common practice in underground openings and tunnels. In addition, in its response to an NRC staff RAI (DOE, 2009bb,ed), the applicant stated that it will monitor the inaccessible openings remotely to detect any progressive deterioration, assess maintenance needs, and promptly implement appropriate maintenance to prevent structural failures that could initiate event sequences or interfere with the applicant's plan to keep waste packages accessible and ventilated through the preclosure period (BSC, 2007ao, Section 7.3; 2008ca). According to the applicant, concrete liners in the exhaust shafts will be inspected for cracks and voids and any evidence of spalling; the exhaust mains and shaft access drifts will be inspected for indications of ground support damage or roof sagging; and shotcrete will be inspected for cracks, delamination, void development, spalling, or chemical alteration (DOE, 2009bb). For the emplacement drifts, the applicant will also monitor convergence of the drift circumference (DOE, 2009bb,ef) to ensure the integrity of equipment operating envelopes specified in the subsurface facility design (e.g., SAR Figure 1.3.4-18). The applicant will monitor the entire length of the openings annually for the first few years and progressively less frequently if the applicant determines that the monitoring frequency could be reduced (DOE, 2009bb). The applicant stated that maintenance of the openings will be performed only as a contingency measure in cases of significant failure or deterioration (DOE, 2009bb). The applicant may monitor areas of failed ground support more frequently to determine when to initiate repair or maintenance. The applicant also stated that maintenance (i) will be scheduled to preclude impacts to repository nuclear safety functions, (ii) may be performed using remotely operated equipment, and (iii) will be preceded by planning and design of remediation activities and controls to assure personnel safety when personnel access to the openings is necessary (DOE, 2009bb).

In summary, the applicant concluded that the inaccessible openings will be stable, and ground support is provided for personnel protection and to provide additional assurance in maintaining repository openings as functional for their intended operations. Monitoring and maintenance plans, as described previously, are intended as a contingency measure to detect any significant deterioration or failure of the ground support system and to facilitate the remediation of any significant deterioration or failure to keep the drift openings functional through the preclosure period.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.7.3.3(II) to review the applicant's design of underground openings in the inaccessible areas of the subsurface facility. The NRC staff's review focused on determining whether the design of the inaccessible openings will satisfy functional requirements that the applicant established. In addition, the NRC staff reviewed the applicant's proposed monitoring and maintenance program to determine whether this program will support performance of the functions of the openings through the preclosure period.

The NRC staff finds that the applicant's design of underground openings in the inaccessible areas of the subsurface facility is acceptable because it is consistent with well-established design procedures as outlined in FHWA, 2009aa and NCHRP, 2011aa. The NRC staff also finds that the applicant's analysis of the design is acceptable because it is based on well-established numerical analysis computer codes such as the FLAC and UDEC computer codes that are commonly used by the geotechnical engineering profession. In response to an NRC staff RAI, the applicant also addressed the potential for some rock to spall at the drift circumference and the need for use of a stainless steel liner, supported by rock bolts anchored to drift walls, to mitigate the spalling effects (DOE, 2009ed). The NRC staff determines that the expectation that the rock will spall or ravel at the drift circumference is consistent with the applicant's description that the lithophysal rock mass is densely fractured with fracture spacing on the order of centimeters [inches], as described in SAR Section 1.1.5.3.1.1 and BSC (2004al, Section 7.3.2). The NRC staff finds that the spalling or raveling can be mitigated using the types of surface protection (perforated stainless steel liner, wire mesh, or shotcrete) included in the applicant's ground-support design because the surface protection will cover the rock surface and will be anchored in areas of the rock that will not be affected by spalling or raveling. Based on its review of the applicant's ground support design, the NRC staff finds that the spacing and penetration length of rock bolts included in the applicant's ground support design will adequately anchor the stainless steel liner, shotcrete, or wire mesh in undisturbed rock.

The effectiveness of rock bolts to anchor surface-protection ground support elements could be undermined if the rock bolts corrode during the preclosure period. The applicant expects stainless steel rock bolts to perform better than carbon steel rock bolts because the stainless steel material will be less susceptible to general corrosion than carbon steel. In response to an NRC staff RAI, the applicant also stated that the environment in the drift will not be conducive to stress corrosion cracking (SCC) of stainless steel rock bolts and that confirmatory studies and tests are planned to verify that SCC will not occur (DOE 2009gu). The NRC staff notes that for SCC to occur, the relative humidity needs to be high enough to have sufficient aqueous environment but also be low enough to have sufficient chloride concentration and no drying out of salts. Given the thermal load and ventilation planned for the drift through the preclosure period, the NRC staff finds that the applicant's statement that the environment in the drift will not be conducive to SCC is acceptable. The relatively higher humidity inside the rock bolt boreholes and the localized stress concentrations in the void zone in the lithophysal rock unit may result in localized SCC of the rock bolt. However, the NRC staff finds that this would have a limited impact, as the openings are expected to be stable without ground support.

The NRC staff evaluated the applicant's emplacement drift stability in connection with the excluded features, events, and processes (FEP). The applicant excluded the thermally induced drift collapse FEP in the postclosure performance assessment (SER Volume 3, Section 2.2.1.2.1, FEP 2.1.07.0A). The DOE drift-stability evaluation, considering combined effects of mechanical, thermal, and time-dependent weakening of rock, resulted in minor rock spalling in lithophysal rock around the periphery of the drift. The NRC staff found, through confirmatory calculations (see SER Volume 3, Section 2.2.1.2.1, FEP 2.1.07.0A) that the drift opening would reach a relatively stable profile after adjusting for stress relief caused by the dislodging of small rock fragments in lithophysal rock. The NRC staff's confirmatory calculations verified stability would be maintained even in an unsupported condition (i.e., no ground support). Considering the potential for minor rock spalling around the periphery of the drift, the NRC staff finds that the applicant's proposed ground support system will provide personnel protection and additional assurance of maintaining repository openings for operations.

To maintain the functional requirements of emplacement drifts, including emplacing drip shields at the end of the preclosure period, the applicant proposed a monitoring and maintenance plan (DOE 2009ea,ef,gk). The applicant's descriptions of monitoring and maintenance plans are acceptable because DOE described how it will (i) monitor rock wall convergence at preselected locations along the openings using convergence pins attached to the rock or fixed laser targets attached to the head of rock bolts, (ii) monitor the deformation of the stainless steel liner using laser scanning at additional selected locations, and (iii) use the convergence data and other available information to determine the need for maintenance to preserve the equipment operating envelopes and meet operational needs. The ability to perform monitoring in these inaccessible areas will be contingent on availability of power and communications provisions enabling remote inspection and observation. Power, communications, and vehicle SSCs required for remote monitoring and maintenance in inaccessible areas are evaluated in SER Sections 2.1.1.2.3.2.3 and 2.1.1.2.3.6.2.

On the basis of the above evaluation, the NRC staff finds that the applicant's information regarding design of underground openings in inaccessible areas of the subsurface facility is adequate.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed above, the NRC staff finds, with reasonable assurance, that the applicant's description and design information for design of the underground openings in inaccessible areas of the subsurface facility meets the requirements of 10 CFR 63.111(d), 63.111(e), 63.112(a) and 63.112(d), because the applicant provided an adequate description and design information for the underground openings in inaccessible areas of the subsurface facility sufficient for the NRC staff to evaluate the PCSA and design.

2.1.1.2.3.7.4 Invert Structure and Rails

The applicant described the invert structure in SAR Section 1.3.4.5. The steel invert structure will provide a platform that supports the emplacement pallets, waste packages, and drip shields (SAR Section 1.3.4.5.1). The invert structure also will provide a platform that will support the crane rail system for operation of the transport and emplacement vehicle (TEV) for emplacement, recovery, and potential retrieval of waste packages, and for operation of the drip shield emplacement gantry (DSEG) and the remotely operated inspection vehicles (ROV). According to the applicant (SAR Section 1.3.4.5.3), the invert structure will be a non-ITS system because it will not be relied on to prevent or mitigate a Category 1 or Category 2 event sequence (SAR Table 1.9-1), and the invert structure was classified as non-ITWI because no credit will be taken for the diffusivity of the invert ballast (SAR Table 1.9-8).

The applicant will use conventional structural methods to design the invert structure (SAR Section 1.3.4.5.6) and indicated that the design will minimize the need for maintenance during the preclosure period (SAR Section 1.3.4.5.2). The invert structure will withstand gravitational, thermal, and seismic loading (SAR Section 1.3.4.5.5) and its performance would not be affected by corrosion during the preclosure period (SAR Section 1.3.4.5.1). The steel invert structure will include transverse beams bolted to four longitudinal beams (SAR Figure 1.3.4-8). The two outermost longitudinal beams at either end of the invert structure section will be attached to and rest on stub columns that transfer the loads to the substrate rock (SAR Figures 1.3.4-9 and 1.3.4-10). The crane rails will be mounted on the two outer longitudinal beams or rail runway beams. After installation of the invert steel structure, the ballast will be placed in lifts and compacted to specifications. The ballast material will be crushed tuff and fill the voids between the drift rock and the invert structure steel frame. Completion of the invert structure assembly will be followed by installation and alignment of the crane rails.

The applicant stated that subsurface facility structures and systems in inaccessible areas (e.g., turnouts and emplacement drifts) will be monitored so that the onset of a condition that may lead to a structural failure will be detected in a timely manner and repaired as needed (DOE, 2009ed).

***Design Criteria and Design Bases***

As part of the design criteria (SAR Section 1.3.4.5.5), the applicant indicated that the invert structure is designed for the appropriate worst-case combinations of construction loads, waste package and pallet loads, drip shield loads, thermal loads, and seismic loads. The applicant also stated that the invert structure is designed with materials that will undergo minimal corrosion during the preclosure period because of the use of a high strength, corrosion-resistant structural steel.

The invert steel structure is designed to accommodate the relatively small structural displacement expected to occur in the emplacement drifts (SAR Section 1.3.4.5.1). Slotted holes will be provided at bolt connections, as well as 1.3-cm [0.5-in] expansion joints between the rail runway beams and 0.64-cm [0.25-in] expansion joints between the longitudinal beams (SAR Figure 1.3.4-10). According to the applicant (DOE, 2009ed), these design features would mitigate potential effects of thermal expansion of the invert steel and rail, preventing buckling of the steel or distortion of the rail.

For the invert ballast, the applicable design criterion will provide a nominally leveled surface that supports the drip shield, waste package, and waste package emplacement pallet for static loads, and that limits degradation of these EBS components associated with ground motion (but excluding faulting displacements) after repository closure, as shown in BSC Table 1 (2008aw).

**NRC Staff's Evaluation**

The NRC staff reviewed the design criteria and design bases the applicant proposed for the design of the non-ITS invert structure and rails, using the guidance in YMRP Section 2.1.1.7.3.1. The NRC staff reviewed the design criteria for the invert structure and rails for normal operating conditions. The NRC staff also reviewed the design information for the invert structure and rails including design features mitigating thermal expansion effects and preventing steel buckling. The NRC staff finds that the design criteria the applicant will use for the design of the invert structure and rails are adequately defined because the applicant: (i) explained that the invert structure is designed for the appropriate worst-case combinations of construction loads, waste package and pallet loads, drip shield loads, thermal loads, and seismic loads; (ii) explained that the invert steel structure is designed to accommodate the relatively small structural displacement expected to occur in the emplacement drifts; (iii) described design features that would mitigate potential effects of thermal expansion of the invert steel and rail, preventing buckling of the steel or distortion of the rail; (iv) explained the design criterion for the invert ballast will provide a nominally leveled surface that supports the drip shield, waste package, and waste package emplacement pallet for static loads, and that limits degradation of these EBS components associated with ground motion; and (v) explained that the design criteria and design bases are consistent with the standard engineering practice for structures of similar functions.

***Design Codes and Standards***

The applicant specified codes and standards it will use in the design of the invert structure in SAR Section 1.3.4.5.8. For instance, the structural steel shapes and plates will conform to ASTM A 588/A 588M–05 (ASTM International, 2005aa), the crane rail will be in accordance with ASTM A 759–00 (ASTM International, 2001aa), the structural steel bolts will conform to ASTM A 325–06 (ASTM International, 2006ae), and welding will be in accordance with AWS D1.1/D1.1M (American Welding Society, 2006aa).

**NRC Staff's Evaluation**

The NRC staff reviewed the design codes and standards the applicant proposed for the design of the invert structure and rails, using the guidance in YMRP Section 2.1.1.7.3.3(II). The NRC staff finds the codes and standards are acceptable because they are in conformance with the standard engineering practices for structures of similar functions.

***Design Loads and Load Combinations***

For the design of the invert structure, the load combinations will include the following loads (SAR Section 1.3.4.5.9.1):

**Gravitational Loads**

Dead loads will include the weight of framing and permanent equipment, and attachments. Live loads will include construction loads, the weight of the heaviest waste package, the pallet's weight, drip shield load, and crane loads and corresponding impact allowances (American Institute of Steel Construction, 1997aa; American Society of Mechanical Engineers, 2005aa).

**Seismic Loads**

Longitudinal beams and transverse support beams of the steel invert structure will be designed to withstand DBGM–2 seismic events {associated to a mean annual probability of exceedance (MAPE) of $5 \times 10^{-4}$}. The TEV rail and rail runway beams will be designed with DBGM–1 seismic loads (MAPE of $1 \times 10^{-3}$), as described in BSC Section 3.2.4 (2007cj). The applicant indicated (SAR Section 1.3.4.5.6) that site-specific acceleration response spectra were developed at the repository horizon in three orthogonal directions. The seismic loads for the invert structure will be computed on the basis of the equivalent static load method in accordance with NRC NUREG–0800, as outlined in NRC Section 3.7.2 (1989ac).

**Temperature Loads**

Transient peak drift wall temperature during off-normal events in the emplacement drifts is not expected to exceed 200 °C [392 °F] (SAR Table 1.3.1-2). Expansion joints will be designed in the longitudinal members of the steel invert structure and the rails in emplacement drifts (BSC, 2007cj) for temperatures up to 200 °C [392 °F].

**NRC Staff's Evaluation**

The NRC staff reviewed the loads and load combinations the applicant proposed for the design of the non-ITS invert structure and rails, using the guidance in YMRP Section 2.1.1.7.3.2. The NRC staff reviewed the design loads and load combinations for the non-ITS invert structure and rails for both normal and off-normal conditions. The NRC staff finds that the loads and load combinations used for the design of the invert structure and rails are acceptable because the applicant: (i) provided dead loads corresponding to the weight of framing and permanent equipment, and attachments; (ii) provided live loads corresponding to construction loads, the weight of the heaviest waste package, the pallet's weight, drip shield load, and crane loads and corresponding impact allowances; (iii) the design for dead and live loads is consistent with the standard engineering practice for the design of similar structures (iv) provided the design basis ground motion for invert structure and rails; (v) explained the seismic loads for the invert structure will be computed on the basis of the equivalent static load method in accordance with NRC NUREG–0800 (Section 3.7.2, NRC 2013ac); and (vi) described expansion joints will be designed in the longitudinal members of the steel invert structure and the rails in emplacement drifts for temperatures up to 200 °C [392 °F] and stated transient peak drift wall temperature during off-normal events in the emplacement drifts is not expected to exceed this temperature.

The NRC staff notes that seismic faults may lead to displacements of several centimeters [inches] for fault events with an MAPE (mean annual probability of exceedance) of $1 \times 10^{-6}$ (SAR Table 2.3.4-55), which is the frequency threshold for subsurface facilities evaluation. Furthermore, the applicant indicated that seismic faulting could occur not only coincident with the location of the known faults, but also elsewhere in the repository (SAR Section 2.3.4.5.5.2.3.2). The applicant stated it will rely on a monitoring and maintenance plan to develop corrective actions for the non-ITS invert structure and rail should seismic fault displacement occur in the invert structure and rail. The NRC staff finds that the applicant's description of its approach to repairing potential damage caused by seismic fault displacement for this non-ITS equipment is acceptable because the applicant will not rely on the continuing function of the invert structure and rail to prevent any event sequence and mitigate any consequences.

***Design of the Invert Structure and Rails***

The applicant stated that the design of the invert structure is in accordance with the design criteria and design bases, codes and standards, and design loads evaluated above in this SER Section 2.1.1.2.3.7.4. In addition, the applicant stated (SAR Section 1.3.4.5.4) that the steel invert structure and rails will not be expected to be subjected to any administrative procedure or procedural safety control (PSC) to prevent event sequences. The applicant intends to monitor the invert structure and rails during the preclosure period but relies on their conservative design (SAR Section 1.3.4.5) to exclude event sequences that involve potential structural failures. The applicant indicated, in its response to an NRC staff RAI on potential event sequences resulting from failure of the invert structure and rails (DOE, 2009ed), that given the conservative design, subsurface facility structures will be monitored remotely so that the onset of a condition that may potentially lead to a structural failure will be detected in a timely manner and repaired, as needed. In response to an NRC staff request that the applicant clarify its plan to ensure adequate functionality of the invert structure and rails through the preclosure period, the applicant stated that its plans will include (i) measurements of the rail alignment and grade using methods such as photogrammetry, laser scanning, or laser targeting of fixed targets; (ii) evaluation of the damage to the invert structure or rails detected through the inspection to determine potential impact to repository operations and need for maintenance, and (iii) development and implementation of a remediation method for each case (DOE, 2009gl).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information regarding the design of the invert structure and rails using the guidance in YMRP Section 2.1.1.7.3.3(II). The NRC staff finds that the design of the invert structure, in accordance with the design criteria and design bases, codes and standards, and design loads previously evaluated, is acceptable because DOE followed acceptable methodology and common industry practice for structures with similar functions. The NRC staff finds that the applicant's description of the monitoring and maintenance plan is acceptable because it discussed (i) the scope of the monitoring activities, (ii) several methods it may use to conduct the monitoring, and (iii) development of remediation methods based on the monitoring findings.

**NRC Staff's Conclusion**

On the basis of evaluations discussed in SER Section 2.1.1.2.3.7.4, the NRC staff finds, with reasonable assurance, that the applicant's description, discussion, and design information for the invert structure and rail meet the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), 10 CFR 63.111(e), 10 CFR 63.112(a), and 10 CFR 63.112(f) because the applicant provided an adequate description, discussion, and design information for the invert structure and rails sufficient for the NRC staff to evaluate the applicant's PCSA and design. In particular, as discussed in the above NRC staff evaluations, the applicant (i) provided the NRC staff with an understanding of the structural capabilities of the invert structure and rails to withstand the effects of operational activities and natural phenomena, (ii) provided the design criteria consistent with applicable codes and standards, and (iii) is in conformance with the standard engineering practices for structures of similar functions.

## 2.1.1.2.4 Evaluation Findings

The NRC staff has reviewed the Safety Analysis Report and other information submitted in support of the license application and with the proposed condition of the construction authorization has found, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3)(i), 10 CFR 63.21(c)(4), and 10 CFR 63.112(a) are satisfied in that DOE has provided an adequate description and design information for the structures, systems, components, equipment, and process activities of the geologic repository operations area. The NRC staff also has found, with reasonable assurance, that the requirements of 10 CFR 63.111(d), 10 CFR 63.111(e), 10 CFR 63.112(a), 10 CFR 63.112(d), and 10 CFR 63.112(f) are satisfied, in that an adequate description, discussion, and design information for the underground opening design, which satisfactorily defines the relationship between design criteria and the performance objectives and which identifies the relationship between the design bases and the design criteria, has been provided.

**Proposed Condition of Construction Authorization**:

DOE shall not, without prior NRC review and approval, accept DOE spent nuclear fuel (SNF) in multicanister overpacks (MCOs) or commercial mixed oxide (MOX) fuel.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these MCOs and MOX fuel at the GROA or (ii) demonstrates, through the PCSA, that the MCOs and MOX fuel can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

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DOE marked the following SAR figures as non-public [Official Use Only]: Figure 1.2.3-2, Figure 1.2.3-3, Figure 1.2.3-4, Figure 1.2.3-5, Figure 1.2.3-6, Figure 1.2.3-7, Figure 1.2.3-8, Figure 1.2.3-9, Figure 1.2.3-10, Figure 1.2.3-11, Figure 1.2.3-12, Figure 1.2.3-13, Figure 1.2.3-14, Figure 1.2.3-15, Figure 1.2.3-16, Figure 1.2.3-18, Figure 1.2.3-26, Figure 1.2.4-2, Figure 1.2.4-3, Figure 1.2.4-4, Figure 1.2.4-5, Figure 1.2.4-6, Figure 1.2.4-7, Figure 1.2.4-8, Figure 1.2.4-9, Figure 1.2.4-10, Figure 1.2.4-11, Figure 1.2.4-13, Figure 1.2.4-14, Figure 1.2.4-41, Figure 1.2.5-2, Figure 1.2.5-3, Figure 1.2.5-4, Figure 1.2.5-5, Figure 1.2.5-6, Figure 1.2.5-7, Figure 1.2.5-8, Figure 1.2.5-9, Figure 1.2.5-10, Figure 1.2.5-11, Figure 1.2.5-12, Figure 1.2.5-13, Figure 1.2.5-14, Figure 1.2.5-15, Figure 1.2.5-16, Figure 1.2.5-17, Figure 1.2.5-18, Figure 1.2.5-33, Figure 1.2.5-66, Figure 1.2.5-67, Figure 1.2.6-2, Figure 1.2.6-3, Figure 1.2.6-4, Figure 1.2.6-5, Figure 1.2.6-6, Figure 1.2.6-7, Figure 1.2.6-8, Figure 1.2.6-9, Figure 1.2.6-10, Figure 1.2.6-11, Figure 1.2.6-12, Figure 1.2.6-13, Figure 1.2.6-17, Figure 1.2.8-2, Figure 1.2.8-3, Figure 1.2.8-4, Figure 1.2.8-5, Figure 1.2.8-6, Figure 1.2.8-7, Figure 1.2.8-10, Figure 1.2.8-11, Figure 1.2.8-12, Figure 1.2.8-13, and Figure 1.2.8-14.

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# CHAPTER 3

## 2.1.1.3 Identification of Hazards and Initiating Events

### 2.1.1.3.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.3 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's ("DOE" or the "applicant") identification of hazards and initiating events in both surface and subsurface facilities of the geologic repository operations area (GROA) at Yucca Mountain during the preclosure period. The objective of the review is to evaluate the applicant's information identifying hazards and initiating events pertaining to the preclosure safety analysis (PCSA) and the GROA design. PCSA is defined in 10 CFR 63.2 as a systematic examination of the site, the design, and the potential hazards, initiating events, and event sequences and their consequences (e.g., radiological exposures to workers and the public).

In this SER section, natural, human-induced, and operational hazards are evaluated, including the technical basis for either inclusion or exclusion of specific hazards and initiating events in the PCSA. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.6 (DOE, 2008ab); supplemental documents referenced in the SAR; and the applicant's responses to the NRC staff's requests for additional information (RAIs). This information addresses how the applicant identified potential hazards and initiating events and screened each hazard and initiating event to assess its potential to initiate an event sequence. The applicant's information included specific hazard identification methodology for each type of hazard, screening criteria, data used, and specific analyses conducted.

### 2.1.1.3.2 Regulatory Requirements

As required by 10 CFR 63.112(b), the PCSA of the GROA must include an identification and systematic analysis of naturally occurring and human-induced hazards at the GROA, including a comprehensive identification of potential event sequences.

As required by 10 CFR 63.112(d), the PCSA of the GROA must also include the technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards in the safety analysis.

The NRC staff reviewed the applicant's identification of hazards and initiating events at the GROA for the PCSA using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa; Section 2.1.1.3). The acceptance criteria are as follows:

- Technical bases and assumptions for methods used for identification of hazards and initiating events are adequate.

- Site data and system information are appropriately used in identification of hazards and initiating events.

- Determination of frequency or probability of occurrence of hazards and initiating events is acceptable.

- Adequate technical bases for the inclusion and exclusion of hazards and initiating events are provided.
- The list of hazards and initiating events that may result in radiological releases is acceptable.

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA (e.g., handling of spent nuclear fuel, criticality controls during storage of spent nuclear fuel, shield doors and interlocks for worker safety from direct radiation of spent nuclear fuel). The applicability of such NRC guidance is discussed in greater detail in the sections where the guidance was used as part of the application or the NRC staff's review.

### 2.1.1.3.3 Technical Review

The NRC staff's review of the applicant's identification of hazards and initiating events in this SER section is integrated with the review of the applicant's site description (SAR Section 1.1) in SER Section 2.1.1.1 and review of the surface and subsurface structures, systems, and components (SSCs) and operational process activities description (SAR Sections 1.2 through 1.5) in SER Section 2.1.1.2. The NRC staff used the information from these sections to evaluate whether the methods used by the applicant for hazard and initiating event identification are consistent with standard industry practices and/or NRC guidance or that nonstandard practices were adequately justified. The NRC staff also evaluated whether the methods selected for hazard and initiating event identification were appropriate for available data and proposed operations, as well as whether assumptions used were well-defined and have adequate technical bases. The NRC staff also confirmed that methods the applicant selected to quantify initiating event frequencies, including uncertainties, were appropriate. The NRC staff further evaluated whether human errors that may lead to radiological doses were adequately identified, including whether adequate human reliability analyses were performed. Finally, the NRC staff confirmed that the technical basis for inclusion or exclusion of specific hazards and initiating events in the PCSA was adequate, which included evaluating the applicant's list of hazards and initiating events from all credible naturally occurring and human-induced events that may result in radiological releases.

The NRC staff used a risk-informed approach to review naturally occurring and human-induced hazards as initiating events of event sequences. The risk-informed evaluation focuses on those initiating events that could potentially lead to radiological doses to workers or the public. Dose performance objectives for workers and the public are specified in 10 CFR 63.111. In 10 CFR 63.2, two event sequence categories and their respective performance objectives are defined. Category 1 event sequences are those that are expected to occur one or more times before permanent closure of the GROA. Category 2 event sequences are those that have at least 1 chance in 10,000 of occurring before permanent closure of the GROA. Event sequences with the probability of occurrence of less than 1 in 10,000 before permanent closure may be screened out from further consideration in the PCSA.

DOE refers to the period before permanent closure as the preclosure period. In SAR Section 1.3.1, DOE defines the preclosure period (also referred to as the period of operations) as the 100-year period of surface and subsurface operations that would occur before permanent closure of the repository. According to DOE, the preclosure period would consist of an initial

50-year period of waste emplacement (including a 24-year period of concurrent repository development) and a subsequent 50-year period of post-emplacement monitoring. For most event sequences, DOE assessed the probability of hazards as initiating events using the full 100-year preclosure period. In these event sequences, the threshold probability for screening is $1 \times 10^{-6}$/year (1/10,000 × 1/100). DOE used the full 100-year preclosure period for these calculations either because the hazards could result in radiological release at any time over the 100-year period, or because doing so was considered by DOE to be a simplifying assumption. In some cases, however, DOE considered the amount of time within the preclosure period that radiological waste would be susceptible to the hazard (called the exposure time). For example, according to DOE, several hazards, such as aircraft crashes or tornadoes, could potentially impact the safe storage of radiological waste only when the waste is present at the surface area of the GROA. Once the waste is emplaced underground, it would be isolated from these types of hazards. Similarly, hazards that initiate accidents, which could then damage waste containers during transportation on the Transport and Emplacement Vehicles (TEV), are, according to DOE, only possible during the time when the waste will actually be in transit. In these cases, DOE considered exposure time during transportation to calculate the initiating event probability. As discussed in the specific reviews of initiating events in this SER section, the NRC staff reviewed the applicant's use of the exposure time to ensure that the resulting initiating event probabilities are not underestimated.

The NRC staff's review also considered uncertainty in the initiating event probabilities, especially in cases in which the event probability was near the threshold probability between Category 1 and Category 2, or just beyond Category 2. Uncertainties in the event probabilities can arise because estimates of the annual probabilities of initiating events are often determined from statistical analyses of available measurable data. If sufficient data are available, event frequency can be defined by a probability distribution that is represented in the PCSA by a single value, often referred to as the measure of central tendency. In most cases, DOE defined the central tendency value by the mean of the probability distribution. In cases where sufficient information was not available to develop a probability distribution of the event occurrence (e.g., seismic and volcanic hazard assessments), DOE relied on expert elicitation to define these probabilities (as described in SAR Sections 1.5.4.2.4 and 2.2.2.1.1 and reviewed by the NRC staff in SER Section 2.1.1.1.3.5).

The NRC staff evaluated the uncertainties associated with event probabilities in order to assess how the event probabilities were defined and whether the probability of the initiating events crossed the probability threshold between Category 1 and Category 2 event sequences or was just beyond Category 2. In the cases where hazards occur with probabilities close to the threshold between Category 1 and Category 2, the NRC staff considered these hazards with regard to both Category 1 and 2 performance objectives. In cases where the event probability was close to but less than the Category 2 probability threshold (1 in 10,000 during the preclosure period), the NRC staff considered the uncertainties associated with the probability estimates in order to ensure that only highly unlikely event sequences (those with likelihoods of less than 1 in 10,000 during the preclosure period) were excluded from the PCSA.

The NRC staff's review of the naturally occurring and human-induced external hazards and initiating events is described in SER Section 2.1.1.3.3.1. The NRC staff's review of the hazards and initiating events from the operational (internal) activities is described in SER Section 2.1.1.3.3.2. The NRC staff's evaluation of event sequences is described in SER Section 2.1.1.4.3, and its evaluation of those event sequences that may lead to significant radiological doses is described in SER Section 2.1.1.5.3.

#### 2.1.1.3.3.1 External Hazards and Initiating Events

The applicant identified external initiating events in SAR Section 1.6.3.2. This information included a list of potential naturally occurring and human-induced external hazards compiled from various sources, and how the applicant screened these hazards for the potential to initiate event sequences that may lead to radiological releases. The NRC staff reviewed and evaluated the naturally occurring and human-induced external hazards by examining the applicant's (i) identification of hazards, (ii) screening criteria for inclusion or exclusion of hazards in the PCSA, and (iii) implementation of the screening criteria.

##### 2.1.1.3.3.1.1 Identification of Hazards

In SAR Section 1.6.3.2, the applicant described a three-step method for identifying external naturally occurring and human-induced hazards for use in the PCSA. In Step 1, the applicant reviewed applicable documents from both nuclear and nonnuclear industries to compile a list of potential external hazards. In Step 2, the applicant narrowed the list of external events identified in Step 1 down to 89 events that DOE considered applicable to the Yucca Mountain repository and that may lead to radiological releases. In Step 3, the 89 external events were then grouped into 13 distinct categories of naturally occurring and human-induced external events based on similarity, as listed in SAR Section 1.6.3.2 and SAR Table 1.6-2. These categories are seismic activity, volcanic activity, nonseismic geological activity, high winds/tornadoes, external floods, lightning, loss of cooling capability event (nonpower cause), external fires, loss of power event, extraterrestrial activity, aircraft crash, nearby industrial/military facility accidents, and onsite hazardous materials release.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.6.3.2 and SAR Table 1.6-2. The NRC staff evaluated whether the applicant provided a comprehensive list of potential naturally occurring and human-induced external hazards and initiating events for screening. The NRC staff concludes that the information in SAR Section 1.6.3.2 and SAR Table 1.6-2 is adequate because the applicant followed guidance for identifying hazards developed for nuclear and chemical facilities. These included (i) NUREG/CR–2300 (ANS/IEEE, 1983aa), which was developed for the identification of external hazards at nuclear power plants; (ii) *Guidelines for Chemical Process Quantitative Risk Analysis* (AIChE,1989aa), which is a chemical industry guidance to identify external hazards; and (iii) *Preclosure Radiological Safety Analysis for Accident Conditions of the Potential Yucca Mountain Repository: Underground Facilities* (Ma, et al., 1992aa), which was developed early in the Yucca Mountain program. The use of these guidance documents is reasonable because the types of external hazards identified for domestic nuclear power plants and industrial chemical facilities are similar to those that might occur at the repository site (e.g., tornadoes as a natural phenomenon and overpressure induced by an explosion of a nearby facility containing explosives as a human-induced hazard). Further, the NRC staff finds that the applicant's list of hazards is also consistent with the list of external hazards considered at nuclear power plants in other countries, as described in International Atomic Energy Agency Standard NS–G–1.5 (IAEA, 2003ab). The NRC staff also finds that the methods used by the applicant to identify hazards and initiating events, including the use of the guidance documents (ANS/IEEE, 1983aa; AIChE, 1989aa), are acceptable given the Yucca Mountain site characteristics identified in Ma, et al. (1992aa). The range of site-specific hazards and initiating events identified in Ma, et al. (1992aa), such as earthquakes, volcanoes, or drift collapse, are included in the environmental conditions addressed in the guidance documents used by DOE.

Based on this review, the NRC staff finds that the initial list for screening of naturally occurring and human-induced potential hazards provided in SAR Section 1.6.3.2 is acceptable for use in the PCSA because (i) the list contained the credible naturally occurring and human-induced events, consistent with available site characterization information; (ii) the applicant followed standard industry practices and NRC guidance to identify hazards; and (iii) the hazards identified are consistent with those identified at other nuclear and non-nuclear facilities.

2.1.1.3.3.1.2 Screening Criteria

The applicant used the criteria in SAR Table 1.6-1 to screen each of the 13 categories of external events to determine whether the event needs to be analyzed further in the PCSA. The applicant's technical bases for the screening process were based on the probabilistic risk assessment processes described in NUREG/CR–5042 (Kimura and Budnitz, 1987aa) and NUREG–1407 (NRC, 1991aa). The screening criteria DOE used were

1. Can the external event occur at the repository? In other words, is it physically realizable?

2. Can the external event occur at the repository with a frequency greater than $10^{-6}$ per year; that is, have a 1 in 10,000 chance of occurring in the 100-year preclosure period?

3. Can the external event (severe enough to affect the repository and its operation) occur at the repository with a frequency greater than $10^{-6}$ per year; that is, have a 1 in 10,000 chance of occurring in the 100-year preclosure period?

4. Can a release that results from the external event severe enough to affect the repository and its operations occur with a frequency greater than $10^{-6}$ per year; that is, have a 1 in 10,000 chance of occurring in the 100-year preclosure period?

The applicant also excluded those potential hazards that were considered to develop slowly during the preclosure period, using Requirement EXT–B1 (Screening Criterion iii) of ANSI/ANS–58.21–2007 (ANS, 2007ab). This requirement states that external events which develop slowly can be excluded because sufficient time would be available to respond or mitigate any adverse consequences. These include such phenomena as soil weathering effects (e.g., denudation, dissolution, erosion, settlement), tectonic activity, disruption of water supplies from underground wells, and loss of the Wet Handling Facility pool water (BSC, 2008ai).

Results from the screening were presented in SAR Table 1.6-2. External event categories that could not be excluded were evaluated further, as initiating events, in the PCSA in SAR Section 1.7.

**NRC Staff's Evaluation**

The NRC staff reviewed the screening criteria of external events provided in SAR Section 1.6.3.4. The NRC staff verified and confirmed that the screening criteria for inclusion and exclusion of events are acceptable because the technical bases for the screening process are derived from NUREG/CR–5042 and NUREG–1407. These two documents provide guidance for developing a risk-informed perspective for identifying and screening external initiating events that could result in core damage at nuclear power plants. They are applicable to the facilities at the GROA because the focus is on a risk-informed process to identify and screen external events. This process is independent of facility type and is consistent with the NRC staff's risk-informed and performance-based review described in the Yucca Mountain Review Plan, NUREG–1804.

The NRC staff also verified and confirmed that the DOE screening criteria to exclude hazards that were considered to develop slowly during the preclosure period are acceptable. The NRC staff concludes that these criteria are acceptable because they are consistent with applicable NRC guidance and standard industry practices [e.g., Kimura and Budnitz, 1987aa; NRC, 1991aa; ANS/IEEE, 1983aa; ANS, 2007ab; American Society of Mechanical Engineers ASME/ANS RA–S–2008 (ASME, 2008aa)]. ASME/ANS RA–S–2008 (ASME, 2008aa) supersedes ANSI/ANS–58.21–2007 (ANS, 2007ab) and has the same requirement in Criterion 5,Table 4-1.8.1.3-2(b) as in EXT–B1 (Screening Criterion iii) of ANSI/ANS–58.21–2007 (ANS, 2007ab).

Finally, based on its review, the NRC staff finds that the screening criteria the applicant developed for inclusion and exclusion of external events are acceptable because they are consistent with the 10 CFR 63.2 definition of event sequences. The specifics of the 13 categories of hazards DOE identified in its screening process are reviewed in the next five SER subsections.

2.1.1.3.3.1.3 Screening Criteria Implementation

Implementation of the screening criteria is described in SAR Section 1.6, specifically in SAR Table 1.6-8. In this SER section, the NRC staff's review of the screening criteria implementation is organized into five hazard types: (i) Geologic Hazards (SER Section 2.1.1.3.3.1.3.1); (ii) Weather-Related Hazards (SER Section 2.1.1.3.3.1.3.2); (iii) Aircraft Crash Hazards (SER Section 2.1.1.3.3.1.3.3); (iv) Nearby Industrial or Military Facility Accidents Hazards (SER Section 2.1.1.3.3.1.3.4); and (v) Other Hazards (SER Section 2.1.1.3.3.1.3.5).

2.1.1.3.3.1.3.1 Geologic Hazards

The applicant provided information on geologic and geotechnical hazards that could affect the repository surface and subsurface areas of the GROA. SAR Table 1.6–8 identified five seismic geologic activity-related and nonseismic geologic activity-related hazards, including five volcanic activity-related hazards.

The NRC staff assessed these geologic and geotechnical hazards according to the following groups, which are discussed in six subsections: (i) Seismic Hazards, (ii) Volcano-Related Hazards, (iii) Slow Geologic Processes, (iv) Hill Slope Geologic Processes (avalanche, landslide, mass wasting), (v) Geologic Processes Affecting Soil Stability, and (vi) Subsurface Drift Degradation Processes. In SAR Table 1.6-8, the applicant also identified "Undetected Geologic Processes" and "Undetected Geologic Features" as potential hazards and did not include them in the PCSA. The NRC staff finds that not including undetected processes and features in the PCSA is acceptable because existence of such features and processes is extremely unlikely, given the detailed site investigations discussed and reviewed in SER Section 2.1.1.1. SAR Table 1.6-8 also includes dissolution as a potential hazard. The NRC staff reviewed dissolution-related hazards in SER Section 2.1.1.3.3.1.3.5, "Other Hazards," as part of its evaluation of geochemical processes.

**Seismic Hazards**

Seismic hazard information as an initiating event is described in SAR Section 1.6 and BSC (2008ai, Section 6.1). The applicant stated that seismic ground motion from an earthquake, and any resulting fault displacement, may damage structures, systems, or components (SSCs), including the surface and subsurface facilities, waste forms, the waste processing systems, and radiation workers. Thus, seismic hazards were included for further analysis as having the potential to initiate event sequences (BSC, 2007bq), as described by the applicant in BSC (2008ai, Section 6.1). As described in SAR Section 1.1.5.2, seismic ground motions with amplitudes significant enough to cause damage have an annual probability of exceedance greater than $10^{-6}$. The applicant excluded liquefaction and lateral spreading of soils, which are hazards that may result from earthquake vibratory ground motions. At Yucca Mountain, the soils are not saturated and the water table is approximately 390 m [1,270 ft] below the ground surface, as described by the applicant in BSC (2007bq, Section 6.1.4.4). Because these hazards require saturated soil conditions, which are not present at Yucca Mountain, these hazards are not likely to occur.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's seismic hazard screening information provided in SAR Section 1.6 and BSC (2008ai, Section 6.1) to confirm that the applicant used appropriate site-specific information and analyses and provided appropriate technical bases to include or exclude seismic hazards as initiators of event sequences.

The NRC staff finds that the inclusion of the seismic hazards for the PCSA by the applicant is acceptable because (i) as described in the screening criteria in SER Section 2.1.1.3.3.1.2, seismic ground motions with amplitudes significant enough to cause damage have an annual probability of exceedance greater than $10^{-6}$ (1 in 10,000 chance within the preclosure period); (ii) seismic hazards can be severe enough to affect the repository and its operations; and (iii) the technical bases for inclusion of the seismic hazards are consistent with site information. Additional details of the NRC staff's review of seismic-related hazards are described in SER Section 2.1.1.1.3.5.2.

The NRC staff also reviewed the information provided by the applicant regarding the exclusion of liquefaction and lateral spreading from the PCSA (BSC, 2007bq). The NRC staff finds that the soils at the proposed GROA are unsaturated (BSC, 2007bq, Section 6.1.4.4), and thus liquefaction will not occur (e.g., Terzaghi, et al., 1996aa). Therefore, the NRC staff finds that the applicant's exclusion of liquefaction and lateral spreading is acceptable because appropriate site-specific information was used to show that these phenomena would not occur.

**Volcanic Activity Hazards**

The applicant described volcanic (igneous) hazards as they relate to the PCSA in SAR Section 1.6.3.4.3 and BSC (2008ai, Section 6.3). These hazards include (SAR Table 1.6.3) volcanic and intrusive activity, encompassing magma intersecting waste emplacement drifts, lava in contact with surface facilities, ash fall from volcanoes, and lahars (mudflows or debris flows from volcanoes).

The applicant estimated the mean frequency of intersection of the subsurface repository by magma or an eruptive conduit to the surface, including the formation of lahars, to be approximately $5 \times 10^{-9}$ per year (SAR Section 1.6.3.4.3), well below the probability limit for exclusion of hazards in the preclosure period of $10^{-6}$ per year. Therefore, the applicant excluded these types of volcanic hazards from the PCSA.

The applicant stated that ash fall hazards could originate from distant volcanic eruptions, such as volcanic eruptions in central California. Ash fall could block overpack vents on the aging pads, accumulate on surface facility roofs, clog heating, ventilation, and air conditioning (HVAC) filters, and cause failure of HVAC systems, resulting in a loss of cooling. The applicant used the ASHPLUME model to estimate the ash fall thickness from a potential eruption (SAR Section 2.3.11.4.1.1.2; BSC, 2004bk). The applicant estimated a mean annual frequency of $6.4 \times 10^{-8}$ for a 10 $g/cm^2$ [20 psf] ash fall on the basis of a probabilistic dispersal of ash fall surrounding Yucca Mountain (BSC, 2008ai) and stated that the roofs of the surface facilities will be designed for a live load of 10.25 $g/cm^2$ [21 psf]. The roof live-loading limit is equivalent to about a 10 to 20-cm [4 to 8-in] thickness of freshly fallen ash, assuming a bulk density of 0.45 $g/cm^3$ [28 $lbs/ft^3$], as discussed in SAR Section 1.6.3.4.3 and responses to RAIs (DOE, 2009ap). Additionally, using the bulk density of ash from the eruption of Mount St. Helens in 1980, the applicant concluded that the ash depth for a 10 $g/cm^2$ [20 psf] live load would be well below the 41-cm [16-in] distance to the bottom of the aging overpack vents. Furthermore, the applicant stated in SAR Section 1.6.3.4.3 that it would conduct maintenance and take remedial actions after an ash fall event to remove ash and unclog the vents of the HVAC system and aging overpacks. Additionally, temporary ventilation systems would be used, if necessary. Consequently, the applicant excluded ash fall from further consideration.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's screening information related to volcanic activity provided in SAR Section 1.6.3.4.3, references therein, and responses to RAIs (DOE, 2009ap). In particular, the NRC staff evaluated the likelihood of magma and lava contacting the repository from volcanic activity near the site, as well as ash fall hazards related to both proximal and distant volcanoes.

The NRC staff finds that the applicant's exclusion of the igneous hazards associated with magma and lava contacting repository facilities both below and above the surface, based on low probability, is acceptable. The basis for this finding is the NRC staff's review, described in SER Sections 2.1.1.1.3.6, 2.2.1.2.1, 2.2.1.3.10, and 2.2.1.3.13, which provide an evaluation of the frequency and probability of volcanic events for the postclosure period, together with the associated uncertainties. For postclosure, the probability of igneous events is scaled to the size of the subsurface area of the GROA, not the size of the entire GROA. The likelihood of an eruptive conduit reaching the surface through the subsurface area of the GROA is estimated by the applicant to be $4.7 \times 10^{-9}$ per year, and this likelihood can then be scaled directly with the potential size of the GROA that could be impacted by the igneous hazard. The size of the GROA, which includes the surface and subsurface areas where waste handling would be conducted, as shown in SAR Figure 1-4, is two to three times larger than the size of the subsurface area of the GROA that could be impacted by the igneous hazard. Considering this scaling, the estimated probability for a conduit reaching the surface anywhere in the GROA relevant to the PCSA would still be well below the $1 \times 10^{-6}$ per year probability screening threshold. Therefore, the NRC staff finds that the applicant adequately estimated the frequency of occurrence of direct volcanic impacts to the repository and that the applicant provided an adequate technical basis to exclude direct igneous-related hazards as initiating events in the PCSA.

The NRC staff reviewed the applicant's assessment of ash loading to surface facilities as described by the applicant in SAR Section 1.6.3.4.3 and the response to the NRC staff RAI (DOE, 2009ap). As described in SER Section 2.1.1.1.3.6, the NRC staff found that the applicant's assessment of the ash-loading hazard to surface facilities is adequate for the following reasons. First, the applicant adequately demonstrated that the probability of a volcanic eruption from basaltic volcanoes in the vicinity of Yucca Mountain is less than $1 \times 10^{-6}$/year. Second, the applicant adequately determined that, before permanent closure, the probability of occurrence of an areal ash fall density greater than 10 g/cm$^2$ [20 psf] from distant volcanoes is well below $1 \times 10^{-6}$/year because the ash thicknesses from a volcanic eruption, as estimated by the applicant using an ash fall distribution model (ASHPLUME), are reasonable. The ASHPLUME model is evaluated by the NRC staff in SER Section 2.2.1.3.13.3.1.1. Based on this evaluation and the NRC staff's evaluation of volcanism at Yucca Mountain, described in SER Section 2.1.1.1.3.6, the NRC staff concludes that DOE provided an adequate basis for assessing the mass and thickness of future ash falls on the repository GROA in the preclosure period.

Also, because the applicant adequately estimated the annual frequency of igneous activity to be below the probability threshold of $1 \times 10^{-6}$ per year for exclusion of hazards in the preclosure period, the NRC staff finds that the applicant provided a sufficient technical basis to exclude ash fall as a potential initiating event. Additionally, the NRC staff finds that the applicant's plan to remove ash from HVAC and aging overpacks to prevent clogging, in the unlikely event of an ash fall, is acceptable because ash removal is a common industry practice.

Finally, the NRC staff considered whether DOE adequately considered uncertainty in the probability estimates. In the DOE analysis, DOE sampled a distribution of probability values for the likelihood of igneous events with a mean value of approximately $2 \times 10^{-8}$ per year and computed the 5$^{th}$ and 95$^{th}$ percentiles of the uncertainty distribution at $7.4 \times 10^{-10}$ and $5.5 \times 10^{-8}$, respectively. Based on the NRC staff's review in SER Sections 2.1.1.1.3.6 and 2.2.1.2.2.3.1, the NRC staff finds that DOE properly evaluated uncertainty in the probability calculation of igneous events. In addition, in SER Section 2.2.1.2.2.3.1, the NRC staff concluded that mean annual probability values outside the range between $1 \times 10^{-6}$ and $1 \times 10^{-9}$ are not consistent with past patterns of activity in the Yucca Mountain region and, thus, are not credible. Therefore, the NRC staff concludes that DOE adequately considered uncertainty in the assessment of the probability of igneous activity in the PCSA.

In summary, the NRC staff verified and confirmed that the applicant's (i) methods selected for determining probability or frequency of occurrence of hazards are appropriate, (ii) frequencies of occurrence of the hazards are valid, (iii) technical bases are adequate because they are consistent with site-specific information, and (iv) evaluation included appropriate consideration of uncertainty. Therefore, the NRC staff concludes that DOE appropriately excluded volcano-related hazards as initiators of event sequences.

**Slow Geologic Processes**

The applicant provided information on potential hazards from slow- or steady-state (noncatastrophic) geologic processes in SAR Section 1.6.3.4.2 and BSC (2008ai, Section 6.2). These processes involve gradual changes to the environment and include such phenomena as continental-scale vertical movements of the Earth's surface; epeirogenic and orogenic diastrophism and tectonic activity (i.e., large-scale folding, faulting, uplift, and depression of the Earth's crust); sedimentation; erosion, including denudation, coastal erosion, and stream erosion; and glaciation and glacial erosion. The applicant further stated that these processes may eventually render some of the waste emplacement areas unsuitable for disposal. The applicant excluded these processes as not having the potential to initiate an event sequence during the preclosure period, because these hazards progress slowly enough over time to allow remediation.

The applicant described sedimentation as the transport and deposition of particles by wind and water. This process occurs unevenly at the site area, with topography playing a major role in the location and amount of sedimentation. The applicant excluded this external hazard from further consideration because the slow rate of progression will provide ample time to consider waste relocation if sedimentation effects pose a hazard (BSC, 2008ai).

The applicant stated that the progression rate of both denudation and erosion will also be slow, allowing sufficient time for remedial actions to be taken to prevent event sequences from developing (SAR Section 1.6.3.4.2). Consequently, the applicant excluded both denudation and erosion from further evaluation as external hazards (SAR Section 1.6.3.4.2; BSC, 2008ai). As there are no coastlines near the repository site, the applicant also excluded coastal erosion as a potential hazard. Currently, there are no intermittent or continuous flowing streams at the site. Consequently, the applicant excluded stream erosion as a potential hazard in the preclosure period.

The applicant stated that the current climatic conditions at the repository site would not allow glacier formation. Therefore, the applicant concluded that glacial erosion and glaciation would not be potential initiators of initiating events at the repository during the preclosure period (SAR Section 1.6.3.4.2; BSC, 2008ai).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on slow geologic processes (steady-state or noncatastrophic) provided in SAR Section 1.6.3.4.2, BSC (2008ai, Section 6.2), and references therein, to examine whether the applicant's technical basis for excluding tectonic activity, sedimentation, erosion, and glaciation as potential initiating events is adequate and consistent with site information. The NRC staff concludes that the applicant showed that these hazards, including tectonic movement, sedimentation, erosion, and denudation, are not credible initiating events, because they would progress at sufficiently slow rates to allow relocation of the waste, if necessary.

Consistent with the NRC staff's review of site information provided in SER Sections 2.1.1.1.3.1, 2.1.1.1.3.2, and 2.1.1.1.3.3, the NRC staff also finds that the site is significantly inland from nearby coastlines, rendering coastal erosion a noncredible event. Further, the NRC staff finds that there is currently no stream or glacier inside the GROA boundary near the planned surface facilities, and climatic conditions are not favorable for formation of a glacier during the preclosure period.

Therefore, the NRC staff concludes that the applicant's exclusion of slow or steady-state geologic processes as initiators of event sequences is acceptable because, consistent with site information, there will be sufficient time to provide an adequate response to address hazards associated with these slow-developing events. The applicant's exclusion of these hazards is consistent with guidance in ASME/ANS RA–S–2008 (ASME, 2008aa), which includes a criterion to justify exclusion of events if they are slow in developing and there is sufficient time to provide an adequate response to address the slow-developing events. Geologic processes that include continental-scale vertical movements of the Earth's surface; epeirogenic and orogenic diastrophism and tectonic activity (i.e., large-scale folding, faulting, uplift, and depression of the Earth's crust); sedimentation; erosion, including denudation, coastal erosion, and stream erosion; and glaciation and glacial erosion are steady state and, therefore, will not change site conditions significantly or at a significant rate over the 100-year preclosure period.

**Hill Slope Processes**

The applicant provided information in SAR Section 1.6.3.4.2 and BSC (2008ai, Section 6.2) on nonseismic geologic hazards. As listed in SAR Table 1.6-8, these hazards include hill slope processes; specifically, avalanche, landslide, and mass-wasting events.

The applicant stated that avalanches, landslides, and mass-wasting events may trigger loose soil, rock, or ice/snow to slide down nearby hill slopes and impact or bury parts of the surface facilities. In SAR Section 1.6.3.4.2, the applicant excluded avalanches as not having the potential to initiate an event sequence because snow and ice do not accumulate at the site. The applicant also excluded mass wasting and, therefore, landslide as a potential initiator of an event sequence because of the absence of suitable topography and geology (BSC, 2008ai). In response to the NRC staff's RAI (DOE, 2009fe), the applicant also stated that the flat topography of the surface GROA was not conducive to generating a mass-wasting event.

In SAR Sections 1.1.7.2.1 and 1.1.7.2.2, the applicant cited a storm-triggered debris flow event that occurred on the south hill slope of Jake Ridge in 1984, which deposited an average thickness of 16 cm [6.3 in] of sediments on the lower hill slope. In response to the NRC staff's RAI (DOE, 2009fe) to justify why a mass-wasting event analogous to the 1984 Jake Ridge event could not be triggered on the eastern slopes of Exile Hill, the applicant stated that, due to topography, the small-scale event at Jake Ridge could not impact the surface GROA facilities that would be built at the base of Exile Hill's eastern slope. In addition, the applicant stated in SAR Section 1.1.4.1.2.2 and DOE (2009fe) that two storm water drainage diversion channels would be constructed to protect the surface GROA and the North Portal from storm water runoff and debris flow emanating from the eastern slopes of Exile Hill. In response to an NRC staff's RAI (DOE, 2009fe), the applicant further stated that sizing and exact placement of the diversion channels will be determined during detailed design.

**NRC Staff's Evaluation**

The NRC staff reviewed the hill slope process information provided in SAR Section 1.6.3.4.2 and BSC (2008ai, Section 6.2), references therein, and responses to RAIs (DOE, 2009fe) to assess whether the applicant used appropriate site-specific information and analysis to exclude avalanche, landslide, and mass-wasting hazards from initiating event sequences.

The NRC staff finds that the applicant's assessment on whether avalanches could be a potential hazard is acceptable because the applicant conducted the assessment using appropriate site-specific data on maximum monthly snowfall and extreme temperature range and terrain slopes in BSC (2008ai). On the basis of these site-specific data, the NRC staff finds that anticipated snowfall near the surface facilities is small. Additionally, the NRC staff finds that the expected temperature ranges do not indicate sufficiently cold and long periods for snow to remain and accumulate to become an avalanche hazard. Therefore, the NRC staff finds the applicant's conclusion that a snow avalanche would not be a potential initiator for event sequences to affect the surface facilities is acceptable.

The NRC staff finds that mass-wasting events analogous to the 1984 Jake Ridge event, although rare, could occur on the eastern slopes of Exile Hill and, thus, cannot be excluded as a hazard by the absence of suitable topography and geology alone, because geologic records and recent observations (e.g., Jake Ridge event in 1984) show mass-wasting events continue to occur at the repository area. However, the applicant provided the design of the diversion channels in DOE (2009fe), which the applicant proposed for mitigating mass-wasting or landslide events. Based on the review of the design of the diversion channels, the NRC staff finds that the applicant can rely on these channels to protect the surface GROA and the North Portal from potential debris flow emanating from mass-wasting events on Exile Hill. Therefore, the NRC staff concludes that landslide and mass-wasting will not be a hazard to the repository surface facilities, because the diversion channels would be able to protect the surface facilities and it is acceptable to eliminate them as an initiating event during the preclosure period.

On the basis of the above evaluation, the NRC staff finds that the applicant's exclusion of hill slope processes that include avalanches, mass-wasting events, and landslides as initiators of event sequences is acceptable because the applicant's technical basis for the exclusion of these hazards is consistent with the site information and design of the diversion channels.

**Geologic Processes Affecting Soil Stability**

The applicant provided information about the potential impact of processes affecting soil stability in SAR Section 1.6.3.4.2 and in BSC (2008ai, Section 6.2). The applicant provided additional information in its responses to the NRC staff's RAIs (DOE, 2009bg,ej,ey) and in BSC (2007ba). As listed in SAR Table 1.6-8, these hazards include settlement, soil shrink–swell consolidation, static fracturing, and subsidence. These hazards can potentially affect the surface facilities by compromising stability and integrity of the surface soil materials. The applicant excluded all these processes as not having potential to initiate an event sequence during the preclosure period, because these hazards would progress slowly over time, allowing necessary remediation actions to be taken, such as relocating waste until a longer term solution could be implemented (BSC, 2008ai).

The applicant stated that soil consolidation and soil shrink–swell due to drying and wetting could result in fissures and cracks in the ground (DOE, 2009ey). The applicant (DOE, 2009ey) stated that any clay-rich soil at the repository site would not be exposed to sufficient wetting and drying due to the arid climate; therefore, any potential hazards associated with soil consolidation from shrink–swell can be eliminated. Additionally, the applicant stated that repository site subsidence would be localized (BSC, 2008ai). The applicant excluded subsidence on the basis of the overall slow progress that would allow necessary remediation actions to be taken.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on the processes affecting soil stability provided in SAR Sections 1.1.5.3.1.1, 1.3.4.2.1, and 1.6.3.4.2; BSC (2008ai, Section 6.2) and references therein; and responses to the NRC staff's RAIs (DOE, 2009bg,ej,ey) to assess whether the technical bases for excluding these processes as external hazards and initiating events are adequate. Also, the NRC staff reviewed the applicant's assessment of potential hazards from settlement of surface facility structures in SER Section 2.1.1.1.3.5.4, where the NRC staff finds that the applicant's information and analyses are adequate to assess the engineering design and performance of the structural foundation related to potential settlement hazard.

The NRC staff finds that the applicant's assessment that the soil is not expected to undergo repeated wetting and drying cycles is acceptable because the climate at the repository site is arid. Therefore, the NRC staff concludes that fissures in the soil or consolidation of the soil mass due to shrink–swell cycles would not be a hazard at the site. The NRC staff also finds that the subsurface facilities will be in relatively competent rock mass at a depth of 300 m [984 ft] or more (SAR Section 1.1.5.3.1.1). Additionally, the emplacement drifts will be constructed at a nominal spacing of 81 m [266 ft] (SAR Section 1.3.4.2.1). This makes the extraction ratio, defined as the ratio of area excavated to total area, small. Because the applicant stated that the excavations will be supported and the extraction ratio is small, the NRC staff finds that the subsurface facilities would not experience massive collapse of the thick {more than 300 m [984 ft]} overlying strata to affect soil stability. Therefore, based on these findings, the NRC staff concludes that the applicant's exclusion of processes affecting soil stability, such as settlement, soil shrink–swell consolidation, static fracturing, and subsidence, as initiators of event sequences, is acceptable, and the technical bases for the exclusion of these hazards are consistent with site information.

**Subsurface Drift Degradation Processes**

The applicant provided information about subsurface drift degradation hazards in SAR Section 1.6.3.4.2 and in BSC (2008ai, Section 6.2). The applicant provided additional information in its response to the NRC staff's RAI (DOE, 2009ey).

The applicant indicated that drift degradation processes included drift degradation, fracturing–fractures (stress-induced fractures), rock deformation, and rockbursts. The applicant assessed the potential degradation of the emplacement drifts during the preclosure period and concluded that drifts will be stable without ground support, based on the drift design (BSC, 2007ai). The applicant further stated that drifts could have spalling of the rock wall; however, such spalling will be mitigated by including a perforated stainless steel liner in the ground support system (DOE, 2009ed). The applicant concluded that these hazards would not cause any adverse effects on the GROA facilities during the preclosure period (BSC, 2008ai).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on potential subsurface drift degradation provided in SAR Section 1.6.3.4.2, BSC (2008ai, Section 6.2), BSC (2007an), and references therein to examine whether the applicant's technical bases for excluding external hazards and initiating events are adequate and consistent with site characteristics. The NRC staff evaluated whether (i) site data were appropriately used to identify drift degradation, fracturing–fractures (stress-induced fractures), rock deformation, and rockbursts as potential hazards; (ii) the hazards could occur within the preclosure period; and (iii) appropriate bases have been provided to exclude the hazard from the PCSA. The NRC staff found that, as evaluated in SER Section 2.1.1.1.3.5.1.1, the applicant used appropriate site characterization data to identify potential drift degradation hazards for waste emplacement and retrieval operations. Based on the NRC staff's evaluation of the DOE characterization, the NRC staff finds that stress-induced deformation and associated fracture formation in the rock mass could potentially contribute to degrading the stability of subsurface excavations during the preclosure period. Rockburst is a sudden release of accumulated strain energy, generally accompanied by violent expulsion of rock blocks from a tunnel or from excavations in deep mines in strong rock masses under a very high stress field. However, the NRC staff finds that rockburst potential would be negligible because of the relatively low stress field in conjunction with relatively softer rock mass, as provided in BSC (2007an, Tables 3-4 and 3-5). Further, the NRC staff's review of underground opening stability during the preclosure period is given in SER Sections 2.1.1.2.3.3 and 2.1.1.2.3.7.3 where the NRC staff found that DOE's design of underground openings in the inaccessible areas of the subsurface facility is acceptable because it is based on empirical rules for ground support system designs and site-specific rock mass mechanical properties. Thus, the NRC staff found it acceptable that DOE's design, along with the monitoring and maintenance plans for underground openings in inaccessible areas of the subsurface facility, can be relied on to ensure that these openings will remain stable during the preclosure period. Therefore, the NRC staff concludes that the applicant's exclusion of subsurface drift degradation processes as initiators of event sequences is acceptable because the technical bases for the exclusion of these hazards are consistent with site information and design of the underground facilities.

2.1.1.3.3.1.3.2 Weather-Related Hazards

The applicant provided weather-related hazards information in SAR Sections 1.6.3.4.4 and 1.6.3.4.6. SAR Table 1.6-8 identified external event categories of high winds, tornadoes, and lightning as weather-related hazards, and includes (i) barometric pressure, (ii) extreme wind, (iii) extreme weather and climate fluctuations, (iv) hurricane (high-wind effects), (v) missile impact, (vi) tornadoes, and (vii) lightning. The NRC staff reviewed weather-related hazards to verify and confirm that the applicant provided adequate technical bases for exclusion of the weather-related hazards that are consistent with the screening criteria discussed in SER Section 2.1.1.3.3.1.2. The NRC staff's review of the applicant's information on exclusion of the weather-related hazards is described in three hazard groups: (i) high winds, (ii) tornadoes and tornado-induced missiles, and (iii) lightning.

**High Winds**

The applicant presented information on high-wind (straight-line wind) hazards in SAR Section 1.6.3.4.4 and BSC (2008ai, Section 6.4) that could result from a number of weather-related phenomena, including rapidly changing barometric pressures, thunderstorms, other extreme weather and climate fluctuations, or hurricanes.

The Yucca Mountain area is classified as a special wind region requiring site-specific data by the ASCE (2006aa). The applicant collected site-specific wind data from Site 1, approximately 1.0 km [0.6 mi] south of the North Portal. The applicant estimated the maximum 3-second gust straight-line wind speed for the Yucca Mountain site to be 193 km/h [120 mph] at an annual frequency of occurrence $10^{-6}$ (BSC, 2007dc). With regard to uncertainty, the applicant estimated the $5^{th}$ and $95^{th}$ percentile values about the mean from this calculation as 169.6 km/h and 208.4 [105.5 and 129.5 mph], respectively. The applicant estimated the straight-line wind speed using the Fisher–Tippett Type I extreme value distribution suggested by Simiu and Scanlan (1996aa) and ASCE/SEI 7–05 (ASCE, 2006aa). The applicant concluded that the design-basis straight-line wind speed, or high-wind speed, is bounded by the design-basis tornado wind speed at an annual frequency of occurrence of $10^{-6}$ (BSC, 2008ai). SAR Section 1.6.3.4.4 specified that the maximum design basis tornado wind speed at the repository site is 304 km/h [189 mph] for ITS structures. As a result, the applicant determined that the potential consequences of straight-line winds are bounded by tornado wind speed, and, therefore, need not be evaluated in the PCSA.

Additionally, the applicant stated that hurricanes and associated high winds are not expected at the repository site, because the nearest sea or ocean is more than 360 km [225 mi] away (BSC, 2008ai). The applicant further stated that, should a hurricane traverse over land toward

Yucca Mountain, a substantial amount of the hurricane energy would dissipate in the intervening mountainous terrain. Additionally, no rivers and estuaries in the intervening areas could serve as pathways to transmit hurricane storm surge to the Yucca Mountain site.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.6.3.4.4 and BSC (2008ai; BSC, 2007dc), including references therein, and the methodologies the applicant used to exclude the wind-related hazards. The NRC staff's review focus was to verify that the applicant appropriately (i) characterized the wind speed and associated frequency of occurrence and (ii) used site-specific data information in wind speed characterization. Specifically, the NRC staff relied on industry standards such as Simiu and Scanlan (1996aa), ASCE/SEI 7–05 (ASCE, 2006aa), ANSI/ANS–2.8 (ANS, 1992ab), and the applicant's response to the NRC staff's RAI (DOE, 2009fe) to evaluate the adequacy of technical bases used to exclude initiating events arising from high winds. Simiu and Scanlan (1996aa) is an industry-recognized reference for determining wind loads in the context of the methods and standards to construct wind-resistant structures. ASCE/SEI 7–05 (ASCE, 2006aa) provides the methods to determine relevant static and dynamic loads (including wind) in the context of general structural design. ANSI/ANS–2.8 (ANS, 1992ab) includes criteria for evaluation of hurricanes at nuclear power plants. These guidance documents are applicable to the PCSA because they provide methods to quantify wind loads that incorporate meteorological data and which are independent of the type of facility being evaluated.

The NRC staff finds that the applicant used an appropriate probabilistic method (the Fisher–Tippett Type I extreme value distribution) to estimate wind speeds and their associated frequencies of occurrence, including uncertainties, because the method is consistent with ASCE/SEI 7–05 (ASCE, 2006aa) and Simiu and Scanlan (1996aa). Additionally, the NRC staff finds that the applicant used appropriate site-specific information {data collected approximately 1.0 km [0.6 mi] south of the North Portal} as input to the calculation to characterize the straight-line wind speed.

The NRC staff reviewed the applicant's information concerning the probability of hurricanes making landfall at Yucca Mountain and finds that the potential is remote because the waters off the coast of California generally are too cold to support a cyclone with hurricane intensity (Vickery, et al., 2011aa). Additionally, the prevailing upper-level wind generally steers the cyclonic systems westward away from the mainland of the United States (Vickery, et al., 2011aa). Although the western part of the United States has experienced tropical storm force winds from remnants of the tropical cyclones, the NRC staff finds that the applicant's assessment that hurricanes generated in the Pacific Ocean would not enhance the straight wind speed significantly is acceptable because the coastline is at least 360 km [225 mi] away with mountains in the intervening region. For comparison, ANSI/ANS–2.8 (ANS, 1992ab) requires that hurricanes should be considered as a potential hazard to a facility site if the site is within 161 to 322 km [100 to 200 mi] from the coastline and if preferential pathways exist.

Based on these findings, the NRC staff concludes that the applicant's exclusion of straight-line winds as initiators of event sequences is acceptable for the following reasons. First, the applicant used appropriate methodology and site-specific data to probabilistically assess straight-line winds (as evaluated in SER Section 2.1.1.1.3.3). Second, the applicant determined that the maximum design basis tornado wind speed exceeds the $1 \times 10^{-6}$/year straight-line wind by a large margin: 193 km/h [120 mph] for straight-line winds versus the maximum design basis tornado wind speed of 304 km/h [189 mph]. Third, structures, systems, and components important to safety will be designed for the maximum design basis tornado wind speed that has an annual frequency of occurrence at the site that is less than $1 \times 10^{-6}$ per year (design basis tornado winds are evaluated in SER Section 2.1.1.1.3.3). Fourth, DOE adequately assessed uncertainties because it developed a statistical analysis of site data, which showed that the 95th percentile straight-line wind speed is well below the maximum design basis tornado wind speed. Therefore, the technical bases for the exclusion of these hazards are adequate and consistent with the site information.

**Tornado/Tornado-Generated Missiles**

The applicant presented information on tornado-related hazards in SAR Section 1.6.3.4.4 and BSC (2008ai, Section 6.4, Attachment A). The evaluation of tornadoes included consideration of the probability of occurrence of tornado winds, failure probabilities of structures and facilities due to extreme winds, the exposure times of critical transportation equipment, and generation of tornado missiles. In particular, the applicant considered the tornado wind hazards as potential initiating events for the following surface structures and vehicles: Canister Receipt and Closure Facility (CRCF), Receipt Facility (RF), Wet Handling Facility (WHF), Initial Handling Facility (IHF), railcar and truck buffer areas, aging pads, and transportation vehicles [site transporters and the transport and emplacement vehicle (TEV)], and ITS structures during and after construction.

To estimate site-specific tornado characteristics at Yucca Mountain (BSC, 2008ai,Table A4), the applicant used the data for the western United States (west of 102° west longitude) given in NUREG/CR–4461 (Ramsdell and Rishel, 2007aa), adjusted for the number of tornadoes observed in the Yucca Mountain region compared to other parts of the western United States. The analysis used mean tornado impact areas and path lengths based on the 5th to 95th percentile distribution of past tornado strike data to estimate the overall mean tornado strike frequency. DOE computed the resulting tornado strike probabilities for each of the surface structures taking into account the dimensions of the structures and adjusted for the 50-year exposure time of the surface-area facilities. DOE made this adjustment for a 50-year exposure time because it concluded that after the 50-year period of waste emplacement and when all the waste is located in the subsurface area of the GROA, radiological release due to event sequences initiated by high-wind hazards are no longer credible.

Based on strike probability analysis, the applicant determined that the annual frequency of a tornado strike during the preclosure period to the surface facilities, including the railcar and truck buffer area and aging pads, is between $1.2 \times 10^{-6}$ and $3.4 \times 10^{-6}$, which exceeds the threshold probability of $1.0 \times 10^{-6}$/year. Therefore, the applicant concluded that a tornado strike is a credible hazard to these facilities and has to be evaluated further in the PCSA. Specifically, the applicant conducted probabilistic failure analyses of the surface facilities (CRCF, RF, WHF, IHF, railcar and truck buffer areas, and aging pads) subjected to tornado winds, as described in the following paragraphs, to assess whether the probability of failure of these surface facilities due to tornado winds was above or below the $1.0 \times 10^{-6}$/year probability threshold.

In contrast to the surface facilities, including the railcar and truck buffer area and aging pads, the applicant determined that a tornado striking a site transporter or a TEV is not a credible hazard. DOE computed the probability of tornado strike for the site transporter and TEV of $7.8 \times 10^{-8}$ and $8.2 \times 10^{-8}$, respectively. This conclusion was based on an evaluation of the exposure time of this equipment to potential tornadoes during TEV operations. The applicant stated that site transporters and the TEV would be exposed to a potential tornado only when they are outside the waste handling buildings or the subsurface facility. During the 50-year operational period, the applicant estimated that the cumulative exposure time for site transporters and the TEV (the transit time in which the equipment would be susceptible to a tornado or a tornado-missile strike) would be 3.9 and 4.1 years, respectively. The applicant based the 3.9-year exposure time for the site transporter on the estimated time needed to move the aging overpacks between the surface facilities and aging pads. Similarly, the exposure time for the TEV was based on the estimated time needed to move waste packages from the surface facilities to the underground facility. The applicant assumed the exposure time for each operation to be 2 hours for the site transporter and 3 hours for the TEV.

To further assess whether tornado hazards from tornado winds and tornado-generated missiles should be included in the PCSA, the applicant determined the failure probabilities of the surface facilities (specifically, CRCF, RF, WHF, IHF, railcar and truck buffer areas, and aging pads) subjected to tornado winds. The likelihood of damage is estimated by calculating the conditional probability of damage from tornado impact and combining this probability with the tornado strike probability. These conditional probabilities of door failures were combined with frequency of tornado strike at the same wind speeds to estimate an overall frequency of door failure due to a tornado. For the CRCF, RF, WHF, and IHF, DOE used data and methodology given in Ramsdell and Rishel (2007aa) to estimate the conditional probability of damage using the overhead doors at the entry vestibules as surrogates for facility damage because DOE considered that the doors are structurally weaker than the entire facility structures. The applicant considered that a structure was damaged if the surrogate for that structure sustained damage because of the maximum tornado design wind speed. The applicant estimated the damage probability of the overhead doors for these facilities by correlating door damage with the wind speed, per the results of the Enhanced Fujita Scale report developed by the Wind and Science Engineering Center at Texas Tech University (Texas Tech University, 2006aa). Of these facilities, the DOE determined that the CRCF has a higher tornado strike failure probability due to its large footprint. The structural failure probability of the surrogate overhead door of the CRCF was estimated to be less than $1 \times 10^{-6}$/year (BSC, 2008ai). Therefore, DOE concluded that the CRCF, RF, WHF, and IHF would not sustain structural damage from tornadoes, because the CRCF analysis is bounding, and thus that tornado-initiated damage to these structures could not initiate event sequences involving these facilities. The surrogates used in the analysis are based on commercial construction, whereas the buildings constructed at the surface area of the GROA will follow nuclear-grade design and construction (see SER Section 2.1.1.7.3.1.1). DOE also noted that most of the large surface facility structures and aging overpacks are designed to withstand the wind effects of 304 km/h [189 mph].

Because realistic surrogates could not be identified for the transportation casks in the railcar and truck buffer area or for aging overpacks sitting on aging pads, the applicant estimated the probabilities of structural failure of a transportation cask and aging overpack subject to a tornado strike at a wind speed determined specifically for the transportation casks and aging overpacks using the information provided in the IAEA Safety Guide (IAEA, 2003ab). In this analysis, the applicant converted the static pressures at failure of a reinforced concrete wall and rugged vessel to an equivalent wind speed using Bernoulli’s equation, and subsequently combined the probability of structural failure at the converted wind speed with the tornado strike probability. On the basis of this analysis, the applicant concluded in BSC (2008ai, Section A3.2) that the probabilities of adverse tornado wind effects on the transportation casks in the railcar and truck buffer area or on aging overpacks sitting on aging pads would be lower than the probability screening threshold of $1 \times 10^{-6}$/year. Therefore, the applicant concluded that transportation casks in the railcar and truck buffer area and aging overpacks sitting on aging pads would not sustain structural damage from tornadoes with an annual probability greater than $1 \times 10^{-6}$, and thus that tornado-initiated damage to these structures could not initiate event sequences involving these facilities. As a result, the applicant excluded these initiating events from the PCSA.

For tornado-generated missiles, the applicant used the classification in Coats and Murray (1985aa) to assess the effects of tornado-generated missiles on ITS structures (BSC, 2008ai, Section A3.3). The applicant stated that heavy missiles (such as utility poles or automobiles) are not possible at the site, because the winds are not strong enough to lift these heavy objects and generate such missiles. DOE also stated that missiles commonly found during concurrent construction and operation were also shown to be unlikely, with a probability of impacting the waste handling facilities to be less than $1 \times 10^{-6}$ (BSC, 2008ai). After the construction is complete, the applicant stated that the tornado-generated missiles would consist of relatively small onsite debris (e.g., branches, pieces of lumber, small beams, pipes). The applicant also concluded that embedded pipes will not become tornado-generated missiles, because the expected tornado wind speed would be too low to dislodge them. The applicant estimated penetration depth of smaller missiles to be significantly less than the wall thicknesses of aging overpacks, waste handling buildings, transportation casks, and TEV. Consequently, the applicant excluded tornado-generated missiles as initiating events in the PCSA.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's assessment of the tornado and tornado-generated missiles hazards, as provided in SAR Section 1.6.3.4.4, BSC (2008ai, Section 6.4 and Attachment A), and references therein. Additionally, the NRC staff reviewed the applicant's response to the NRC staff RAI (DOE, 2009ey). The NRC staff considered the guidance in NUREG–0800 (NRC, 2007al, Section 3.5.1.4), Regulatory Guide 1.76 (NRC, 2007ai), and Regulatory Guide 1.200 (NRC, 2009af). Regulatory Guide 1.76 is based on the data and analysis of wind speeds in NUREG/CR–4461 (Ramsdell and Rishel, 2007aa), and provides guidance in selecting design-basis tornado and design-basis tornado-generated missiles for nuclear power plants. NUREG/CR–4461 uses characteristics of tornadoes reported in the contiguous United States from January 1950 through August 2003 to determine strike probabilities and maximum wind speed for use in nuclear power plant designs. NUREG–0800, Section 3.5.1.4, provides guidance on how to review missiles generated by tornadoes and extreme winds for nuclear power plants. Regulatory Guide 1.200 provides guidance on determining the technical adequacy of probabilistic risk assessments. Although these guidance documents were originally developed for nuclear power plants, they are applicable to the surface area of the GROA because they include acceptable methodologies for assessing natural phenomena and extreme weather, which are independent of the type of facility being evaluated.

The NRC staff finds that the applicant's use of the 50-year exposure time to estimate the tornado strike frequencies for the GROA facilities is acceptable because this exposure time is consistent with the expected duration of operations at the surface area of the GROA. Use of the exposure time in the frequency and probability calculations is based on a risk-informed approach, where the period of time the radiological material is exposed to the hazard is taken into consideration. For operations and activities related to handling radioactive waste at the surface area of the GROA, the exposure of waste to tornadoes would be limited to the waste emplacement period (the first 50 years of the preclosure period) because tornadoes would only initiate event sequences that would lead to radiological release when waste is present at the surface of the GROA. For the second 50 years of the preclosure period (monitoring), waste would no longer be present at the surface because it would have been emplaced underground where it would be shielded from any potential events resulting from high winds. Similarly, the NRC staff also finds the uses of the estimated 3.9-year exposure time for the site transporter and the 4.1-year exposure time for the TEV acceptable because these exposure times represent the amount of time the waste would be susceptible to the hazard. These exposure times are consistent with the throughput values in SAR Table 1.7-5.

The NRC staff finds the methodology and wind speed data used by the applicant to assess the tornado hazard at the repository are appropriate because the assessment relied on Regulatory Guide 1.76, NUREG/CR–4461, and NUREG–0800, as well as site-specific meteorological data, as evaluated in SER Section 2.1.1.1.3.3. Because the probability of a tornado strike is greater than the $1 \times 10^{-6}$ threshold, the NRC staff finds DOE's conclusion that tornado strikes are a credible hazard acceptable. The NRC staff finds the use of conditional probability of damage from tornado impact, combined with the tornado strike probability, acceptable because this is a standard method to probabilistically evaluate the potential for component or structural failure and is consistent with NRC guidance for probabilistic risk assessments (NRC, 2009). The NRC staff notes that the technical basis for the maximum tornado design basis wind speed of 304 km/h [189 mph] is described and evaluated in SER Section 2.1.1.1.3.2. The probability of the maximum tornado design basis wind speed is $1 \times 10^{-7}$, which is below the $1 \times 10^{-6}$/year probability threshold. Because DOE used the probability of the maximum design basis wind speed of $1 \times 10^{-7}$ in its failure probability analysis, the NRC staff finds that DOE's analysis is conservative. The NRC staff also concludes that the statistical approach used by DOE to calculate tornado strike frequencies, which was based on the 5$^{th}$ to 95$^{th}$ percentile distribution of past tornado strike data, adequately considers the uncertainties associated with the frequency calculations.

The NRC staff finds the approach of using the overhead doors as surrogates to estimate the damage potential for CRCF, RF, WHF, and IHF acceptable because the surrogates used are structurally weaker than the rest of the reinforced concrete structures that comprise the waste handling facilities. Consequently, this approach will lead to a conservative evaluation. The NRC staff also finds DOE's approach for estimating failure probabilities of transportation casks and aging overpacks acceptable for the following two reasons. First, the applicant converted the static pressures at failure to an equivalent wind speed using Bernoulli's equation, which is based on fundamental fluid mechanics. Second, DOE subsequently combined the probability of structural failure at the converted wind speed with the tornado strike probability, resulting in a bounding probability of failure.

The NRC staff finds DOE's failure probability analyses of the CRCF, RF, WHF, and IHF acceptable because the analysis used the degree of damage and damage indicators developed in the Texas Tech University Enhanced Fujita Scale report (Texas Tech University, 2006aa). This updated scale that correlates wind speeds and expected degrees of damage was developed based on input from a broad cross section of civil engineers, meteorologists, and atmospheric scientists and has been accepted for use by the National Oceanographic and Atmospheric Administration and the National Weather Service. The NRC staff also finds DOE's conclusion that tornado damage for these facilities can be excluded as an initiating event in the PCSA acceptable because the calculated tornado-initiated failure probabilities, based on the Enhanced Fujita Scale, are less than $1 \times 10^{-6}$/year.

The NRC staff also finds that the applicant's use of IAEA Safety Guide (2003ab) data to assess damage to transportation casks in railcar and truck buffer areas and aging overpacks sitting on aging pads is acceptable. This analysis is based on the failure probability of a material with a given thickness correlated with wind speed. These IAEA data are internationally accepted and used in the design of nuclear power plants. They are applicable to analogous facilities and activities proposed for the GROA because they are being used to assess structural damage from wind and wind-generated missiles, which is a facility-independent analysis. Thus, the NRC staff concludes that adverse tornado wind effects on transportation casks in the railcar and truck buffer area and aging overpacks sitting on aging pads are below the probability threshold of $1 \times 10^{-6}$/year.

The NRC staff reviewed the applicant's assessment that heavy tornado missiles (e.g., automobiles, utility poles) would not be generated at wind speeds exceeding those in BSC (2008ai, Table A4) based on Coates and Murray (1985aa, Table 9). In addition, analysis presented in BSC (2008ai) showed that the annual frequency of lighter missiles (e.g., tree branches, pieces of lumber, small beams, pipes) affecting safety-related structures, systems, and components would be below the threshold frequency. The NRC staff finds that the applicant's analysis is bounding because the applicant designed all ITS structures to withstand a Spectrum II tornado missile strike (SAR Table 1.2.2-1), as defined in NUREG–0800 (NRC, 1981ad, Section 3.5.1.4). Spectrum II tornado missiles have equal or greater mass than those recommended in Regulatory Guide 1.76 (NRC, 2007ai), which is applicable to tornado analyses in the same geographic region as Yucca Mountain. Based on this information, the NRC staff finds that surface facilities of the surface area of the GROA would withstand potential tornado missiles that may be generated at the site, and, therefore, the applicant's exclusion of event sequences initiated by tornado missiles is acceptable.

In summary, based on the foregoing evaluation, the NRC staff finds that the applicant's exclusion of tornadoes and tornado-generated missiles hazards as initiators of event sequences in the PCSA at the GROA is acceptable because the technical bases for the assumptions and methods used to identify tornado hazards as initiating events is based on applicable NRC and industry guidance. The applicant's probability analyses relied on acceptable site- and facility-specific data, incorporating all the surface facilities that handle radiological waste. The event probabilities used to exclude wind hazards from the PCSA are based on acceptable site and facility data, engineering failure analyses, NRC guidance documents, and industry practice. Therefore, the NRC staff concludes that the technical bases for the exclusion of high-wind hazards are adequate because they are consistent with site information and include an adequate consideration of uncertainty.

**Lightning**

The applicant presented the lightning hazard information in SAR Section 1.6.3.4.6 and BSC (2008ai, Section 6.6). On the basis of lightning strike data collected over a 3,600-$km^2$ [1,400-$mi^2$] region around Yucca Mountain between 1991 and 1996, the applicant determined that the strike density ranges from 0.06 to 0.4 strikes/$km^2$/yr [0.16 to 1.04 strikes/$mi^2$/yr] (BSC, 2008ai). A National Oceanic and Atmospheric Administration (NOAA) report cited in BSC, (2008ai) reported a strike density for the Yucca Mountain area of 0.2 flashes/$km^2$/yr [0.52 flashes/$mi^2$/yr]. Assuming the protected area of the GROA is 2.7 $km^2$ [1.04 $mi^2$], the applicant estimated a lightning strike rate of 0.54 lightning strikes/yr (SAR Section 1.6.3.4.6; BSC, 2008ai). Because this annual strike rate exceeds the $1 \times 10^{-6}$/year threshold probability for excluding lightning strikes from the PCSA, the applicant concluded that this is a credible hazard which needed to be considered further. DOE stated that ITS structures and systems are designed to withstand a lightning strike (SAR Section 1.6.3.4.6; BSC, 2008ai). Based on the DOE design of SSC and an analysis of the effects of lightning strikes on representative transportation casks, aging overpacks, and TEVs (BSC, 2008ai, Appendix B) that showed lightning could not damage these SSC ITS nor breach containment, DOE concluded that radioactive release could not occur as a result of lightning strikes. Based on this conclusion, DOE concluded that lightning strikes can be excluded as initiating events in the PCSA.

As part of the proposed design of ITS structures and systems, the applicant stated that it would install a lightning protection system for buildings and outdoor elevated structures (BSC, 2007av; BSC, 2008ai) in accordance with (i) National Fire Protection Association (NFPA) 780–2004 (NFPA, 2004aa), which provides lightning protection system installation guidelines for a variety of structures; (ii) Underwriters Laboratories 96A (Underwriters Laboratories, 2005aa), which provides guidelines on how to develop and install a lightning protection system; and (iii) Regulatory Guide 1.204 (NRC, 2005ad), which is NRC guidance on lightning protection for nuclear power reactors. In addition, the applicant noted that lightning strikes may be mitigated by reinforcing bars in reinforced concrete structures (known as the Faraday cage effect). The applicant stated that placing waste forms within the reinforced concrete structures lowers risk (BSC, 2008ai). According to the applicant, the Faraday cage effect can be relied on to help protect the railcar and truck buffer area and aging facility from lightning strikes (BSC, 2008ai). The applicant also recognized that casks and canisters may be vulnerable to a lightning strike during transportation between different site facilities and protected areas, in addition to any side flashes generated when lightning strikes the lightning safety system. Therefore, the applicant analyzed the effects of a direct lightning strike on a representative transportation cask, aging overpack, and TEV, and showed that in a worst-case lightning strike, the pit depth would be less than 3 mm [0.1 in] and the average interior wall temperature under the strike point would not exceed 570 °C [1,058 °F] if the wall has at least 12 mm [0.47 in] of metal (BSC, 2008ai). As the walls of an aging overpack, transportation cask and canister, and TEV would be thicker than 12 mm [0.47 in], the applicant concluded that there would be no breach of containment resulting in radioactive release.

**NRC Staff's Evaluation**

The NRC staff reviewed lightning information the applicant provided in SAR Section 1.6.3.4.6, BSC (2008ai, Section 6.6), and references therein. Specifically, the NRC staff evaluated how the applicant excluded initiating events arising from lightning strikes that could affect an ITS SSC in addition to design features incorporated to withstand a lightning strike. The NRC staff's review was to assess whether the applicant appropriately estimated lightning strike frequency and provided design features for ITS SSCs to reduce lightning strike potential and withstand a lightning strike without any radiological consequences. As appropriate, the NRC staff relied on guidance and information provided in NFPA 780–2004 (NFPA, 2004aa), Underwriters Laboratories 96A (Underwriters Laboratories, 2005aa), and Regulatory Guide 1.204 (NRC, 2005ad). All three guidance documents are applicable to the GROA because lightning is a common hazard for both the repository facilities and nuclear power plants, and the analysis is facility independent.

The NRC staff finds that the applicant appropriately estimated annual lightning strike frequency because the lightning strike data at the Yucca Mountain region that the applicant used consisted of site-specific recordings as well as recordings of lightning strikes across the Nevada National Security Site (NNSS, formerly the Nevada Test Site). The NRC staff evaluated and found acceptable DOE's lightning data in SER Section 2.1.1.1.3.3. The NRC staff also finds that the special design features the applicant proposed to install on ITS SSCs to reduce the potential for lightning strike are acceptable because the lightning protection system is designed following industry standard codes and NRC guidance on lightning protection (NFPA 780–2004, Underwriters Laboratories 96A, and Regulatory Guide 1.204). Additionally, the NRC staff finds that the transportation cask, aging overpack, and the TEV would be able to withstand a lightning strike without resulting in event sequences because the applicant's analysis, given in BSC (2008ai, Attachment B), demonstrates that a lightning strike would be insufficient to cause a breach of the containment. Therefore, the NRC staff finds that the technical basis for exclusion of lightning strikes as initiating events in the PCSA is acceptable.

2.1.1.3.3.1.3.3 Aircraft Crash Hazards

The applicant provided information involving aircraft crash hazards in SAR Section 1.6.3.4.1, BSC (2007ak,ap), and responses to the NRC staff's RAIs (DOE, 2008ah, DOE, 2009fh,fi). DOE's assessment was based on a two-step approach. In the first step, DOE analyzed all potential hazards from airborne activities within 160 km [100 mi] of the potential GROA, using the North Portal as a point of reference for distance measurements. The hazards that posed a potential risk were then used in the second step of its analysis. In this second step, DOE quantified the frequency of aircraft crash hazards at the GROA based on historical flight and crash data from the Federal Aviation Administration and the U.S. Air Force. In the frequency analysis, DOE took credit for several flight restrictions and flight operational constraints over the repository that limited aircraft activity. DOE's frequency analysis assumes that these constraints would be implemented prior to operations (SAR Section 1.6.3.4.1). Based on results of this frequency analysis, DOE excluded aircraft crashes as an initiating event in the PCSA.

The applicant listed potential sources of aircraft-related hazards within 160 km [100 mi] of the North Portal (SAR Section 1.6.3.4.1; BSC, 2007ap). These hazards included flights to and from nearby civilian, DOE-controlled, and military airports, including those through the Beatty Corridor; federal airways; military training routes and areas; air refueling routes; restricted airspace and military operating areas of the Nevada Test and Training Range (NTTR); and the restricted airspace over the NNSS. As shown in BSC (2007ap, Figure 6-1), the restricted airspace above the NNSS is subdivided into two federal airways: R–4808N and R–4808S. The applicant stated in BSC (2007ap) that DOE controls Airspace R–4808N, which is subdivided into R–4808A, R–4808B, R–4808C, R–4808D, and R–4808E. The repository surface facilities are located beneath Airspace R–4808E, as shown in BSC (2007ap, Figure 6-1). Airspace R–4808S is jointly controlled by DOE (NNSS), Nellis Air Traffic Control Facility, and Federal Aviation Administration (FAA) Los Angeles Air Route Traffic Control Center (BSC, 2007ap).

In the first step of the analysis, DOE used methodology from DOE–STD–3014–2006 and NUREG–0800 to exclude all but three potential aircraft hazards on the basis of distance and flight-frequency criteria (BSC, 2007ap). Based on results from this first step, DOE identified three potential hazards that required further evaluation: (i) helicopter flights near the GROA, (ii) small military aircraft flying in the NNSS and the NTTR, and (iii) aircraft transiting the Beatty Corridor. For the second two potential hazards, DOE conducted an aircraft crash frequency analysis. For helicopter flights, DOE did not perform a crash frequency analysis, because DOE assumed that an operational requirement prohibiting any helicopter flights within 0.8 km [0.5 mi] of waste handling facilities, aging pads, and other relevant areas that handle SNF and high-level radioactive waste (BSC, 2007ak, Section 3.3.3) would be in place prior to construction and during operations (SAR Section 1.6.3.4.1). Additionally, DOE stated that helipads would be located 0.8 km [0.5 mi] away from the relevant surface facilities/areas, as shown in BSC (2007ak, Table 1).

For evaluation of small military aircraft flying in the NNSS and NTTR, DOE considered two cases: flights over the flight-restricted area around the GROA [Case (ii)a] and flights operating in the NTTR but outside the flight-restricted area [Case (ii)b]. For the crash frequency of small military aircraft overflying the flight-restricted area around the GROA [Case (ii)a], the applicant relied on data from U.S. Department of the Air Force (2007aa) from 1990 through 2006. According to the applicant, aircraft conducting these flights are required by the U.S. Air Force to be in a normal flight mode, not conducting maneuvers or other activities (BSC, 2007ak). In addition, in SAR Section 1.6.3.4.1, the applicant states an assumed flight restriction, in which flights by fixed-wing aircraft within the restricted airspace and below 4,267 m [14,000 ft mean sea level] are prohibited. In DOE (2008ah, 2009fi), the applicant used the NUREG–0800 (NRC, 2010ab, Section 3.5.1.6) formula to calculate the annual crash frequency due to the assumed maximum 1,000 annual flights permitted over the flight-restricted airspace. The estimated annual crash frequency onto the surface area of the GROA was $8.1 \times 10^{-7}$.

The applicant used a Bayesian analysis of the crash rate of military aircraft in the NTTR to estimate the annual crash frequency that would result from small military aircraft operating in the NTTR but outside the flight-restricted area [Case (ii)b]. The Bayesian density is derived from 18 crashes observed in the NTTR and military operations area over the 16-year period between May 1990 through December 2006 and the applicable areas of the NTTR and military operations area. The estimated frequency of crashes from military flights outside the flight-restricted airspace was $9.4 \times 10^{-7}$ crashes/yr (DOE, 2009fi).

The third type of potential aircraft hazards that DOE evaluated further were flights through the Beatty Corridor. The Beatty Corridor is defined in BSC (2007ak, Section 3.2.8) as an equivalent [26-mile]-wide band with edges parallel to the Nevada–California border, passing within equivalent [5 mile] of the North Portal, at its closest. BSC (2007ak, Section 3.2.10, Table 2) lists annual flight traffic transiting the Beatty Corridor from small military, large military, general aviation, air taxi, and air carrier flights. In BSC (2007ak, Section 6.5.1), the applicant calculated the annual frequency of crashes onto the repository facilities from flights using this corridor with the model of Solomon (1988aa). As given by Solomon (1988aa), the exponential decay rate of areal flight density with distance from the center line of the airway toward the edge of the airway differs for each type of aircraft. In addition, the applicant increased the number of flights through the Beatty Corridor by 400 percent to account for uncertainties in flight count and traffic growth (BSC, 2007ak, Section 3.2.10). Crash rates of aircraft flying through the Beatty Corridor per flight mile were taken from Kimura, et al. (1996aa) using information from the National Transportation Safety Board and listed in BSC (2007ak, Table 15). The applicant used military aircraft crashes from Kimura, et al. (1996aa) for both normal and special flight modes, including updated mishap information from the U.S. Department of the Air Force, as described in BSC (2007ak, Attachment IV).

Because aircraft crash hazard frequency also depends on the area of the GROA that could be adversely affected by an aircraft crash or related hazard, DOE included a factor in its calculations to account for the effective target area. Using the equations from DOE–STD–3014–2006 (DOE, 2006aa), the applicant calculated the effective areas of structures in the GROA that may contain radioactive waste, including various handling facilities, rail and truck staging areas, and the aging pads. Results of these total effective area calculations were given in BSC (2007ak, Table 19) for various aircraft and range from 0.85 $km^2$ [0.33 $mi^2$] for small military aircraft to 1.89 $km^2$ [0.73 $mi^2$] for large commercial aircraft. These values are used in the frequency calculation to scale the aircraft crash rate to the surface area of the GROA where an aircraft crash could potentially result in an accidental radiological release. After combining the crash frequencies from flights through the Beatty Corridor and military flights over and outside the flight-restricted airspace, the applicant estimated the cumulative annual frequency of aircraft crashing onto the effective target area to be $1.78 \times 10^{-6}$. According to the applicant, this combined crash frequency meets the screening criterion of less than $2.0 \times 10^{-6}$ per year (BSC, 2007ak). This screening criterion of $2.0 \times 10^{-6}$ per year is based on a 50-year operational period for the surface facilities, as stated in SAR Section 1.6.3.4.1 and BSC (2007ak). The frequency value, given a 50-year-exposure time in the 100-year preclosure period, is equivalent to $9 \times 10^{-7}$ over the preclosure period ($1.78 \times 10^{-6}$/year × 50 years ÷ 100 years).

The DOE frequency analysis credited a number of flight restrictions and operational constraints over the GROA. First, although DOE included aircraft crash hazards from military flights through the NNSS and NTTR in the second step of its analysis, potential hazards from associated military operations, such as ordnance delivery, dropped objects, ground-to-ground missile testing, and radar and communications jamming, were excluded from the frequency analysis based on assumed operational controls regarding flight-restricted airspace and activity constraints (SAR Section 1.6.3.4.1). Second, in BSC (2007ak, Section 3.3.1), the applicant assumed a flight-restricted airspace surrounding the North Portal with a radius of 9 km [4.9 nautical mi or 5.6 statute mi] extending from the ground surface to 4,267 m [14,000 ft] above mean sea level. DOE assumes that only 1,000 overflights by military aircraft would be allowed annually, as described in BSC (2007ak, Section 3.3.2). The applicant stated that these flights would be in normal flight mode; therefore, no tactical maneuvering would be allowed, as detailed in BSC (2007ak, Section 3.3.2). Third, in BSC (2007ap, Section 6.1.2), and in responses to RAIs (DOE, 2008ah), DOE stated that military aircrafts are allowed to transit R–4808N, but that pilots must observe certain avoidance areas such as 1-nautical-mile radius no-fly areas over the Device Assembly Facility and Bren Tower. As described in responses to RAIs (DOE, 2008ah), the U.S. Department of the Air Force Warfare Center at Nellis Air Force Base, Nevada, revised Air Force Instruction 13–212, Volume 1, Addendum A, to include flight restrictions that, according to DOE, would be implemented before waste is onsite. The applicant stated that it will implement these flight restrictions through the National Nuclear Security Administration, Nevada Site Office. The applicant also stated in its responses to RAIs (DOE, 2008ah) that the actual controls for restricting flights over the repository area (SAR Section 5.8.3) are not fully developed and that it will develop these controls to restrict maneuvering and other activities. According to the RAI response, the final plan will include (i) the means to inform the affected organizations about the proposed flight-restricted airspace; (ii) the means to monitor the annual number of flights over the flight-restricted airspace, to inform affected organizations when the annual limit is being approached, and to restrict overflights when the limit is reached; and (iii) the means to ensure that restricted activities do not occur in the flight-restricted airspace.

**NRC Staff’s Evaluation**

The NRC staff reviewed the information on aircraft crash hazards in SAR Section 1.6.3.4.1, BSC (2007ak,ap), and responses to the NRC staff’s RAIs (DOE, 2008ah; DOE,2009fh,fi). Specifically, the NRC staff evaluated DOE’s (i) methodology to develop the aircraft crash hazard frequency analysis, (ii) consideration of all potential hazards within 160 km [100 mi] of the GROA to determine which hazards merit more detailed evaluation (the first step in DOE’s two-step approach), (iii) information used to support DOE’s frequency calculation (the second step in DOE’s two-step approach), and (iv) set of flight restrictions and operational constraints assumed by DOE as credits in the frequency analysis in order to limit certain types of aircraft crashes and thereby reduce the overall likelihood of an aircraft crash hazard at the GROA.

The NRC staff finds the methodology used by DOE to evaluate the frequency of an aircraft crash acceptable because DOE’s methodology follows guidance in both DOE–STD–3014–2006 (DOE, 2006aa) and NUREG–0800, Section 3.5.1.6 (NRC, 2010ab). DOE–STD–3014–2006 was established by the DOE “to provide a sound, technically justifiable, and consistent approach to analyzing the risk posed by an aircraft crash into a facility containing radioactive or hazardous chemical materials."  This standard has been used by NRC and DOE to evaluate aircraft crash hazards at numerous nuclear facilities across the United States, including by the NRC staff in evaluating aircraft crash hazard frequency for the GE–Hitachi Global Laser Enrichment Facility in North Carolina (NRC, 2012ab, NUREG–2120).  The NRC staff also used this standard to support aircraft hazard evaluations for nuclear power plants, including for safety evaluations of spent fuel pools at decommissioned nuclear power plants (NRC, 2001ag, NUREG–1738), which are analogous to the wet handling facility proposed for the surface of the GROA.  Finally, the NRC staff finds that it is appropriate to use guidance and methodologies from NUREG–0800 to assess aircraft hazards, which were developed for nuclear power plants, because the likelihood of aircraft crash hazards does not depend on the type of nuclear facility being evaluated.

In evaluating the first step of DOE's two-step approach, the NRC staff reviewed the information related to flights to and from nearby civilian, DOE-controlled, and military airports, including those through the Beatty Corridor; federal airways; military training routes and areas; air refueling routes; restricted airspace and military operating areas of the NTTR; and the restricted airspace over the NNSS, as provided in SAR Section 1.6.3.4.1, BSC (2007ak), and references therein.  The NRC staff finds that the applicant appropriately excluded any hazards associated with aircraft crashes except for (i) helicopters, (ii) small military aircraft in the NNSS and the NTTR, and (iii) aircraft transecting the Beatty Corridor.  Specifically, the NRC staff finds that flights landing at and taking off from the civilian, DOE, and military airports that were considered in the first step of DOE's two-step approach were properly eliminated from consideration based on distance and flight frequency criteria specified in NUREG–0800 (NRC, 2010ab, Section 3.5.1.6).  According to this guidance, none of these airports have a sufficient number of operations, given their distance from the GROA, to pose a credible hazard.

In evaluating the second step of DOE's two-step approach, the NRC staff reviewed DOE's additional analysis for (i) helicopters; (ii) small military aircraft in the NNSS and the NTTR; and (iii) aircraft transecting the Beatty Corridor, which were not eliminated from consideration in the first step.  For helicopter crashes, the NRC staff evaluated DOE's assumption that procedural safety controls (PSCs) would be in place to maintain a minimum separation distance of 0.8 km [0.5 mi] from relevant surface facilities (SAR Table 1.9-10, PSC-18).  The NRC staff finds use of PSCs acceptable because maintaining a separation distance of more than 0.4 km [0.25 mi], which is recommended in DOE Standard DOE–STD–3014–2006 (2006aa), will prevent helicopters from crashing into GROA facilities that would handle SNF and high-level radioactive waste.

The NRC staff also reviewed the information the applicant provided on crashes of military aircraft flights over flight-restricted airspace in BSC (2007ak) and in responses to the NRC staff's RAIs (DOE, 2008ah; DOE,2009fi).  The NRC staff finds the use of the methodology from NUREG–0800 (NRC, 2010ab, Section 3.5.1.6) to estimate crash frequency from small military aircraft overflying the flight-restricted area is acceptable.  This method evaluates aircraft crash frequency using several conservative assumptions, including an assumed uniform distribution of flight paths through the entire width of the corridor, and an assumption that any crashes that originate within the flight paths of the corridor will result in a crash within the corridor.  More realistically, some crash trajectories that initiate in the corridor could exit the corridor before impact.  Therefore, the NRC staff finds that the estimated crash frequency of aircraft transiting the flight-restricted airspace onto the surface area of the GROA is acceptable.  The NRC staff concludes that the crash frequency would be less than or equal to the applicant's estimated $8.1 \times 10^{-7}$ crashes per year because of the aforementioned conservatisms.  The NRC staff notes that, as described later in this discussion, this conclusion is based on DOE's assumed flight restrictions, in which military aircraft must fly in normal flight mode and are limited to 1,000 overflights per year. DOE credits these assumptions in its frequency calculation.

The NRC staff reviewed the information in BSC (2007ak, Section 3.2.14) and responses to RAIs (DOE, 2009fi) that the applicant provided on crashes of military flights that originated outside flight-restricted airspace but crashed into the GROA. The NRC staff concludes that the applicant's use of a Bayesian analysis to evaluate the crash frequency for this type of hazard is acceptable because this approach is consistent with standard probabilistic risk assessments (PRA) used for nuclear power plants, as described in NUREG/CR–6823 (Atwood, et al., 2003aa). This NUREG/CR provides guidance on sources of information and methods for estimating the parameters used to determine the frequencies and probabilities of various events modeled in PRAs and for quantifying the uncertainties in the estimates. This includes a determination of both facility-specific and generic estimates for initiating event frequencies. This guidance is appropriate for use in evaluating aircraft crash hazard frequency at the GROA because the methodologies do not depend on the type of nuclear facility being evaluated.

Further, the NRC staff finds that the applicant's estimation of crash frequency from flights originating outside flight-restricted airspace is conservative for the following reasons. First, DOE did not take credit for the distance from the boundary of the flight-restricted area to the North Portal. For this crash scenario, a damaged aircraft would have to travel a minimum of 9 km [5.6 mi] (the radius of the flight-restricted area) to impact the surface area of the GROA. To calculate crash density, DOE used all historical small military aircraft crash data for crashes in a 16.5-year period (BSC, 2007ak, Attachment III). However, according to BSC (2007ak, Attachment III) only about 25 percent of these crashes originated from engine failure, and based on the NRC staff's technical judgment, only engine failure mishaps would result in a crash that could reach the GROA. Crashes initiated by other causes, such as flying directly into the ground, mid-air collisions, or loss of control, would result in a crash much closer to the point at which the mishap originated rather than a trajectory traversing the 9-km [4.9-nautical mi or 5.6-statute mi] flight-restricted airspace radius. Moreover, the applicant indicated that, in case of engine failures, military pilots follow an explicit set of procedures to try to prevent a crash. These procedures include zooming up to gain altitude, gliding, pointing the aircraft toward the nearest airfield, and attempting engine restart. As a result of these procedures and considering where the aircrafts operate, the NRC staff concludes that pilots experiencing engine failures will generally not be steering their aircraft toward the surface area of the GROA.

The third potential type of hazard that DOE considered in the second step of DOE's two-step approach involves crashes from flights through the Beatty Corridor. The NRC staff reviewed the information the applicant provided in BSC (2007ak, Section 6.5.1) on crashes using the method proposed by Solomon (1988aa). The Solomon method used by the applicant is an alternative to the method in NUREG–0800 (NRC, 2010ab, Section 3.5.1.6). The staff finds use of this method acceptable because (i) similar to the NUREG–0800 Section 3.5.1.6 (NRC, 2010ab) model, the Solomon model assumes that the flights in an air corridor follow a straight-line path; (ii) the model is a nuclear-industry-recognized method for evaluating aircraft crash hazards; and (iii) the Solomon model provides specific factors to account for differences in the flight density for commercial air carriers, general aviation, and military aircraft, based on information from the National Transportation and Safety Board and Naval Safety Center. In addition, the NRC staff also evaluated DOE's estimate of flight traffic in the Beatty corridor. The NRC staff concludes that increasing estimated flight traffic through the Beatty corridor by 400 percent is conservative, especially when compared to current FAA projections (FAA, 2014aa). Therefore, the NRC staff finds that DOE's estimated annual crash frequency of aircraft transiting through the Beatty Corridor onto the surface area of the GROA is acceptable.

Because aircraft crash hazard frequency also depends on the area of the GROA that could be adversely affected by an aircraft crash or related hazard, the NRC staff reviewed information the applicant provided in BSC (2007ak) on the effective target area of GROA facilities. The NRC staff finds that the applicant's estimation of the effective aircraft crash target areas is appropriate because the estimation is based on the standard approach provided in DOE–STD–3014–2006 (DOE, 2006aa, Appendix B) and is consistent with the guidance in NUREG–0800 (NRC, 2010ab, Section 3.5.1.6).

On the basis of the above evaluation of DOE's information, the NRC staff concludes that DOE provided an adequate technical basis for exclusion of aircraft crash hazards as an initiating event in the PCSA. In particular, the NRC staff concludes that DOE's methodology is acceptable because it is consistent with applicable NRC guidance and standard industry practices. This methodology identified crashes related to helicopters, small military aircraft in the NTTR and NNSS, and aircraft transecting the Beatty Corridor to be the only types of aircraft activity that contribute to the aircraft crash hazard assessment. After eliminating helicopter hazards due to PSCs, the applicant calculated a crash frequency of $1.78 \times 10^{-6}$/year, which included crashes from small military aircraft and aircraft transecting the Beatty corridor.

Exclusion of aircraft crash hazards from the PCSA is based on a risk-informed approach, where the exposure time of the radiological material is considered. For most event sequences, performance is assessed by DOE and evaluated by the NRC staff against the 100-year preclosure period because radiological release could occur as a result of the hazard any time during that period. However, operations and activities related to handling radioactive waste at the surface area of the GROA would be limited to the waste emplacement period (the first 50 years of the preclosure period). Considering the type of hazard posed by an aircraft crash, the NRC staff determines that aircraft crash hazards could only initiate event sequences that would lead to radiological release when waste is present at the surface of the GROA. For the second 50 years of the preclosure period (monitoring), waste would no longer be present at the surface because it would have been emplaced underground where it would be shielded from any potential adverse impacts resulting from an aircraft crash. As a result, the NRC staff concludes that aircraft crashes do not present a credible hazard with regard to radiological release after waste has been emplaced. Therefore, the NRC staff determines that use of an exposure time of 50 years (the period of waste emplacement) is appropriate for the probability calculation used to exclude aircraft crash hazards from the PCSA. The effective probability of aircraft crash onto the site is then calculated as $9 \times 10^{-7}$/year ($1.78 \times 10^{-6}$/year $\times$ 50 years $\div$ 100 years). The NRC staff finds that DOE's basis for exclusion of aircraft crash hazards from the PCSA is acceptable because it is less than the threshold probability of $1 \times 10^{-6}$/year (1 in 10,000 chance over the preclosure period).

As described in SAR Section 1.6.3.4.1, DOE states that its "frequency analysis credits a flight restricted airspace and operational constraints over the repository." In particular, DOE assumes that only 1,000 overflights per year of fixed-wing aircraft will be permitted in the flight-restricted airspace above the repository and that flights will be conducted in normal flight mode. In addition to the aforementioned operational constraints and flight restrictions, DOE also credited operational constraints regarding potential hazards related to ordnance delivery, dropped objects, ground-to-ground missile testing, and radar and communications jamming (BSC, 2007ap, Sections 6.5, 6.6, and 6.8). The NRC staff finds that, with consideration of the assumed operational constraints for these activities, DOE's conclusion that the activities would be limited to allowed ranges, and therefore would not constitute a hazard to the surface area of the GROA, is acceptable.

In total, DOE credits six flight restrictions and operational constraints in its frequency analysis (SAR Section 1.6.3.4.1) in the restricted airspace. These are (i) prohibiting fixed-wing flights below 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (ii) 1,000 overflight limit per year for fixed wing aircraft above 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (iii) overflights are limited to straight and level flights (i.e., maneuvering is not permitted); (iv) carrying ordnance is prohibited within 9 km [5.6 mi] of the North Portal; (v) electronic jamming activities are prohibited within 9 km [5.6 mi] of the North Portal; and (vi) helicopters are not permitted within 0.8 km [0.5 mi] of facilities that process, stage, or age nuclear waste forms. Because DOE's identification of hazards and initiating events and associated probabilities and subsequent exclusion of aircraft crash hazards as an initiating event in the PCSA assumes these flight restrictions and operational constraints as part of its technical bases, the NRC staff proposes a condition of construction authorization. This proposed condition of construction authorization would require DOE to provide written notification that the agreements for these restrictions and operational constraints are in place before commencement of construction to confirm that the technical bases for exclusion of aircraft crash hazards at the GROA from the PCSA that DOE provided in accordance with 10 CFR 63.112(d) remain valid.

**Proposed Condition of Construction Authorization**:

> DOE shall provide the NRC staff written notification that the agreements for the six flight restrictions and operational constraints that DOE credits in its frequency analysis (SAR Section 1.6.3.4.1) are in place before commencement of construction to confirm that the technical bases for exclusion of aircraft crash hazards at the GROA from the Preclosure Safety Analysis (PCSA) that DOE provided in accordance with 10 CFR 63.112(d) remain valid. These restrictions and operational constraints are (i) prohibiting fixed-wing flights below 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (ii) 1,000 overflight limit per year for fixed-wing aircraft above 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (iii) flights are limited to straight and level flights (i.e., maneuvering is not permitted); (iv) carrying ordnance is prohibited within 9 km [5.6 mi] of the North Portal; (v) electronic jamming activities are prohibited within 9 km [5.6 mi] of the North Portal; and (vi) helicopters are not permitted within 0.8 km [0.5 mi] of facilities that process, stage, or age nuclear waste forms.

2.1.1.3.3.1.3.4 Nearby Industrial or Military Facility Accidents Hazards

The applicant identified nearby industrial and military facilities and associated activities in SAR Sections 1.6.3.4.8 and 1.1.1.3. Additional information and analyses were provided in BSC (2008an) and DOE (2009fe). On the basis of guidance provided in NUREG–0800 (NRC, 2007aj, Sections 2.2.1 and 2.2.2), the applicant described all facilities and activities within 8 km [5 mi] of the repository. Additionally, following the guidance in NUREG–0800 (NRC, 2007aj, Sections 2.2.1 and 2.2.2), facilities and activities at distances greater than 8 km [5 mi] from the repository that could affect the safety-related features at the repository facilities were described in SAR Section 1.6.3.4.8 and BSC (2008an). BSC (2008an, Figure 1) provided the location of these facilities. NNSS land use information in this figure was from the final environmental impact statement for the test site and offsite locations (DOE, 1996ab). Locations of the active mines were from Driesner and Coyner (2006aa). The applicant used this information and the analysis as the bases for evaluating activities at these nearby facilities that could pose a potential hazard to the repository during the preclosure period and initiate an event sequence. The NRC staff reviewed the following hazards identified by the applicant from nearby industrial or military facility accidents: (i) induced air overpressure, (ii) induced seismic motion, (iii) release of radiological materials and toxic chemicals from nearby facilities, (iv) waste management program, (v) mining, (vi) release of onsite hazardous material, and (vii) turbine-generated missiles.

The applicant considered shipwreck as a potential hazard in its assessment of nearby industrial and military facility accidents (SAR Table 1.6-8; BSC, 2008an). Because the repository site is far from any seashore, the applicant excluded a shipwreck hazard from affecting preclosure operations because it is not a credible event. The NRC staff concludes that shipwrecks are not a credible event, because of the distance from the GROA to the nearest coastline {e.g., greater than equivalent [225 mi] from the GROA to the California Coast}. The applicant also evaluated potential hazards associated with commercial rocket launch and retrieval operations proposed by Rocketplane Kistler. However, no operational facilities exist, and the National Aeronautics and Space Administration (NASA) terminated the contract for using Areas 18 and 19 for launching and recovering reusable rockets in 2007. The NRC staff, therefore, concludes that commercial rocket launches are not a credible hazard and can be excluded from the PCSA. The applicant also listed fog caused by an industrial accident as a potential external hazard (SAR Table 1.6-8; BSC, 2008ai). The NRC staff concludes that such a fog would not be a credible hazard and can be excluded from the PCSA because no accidents were identified that would produce fog at the site.

**Induced Air Overpressure**

The applicant provided information on air overpressure hazards resulting from explosive and flammable materials within the GROA and at facilities or activities within the adjoining NNSS in SAR Sections 1.1.1.3 and 1.6.3.4.8, BSC (2008an), and DOE (2009fe). The applicant cited Regulatory Guide 1.91 (NRC, 1978ac), which provides guidance for evaluating postulated explosions at nearby facilities and transportation routes near nuclear power plants. In particular, the applicant relied on the analysis in Regulatory Guide 1.91, which determined that, conservatively, 6.9 kPa [1.0 psi] is the level below which no significant damage to SSCs from an explosion is expected to occur. In addition, the applicant relied on Equation 1 in the Regulatory Guide to determine the minimum safe distance. According to the guidance, the minimum safe distance is the distance from an explosion that would result in an incident overpressure less than or equal to 6.9 kPa [1.0 psi]. SAR Section 1.6.3.4.8 and BSC (2008an) identified only the Rail Equipment Maintenance Yard within 8 km [5 mi] of the repository that would store a substantial amount of flammable materials and pose an air overpressure hazard. This yard, located 3.2 km [2 mi] from the GROA boundary, would store diesel fuel in a 189,271-L [50,000-gal] tank. Assuming the diesel fuel undergoes a vapor-cloud explosion, the applicant (BSC, 2008an) estimated that an explosion of all the diesel fuel in the tank would produce an air overpressure of 6.9 kPa [1 psi] at a distance of 52 m [170 ft] for the nominal case, and 166 m [546 ft] for the bounding case. The nominal case assumes an explosion efficiency of 3 percent, and the bounding case assumes 100 percent explosion efficiency. The applicant stated that as trains with loaded transportation casks will not travel closer than 183 m [600 ft] to this tank, no damage to the transportation casks will be expected from such an explosion.

On the basis of information on activities conducted at different facilities in the NNSS, the applicant (BSC, 2008an) identified that activities at the following facilities may pose an induced-air-overpressure-related hazard to the repository: Device Assembly Facility; Area 27 Complex; U–1a Complex/Lyner Complex; Big Explosives Experimental Facility; Nevada

Energetic Materials Operations Facility; Next Generation Radiographic and Magnetic Flux Compression Generation Facilities; Area 11 Explosive Ordnance Disposal Unit; and testing and training exercises with small arms, artillery, guns, and rockets. The facilities identified by the applicant are situated at least 32 km [20 mi] from the repository. Using the methodology given in Regulatory Guide 1.91 (NRC, 1978ac), the applicant estimated that 5,900 kt [$1.3 \times 10^{10}$ lb] of trinitrotoluene (TNT) would be necessary to develop an air overpressure of 6.9 kPa [1 psi] at a distance of 32 km [20 mi] from the repository surface facilities. The applicant (BSC, 2008an) stated that 92 kt [$2 \times 10^{8}$ lb] would most likely exceed inventories of TNT within the NNSS. Therefore, the applicant concluded that insufficient material is available to generate an explosion at the NNSS facilities that would produce an air overpressure of 6.9 kPa [1 psi] at the GROA. Consequently, the applicant excluded induced air overpressure from the PCSA.

Lathrop Wells Road is the closest road to the GROA, located approximately 11 km [7 mi] to the south–southeast. Some hazardous materials are transported over this road to support the Work for Others Program, which is a program hosted by the applicant, where federal agencies share resources and facilities in NNSS for various military training exercises and research and development projects (BSC, 2008an). Additionally, U.S. Highway 95 is used to haul significant quantities of munitions, propellants, explosives, and radioactive materials. At its closest point to the repository, U.S. Highway 95 is approximately 21 km [13 mi] away. There are no transportation railway lines within 32 km [20 mi] of the repository. The applicant stated that it will construct a new rail line connecting the repository operations area with the commercial line. The applicant concluded that as the road and railway transportation routes are sufficiently far from the repository, a transportation accident resulting from an explosion would not pose significant adverse effects to the repository facilities and operations (BSC, 2008an).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's induced-air-overpressure information provided in SAR Sections 1.1.1.3 and 1.6.3.4.8 and BSC (2008an). Specifically, the NRC staff evaluated the description, quantity, and distance of the facility handling or storing the explosive materials from the repository facilities to estimate the potential for a hazard caused by induced air overpressure and whether this type of hazard can be excluded from the PCSA.

The NRC staff finds that the applicant's assessment of the damage potential of air overpressure from accidental explosion of stored diesel fuel in the GROA and other explosive materials in the NNSS is acceptable because this assessment was based on applicable guidance in Regulatory Guide 1.91 (NRC, 2013af). This Regulatory Guide is applicable to the repository because it provides a methodology for evaluating postulated explosions at nearby facilities and transportation routes near nuclear power plants. The methodology, which factors in the size of the explosion and the distance from the explosion to the site, is independent of the type of facility being evaluated. Specifically, the applicant used the peak positive incident-air-overpressure criterion of Regulatory Guide 1.91 (NRC, 2013af) to assess the separation distance (or alternatively, safe quantity) of explosives that would not exceed the safe air overpressure of 6.9 kPa [1 psi]. This Regulatory Guide specifies that below this overpressure, no significant damage to any ITS SSC is expected as the additional load imposed on them is insignificant. The NRC staff verified that the applicant correctly converted the diesel fuel to an equivalent amount of TNT explosive. This TNT equivalency is a standard methodology many organizations use (e.g., U.S. Departments of Army, Navy, and the Air Force, 1990aa). Additionally, the NRC staff finds that the applicant's assumption that the entire tank would be filled with diesel vapor at the upper flammable limit is acceptable because this is a bounding assumption that is consistent with Regulatory Guide 1.91 (NRC, 2013af). The NRC staff also verified that an explosion of a diesel tank located at the rail maintenance yard would not produce an air overpressure greater than the safe air overpressure limit of 6.9 kPa [1 psi], given the distance from the rail yard to the GROA. Therefore, the NRC staff finds that the applicant's conclusion that an explosion of the diesel tank can be excluded from the PCSA is acceptable.

The NRC staff independently verified the distances of the facilities within the Nevada Test Site (NTS) by examining BSC (2008an, Figure 1) and other NNSS maps. The NRC staff also reviewed the fact sheet for the Device Assembly Facility (DOE, 2010ao) to assess any hazards to the GROA from an accident at this facility. On the basis of its review of this fact sheet, the NRC staff finds that each of the five cells in the Device Assembly Facility can handle a maximum of 250 kg [550 lb] of TNT. Additionally, the NRC staff estimated the quantity of TNT-equivalent explosives using the approach in Regulatory Guide 1.91 (NRC, 2013af). The NRC staff finds that the amount of TNT present at the Device Assembly Facility is insufficient to exceed the amount of TNT needed to produce 6.9 kPa [1 psi] of overpressure 32 km [20 mi] away {5,900 kt [$1.3 \times 10^{10}$ lb]}, which is consistent with DOE's analysis. Additionally, on the basis of Regulatory Guide 1.91 (NRC, 2013af, Figure 1), the NRC staff finds that the safe distance for the detonation of 1,860 kg [4,100 lb] of TNT-equivalent explosive at the Area 11 explosive ordnance disposal facility is 220 m [720 ft] and detonation of this quantity would not damage ITS SSCs at the repository from the generated air overpressure due to the relatively large distance between the two facilities, which is 32 km [20 mi]. Therefore, the NRC staff concludes that the amount of explosive material available to produce an explosion is significantly less than the amount that would be required to exceed an overpressure of 6.9 kPa [1 psi] at the GROA, given the distance between the nearest potential explosion site and the GROA.

For transported explosive materials, DOE calculated that 92 kt [$2.0 \times 10^8$ lb] of TNT-equivalent material would have to explode at a distance of 8.0 km [5 mi] from the GROA in order to produce an air overpressure in excess of the safe overpressure limit of 6.9 kPa [1 psi] (BSC, 2008an, Table 3). Based on Regulatory Guide 1.91 (NRC, 2013af, Figure 1), the maximum amount of solid hazardous cargo that can be transported in a single truck is 23,000 kg [50,000 lb]. A single railcar can carry a maximum of 60,000 kg [132,000 lb] of explosives. Because these amounts are several orders of magnitude smaller than the calculated amount of available material needed for such an explosion, the NRC staff concludes that an accidental explosion on the nearby highway or rail route would not generate a strong enough induced air overpressure to damage any ITS SSCs at the repository facilities. Additionally, the nearest road to the GROA is the Lathrop Wells Road, which is located further away than the 8.0-km [5.0-mi] distance used in the calculation.

In summary, based on the evaluation discussed above, the NRC staff concludes that (i) the applicant used appropriate regulatory guidance to estimate the induced air overpressure at the GROA from potential explosions at nearby industrial and military facilities and (ii) any explosions at the NNSS facilities or from nearby truck and rail traffic would not exceed an overpressure of 6.9 kPa [1 psi], and, therefore, would not damage ITS SSCs at the GROA. This is because the GROA is too distant from these facilities or transportation routes to be adversely impacted by postulated explosions. Therefore, the NRC staff concludes that DOE's technical basis for excluding explosions at the NNSS or along nearby transportation routes as initiating events in the PCSA is acceptable.

### Induced Seismic Motion

The applicant provided information on hazards from induced seismic motion from activities at the adjoining NNSS in SAR Sections 1.6.3.4.8 and 1.1.1.3, BSC (2008an), and DOE (2009fe). The applicant identified activities at several facilities at the NNSS that can generate ground motion from underground explosions. These underground explosions could arise from conventional weapons demilitarization or blasting at nearby mines, or from activities related to stockpile stewardship and the damaged nuclear weapons program. These explosions could be potentially hazardous to the repository facilities.

Stockpile management includes operations to store and maintain the nuclear weapons stockpile. Experiments and testing of nuclear devices were previously conducted in NNSS Areas 1 through 10 for continued stewardship of the nuclear weapons' stockpile, but the applicant stated that these activities are no longer authorized. If limited underground nuclear testing commences, Yucca Flat Area (Area 6) and Pahute Mesa Area (Areas 19 and 20) would likely be selected (BSC, 2008an). The applicant concluded that the ground motions at Yucca Mountain from nuclear tests would be bounded by moderate to large earthquakes in the region (magnitudes of 6.5–7.5), which will represent the controlling ground motions for the Yucca Mountain site, on the basis of the 14-year test data (Walck, 1996aa). The applicant indicated that the ground motions measured at rock and soil sites near the repository and at the NNSS from a nearby moderate to large earthquake had larger amplitudes than the underground nuclear explosions. Additionally, secondary seismic effects, associated with coseismic release of strain, aftershocks, and cavity collapse, are not significant at distances beyond about 10 km [6 mi] from the explosion, even from the largest of the underground nuclear tests (BSC, 2008an). The applicant also stated that activities of the damaged nuclear weapons program at the rehabilitated G-tunnel in Area 12, approximately 40 km [25 mi] from the repository, would also not affect the repository facilities, because the explosion-generated ground motions would be bounded by the design earthquake ground motions. Similarly, the applicant stated that destruction of obsolete conventional munitions, pyrotechnics, and solid rocket motors at the X-tunnel, approximately 16 km [10 mi] from the repository, and at the Nonproliferation Tests and Evaluation Complex in Area 5, approximately 40 km [25 mi] from the repository, would not impact the repository, because their equivalent explosive quantities will be significantly smaller than those of a nuclear blast.

### NRC Staff's Evaluation

The NRC staff reviewed the induced seismic motion information provided in SAR Sections 1.1.1.3 and 1.6.3.4.8, BSC (2008an), and responses to RAIs (DOE, 2009fe). Additionally, the NRC staff reviewed the analysis of induced seismicity provided in Walck (1996aa). Specifically, the NRC staff evaluated the description, quantity, and distance of the potential nuclear explosions and mine blasting to the repository facilities to assess the potential impacts that induced ground motion may have on the repository. Induced seismic motion at the repository may have damaging effects similar to earthquake-induced seismic motion.

The NRC staff finds that ground motions generated by underground nuclear blasts from 1977 through 1990 at the NNSS, measured at stations of rock and soil near the repository site, yielded ground motions with smaller amplitudes than those from an equivalent earthquake. Therefore, the NRC staff finds the applicant's assessment that potential effects on the repository from induced seismic motion from nuclear blasts at the NNSS would be bounded by effects from moderate to large earthquakes in the region acceptable.

As observed in underground nuclear tests, secondary seismic effects were not significant at distances exceeding 10 km [6 mi] (DOE, 1996ab). The repository facilities will be more than 24 km [15 mi] away from any potential test areas (the closest area being Area 6). On this basis, the NRC staff finds acceptable the applicant's conclusion that secondary seismic effects from future underground nuclear tests would not be credible hazards to the repository. Similarly, the NRC staff finds that activities associated with damaged nuclear weapons at G-tunnel would not pose a hazard to the repository facilities, because it is located 40 km [25 mi] away. The NRC staff finds that the ground motions that obsolete munitions, pyrotechnics, and solid rocket motors can generate would also be bounded by the ground motion from earthquakes, as the explosive amount involved would be significantly less than an underground nuclear blast. On the basis of this evaluation, the NRC staff finds that the applicant's assessment that induced seismic motions from underground explosions of nuclear and conventional explosives will not initiate an event sequence at the repository is acceptable.

Based on these findings, the NRC staff concludes that (i) the estimated ground motion at the GROA from underground nuclear explosions would be bounded by the design earthquake ground motion expected at the repository site and (ii) the repository facilities are a sufficient distance away from the nuclear explosion sites that they would not be affected by the secondary seismic effects from such blasts. Therefore, the NRC staff finds that DOE's technical basis for exclusion of induced seismic effects from the underground mining or nuclear blasts as an initiating event in the PCSA is acceptable.

**Release of Radiological Materials and Toxic Chemicals from Nearby Facilities**

The applicant provided information on hazards from released radiological materials and toxic chemicals from the NNSS facilities in SAR Sections 1.6.3.4.8 and 1.1.1.3, BSC (2008an), and DOE (2009fe). The applicant identified several facilities at the NNSS that use or will use radiological materials or toxic chemicals (including biological stimulants) based on the description and activities conducted therein. These facilities included the Joint Actinide Shock Physics Experimental Research (JASPER) Facility, the Criticality Experiments Facility, the Radiological/Nuclear Countermeasures Test and Evaluation Complex, the Nonproliferation Test and Evaluation Complex, and the Storage and Disposal of Weapons–Usable Fissile Materials (SAR Section 1.6.3.4.8; BSC, 2008an). The applicant evaluated whether these activities could result in a radiological or chemical release that could impact the GROA.

In the JASPER Facility, located approximately 32 km [20 mi] from the repository, a gas gun is used to shoot projectiles at radiological target materials in shock physics experiments. In BSC (2008an) and DOE (2009fe), the applicant concluded, using an analysis conducted by Lawrence Livermore National Laboratory, that the worst consequences to the environment from these experiments would be minor local contamination from radioactive materials and; therefore, there is no adverse consequence to the repository.

The applicant identified that nuclear criticality activities currently performed at Technical Area 18 of the Los Alamos National Laboratory in New Mexico would be relocated to the western section of the Device Assembly Facility in Area 6. This section is designated the Criticality Experiments Facility (BSC, 2008an). On the basis of the final environmental impact statement for this relocation, the applicant stated that noninvolved workers (those not directly involved with the handling of radioactive materials) at Technical Area 18 would receive a minimal radiation dose from an accident at that facility. The distance between the GROA and Technical Area 18 in Area 6 is more than 32 km [20 mi], which, according to the applicant, is a sufficient distance away that radiation from an accident would not affect the GROA. Consequently, the applicant concluded that an accident at Technical Area 18 would not be a hazard to the repository facilities (BSC, 2008an).

The Radiological/Nuclear Countermeasures Test and Evaluation Complex is being constructed approximately 32 km [20 mi] away from the repository to conduct activities related to combating terrorism. The applicant stated that this facility, classified as a Hazard Category 2 nuclear facility (potential for onsite consequences), could use up to 50 kg [110 lb] of highly enriched uranium and other special solid nuclear materials. All radioactive materials would either be sealed or encased in metal cladding. The applicant stated that the activities at this complex would not release any radioactive materials (BSC, 2008an) and, consequently, will not be a hazard to the repository operations.

The Nonproliferation Test and Evaluation Complex in Area 5 of the NTS, approximately 40 km [25 mi] from the repository, tests large- and small-scale release of hazardous and toxic materials and biological simulants in a controlled environment. Most tests are conducted when the wind is blowing away from the repository site (BSC, 2008an). On the basis of the distance from the repository, the applicant (BSC, 2008an) concluded that there would not be any impact on the repository and its operations.

DOE stated that two proposed options for storage of fissile materials from dismantling of nuclear weapons in the NNSS have been investigated: (i) construction for a new storage facility near the Device Assembly Facility or (ii) utilization of one of the horizontal event tunnels (BSC, 2008an). The applicant stated that storage activities of the fissile materials would not impact the repository, because of the relatively large distances between these sites and the GROA.

**NRC Staff's Evaluation**

The NRC staff reviewed the information regarding potential radiological material and toxic chemical releases from the NNSS facilities provided in SAR Sections 1.1.1.3 and 1.6.3.4.8, BSC (2008an), and responses to RAIs (DOE, 2009fe). The NRC staff also reviewed environmental impact statements or environmental assessments that BSC (2008an) referred to in its assessment. Specifically, the NRC staff evaluated the type, quantity, and distance to the repository from each NNSS facility that handles these materials to assess the potential for hazards to the operations at GROA facilities during the preclosure period.

In the JASPER Facility, radionuclides are used as the target materials in the experiment. The NRC staff finds that the Lawrence Livermore National Laboratory study (as reported in DOE, 2002ab) showed that the risk to the public from an accident at this facility would be negligible and the worst possible consequence would be minor local {within 30 m [100 ft]} contamination. The NRC staff, therefore, finds the applicant's assessment that the dose to a repository worker would be negligible from an accident at the JASPER Facility is acceptable because the distance to the GROA is 32 km [20 mi] and, consequently, can be excluded from the list of initiating events that may result in radiological releases.

The NRC staff reviewed the final environmental impact statement for relocating the activities in Technical Area 18 of the Los Alamos National Laboratory to the Criticality Experiments Facility at Area 6 in the NNSS (DOE, 2002ac). The NRC staff finds that the highest risk of a latent cancer fatality of a noninvolved worker at a distance of 100 m [330 ft] would be on the order of $10^{-9}$ per year. Therefore, the NRC staff finds that the applicant's assessment that an accident at this facility (when it begins operating in the future) would not initiate an event sequence at the repository is acceptable because of the distance from Area 6 to the GROA and the small probability of latent cancer risk and, consequently, can be excluded from the list of initiating events that may result in radiological releases.

The NRC staff reviewed DOE (2004ac) for the Radiological/Nuclear Countermeasures Test and Evaluation Complex and finds that this facility is classified as a Hazard Category 2 nuclear facility. Therefore, the hazard analysis of this facility showed that any potential consequence of unmitigated releases of radioactive and chemical materials would be limited to onsite only (DOE, 1992aa). Therefore, the NRC staff finds that the applicant’s assessment regarding a potential accident at this facility would not pose a hazard to the repository facilities or initiate an event sequence is acceptable because this facility is 32 km [20 mi] away from the repository and, consequently, can be excluded from the list of initiating events that may result in radiological releases.

The NRC staff reviewed the final environmental assessment for release of biological simulants and chemicals at the NNSS (DOE, 2004ad). The biological simulants mimic the behavior and/or other identifiable characteristics of the agents used in biological weapons, but not the severe adverse health effects associated with higher risk biological agents. The NRC staff notes that release of low concentrations of chemicals and biological simulants is permitted in Area 5 and other areas of the NNSS (DOE, 2004ad; BSC, 2008an). Based on its evaluation of this information, the NRC staff finds in DOE (2004ad) that the released materials were not detectable beyond the NNSS boundaries and did not affect the involved and noninvolved workers or members of the public. The NRC staff, therefore, concludes that released biological simulants and chemicals would not affect the repository workers because of the large distance and atmospheric dispersion and, consequently, can be excluded from the list of initiating events that may result in radiological releases.

The NRC staff reviewed information regarding Storage and Disposal of Weapons–Usable Fissile Materials at the NNSS (BSC, 2008an). The NRC staff also evaluated the information in DOE (1996ab) regarding the sites under consideration for long-term storage of fissile materials from weapons as a part of the nation’s nuclear weapons dismantling processes. Two potential sites within the NNSS for storage of the fissile materials are the Device Assembly Facility, approximately 32 km [20 mi] from the repository site, or one of the horizontal event tunnels. The NRC staff finds that any seismic ground motions that may generate from an accidental explosion of stored fissile materials will be bounded by the induced seismic motion from activities under the nation’s stockpile management program of nuclear weapons. The NRC staff, therefore, finds the applicant’s assessment that activities associated with storage of fissile materials from dismantled weapons, if the facility becomes operational in the future, would not impact the repository during the preclosure period is acceptable because the distance to the GROA is at least 32 km [20 mi] and, consequently, can be excluded from the list of initiating events that may result in radiological releases.

In summary, based on these NRC staff findings, the NRC staff concludes that a worker at the repository facilities would receive negligible dose consequences from potential radioactive and toxic chemical releases at NNSS facilities due to the large distance between the GROA and nearby facilities that may release radioactive and toxic chemicals. Therefore, the NRC staff concludes that DOE’s technical basis to exclude release of radiological materials and toxic chemicals from nearby facilities as an initiating event in the PCSA is adequate.

**Waste Management Programs**

The applicant provided information on hazards from waste management programs at the NNSS facilities in SAR Sections 1.1.1.3 and 1.6.3.4.8, and BSC (2008an). The primary mission of the waste management programs is to dispose of low-level radioactive waste (LLW) generated at the NNSS and from other DOE-approved waste generators (BSC, 2008an).

According to the applicant, NNSS Areas 3, 5, and 6 are at least 32 km [20 mi] from the repository. The LLW is disposed of in seven subsidence craters generated from underground nuclear tests in Area 3 and buried in shallow pits and trenches in Area 5. Low-level and mixed waste effluent, generated at the Nevada Environmental Management and Defense program, is treated at the Liquid Waste Treatment System facilities in Area 6.

**NRC Staff's Evaluation**

The NRC staff reviewed the information in SAR Sections 1.1.1.3 and 1.6.3.4.8, and BSC (2008an) regarding waste management programs at the NNSS facilities. The NRC staff finds that the LLW is disposed of in subsidence craters in Area 3 and shallow pits and trenches in Area 5. As the distance to both sites from the repository exceeds 40 km [25 mi], the NRC staff finds that LLW disposal would not pose a credible hazard to the repository facilities. At the Liquid Waste Treatment System facilities in Area 6, the waste is stored in double-walled steel tanks fitted with a leak detection system (DOE, 1996ab). The NRC staff finds that any leak at this facility would not pose a hazard to the repository and its operations, because of the distance between Area 6 and the GROA, which is at least 32 km [20 mi].

Based on the evaluation discussed above, the NRC staff concludes that a worker in the repository facilities would not be affected by LLW disposal in Areas 3, 5, and 6, because of large distances between the GROA and these NNSS areas. Therefore, the NRC staff finds DOE's technical basis adequate to exclude LLW disposal at the NNSS as a hazard from the PCSA.

**Mining**

The applicant provided information on hazards from mining-related activities near the repository facilities in SAR Sections 1.6.3.4.8 and 1.1.1.3, and BSC (2008an). According to the license application, there were no mining claims in the repository and Public Land Orders precluded mining claims in the controlled area. Although there were unpatented mining claims at the southern edge of the proposed land withdrawal area, they were outside the 8-km [5-mi] zone. Trucks from the IMV Nevada Mine, located beyond the 8-km [5-mi] zone, use U.S. Highway 95 and State Highway 373, which are more than 16 km [10 mi] from the repository. The applicant stated that the Cind–R–Lite Company owns approximately 4,047 $m^2$ [200 acres] within the proposed land withdrawal area and extracts materials from the cinder cone to manufacture lightweight concrete blocks. This operation is approximately 11 km [7 mi] from the repository. The applicant stated that there are no sand or gravel quarrying operations within an 8-km [5-mi] radius of the repository, and any activities that may cause significant impact on the repository will not be permitted (BSC, 2008an). Therefore, the applicant concluded that even the nearest existing mining operation would not have any impact to the repository and its operation, because of the large distance from the mining site to the repository facilities. Furthermore, DOE stated that it does not expect new mining-related activities, because no significant sources of oil or gas have been found in southern Nevada or in adjacent areas of California and Arizona. According to the applicant, the potential for oil and natural gas deposits near Yucca Mountain is low (BSC, 2008an). Other energy sources, such as tar sand, oil shale, and coal, are not known to exist in the Yucca Mountain area. Potential uses of the GROA for activities other than repository operations are evaluated by the NRC staff in SER Section 2.5.9, and controls to restrict land use and access to the GROA are evaluated in SER Section 2.5.8.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Sections 1.1.1.3 and 1.6.3.4.8, and BSC (2008an) regarding the hazards from mining-related activities near the repository facilities. Specifically, the NRC staff evaluated the locations of the nearby mines and description of their activities to assess the potential hazards. Additionally, site characterization information regarding the existing mining operations is evaluated in SER Sections 2.1.1.1.3.1 and 2.1.1.1.3.9. The NRC staff finds that information about the locations of the nearby active mines the applicant used to assess mining-related hazards is acceptable because this information was derived from an authoritative document on mines in Nevada (Driesner and Coyner, 2006aa).

The NRC staff also evaluated the applicant's assessment that the nearby mining operations will not pose a hazard to the repository operations using guidance from NUREG–0800 (NRC, 2007aj, Sections 2.2.1 and 2.2.2). NUREG–0800 provides guidance for identification of potential hazards in the vicinity of a nuclear power plant. This is applicable to evaluation of hazards at the repository because the effects from mining on safe handling of waste at the GROA are analogous to the effects of waste handling on nuclear power plants. This guidance recommends that facilities and activities located within 8 km [5 mi] that could potentially pose a hazard should be evaluated, and facilities or activities at larger distances should only be considered if they have the potential for affecting facility safety. Because all mining activities described by the applicant are located farther than 8 km [5 mi] away from the repository, and the mining activities at this distance are not of a nature that would affect repository safety, the NRC staff finds that excluding mining operations as a hazard from the PCSA is acceptable. DOE also stated that activities that may cause significant impact on the repository will not be permitted and that it does not expect any new mining-related activities in the vicinity of Yucca Mountain. The NRC staff concludes that DOE's expectation about no new mining activities is reasonable because there are no known significant sources of oil or natural gas in the Yucca Mountain area.

Based on these findings, the NRC staff concludes that the applicant's assessment in which activities at nearby mines can be excluded as a hazard or initiating event in the PCSA is acceptable because existing mining activities are situated at a sufficient distance from the repository and new mining activities are not expected nor, according to DOE, will they be permitted.

**Release of Onsite Hazardous Materials**

The applicant assessed hazards to the repository facilities associated with potential onsite release of hazardous materials in SAR Section 1.6.3.4.9 and BSC (2008an, Section 6.11). The applicant conducted the screening analysis following Regulatory Guide 1.78 (NRC, 2001af) on hazards from release of hazardous materials at nearby facilities.

The applicant stated that chlorine and helium are the two chemicals of those listed in Regulatory Guide 1.78 (NRC, 2001af, Table 1) that will be stored onsite (SAR Section 1.6.3.4.9; BSC, 2008an). According to the applicant, chlorine tablets (in the form of solid calcium hypochlorite tablets) will be used for the water treatment system, and helium will be used as an inert gas in the waste containers. Additionally, argon, a potential asphyxiant, will be stored onsite (SAR Section 1.6.3.4.9). The applicant stated that helium and argon gases will be supplied to the repository surface facilities from gas bottles, storage tanks, or mobile tube trailers located outside the buildings. Any released gases would disperse into the atmosphere. Additionally, the applicant stated that solid chlorine does not pose a hazard to the facility personnel, as it cannot become airborne (SAR Section 1.6.3.4.9; BSC, 2008an). Any release of diesel fumes from the storage tanks will be localized. The applicant also stated that if any operation room needed to be abandoned because of inhabitable conditions from a release of chemicals, remote monitoring equipment installed at the repository facilities would continue to monitor the safety-related functions. Consequently, the applicant concluded that an accidental release of hazardous materials would not affect the safety-related functions of the repository due to paucity of onsite hazardous chemical sources.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.6.3.4.9 and BSC (2008an, Section 6.11) on potential release of onsite hazardous materials. The NRC staff finds acceptable the applicant's screening analysis of potential onsite release of hazardous materials because the applicant relied on NRC guidance in Regulatory Guide 1.78 (NRC, 2001af) to identify the hazardous materials. The NRC staff finds that use of solid chlorine tablets for water treatment would not affect personnel at other locations, because the chlorine will not become airborne. The NRC staff also finds that the applicant's assessment that helium and argon gas releases would not pose a hazard to the surface facilities is acceptable because any releases of gaseous materials would be readily dispersed into the atmosphere. Both argon and helium are common industrial gases with well-known handling procedures. The NRC staff also finds that any diesel fumes from a spill will not affect operations at other locations, because any such releases would be localized. In addition, remote monitoring of safety-related functions would continue if any operation room had to be abandoned because of a chemical leak.

Based on these findings, the NRC staff concludes that DOE provided an adequate technical basis to exclude accidental release of hazardous materials affecting safety-related functions of the repository as a potential initiating event from the PCSA.

**Turbine-Generated Missiles**

The applicant assessed hazards to the repository facilities associated with potential turbine-generated missiles in SAR Table 1.6-8 and BSC (2008an). SAR Table 1.6-8 identified turbine-generated missiles as a potential hazard for the repository facilities. The applicant (BSC, 2008ai) stated that the hazard from turbine missiles is generally associated with large turbines in nuclear power plants. The applicant excluded this hazard as a potential initiator of event sequences (SAR Table 1.6-8; BSC, 2008an) because there are no nuclear or fossil power plants near the repository facilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Table 1.6-8 and BSC (2008an) on hazards related to turbine-generated missiles. The NRC staff finds that failure of the massive rotor of a turbine with high rotational speed may generate high-energy missiles that affect ITS SSCs. Additionally, the NRC staff independently verified, by reviewing maps of the area, that there are no other large power plants within 8 km [5 mi] of the repository that use large turbines. Therefore, the NRC staff finds that the applicant's exclusion of turbine missile hazards as a potential initiator of event sequences in the PCSA is acceptable because there are no nuclear or fossil power plants near the repository facilities.

2.1.1.3.3.1.3.5 Other Hazards

The NRC staff's review of the remaining external hazards and initiating events (SAR Table 1.6-8) is described using the following groups:

- External floods
- Loss of power
- Loss of cooling capability
- External fire
- Explosions
- Extraterrestrial activity
- Waste and rock interaction, geochemical alterations, and dissolution
- Perturbation of groundwater system
- Undetected past human intrusions
- Security-related hazards (namely, sabotage, terrorist attack, and war)

Improper design- or operation-related hazards deal with operational activities in the repository facilities and have been reviewed as a part of the review of internal hazards in SER Section 2.1.1.3.3.2.

**External Floods**

The applicant provided information on external flooding at the repository facilities in SAR Section 1.6.3.4.5, SAR Table 1.6-3, and BSC (2008ai, Section 6.5). In addition, the flood hazard curve developed for the GROA is described in BSC (2008cd). The applicant identified 15 events that may cause external floods at the repository site: dam failure, external flooding, extreme weather and climate fluctuations, high lake level, high tide, high river stage, hurricane, ice cover, rainstorm, river diversion, seiche, snow, storm surge, tsunami, and waves. Because no rivers or streams flow past the site, there are no upstream dams. Therefore, the applicant stated that dam failure, river diversion, flooding due to ice cover, and high river stage cannot occur at the GROA. The applicant stated that the repository is approximately 360 km [225 mi] from the nearest body of water large enough to support standing waves, and the mountainous terrain between the Pacific Coast and the Yucca Mountain region prevents flooding effects due to a hurricane, high tide, seiche, tsunami, aquatic waves, or storm surge from occurring at the GROA. The applicant stated that permanent reservoirs and lakes in the vicinity of the repository are Crystal Reservoir, Lower Crystal Marsh, Horseshoe Reservoir, and Peterson Reservoir. These are small, artificial impoundments located approximately 51 km [32 mi] south-southeast of Yucca Mountain and at a lower elevation than the GROA. Thus, the applicant determined that external flooding because of high lake level or dam failure cannot occur at the GROA.

The applicant further evaluated external flooding resulting from severe rainstorms that could occur at Yucca Mountain. The applicant evaluated rainstorm as a bounding case for potential flooding resulting from storm precipitation because potential flooding due to melted snow and ice was determined to be less severe and less frequent than from severe rainstorms. In the process of developing the flood hazard curve for the GROA, the applicant estimated a probable maximum precipitation using the annual precipitation values from NOAA. Additionally, the applicant estimated a probable maximum flood (PMF) for the GROA based on flood peak simulations using the HEC–1 hydrologic model (U.S. Army Corps of Engineers, 1985aa). According to the applicant, average annual precipitation at the NNSS is less than 81 cm [10 in]. The maximum daily precipitation within 50 km [31 mi] of Yucca Mountain was projected to be less than 13 cm [5 in]. The 6-hour probable maximum precipitation for the GROA was estimated at approximately 30 cm [12 in].

The applicant's flood hazard analysis estimated the million-year flood flow (i.e., with annual exceedance probability of $10^{-6}$) at 1,133 $m^3/s$ [40,000 $ft^3/s$], while the diversion channels, levees, and other flood protection features at the GROA will be designed with a capacity of up to 1,557 $m^3/s$ [55,000 $ft^3/s$] to divert flood flow. To ensure that these features maintain their intended functions, the applicant stated that it will implement a standard maintenance practice on the flood protection features. The applicant estimated Probable Maximum Flood (PMF) frequency as $1.1 \times 10^{-9}$ per year (BSC, 2008ai). The applicant, therefore, excluded external floods from further consideration because the PMF frequency was determined to be less than $10^{-6}$ per year.

**NRC Staff's Evaluation**

The NRC staff evaluated information the applicant provided in SAR Section 1.6.3.4.5, SAR Table 1.6-3, and BSC (2008ai, Section 6.5) on external flood hazards, and its reference describing the flood hazard curve (BSC, 2008cd). The NRC staff finds that the applicant provided appropriate site-specific data on rainfall, as evaluated by the NRC staff in Section 2.1.1.1.3.4. The NRC staff further finds that the applicant's flood hazard analysis and hydrologic model are acceptable because they are based on the HEC-1 model, which is evaluated and found acceptable in SER Section 2.1.1.1.3.4. Because the estimated PMF frequency is well below a $1 \times 10^{-6}$/year threshold probability, the NRC staff finds that DOE adequately excluded external floods as initiating events in the PCSA. In addition, the NRC staff notes that the estimated million-year flood flow (i.e., with annual exceedance probability of $10^{-6}$) of 1,133 $m^3/s$ [40,000 $ft^3/s$] is below the applicant's diversion channel design capacity, as evaluated in SER Section 2.1.1.7.3.1.3, and therefore, the NRC staff finds that the diversion channels have the capacity to accommodate a one million-year flood. Additional details of the NRC staff's review of the flood control features at the GROA provided by the applicant are in SER Section 2.1.1.7.3.1.3.

Based on the evaluation discussed above, the NRC staff finds that the exclusion of the external flood hazard from the PCSA is acceptable because (i) methods selected for determining probability or frequency of occurrence of the hazard are appropriate and (ii) the technical bases are consistent with the site information, flood management design, and standard industry practice.

**Loss of Power**

The applicant provided information on loss of power to the repository facilities in SAR Section 1.6.3.4, SAR Table 1.6-8, and BSC (2008ai, Section 6.7). The applicant identified loss of electrical power to be an initiating event in the repository facilities. In addition to grid failure, several natural hazards were identified that could cause loss of offsite and/or onsite power: extreme weather and climate fluctuations, frost, hail, and sand or dust storms.

The applicant (BSC, 2008ac,as,be,bk,bq) estimated the frequency of a loss of electrical power event occurring at the Yucca Mountain facilities to be $3.6 \times 10^{-2}$ per year using the estimated mean frequency for the entire United States from 1986 through 2004 given in NUREG/CR–6890 (Eide, et al., 2005aa). This estimated frequency included plant, switchyard, grid, and weather-related information. The applicant estimated the annual frequency of the initiating event due to a loss of electrical power lasting more than 24 hours to be $3.2 \times 10^{-2}$ per year (BSC, 2008ac,as,be,bk,bq), assuming that the waste handling operations would continue for the first 50 years of the preclosure period. The applicant concluded that a loss of external power event is expected to be a normal occurrence during the 50-year period of operations because the annual probability is more than $10^{-6}$ (Category 1 event) and assessed the hazard for the potential to cause a radiological release in the GROA facilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.6.3.4, SAR Table 1.6 8, and BSC (2008ai, Section 6.7) on loss of power to confirm that the applicant used appropriate site-specific information and analyses to include loss of power as an initiating event. The applicant's use of NUREG/CR–6890 (Eide, et al., 2005aa) to estimate the frequency of loss of electric power and likelihood of a loss of power event lasting for more than 24 hours is acceptable because this NUREG/CR incorporates 19 years (1986 through 2004) of loss of offsite electric power event data from U.S. nuclear power plants. The data in NUREG/CR–6890 (Eide, et al., 2005aa) are applicable to the repository because loss of offsite power is an external event that would affect nuclear power plants or repository facilities similarly. As identified in NUREG/CR–6890 (Eide, et al., 2005aa), there are significant geographical differences in grid-related outage events among different areas of the country. For the period of study, the western region, in which Yucca Mountain is located, showed a grid-related outage performance frequency of $4.18 \times 10^{-2}$ per year. This frequency is more than double the national mean outage frequency of $1.86 \times 10^{-2}$ per year. However, use of the frequency of loss of power for the western region instead of the entire country, resulting in an extended (720 hr) power loss event, does not change the frequency-based categorization (Category 1 or 2) of the load drop due to brake failure event sequences, for example, in the CRCF (BSC, 2008ac). Therefore, the use of the western region outage performance frequency value in the PCSA is acceptable. The NRC staff finds that the inclusion of the loss of power hazard as an initiating event in the PCSA is acceptable because the occurrence of a loss of power event has a greater than 1 in 10,000 chance of occurring before permanent closure. In addition, this initiating event can be severe enough to affect the safety and operations of the repository. Further, the NRC staff finds that methods the applicant selected for determining frequency of occurrence of the loss of power are appropriate because they are based on historical data, as documented in NUREG/CR–6890 (Eide, et al., 2005aa). Therefore, the NRC staff finds that the technical bases for inclusion of the loss of power hazard are appropriate and consistent with site information.

**Loss of Cooling Capability**

The applicant provided information on loss of cooling capability to the repository waste handling facilities in SAR Section 1.6.3.4.7 and BSC (2008ai, Section 6.8). Water supply at the repository facilities may be disrupted due to the following events: dam failure; extreme weather and climate fluctuations, including drought, high summer temperature, and low winter temperatures; presence of fungus, bacteria, and algae; ice cover; low lake level; low river level; river diversion; and sandstorm (BSC, 2008ai). The applicant stated that three underground wells will supply water to the repository through a 3,217,600-L [850,000-gal] storage tank. The storage tank will feed the delivery systems for deionized water for the fuel-handling pool, fire suppression systems, potable water supplies, and HVAC cooling towers. Because the water supply for the Yucca Mountain repository will be obtained from groundwater sources, the applicant concluded that dam failure, ice cover, low lake level, low river level, and river diversion would not result in loss of cooling capability at the repository facility (BSC, 2008ai). Additionally, the applicant stated that sandstorm or dust storms would not cause blockage to the water supply infrastructure, because they are located either underground or covered. The applicant further concluded that hazards that could affect the water supply at the GROA included (i) climate fluctuations and droughts severe enough to disrupt groundwater sources; (ii) extreme weather, especially freezing temperatures; and (iii) bacteria or algae growth that could reduce or block water flow (BSC, 2008ai).

Analyses of pool operations (BSC, 2008cn) indicated that, without makeup water, it would take at least 180 days to evaporate enough water from the WHF pool to compromise radiation-protection shielding. Also, the applicant stated that waste forms do not exceed temperature limits for 30 days after the loss of HVAC cooling (BSC, 2007dd). Therefore, the applicant concluded that, if water supply is disrupted due to a pipe freezing and rupturing, there would be sufficient time to arrange for alternate sources of cooling water. Consequently, the applicant excluded the loss of cooling capability as a potential initiator of event sequences (SAR Section 1.6.3.4.7; BSC, 2008ai).

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.6.3.4.7 and BSC (2008ai, Section 6.8) on loss of cooling capability to the repository waste-handling facilities due to disruption of the water supply to the WHF pool and HVAC system as a result of dam failure, ice cover, low lake or river level, and river diversion. The NRC staff finds that these events do not pose credible hazards, because cooling will be supplied from underground wells. Consequently, extreme weather, especially freezing temperatures, will have negligible effects. Because change in groundwater supply occurs gradually, sufficient time would be available to identify alternate water sources. In addition, reduced or complete blockage of water flow due to bacteria or algae growth will occur gradually and would be detected. The NRC staff also finds that sufficient time would be available for any remedial actions or, in an extreme case, identifying alternate water sources. This is consistent with industry standard ASME/ANS RA–S–2008 (ASME, 2008aa), which includes a criterion to justify exclusion of events if they are slow in developing and there is sufficient time to provide an adequate response to address the slowly developing events. Furthermore, the NRC staff finds that the applicant's respective 30- and 180-day estimates for how long the WHF pool and HVAC systems can operate without makeup water before compromising safety functions are acceptable because the estimates are based on appropriate facility descriptions, and the assumptions and input data for these types of facilities are well established, as they are from other types of NRC licensed facilities. These time estimates are also consistent with the NRC staff's finding that sufficient time would be available, even in an extreme case, to restore the water supply before safety functions would be compromised. Based on these findings, the NRC staff finds that the applicant's exclusion of loss of cooling capability as an initiating event in the PCSA is acceptable because DOE provided adequate technical bases that are consistent with site information and industry practice.

**External Fire**

The applicant provided information on external fires at the repository facilities in SAR Section 1.6.3.4.10. BSC (2008ai, Section 6.12) provided additional information on the estimated annual frequency of ignition. In its response to the NRC staff's RAI (DOE, 2009fa), the applicant summarized an analysis to establish the firebreak width necessary to keep the ITS SSCs safe from potential wildfires.

The applicant indicated that the U.S. Forest Service collected information on wildfires from 1970 through 2000 on the basis of Bailey ecoregion divisions (BSC, 2008ai). The repository site belongs to the temperate desert or tropical/subtropical desert division. The U.S. Forest Service database had 2,391 fires in the 30-year period on 39,210 $km^2$ [15,139 $mi^2$] of U.S. Forest Service land in the temperate desert division. Assuming a uniform density, this translates to $5.5 \times 10^{-3}$ fires/yr in the GROA with a surface area of 2.7 $km^2$ [1.0 $mi^2$] (BSC, 2008ai). The applicant (SAR Section 1.6.3.4.10; BSC, 2008ai, Section 6.12) proposed a separation distance of 10 m [33 ft] that would be maintained vegetation-free between fuel sources (brush and vegetation) and the structures, as recommended in NFPA 1144 (NFPA, 2008ab). The applicant indicated that it would use administrative controls as a part of the Fire Protection Program (SAR Section 1.4.3.5 and Table 5.10-3) to maintain this noncombustible buffer zone.

The applicant calculated the heat release from a postulated fire located near the corner of an aging pad (DOE, 2009fa) using methods outlined in the Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering (SFPE, 1995aa). The applicant stated that the corner of an aging pad was selected because this configuration will produce exposure from two directions and will produce a conservative incident heat flux profile. The estimated radiative heat flux was 0.89 $kW/m^2$ [0.078 $Btu/ft^2$-sec]. Radiant heat flux to planar structures, such as buildings (exposure to one side only), would be lower. Because the minimum critical heat flux needed to ignite certain types of paper and wood products is 10 $kW/m^2$ [0.88 $Btu/ft^2$-sec], the applicant concluded that the noncombustible aging overpacks and waste handling facilities would not sustain any damage from the postulated fire when separated with a 10-m [33-ft] vegetation-free buffer zone. The applicant further concluded that the buffer zone width would be sufficient so that even structures in the most vulnerable locations (e.g., loaded aging casks on the corner of an aging pad) would not sustain significant damage from these vegetation fires and would be capable of maintaining their intended safety functions. On the basis of these results, the applicant excluded external fire as a potential initiator of event sequences in the repository facilities because the ITS SSCs would be surrounded by a vegetation-free buffer zone that provides protection from approaching wildfires, and the ITS SSCs have sufficient capacity to resist the effects of these fires.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Section 1.6.3.4.10, references therein, and response to the NRC staff's RAI (DOE, 2009fa). Specifically, the NRC staff evaluated the description of potential vegetation characteristics near the repository site and the screening analysis for initiating events arising from fires (vegetation or wildfire) that might originate outside the GROA.

The NRC staff finds that the applicant's use of U.S. Forest Service data on wildfires, classified by ecoregion, is acceptable to estimate the annual frequency of wildfire in the Yucca Mountain region because these data are site specific and from a reliable source. The NRC staff also finds that the applicant's methodology using the U.S. Forest Service data (BSC, 2008ai) is acceptable because it is a common methodology used by other government agencies for wildfire hazard management, climate studies, and research purposes.

The NRC staff independently estimated the annual frequency of wildfires on the basis of the Bailey tropical/subtropical desert division, which translates to $1.61 \times 10^{-4}$ fires per year in the

GROA {2,379 fires in 30 years in 13,306 $km^2$ [5,137 $mi^2$] of U.S. Forest Service land in the tropical/subtropical desert division translate to 2.7 $km^2$ [1.0 $mi^2$] of GROA land}. Although the estimated frequency based on the repository site falling under the tropical/subtropical desert division results in a higher frequency than the frequency provided by the applicant (based on temperate desert division classification), both estimates put the annual frequency of wildfire occurrence into a Category 2 initiating event. The NRC staff finds that the applicant's approach of estimating incident heat flux to an aging overpack located at the corner of the aging pad is an acceptable methodology for the site and a conservative methodology for estimating exposure to waste handling facilities because an aging cask on an aging pad will be more vulnerable to wildfire than the waste handling facilities at the repository, given the location and exposure configuration.

The NRC staff also finds the applicant's conclusion that vegetation at the repository site represents a light fire load is acceptable because the biomass of living and dead vegetation around the aging pad area is 0.2 $kg/m^2$ [0.04 $lb/ft^2$], substantially lower than the limit for the "light" fuel load of 0–34 $kg/m^2$ [0–7 $lb/ft^2$], defined in NFPA 80A (NFPA, 2007ag, Section 4.3.5.2). The fire protection industry-standard methodology (SFPE, 1995aa) was used in the applicant's analysis and the NRC staff finds the applicant's approach to be acceptable. The NRC staff finds that the calculation the applicant provided made several conservative assumptions. These conservative assumptions included (i) use of a radiative fraction of 40 percent of the total heat released by a vegetation fire as compared to 20 to 40 percent, per the Fire Protection Handbook (NFPA, 2003ac) and (ii) use of configurations with casks located on a corner of an aging pad, where the cask is expected to receive heat flux from two directions, to estimate incident radiative heat flux. Therefore, the NRC staff finds that the applicant appropriately analyzed the potential for external wildfire sources within the surface facilities area.

The NRC staff also reviewed the cask fire fragility analysis presented in BSC (2008ac, Attachment D, Section D2) with respect to the potential heat exposures from external fires. The NRC staff finds acceptable the applicant's conclusion that a higher intensity fire exposure with a substantially longer duration than specified in the SFPE Handbook (1995aa) would be required before the shielding material on an aging overpack is compromised. This is because more than 34.5 cm [13.6 in] of concrete overpack would have to spall before firefighters or other personnel would be exposed to radiation (BSC, 2008ac, Section D2.2.3.1).

Based on the evaluation discussed above, the NRC staff finds that the applicant's exclusion of the external fire hazard as a potential initiator of event sequences in the repository facilities is acceptable because (i) a vegetation-free buffer zone will be established to protect ITS SSCs from wildfires; (ii) this buffer zone is sufficiently large such that the estimated radiative heat flux {0.89 $kW/m^2$ [0.078 $Btu/ft^2$-sec]} is substantially smaller than the minimum heat flux {10 $kW/m^2$ [0.88 $Btu/ft^2$-sec]} (SFPE, 1995aa) necessary to ignite certain types of paper and wood products; and (iii) a higher intensity fire exposure with a substantially longer duration would be required to damage an aging overpack to a point where the damage would cause radiation exposure to workers and firefighters.

### Explosions

The applicant provided information on explosion hazards to the repository facilities in SAR Section 1.7.1.2.2 and BSC (2008au, Section 6.0.5). The applicant stated that Area 70A will have a diesel oil storage tank capacity of 454,250 L [120,000 gal]. This tank will be supplied by a 37,850-L [10,000-gal] tanker truck. As described in BSC (2008au, Section 6.0.5), the applicant analyzed the air overpressure generated by an accidental explosion of either the storage tank or the tanker truck, following Regulatory Guide 1.91 (NRC, 1978ac). Regulatory Guide 1.91 states that an air overpressure of 6.9 kPa [1 psi] or less would not have an adverse effect on ITS SSCs. In addition, the NRC-approved transportation casks are designed to withstand an external air overpressure of 140 kPa [20 psi], consistent with NUREG–1617 (NRC, 2000aj, Section 2.5.5.4) and as per 10 CFR 71.71(c)(4). Therefore, the applicant evaluated whether the waste handling facilities and transportation casks would be subjected to air overpressures larger than 6.9 kPa [1 psi] and 140 kPa [20 psi], respectively, as a result of an explosion of either the storage tank or the tanker truck.

The postulated event in both the storage tank and tanker truck is vapor-cloud explosion. As documented in BSC (2008au), the applicant analyzed the effects of vapor-cloud explosions by converting the diesel fuel to equivalent trinitrotoluene (TNT) mass and assuming the entire volume of available diesel fuel participated in the explosion process. According to the applicant's calculation, the Area 70A storage tank would develop an overpressure of 6.9 kPa [1 psi] at a distance between 51 and 164 m [168 and 539 ft]. The tanker truck would develop a 6.9 kPa [1 psi] overpressure at a distance between 23 and 72 m [74 and 237 ft]. The applicant concluded that a diesel storage tank explosion in Area 70A would not cause any adverse effect in the waste handling facilities, because the distance between the Area 70A storage tank and the waste handling facilities would exceed 164 m [539 ft] (BSC, 2008au). Similarly, the applicant concluded that an explosion of the tanker truck would not cause any adverse effects on the waste handling buildings due to the distance between the tanker truck and the buildings.

The applicant performed a separate analysis to determine the potential impact of air overpressure on nearby transportation casks. The distance between the routes the tanker truck would use to reach the Area 70A storage tank would be more than 46 m [150 ft] from the nearest transportation cask. Because an overpressure of 140 kPa [20 psi] caused by an accidental explosion of the tanker truck is not expected to propagate beyond 46 m [151 ft], the applicant concluded that the transportation cask would not suffer any adverse effects from a tanker vapor-cloud explosion (BSC, 2008au). Therefore, the applicant excluded explosion hazards as a potential initiator of event sequences for repository facilities and transportation casks in the PCSA because the repository facilities and transportation casks will be located sufficiently far away from the explosion sources to incur any explosion-related damage.

Additionally, BSC (2008au, Table 6.0-2) excluded any damage to the site transporter, cask tractor, cask transfer trailer, or site prime mover from the onboard fuel tank explosions because the fuel tanks will be made of low-melting-temperature materials. As a result, the fuel tanks are unlikely to contain a vapor overpressure during a heating event because their construction would preclude conditions resulting in an explosion. Design of the site transporter, cask tractor, cask transfer trailer, and site prime mover has been reviewed in SER Section 2.1.1.7.3.5.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's assessment of potential explosion hazards in SAR Section 1.7.1.2.2 and BSC (2008au, Section 6.0.5), with the focus on the description of explosion sources and the associated analysis supporting the exclusion of initiating events arising from explosions. The NRC staff finds that the applicant's methodology to estimate the safe standoff distance from an explosion at the storage tank in Area 70A and the refueling tanker truck along its course of travel within the site is acceptable because the applicant followed applicable guidance in Regulatory Guide 1.91 (NRC, 1978ac).

The NRC staff finds that the applicant's exclusion of explosion events at the storage tank site or along the refueling tanker route as initiating events is acceptable because the standoff distances between facilities or transportation casks of concern and potential explosion sources are sufficiently large enough so that any potential air overpressure from explosions would dissipate before reaching facilities or transportation casks. The NRC staff also finds that the applicant's selection of low-melting-temperature materials to construct fuel tanks in cask tractor, cask transfer trailer, or site prime mover acceptable because fuel tanks made of low-melting-temperature materials will not support an overpressure leading to an explosion. SER Section 2.1.1.7.3.5 contains information regarding the NRC staff's review of the design of the site transporter, cask tractor, cask transfer trailer, and site prime mover. Therefore, the NRC staff finds that the applicant's exclusion of explosion hazards as a potential initiator of event sequences in the PCSA is acceptable because (i) acceptable methodology has been used to assess the safe standoff distances from an explosion, (ii) appropriate facility-specific information was used to assess the potential for an initiating event, (iii) separation distances between the ITS structures and systems and the potential explosion sources are always larger than the safe distances necessary to withstand the air overpressure generated from the explosion, and (iv) technical bases provided to exclude explosion as a potential hazard are adequate and consistent with facility information.

**Extraterrestrial Activity**

The applicant provided information on extraterrestrial activity at the repository site in SAR Section 1.6.3.4.11 and SAR Table 1.6-8. The applicant also discussed the potential impact on the repository facilities by the extraterrestrial objects in BSC (2008ai, Section 6.13). An asteroid is an extraterrestrial object with a size greater than 50 m [164 ft] (BSC, 2008ai) and can cause significant damage; however, the applicant indicated that the frequency of asteroid impacts is relatively small. The return periods for smaller asteroids are hundreds to thousands of years (BSC, 2008ai); therefore, the applicant stated that asteroids will not be credible hazards during the preclosure period. Comets are small objects orbiting the Sun. The nucleus of a comet is a loose collection of ice, dust, and small rock particles. If a comet enters the Earth's atmosphere, it would break up at higher altitudes due to lower density and unconsolidated composition (BSC, 2008ai). A meteorite is an object originating in outer space that survives travel through the Earth's atmosphere and impacts the Earth's surface (BSC, 2008ai). The applicant assumed that meteorites are less than 50 m [164 ft] in diameter and categorized meteorites into three classes, based on their composition, to assess the hazards: (i) iron meteorites, about 5 percent of the total meteorites found; (ii) hard stone meteorites, about 4 to 18 percent of the total meteorites found, on the basis of their initial mass (which is related to the size); and (iii) soft stone and ice meteorites for the remaining population (BSC, 2008ai).

The Earth's atmosphere acts as a shield against meteorites. Most meteorites disintegrate while descending through the Earth's atmosphere due to frictional heating. Both iron and hard stone meteorites smaller than approximately 10 kg [22 lb] tend to burn up in the atmosphere and will not impact the Earth's surface. Iron meteorites smaller than 100,000 kg [110 T] may impact the Earth; those larger than 100,000 kg [110 T] tend to break up in the atmosphere. Hard stone meteorites with masses greater than 10 to 1 million kg [0.01 to 1,102 T] tend to fragment in the atmosphere. Soft rock and ice meteorites would burn up or disintegrate at even higher altitudes than iron and hard stone meteorites (BSC, 2008ai). Although larger stone and iron meteorites may break up upon entering the Earth's atmosphere, the resulting fragments may have sufficient velocity to cause significant damage. The applicant, therefore, considered the iron and hard stone meteorites within the size range of 10 to 1,000 kg [22 to 2,204 lb] for potential impact to the repository (BSC, 2008ai).

Using information on the number of meteorites impacting the Earth's surface from Bland and Artemleva (2006aa, Table 2), the applicant estimated that iron meteorites in the range of 10 to 1,000 kg [22 to 2,204 lb] will have a GROA impact frequency of $1.8 \times 10^{-7}$ to $5.8 \times 10^{-10}$ per year (BSC, 2008ai). Hard stone meteorites will impact the GROA at a frequency varying between $6.4 \times 10^{-7}$ and $1.2 \times 10^{-9}$ per year for the same mass range. Meteorites with larger mass would have a lower annual probability of striking the GROA. On the basis of the estimated annual frequency, the applicant concluded that meteorite strike would not initiate event sequences at the repository during the preclosure period (SAR Section 1.6.3.4.11; BSC, 2008ai).

Additionally, approximately 17,000 tracked space objects (man-made objects) re-entered the Earth's atmosphere from 1957 through 1999 (BSC, 2008ai). Most of these objects burned up completely before reaching the Earth's surface; however, a small portion of them may reach the Earth's surface and cause damage. The applicant (BSC, 2008ai) estimated that about one object reenters the atmosphere every day and one to two objects with a 1-$m^2$ [11-$ft^2$] radar cross section reenter the atmosphere each week. The applicant assumed, for conservatism, 4 objects with radar cross sections exceeding 1 $m^2$ [11 $ft^2$] per week or up to 210 objects per year re-enter the Earth's atmosphere. Assuming that the space debris impacts the surface facilities at a 90° angle, the applicant estimated the total area of affected surface facilities to be 0.31 $km^2$ [0.12 $mi^2$]. The applicant stated that the probability that space debris would strike a surface facility is $1.3 \times 10^{-5}$ impacts over the operational period of 50 years, which is equivalent to an annual probability of $2.52 \times 10^{-7}$ (beyond Category 2). Thus, DOE excluded it from further consideration in the PCSA (BSC, 2008ai).

**NRC Staff's Evaluation**

The NRC staff reviewed the information and analyses provided by the applicant in SAR Section 1.6.3.4.11 and SAR Table 1.6-8, and references therein, on extraterrestrial activity presented to exclude impact of extraterrestrial objects with safety-related structures as initiating events.

The NRC staff notes that the applicant performed the meteorite impact analysis using the following assumptions: (i) meteorites fall randomly on the Earth's surface; (ii) the number of meteorites that fall to the Earth's surface would remain constant for at least the operational period of 50 years; and (iii) the size distribution and proportion of iron, hard stone, and soft stone/iron meteorites that fall to the Earth's surface remain constant over the same period. The NRC staff finds that these assumptions are appropriate for the type of analysis the applicant conducted because (i) meteorite impact with the Earth is a rare event without any correlation (i.e., meteorites fall randomly over the Earth's surface); (ii) available data do not show a significant change in the rate of impact, especially over a 50-year period; and (iii) this methodology is used to assess the potential safety of nuclear power plants from a meteorite strike (Solomon, et al., 1975aa). The NRC staff finds that this methodology (Solomon, et. al., 1975aa) is applicable to this evaluation of hazards to the GROA because meteorite strikes may occur independent of the facility type. The NRC staff also finds that the sources of meteorite information (e.g., Bland and Artemleva, 2006aa; Ceplecha, 1994aa) are appropriate for the analysis, as they are from established literature. Additionally, the NRC staff finds that the size range used in the analysis is acceptable because large-sized meteorites would break up into smaller sized objects upon entering the atmosphere. On the other hand, smaller objects would burn in the atmosphere and may not reach the Earth's surface. The largest stony meteorite recovered is smaller than 500 kg [1,102 lb] (Hills and Goda, 1993aa). Few meteorites that strike the Earth annually are large enough to create large impact craters. On the basis of information on the proportion of different types of meteorites striking the Earth, the NRC staff finds that the analysis presented in BSC (2008ai, Section 6.13) is appropriate. The NRC staff finds that the applicant's exclusion of a meteorite strike as an initiating event in the PCSA is acceptable because the estimated annual frequency of a meteorite striking the GROA is less than $10^{-6}$.

The NRC staff also finds that the applicant's use of data on space objects with radar cross sections larger than 1 $m^2$ [11 $ft^2$], which are tracked more closely by the U.S. Space Command until atmospheric reentry, is acceptable because the U.S. Space Command is the federal agency entrusted with the mission to track space objects. The NRC staff also finds that the applicant's use of data on space objects larger than 1 $m^2$ [11 $ft^2$] is reasonable because smaller objects may not cause significant damage to any hardened structure to be used in the repository facilities. The NRC staff notes that DOE's analysis assumed four objects larger than 1 $m^2$ [11 $ft^2$] would reenter the Earth's atmosphere each week. The NRC staff finds that this assumption is conservative because the number is double what has been observed by the U.S. Space Command. The NRC staff independently reviewed data on space debris (e.g., Klinkrad, et al., 2001aa). The U.S. Space Command currently tracks about 8,500 unclassified objects. The size of these objects varies from about 10 cm [3.9 in] in low-Earth orbit to about 1 m [33 ft] at geostationary altitudes. Approximately one to two objects greater than 1 $m^2$ [11 $ft^2$] in size reenter the Earth's atmosphere per week. Therefore, the NRC staff finds that the applicant's analysis is conservative, with respect to the space data the NRC staff independently reviewed (Klinkrad, et al., 2001aa). Therefore, the NRC staff finds that the applicant's exclusion of potential extraterrestrial objects as event sequence initiators in the PCSA is acceptable because (i) methods selected to assess the hazards are appropriate for the available data, (ii) appropriate properties of the meteorites and space objects were considered to assess the potential hazards, (iii) acceptable methodologies have been used to exclude the hazards, (iv) the frequencies estimated for strikes from a meteorite and a space object are conservative, and (v) technical bases used to exclude these hazards are adequate.

**Waste and Rock Interaction, Thermal Loading, Geochemical Alterations, and Dissolution**

The applicant provided information on waste and rock interaction, geochemical alterations, thermal load, and dissolution at the repository site in SAR Table 1.6-8, BSC (2008ai, Section 4.4), and DOE (2009ey). The applicant excluded waste and rock interaction as an external hazard in SAR Table 1.6-because any potential interaction between waste released from a waste package and repository rock could only occur following final disposal. Therefore, this potential interaction is applicable only to the postclosure timeframe. With regard to thermal loading, the applicant stated in SAR Section 1.3.5 that forced ventilation during the preclosure period would moderate any temperature rise. The applicant identified in SAR Section 1.1.8.4 that geochemical alteration and dissolution are slow-acting geological processes, and heat from the waste package and forced ventilation during the preclosure period would further limit geochemical alteration and dissolution in the rock by drying out the near-field rock mass and moderating the temperature rise.

**NRC Staff's Evaluation**

The NRC staff reviewed the information and technical basis the applicant provided in SAR Section 1.3.5, SAR Table 1.6-8, BSC (2008ai, Section 4.4), and responses to RAIs (DOE, 2009ey) on waste and rock interaction, thermal loading, geochemical alterations, and dissolution hazards, with a focus on the information and rationale to exclude waste and rock interaction, thermal loading, geochemical alterations, and dissolution as initiating events that could affect the repository. The NRC staff also evaluated the process-level model analysis on temperature and relative humidity distribution, including dryout zones under ambient ventilated conditions and observations in the Exploratory Studies Facility, as described in BSC (2004bg, Section 6.6) and SNL (2008aj, Section 7.5.2). Additionally, the NRC staff reviewed the information the applicant provided on dissolution in the postclosure screening analysis of features, events, and processes in SAR Table 2.2-5 (solubility, speciation, phase changes, precipitation/dissolution) and in SAR Section 2.3.5.3.3 to confirm that the applicant's models of dissolution processes in the preclosure and postclosure periods are consistent with each other.

The NRC staff finds that the applicant used appropriate site data on mineral dissolution and actual observations, including model prediction, to determine that the rates of progression of these processes are too slow to be hazards during the preclosure period. Therefore, the NRC staff finds that the applicant's exclusion of waste and rock interaction, thermal loading, geochemical alterations, and dissolution, as initiators of event sequences, is acceptable because (i) these processes are too slow to be hazards during the preclosure period and (ii) the technical bases for the exclusion of these hazards are adequate and consistent with the site information and the industry standard ASME/ANS RA–S–2008 (ASME, 2008aa), which includes a criterion to justify exclusion of events if they are slow in developing and there is sufficient time to provide an adequate response to address the slowly developing events.

**Perturbation of Groundwater**

The applicant provided information on perturbation of groundwater at the repository site in SAR Table 1.6-8, BSC (2008ai, Section 4.4), and DOE (2009ey). The applicant determined that the hazard associated with perturbation of groundwater or availability of groundwater in the long term would not initiate event sequences, because there would be sufficient time to develop alternate sources for additional water demand at the repository facilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Table 1.6-8, BSC (2008ai, Section 4.4), and DOE (2009ey) on the hazard associated with perturbation of groundwater, and compared the potential hazard of groundwater perturbation with conventional models of hydrologic responses to pumping water from unconfined aquifers and groundwater basins, as detailed in Freeze and Cherry (1979aa, Sections 8.3 and 8.10). The NRC staff finds that the applicant's description and technical basis for excluding the perturbation of groundwater are acceptable because groundwater perturbation is a slow process, as described in Freeze and Cherry (1979aa), and there would be sufficient time for DOE to seek alternate source(s) of water for the GROA. This is consistent with industry standard practice in ASME/ANS RA–S–2008 (ASME, 2008aa), which includes a criterion to justify exclusion of events if they are slow in developing and there is sufficient time to provide an adequate response to address the slowly developing events. Therefore, the NRC staff finds that the exclusion of groundwater perturbation as an initiating event is acceptable because the technical bases are adequate and consistent with the site and system information, and exclusion is consistent with the industry standard (ASME, 2008aa).

**Undetected Past Human Intrusions**

The applicant provided information on undetected past human intrusions at the repository site in SAR Table 1.6–8 and BSC (2008ai, Section 4.4). The applicant described undetected human intrusions as potential hazards associated with undiscovered boreholes or mine shafts. Any undetected boreholes or mine shafts that are directly connected to the subsurface facilities may act as direct conduits for radionuclide release, in addition to preferential flow paths for air and water for wastes already disposed. Therefore, the applicant considered undetected human intrusions (open site investigation boreholes or open mine shafts) in a screening of relevant features, events, and processes for the postclosure period in SAR Table 2.2-1 and SNL (2008ab, Table G-1). The applicant classified undetected past human intrusion as a Yucca Mountain unique hazard in BSC (2008ai, Section 4.4) because either boreholes or open mine shafts would be detected during repository construction or erosion of the borehole would proceed too slowly to affect the repository facilities during the preclosure period.

**NRC Staff's Evaluation**

The NRC staff reviewed the information and technical basis provided in SAR Table 1.6–8 and BSC (2008ai, Section 4.4) on hazards associated with undetected past human intrusions. The NRC staff finds that the applicant's information and technical basis to exclude this hazard as an initiating event from the PCSA are acceptable because either (i) signs of past human intrusion would be detected during repository construction or (ii) erosion of the condition would proceed too slowly to affect the repository facilities during the preclosure period. Therefore, the NRC staff finds that the exclusion of undetected past human intrusions as a hazard in the PCSA is acceptable because the technical bases are adequate and consistent with the site and system information.

**Security-Related Hazards**

The applicant identified security-related external events (i.e., sabotage, terrorist attack, and war) in SAR Table 1.6-8 and BSC (2008ai, Section 4.4) to be outside the scope of the PCSA. The applicant stated that safeguards and security systems are assessed within the physical security criteria in 10 CFR 73.51 (BSC, 2008bu).

**NRC Staff's Evaluation**

The NRC staff finds that DOE's exclusion of security-related events from the PCSA is acceptable because DOE's basis for exclusion is consistent with the requirements for the PCSA in 10 CFR 63.112. As described in 10 CFR 63.112(b), the PCSA identifies SSCs important to safety based, in part, on a systematic evaluation of the naturally occurring and human-induced hazards and initiating events, including action or inaction of operating personnel that could lead to dose consequences. The performance objectives for physical protection of spent nuclear fuel and high-level waste are provided in 10 CFR 73.51. Pursuant to 10 CFR 63.21(b)(3), the applicant provided a description of its security measures for physical protection in the SAR, General Information, Section 3. The NRC staff's evaluation of the applicant's description of physical security measures is documented in SER Section 1.3 of NUREG–1949 (NRC, 2010aa), where the NRC staff finds that the applicant will implement a physical protection program for SNF and HLW that includes physical protection, a safeguard contingency plan, and a security organization personnel training and qualification plan that complies with 10 CFR 73.51.

**NRC Staff's Conclusion on External Hazards and Initiating Events**

On the basis of the NRC staff's evaluation of the applicant's information and analyses on identification of external hazards and initiating events pertaining to the preclosure period, the NRC staff concludes, with reasonable assurance, that the regulatory requirements of

10 CFR 63.112(b) and 10 CFR 63.112(d) are satisfied, with regard to external hazards. The applicant has adequately identified the potential geologic, weather-related, aircraft crash, accidents at nearby industrial and military facilities, and other hazards to the surface and subsurface operations in the GROA. The associated annual frequency of occurrences of these external hazards and initiating events, including consideration of uncertainties, as applicable, has been provided based on methodologies consistent with acceptable guidance documents and industry practices. Adequate technical bases for inclusion or exclusion of the hazards and initiating events have been provided by the applicant.

#### 2.1.1.3.3.2 Operational (Internal) Hazards and Initiating Events

The applicant identified internal hazards and initiating events at the GROA in SAR Section 1.6.3.1. These hazards and associated initiating events are internal to the processes or operations and are generally associated with failure of equipment, either system or component, and human-initiated events. The NRC staff's review of the identification of internal initiating events is described in SER Section 2.1.1.3.3.2.1, followed by the review of the applicant's quantification and screening of (i) equipment and human-induced failures at surface facilities (SER Section 2.1.1.3.3.2.2), (ii) subsurface facilities (SER Section 2.1.1.3.3.2.3), (iii) fire hazards (SER Section 2.1.1.3.3.2.4), (iv) internal flood hazards (SER Section 2.1.1.3.3.2.5), and (v) criticality hazards (SER Section 2.1.1.3.3.2.6).

##### 2.1.1.3.3.2.1 Identification of Internal Initiating Events

The applicant described how it identified internal (operational) initiating events in SAR Section 1.6.3.1. The applicant provided additional details regarding the process to identify internal initiating events in Section 4.3.1 of the surface facility event sequence development analysis documents (BSC, 2008ab,bo,ao,bd) and documented the analysis results in Section 6.1.3 of the same documents (BSC, 2008ab,bo,ao,bd). Similarly, the process for identification of internal initiating events of the subsurface facilities is given in the event sequence development analysis document (BSC, 2008bj, Section 4.3.1), and the analysis results are documented in BSC (2008bj, Section 6.1.3). The applicant first described the facility operations and processes in the process flow diagrams. The applicant then developed initiating events using the master logic diagram (MLD) approach described in the American Nuclear Society/Institute of Electrical and Electronics Engineers (1983aa), Stamatelatos, et al. (2002aa), and the Electric Power Research Institute (EPRI, 2004aa). An MLD systematically relates the loss of top-level safety functions to lower level failure events via a hierarchical, top-down decomposition of safety systems. The applicant provided MLDs to identify potential hazards at each process step in Attachment D of BSC (2008ab,bo,ao,bd). The applicant then described how it verified the list of initiating events following the hazard and operability study (HAZOP) methodology described in American Institute of Chemical Engineers (1989aa) and Knowlton (1992aa). As the American Institute of Chemical Engineers (1989aa) described, a HAZOP is a systematic review of a process or operation to determine whether process deviations can lead to undesirable consequences. The applicant provided tables of HAZOP deviations in Attachment E of BSC (2008ab,bo,ao,bd,bj). The applicant compiled the list of internal initiating events in SAR Table 1.6-3 from Table 10 in each of the event sequence development analysis documents (BSC, 2008ab,bo,ao,bd,bj). As documented in Section 6.2 and in Attachment F of BSC (2008ab,bo,ao,bd,bj), the applicant then grouped individual initiating events that were associated with similar operations and with the same system response for further analysis. The applicant developed fault tree models to analyze the failures of either an individual or a group of structures, systems, and components and/or human actions to quantify the system failure probabilities within the grouped operations as they were included in the PCSA.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's identification of initiating events to determine whether the methodologies used were appropriate for identifying initiating events that could lead to risk-significant event sequences. The NRC staff reviewed the discussions of the methodology provided in SAR Section 1.6.3.1 and in Section 4.3.1 of BSC (2008ab,bo,ao,bd,bj), and then examined Chapter 6 and Attachments D and E of BSC (2008ab,bo,ao,bd,bj) to evaluate how the applicant applied these methodologies.

The NRC staff finds that the applicant's use of MLDs and HAZOP studies is acceptable because these are standard practices used by the nuclear and chemical industries to identify initiating events [NUREG/CR–2300 (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa)]. The NRC staff finds that additional events, beyond those identified in SAR Table 1.6-3, were included in the fault trees that the applicant used to quantify groups of initiating events. These events were either identified directly in the fault tree or were identified through the evaluation of human reliability. The NRC staff finds that the applicant's use of fault trees and human reliability analysis to develop potential initiating events is acceptable because they are standard industry practices for modeling contributors to system failure in nuclear and other industries [e.g., NUREG/CR–2300 (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa)].

The NRC staff conducted a risk-informed review of the applicant's identification of initiating events to determine whether site data and system information were appropriately used in the identification of internal initiating events (i.e., would not underestimate risk). The NRC staff selected several initiating events based on their risk potential (e.g., initiating events with potential to pose significant consequences to the public and worker safety or that could have high annual frequency of occurrence close to the boundaries between Category 1 and Category 2, or Category 2 and beyond Category 2) for review. The NRC staff mapped the selected events from the MLD and HAZOP tables into the fault trees and then examined how the applicant included the events in the fault trees.

The NRC staff finds that the applicant's process flow diagrams (PFDs) were developed at a level of sufficient detail to identify challenges at different process steps. The applicant used the term "challenge" to refer to an accident or event that may cause damage, such as a drop or impact to a cask. As described in Section 6.1.2 and Attachment B of BSC (2008ab,bo,ao,bd,bj), the PFDs included the type of equipment used in each process step and how the equipment would be used. For example, the applicant used process flow descriptions to describe how a particular process step would be conducted (e.g., how the canister transfer machine will transfer a waste canister from a transportation cask to a waste package and how many crane lifts or slide gate operations would be required to carry out a particular step).

The NRC staff finds that the applicant appropriately considered other operating modes, such as maintenance activities, in identifying initiating events. For example, the applicant stated that maintenance would not be performed on equipment in operation, and several fault trees included the possibility of the failure to reset the systems following maintenance. Similarly, for mechanical systems (e.g., cranes), the NRC staff finds that the applicant included maintenance-related failures when quantifying failure rate estimates using empirical data.

The NRC staff, therefore, finds that the applicant's information on identification of internal initiating events is acceptable because (i) methods used for hazard and initiating event identification are consistent with standard industry practices; (ii) methods selected for hazard and initiating event identification are appropriate for the available data on the site and geologic repository operations area; (iii) assumptions used to identify human-induced hazards and initiating events are well defined, have adequate technical bases, and are supported by information on the site and its structures, systems, components, equipment, and operational processes; and (iv) the identification of human-induced hazards encompasses relevant aspects of the geologic repository operations area radiological systems and all modes of operation. Therefore, the NRC staff finds that the applicant's information on identification of internal initiating events is acceptable.

2.1.1.3.3.2.2 Quantification of Initiating Event Frequency for Equipment and Human-Induced Failures at Surface Facilities

2.1.1.3.3.2.2.1 Grouping and Screening of Initiating Events at Surface Facilities

The applicant discussed grouping of initiating events identified in SAR Table 1.6-3 and in Section 4.3.4.4 of the surface and subsurface facility event sequence development analysis documents (BSC, 2008ab,bo,ao,bd,bj). The applicant stated that events from the MLDs which involved the same SSCs, operations response, and pivotal event system response were grouped. The applicant documented these groupings in Section 6.2 and in Attachment F of BSC (2008ab,bo,ao,bd,bj). For categorization purposes, these initiating event groups were further combined with events that were related to the same operational area/activity and that led to the same end state.

The applicant discussed screening of initiating events in SAR Section 1.7.1.2.1 and Section 6.0 of the surface and subsurface facility event sequence reliability and categorization analysis documents (BSC, 2008ac,as,be,bq,bk). The applicant identified criteria for exclusion of initiating events based on design features and by subsuming less significant events into existing events. The applicant listed excluded internal events in SAR Table 1.7-1 and Table 6.0-2 of BSC (2008ac,as,be,bq,bk).

**NRC Staff's Evaluation**

The NRC staff reviewed the information on initiating event grouping and screening provided in SAR Sections 1.7.1.2.1 and 6.0 to determine whether adequate technical bases for the inclusion and exclusion of internal initiating events were provided. The NRC staff notes that the applicant used grouping and screening, as discussed in this section, to evaluate initiating events at surface facilities to facilitate consideration of the large number of initiating events. This approach was not necessary in the other three evaluation sections (subsurface, fire, and criticality) where there was not a similarly large number of initiating events to consider. Consistent with the risk-informed review approach discussed in SER Section 2.1.1.3.3, the NRC staff reviewed the initiating event identification for the CRCF (BSC, 2008ab,ac) and WHF (BSC, 2008bo,bq) because these facilities handled the largest quantities of packaged and unpackaged waste and, thus, are sufficiently bounding. In particular, the NRC staff selected the CRCF as a representative facility because it handles canistered waste, and the combined throughput for the three proposed CRCFs is higher than that for any other surface facility (SAR Table 1.7-5). In addition, the NRC staff selected the WHF because this facility handles both canistered and uncanistered spent fuel. The NRC staff mapped the selected events from the MLDs and HAZOP deviation tables into the fault trees. The NRC staff examined whether the events were adequately included in the fault trees to quantify the initiating event group represented by the “small bubbles” in Attachment F of BSC (2008ab,bo,ao,bd,bj).

Consistent with the approach outlined by the applicant for grouping of the “small bubble” initiating event groups, the NRC staff further grouped these small bubbles according to the type of challenge they posed, corresponding to the top event of the fault trees, as documented in Attachment B of BSC (2008ac,as,be,bq,bk), and the system response event trees to which the initiating event fault trees were assigned in Attachment A of BSC (2008ac,as,be,bq,bk). For mechanical challenges to containment, the NRC staff used the challenges identified in Table 6.3-7 of BSC (2008ac,as,be,bq) as a starting point to form these groups and then used the passive reliabilities identified in Table 6.3-8 of these documents to link the initiating events represented by the fault trees to the system responses.

On the basis of its review of the applicant’s information on grouping and the NRC staff’s independent grouping, the NRC staff finds that the applicant’s grouping of the initiating events from the MLD on the basis of safety function and similarity of challenges to safety systems, as discussed in Section 6.2 and Attachment F of BSC (2008ab,bo,ao,bd), is acceptable because the applicant’s approach is consistent with the guidance in NUREG/CR–2300, Chapter 3, “Accident-Sequence Definitions and System Modeling” (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa). This guidance describes a probabilistic risk assessment methodology, including the fault tree method used to characterize system fault logic and techniques for assessing the initiating event probability at nuclear power plants. This is applicable to the PCSA because fault tree modeling is a general technique that is not specific to a particular type of facility.

The NRC staff finds that the applicant’s screening process for initiating events based on design is acceptable because the applicant subsumed less significant events into existing events (thereby using a bounding approach) and identified specific equipment design features that would prevent the event from occurring. For example, in the list of nuclear safety design bases provided in BSC (2008ac, Table 6.9-1), the applicant designed the site prime mover fuel tank to preclude explosions. Thus, because the site prime mover fuel tank cannot explode, it cannot initiate an event sequence that would lead to a potential cask breach and is, therefore, acceptably screened out. Additionally, for the WHF, the applicant stated in BSC (2008bo, Attachment A) that the decontamination pit cover will be strong enough to prevent a transportation cask or shielded transfer cask from penetrating the cover and falling into the pit so as to screen consideration of higher drop heights for these casks.

The NRC staff, therefore, finds that the applicant’s initiating event grouping and screening is acceptable because (i) the applicant identified events that comprise a wide selection of initiating events and grouped these events using a process consistent with applicable NRC guidance and standard industry practices and (ii) the applicant’s groups of initiating events covered a broad spectrum of challenges to relevant safety functions, such as shielding, containment, and criticality to encompass a range of potential challenges. Therefore, the NRC staff finds that the technical basis for grouping and screening of internal initiating events at surface facilities is acceptable.

2.1.1.3.3.2.2.2 Quantification of Initiating Events at Surface Facilities

The applicant’s approach for quantifying initiating events was discussed in SAR Section 1.7.2 and Sections 4.3 and 6.2.1 of the surface facility event sequence reliability and categorization analysis documents (BSC, 2008ac,as,be,bq) for the Canister Receipt and Closure Facility

(CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), and Wet Handling Facility (WHF). The applicant quantified initiating events by developing fault trees for groups of the initiating events listed in Table 1.6-3 of the SAR rather than quantifying each individual initiating event. These groups were identified in Attachment F of the surface facility event sequence development analysis documents (BSC, 2008ab,bo,ao,bd) for the CRCF, IHF, RF, and WHF. Each of the groups is mapped to a fault tree or basic event in Attachment A of BSC (2008ac,as,be,bq) for the CRCF, IHF, RF, and WHF.

The applicant estimated equipment reliability either by a direct probability assignment, based on available historical data, or by modeling a system and its components using fault trees. Fault trees were parameterized by empirical data derived from standard equipment reliability databases or estimates of human error on the basis of a human reliability analysis. The applicant stated that it used fault trees, rather than direct probability assignments, to model potential faults in complex machinery, for which no historical data exist at the system level. As discussed in SAR Section 1.7.2.1 and in Sections 4.3.2.1 and 6.2.2 of BSC (2008ac,as,be,bq) for the CRCF, IHF, RF, and WHF, the applicant developed fault trees following the process described in NUREG–0492 (NRC, 1981ab). Fault trees for particular components were provided in Attachment B of BSC (2008ac,as,be,bq).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's quantification of initiating events provided in SAR Section 1.7.2 to determine whether the applicant acceptably determined the frequency of occurrence of internal initiating events and adequately considered uncertainties. The NRC staff reviewed the applicant's information in two steps: (i) selected a subset of event sequences to encompass a range of initiating event types and facilities and examined how the initiating events for these sequences were quantified and (ii) reviewed the initiating event quantification for CRCF (BSC, 2008ab,ac) and for WHF (BSC, 2008bo,bq). The NRC staff selected these two facilities as part of its risk-informed review to provide a representative sample of handling operations of both canistered wastes and canistered and uncanistered spent fuel. The NRC staff mapped selected events from the MLD and HAZOP deviation tables into the fault trees. The NRC staff examined how these events were included in the fault trees to quantify the initiating event group represented by the small bubbles in Attachment F of BSC (2008ab,bo,ao,bd). In particular, the NRC staff examined how the applicant quantified the basic events used to quantify these fault trees and then examined the frequencies within the initiating event groups (as described in SER Section 2.1.1.3.3.2.2.1) to determine whether the results of the quantification were reasonable and consistent with standard industry practices.

The NRC staff reviewed the applicant's discussion pertaining to fault trees, using the guidance in NUREG/CR–2300 (ANS/IEEE, 1983aa), NUREG–0492 (NRC, 1981), and HLWRS–Interim Staff Guidance–02 (HLWRS–ISG–02) (NRC, 2007ab), and finds that the applicant used acceptable methodologies to quantify initiating events for the following reasons. NUREG/CR–2300 (ANS/IEEE, 1983aa) describes fault tree analysis in terms of qualitatively characterizing the system fault logic such that quantification of the top event probability is obtained by using the Boolean logic represented in the fault tree model. The applicant used fault tree analysis to quantify the failure probability pertaining to the top event, which, in this case, is the initiating event probability. The NRC staff finds that the applicant's use of a fault tree analysis to model the failure probability of a system, where a direct probability assignment cannot be made due to limited or unavailable empirical data, is acceptable because it follows HLWRS–ISG–02 (NRC, 2007ab).

**Equipment Reliability**

The applicant documented the quantification of active component reliability in SAR Section 1.7.2.2 and in Sections 4.3.3 and 6.3.1, Table 6.3-1, Attachment C, and the supporting files in Attachment H of BSC (2008ac,as,be,bq) for the CRCF, IHF, RF, and WHF. The applicant estimated equipment reliability (either directly assigned or modeled using fault trees) using databases containing equipment reliability information and data using databases, such as NUREG–1774 (NRC, 2003ai), NPRD–95 (Denson, et al., 1994aa), and Savannah River Site Generic Database Development (Blanton and Eide, 1993aa). The applicant estimated the probability of failure for various components and then used these estimates to develop initiating event probabilities as part of its PCSA.

For most of the cranes used in the surface facilities {i.e., the 181-tonne [200-ton] cask handling cranes, the waste package handling cranes, jib cranes, and the spent fuel transfer machine}, the applicant conducted a direct quantification of the failure probability based on analysis of empirical data on crane drops taken from NUREG–1774 (NRC, 2003ai) and NUREG–0612 (NRC, 1980aa). The applicant documented the failure probability estimates for these cranes in Attachment C.1.3 of BSC (2008ac,as,be,bq).

The applicant modeled the reliability of equipment, for which no historical data exist at the system level [e.g., the canister transfer machine (CTM), the cask transfer trolley], by developing fault trees with sufficient detail at the component level to allow the use of industry experience with similar components. For example, the applicant quantified the failure of crane load cells pressure sensors using selected pressure sensors data from the NPRD–95 (Denson, et al., 1994aa) database.

SAR Section 1.7.2.2 described how the applicant developed active system or component reliabilities and defined an active system or component as one that changes position and, by doing so, modifies the system behavior. The applicant described, in Section C1 of BSC (2008ac,as,be,bq), the process for matching component-level design features from the design and failure modes from the PCSA to failure data from the selected reliability databases. The applicant then used the Bayesian analysis to combine information from multiple data sources to develop failure probability distributions for active systems and components and documented these Bayesian analyses in Mathcad files with the supporting documents. In applying the Bayesian techniques, the applicant used a parametric empirical Bayes method (Siu and Kelly, 1998aa; Droguett, et al., 2004aa). Additionally, the applicant used the alpha factor method described in NUREG/CR–5485 (Mosleh, et al., 1998aa) to quantify common-cause failures. The applicant stated that in some cases, even if more than one data source was available, a component's failure probability was quantified by selecting one failure distribution. The applicant explained that this approach was selected when the use of Bayesian analysis with multiple similar estimates would yield an unrealistically narrow distribution. The applicant used this approach to quantify, for example, the interlock failure on demand. The applicant indicated that it used the single data source yielding the most diffuse information, which would produce the largest uncertainty, and the median of the five data sources as representative of the mean failure rate (DOE, 2009dy). The applicant selected one of the five distributions having a peak value that coincided with the combination distribution peak for interlock failure on demand. For cases with only one data source (e.g., air handling unit failure to run and pressure sensor failure on demand), the applicant used a single data source and then updated the value using a Jeffreys noninformative prior distribution, in accordance with NUREG/CR–6823 (Atwood, et al., 2003aa). The component reliability values provided in Attachment C of BSC (2008ac,as,be,bq) for the CRCF, IHF, RF, and WHF were computed and documented using a combination of spreadsheets and Mathcad files that the applicant included as Attachment H in BSC (2008ac,as,be,bq).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's quantification of active component reliability provided in SAR Section 1.7.2.2 and in Sections 4.3.3 and 6.3.1, Table 6.3-1, Attachment C, and the supporting files in Attachment H of BSC (2008ac,as,be,bq) for the CRCF, IHF, RF, and WHF to determine whether empirical analyses and modeling techniques were used appropriately to estimate equipment reliability and whether uncertainty in the reliability estimates had been addressed adequately. The NRC staff examined the applicant's approach to determine selected system failure probability distributions for the CRCF and intrasite facilities. The NRC staff focused its review on component failure distributions for the CRCF because the CRCF equipment reliability analysis is representative for the remaining surface facilities.

The NRC staff compared the applicant's estimated crane failure frequencies to those documented in NUREG–1774 (NRC, 2003ai) and NUREG–0612 (NRC, 1980aa). NUREG–1774 (NRC, 2003ai) describes the results of a detailed review of crane operating experience at U.S. nuclear power plants from 1968 through 2002. NUREG–0612 (NRC, 1980aa) provides the NRC staff's review of the handling of load drops at nuclear power plants and recommendations on actions to be taken to assure safe handling of heavy loads. Both NUREGs are applicable to GROA operations due to the similarity of heavy load lifting at the GROA and nuclear power plants.

The NRC staff finds that the estimated failure frequencies (in the range of $10^{-5}$ to $10^{-4}$ per transfer) for the cask handling crane and the CTM (SAR Tables 1.9-3 for the CRCF and 1.9-4 for the WHF) are reasonable because the estimates are within the range of values reported in (i) NUREG–0612, Appendix B.1.1.2.3 (NRC, 1980aa) of approximately $10^{-5}$ and $1.5 \times 10^{-4}$ per lift for non-single-failure-proof cranes, on the basis of U.S. Navy experience in 1977 and (ii) NUREG–1774 (NRC 2003ai) for very heavy load drops (3 drops out of 54,000 lifts, or approximately $6.0 \times 10^{-5}$ per transfer). The NRC staff finds that the applicant's estimated spent fuel transfer machine drop rate (in the range of $5.0 \times 10^{-6}$ per transfer) is reasonable because it is consistent with the data on fuel-handling drops from NUREG–1774 (NRC, 2003ai).

The NRC staff finds that the use of Bayesian analysis to quantify uncertainty of the estimated reliability using data from multiple sources is acceptable because this technique is a standard technique identified in NUREG/CR–6823 (Atwood, et al., 2003aa), which provides guidelines on sources of information and methods for estimating parameters used in probabilistic risk assessment models and for quantifying uncertainties in the estimates. The NRC staff finds that the use of a Jeffreys noninformative prior distribution to estimate the uncertainty in the reliability estimate for cases with only one data source with no uncertainty data is also reasonable because this approach is also consistent with NUREG/CR–6823 (Atwood, et al., 2003aa). Similarly, the NRC staff finds that the applicant's use of the alpha factor method described in NUREG/CR–5485 (Mosleh, et al., 1998aa) to quantify common cause failure is acceptable because this method is a standard method capable of handling various levels of redundancy. NUREG/CR–5485 (Mosleh, et al., 1998aa) provides guidelines in modeling common-cause failure events in commercial nuclear power plants and is applicable to GROA operations because the applicant's modeling techniques, such as fault tree modeling, are independent of the facility type.

In some cases, the applicant selected a single data source to represent the uncertainty (i.e., standard deviation) of existing data of equipment failure rate. The applicant stated that, in such cases, it selected the data source that would yield the largest uncertainty but considered available data sources to estimate a representative mean failure rate (DOE, 2009dy). The NRC staff finds the applicant's approach of selecting a large uncertainty (i.e., large standard deviation) appropriate because the applicant assumed failure rates to follow a log-normal distribution (which is a commonly used distribution to describe failure rates). The NRC staff finds that this approach tends to overestimate failure rates when sampled from a distribution with a large standard deviation and is, therefore, conservative.

The NRC staff finds, on the basis of examination of the spreadsheets provided in support of the active component reliability database in Attachment H of BSC (2008ac,as,be,bq), that the applicant generally selected data sources that reflected comparable component types and failure modes from the comparable facilities' (e.g., Savannah River Site) reliability databases. The NRC staff finds that the applicant's use of the available reliability databases [NUREG–1774 (NRC 2003ai), NPRD–95 (Denson, et al., 1994aa)] to estimate equipment reliability is acceptable because these databases contain reliability data of components used in comparable facilities. NPRD–95 (Denson, et al., 1994aa) documents the reliability data for nonelectronic parts (e.g., electrical, mechanical) from government and nongovernment sources.

The NRC staff notes that the calculations documented in the spreadsheets and Mathcad files did not appear to consistently match the calculations described in NUREG/CR–6823 (Atwood, et al., 2003aa) for Bayesian updating and in Mosleh, et al. (1998aa) for estimating the impact of common-cause failures. For example, the calculations in the spreadsheet documenting the pressure sensor failure on demand did not match the calculation described in NUREG/CR–6823 (Atwood, et al., 2003aa) used to estimate the shape factor for a beta distribution. Likewise, the estimation of the reliability for the exhaust fans and air-handling units described in BSC (2008ac, Section C3) differed from the calculations described in NUREG/CR–5485 (Mosleh, et al., 1998aa, Table 5-4). However, the NRC staff finds that these inconsistencies would have negligible impact on the results in the PCSA in support of the facility design because the inconsistencies are numerically insignificant, and as such would not affect event sequence categorization.

The NRC staff notes that, for systems that the applicant modeled using fault trees, the applicant developed data from databases that are in wide use for reliability engineering and developed the design to a level of detail that allowed identification of components and failure modes from these databases. For example, the NRC staff finds that the applicant generated estimates of crane reliability that are consistent with empirical data documented in NUREG–0612 (NRC, 1980aa) and NUREG–1774 (NRC, 2003ai).

**Human Reliability**

The applicant described the approach to assessing human reliability in SAR Section 1.7.2.5 and in Attachment E and Sections 4.3.4 and 6.4 of BSC (2008ac,as,be,bq). The list of resulting human failure events was included in Tables 6.4-2 and 6.4-1 of BSC (2008ac,as,be,bq).

The applicant outlined a nine-step approach for conducting the human reliability analysis that it considered to be consistent with HLWRS–ISG–04 (NRC, 2007ad), ASME RA–S–2008 (American Society of Mechanical Engineers, 2008aa, Section 1-4.3), and NUREG–1624 (NRC, 2000ai). The applicant's approach is an iterative process that begins with a definition of the scope of the analysis, works through an identification of potential human failure events, conducts a preliminary analysis for initial quantification of the human failure event, and performs a detailed analysis of human failure events deemed to be of high significance. The applicant stated that it quantified human failure events by selecting from four possible quantification methods: (i) Cognitive Reliability and Error Analysis Method (Hollnagel, 1998aa); (ii) Human Error Assessment and Reduction Technique (Williams, 1986aa) and Nuclear Action Reliability Assessment (NARA) (Corporate Risk Associates Ltd., 2006aa); (iii) Technique for Human Error Rate Prediction (NUREG/CR–1278) (Swain and Guttmann, 1983aa); and (iv) A Technique for Human Event Analysis (ATHEANA, NUREG–1624) (NRC, 2000ai). The applicant discussed the general human reliability assessment method selection in Attachment E of BSC (2008ac,as,be,bq), including selecting four human failure event quantification methods to treat operator errors. The applicant's specific selection of human reliability assessment quantification methods for specific human failure events was described in conjunction with the analysis of these human failure events. The results of the human error analysis were a list of human failure events that were incorporated into the PCSA as basic events in the SAPHIRE model.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's quantification of human failure events provided in SAR Section 1.7.2.5, references therein, and Attachment E and Sections 4.3.4 and 6.4 of BSC (2008ac,as,be,bq), to examine whether empirical analyses and modeling techniques were used appropriately to estimate human reliability, and whether uncertainty in the reliability estimates was addressed adequately. Specifically, the NRC staff's review was to evaluate the applicant's treatment of operator errors to assess whether (i) the methodology the applicant used to assess the potential for operator errors was acceptable and (ii) the applicant acceptably implemented its methodology.

To determine whether the applicant's human reliability analysis process is acceptable, the NRC staff examined the detailed description of the process in Attachment E of BSC (2008ac,as,au,be,bk,bq) and finds that the applicant used a comprehensive human reliability analysis process, consistent with human reliability analysis processes and HLWRS–ISG–04 (NRC, 2007ad). The process the applicant used was developed for the ATHEANA human reliability analysis method (NRC, 2000ai). NUREG–1624 states that ATHEANA was developed to increase the degree to which an HRA can represent the kinds of human behaviors seen in accidents and near-miss events at nuclear power plants and at facilities in other industries that involve similar kinds of human/system interactions, such as those at the GROA. The NRC staff also finds that the process included additional steps not typically addressed in detail for nuclear power plant human reliability analyses. For example, the applicant explicitly included a detailed search process to identify human failure events. Furthermore, the applicant's process also identified human-induced initiating events, as identified in HLWRS–ISG–04 (NRC, 2007ad), as a potentially important aspect to be considered for preclosure operations. The NRC staff finds the applicant's methodology acceptable because the methodology is consistent with HLWRS–ISG–04 guidance and included additional levels of detail.

To determine whether the applicant selected the human failure event quantification methods appropriately, the NRC staff examined Appendix E.IV of BSC (2008ac,as,au,be,bk,bq), as well as discussion on the preclosure design, potential operating characteristics, and potential operator vulnerabilities described in Sections E.4 and E.5 of the same documents. The NRC staff notes that the applicant decided to use four existing human reliability analysis methods to analyze operator errors by comparing the operations at the repository facilities and nuclear power plants, capabilities of available human reliability analysis methods, and characteristics of expected operator errors for the repository facilities. The NRC staff finds that the applicant's choices are acceptable because relevant factors to these decisions were identified and are consistent with the NRC guidelines in HLWRS–ISG–04 (2007ad). Additionally, DOE's methodology is consistent with NUREG–1792 (NRC, 2005ae), which provides that human reliability analysis methods be selected after analysts identify the factors that most influence operator performance, matching these factors with the human reliability analysis methods that best represent them in human reliability analysis quantification.

To determine whether the results of the applicant's process for identifying human failure events to include in its PCSA are acceptable, the NRC staff examined the results documented in Table E7-1 of BSC (2008ac,as,au,be,bk,bq), as well as qualitative analyses of potential human failure events in Appendices A, B, and E of BSC (2008ac,as,au,be,bk,bq). The NRC staff finds that the applicant considered a broad range of potential operator errors to include in the PCSA. The NRC staff finds that this is acceptable because the identification process described was thorough and the justifications provided for excluding selected operator actions were logical and consistent with operations at the GROA.

To determine whether the results of the applicant's qualitative analyses are acceptable, the NRC staff examined the results of the qualitative analysis documented in BSC (2008ac,as,au,be,bk,bq) and the applicant's response to the NRC staff's RAI (DOE, 2009dy). Based on its evaluation, the NRC staff finds that the applicant provided adequate qualitative analyses because the applicant's analyses produced general and human failure event-specific results consistent with site and system information, and they are, therefore, appropriate for use in the PCSA.

To determine whether the applicant's treatment of dependencies is acceptable, the NRC staff examined the general discussion of dependency treatment in Section E.3.3 of BSC (2008ac,as,au,be,bk,bq), as well as the applicant's treatment for specific human failure events, as applicable. The NRC staff finds that the applicant's approach is acceptable because DOE used a traditional dependence approach documented in NUREG/CR–1278 (Swain and Guttmann, 1983aa) for treating dependencies. NUREG/CR–1278 is appropriate for use because it provides the modeling and information necessary for the performance of human reliability analysis as a part of probabilistic risk assessment of nuclear power plants. It is applicable to the PCSA because many of the operations at the GROA are analogous to those analyzed for nuclear power plants. Further, the approach is acceptable because in discussing the quantification for specific human failure events, the applicant identified when and why dependencies should be modeled, as well as mechanisms for potential dependencies specific to the operations at the GROA.

To determine whether the applicant's selection of human reliability analysis quantification methods for specific human failure events is appropriate, the NRC staff reviewed both generic and specific human failure event qualitative inputs and compared these inputs with how the applicant represented these factors in its quantification methods used to select specific human reliability analyses. The NRC staff finds that the applicant appropriately identified the specific inputs that were used in detailed human reliability analysis quantification methods and described how these were related to each contribution to a human failure event probability. In addition, the NRC staff finds that the quantification method appropriately addressed the relevant error modes for the operator action. For example, in Section E.6.5.3.4.4.5 of BSC (2008ac,as,au,be,bk,bq), the applicant first described the scenario (or elements that contribute to the human failure event) and then described how the scenario might occur. The elements of this discussion were then related to a specific method [e.g., Nuclear Action Reliability Assessment (NARA) (Corporate Risk Associates Ltd., 2006aa)] on the basis of contributing elements and attributes of the various detailed human reliability analysis methods (BSC, 2008ac,as,au,be,bk,bq).

To determine whether DOE's application of specific human reliability analysis quantification methods is acceptable, the NRC staff examined how the various qualitative analysis results were represented in the selection of inputs to specific detailed human reliability analysis quantification methods. The NRC staff finds that the applicant described the analyst choices for specific inputs to detailed human reliability analysis quantification methods, and the NRC staff finds this acceptable because relevant and expected error mechanisms were reflected, the methods were applied consistently and as intended by the authors of the methods, qualitative analysis inputs were appropriately reflected, and the human reliability analysis is consistent with HLWRS–ISG–04 (NRC, 2007ad). For example, in Section E.6.5.3.4.4.5 of BSC (2008ac,as,au,be,bk,bq), the applicant identified the key inputs to the detailed human reliability analysis quantification method (e.g., generic task type and error-producing condition selections for the NARA method) and described the scenario-specific aspects that underlie how these inputs were assessed.

The NRC staff finds that the applicant's human reliability analysis process is acceptable because (i) human reliability analysis methods selected are consistent with the standard industry practices and NRC guidance [e.g., HLWRS–ISG–04 (NRC, 2007ad)] and (ii) human errors that may lead to radiological consequences were adequately identified, and adequate human reliability analyses were performed.

In summary, the NRC staff finds that empirical analyses and modeling techniques were used appropriately to estimate human reliability and that uncertainty in the reliability estimates was addressed adequately because (i) the applicant's process for identifying human failure events to include in its PCSA is acceptable; (ii) the results of the applicant's qualitative analyses, both for the facility as a whole and for specific activities, locations, and environments are adequate and the applicant's qualitative analyses are acceptable; (iii) the applicant's treatment of dependencies is acceptable; (iv) the applicant's selection of human reliability analysis quantification methods for specific human failure events is appropriate; and (v) the applicant's application of specific human reliability analysis quantification methods is acceptable. The NRC staff, therefore, finds that the applicant used the human reliability analysis methodologies appropriately.

**Overall Initiating Event Frequency**

The applicant presented the results of fault tree analyses for initiating events in Attachment F of BSC (2008ac,as,be,bq). The applicant performed the fault tree analyses for quantifying initiating event frequencies using the software SAPHIRE files, as described in Attachment H of BSC(2008ac,as,be,bq).

**NRC Staff's Evaluation**

The NRC staff examined the results of the applicant's analyses of initiating event frequency provided in Attachment F of BSC (2008ac,as,be,bq). In particular, the NRC staff examined the SAPHIRE files in Attachment H of BSC (2008ac,as,be,bq) to evaluate the applicant's numerical estimates of the initiating event fault trees.

The applicant used direct probability assignments, where it was able to obtain empirical failure data on systems that it considered as analogous to those intended for use at the GROA facilities. In addition, the applicant used SAPHIRE to model systems with no direct analog to the GROA operations and quantified the initiating event failure rates using fault trees in these models. SAPHIRE (Kvarfordt, et al., 2005aa) has been developed for NRC use and is an acceptable software platform for probabilistic safety analysis, such as fault tree and event tree development. The models combined estimates of equipment and human errors that could lead to unintended radiological exposures. The NRC staff finds that the use of fault trees, as described in NUREG–0492 (NRC, 1981ab), for quantifying initiating events is acceptable because fault tree analysis is a standard industry practice that was developed by NRC. NUREG–0492 (Fault Tree Handbook) was developed by NRC to set forth a set of recommendations and the methodology for constructing and evaluating fault/event trees in risk assessments, such as those used to evaluate potential initiating events at the GROA.

The NRC staff finds that the applicant estimated events associated with relatively low energy mechanical impacts (e.g., low-speed collisions or side impacts) to occur with a relatively high frequency (on the order of a few events out of every thousand transfers). Data provided by the applicant indicated that the reliability of the waste containers against such challenges is very high, with failure frequencies less than 1 in every 100 million challenges, such that the probability of a breach would be very low.

The NRC staff notes that the applicant estimated frequency of tip overs, drops of heavy objects onto canisters, and flat-bottom drops of a waste container (all of which would result in higher energy mechanical impacts on the canisters, which are less resilient) to occur on the order of a few events out of every 100,000 transfers. The NRC staff finds that these estimates were dominated by estimates of crane reliability. The NRC staff finds that these estimated frequencies are reasonable because they are consistent with empirical data on crane reliability from nuclear industry experience, as documented in NUREG–0612 (NRC, 1980aa) and NUREG–1774 (NRC, 2003ai).

The NRC staff notes that the applicant estimated canister shearing-type impacts to occur with very low frequencies (on the order of $7.0 \times 10^{-9}$ per transfer). The applicant included this frequency in its controlling parameter for spurious movement of a CTM in SAR Tables 1.9.1-3 for the CRCF and 1.9.1-4 for the WHF. The applicant associated shearing-type impacts with multiple equipment and human failures. The NRC staff finds the applicant's frequency for shearing-type impacts acceptable because the applicant (i) showed in its fault tree model and human reliability analysis that multiple equipment and human failures would have to occur to have a shearing-type impact and (ii) demonstrated in its fault tree analysis that these failures, when combined, would lead to this low frequency.

The NRC staff notes that the applicant estimated event frequencies in the WHF that were similar to those estimated for the CRCF for analogous canister-handling operations. The NRC staff finds that the frequencies of damage (on the order of $5.0 \times 10^{-6}$ per transfer) the applicant estimated due to drops during spent fuel transfers in the WHF are acceptable because these frequencies are consistent with empirical data from nuclear industry experience, as documented in NUREG–0612 (NRC, 1980aa) and NUREG–1774 (NRC, 2003ai).

The applicant estimated that events involving a loss of shielding due to equipment or human error occur relatively infrequently (on the order of only a few events every 100,000 transfers). The applicant's calculated frequencies for loss of shielding events included multiple human errors and equipment (e.g., interlock) failure. Therefore, the NRC staff finds that the applicant adequately addressed direct exposure event sequences because the event sequence frequencies occur below a frequency of 1 per 100 years (below Category 1). Because DOE demonstrated that these events involving loss of shielding due to equipment or human error are below Category 1, the NRC staff finds the applicant adequately included these types of events as Category 2 event sequences in the PCSA.

In their evaluation (DOE, 2009dx), DOE stated that it would add an interlock function to the existing interlocks on the canister transfer machine in the CRCF in order to further reduce the probability of an operator error leading to a direct exposure. NRC staff finds that the addition of this interlock function would effectively reduce the initiating event probability by several orders of magnitude, as shown in Table 4 (DOE, 2009dx).

In summary, the NRC staff finds that the applicant's quantification of internal initiating events for surface facilities is adequate because (i) the applicant's estimates are consistent with standard industry practices for analogous operations; (ii) the applicant used standard methods and standard databases to estimate mechanical and human errors; (iii) methods selected for determining the probability or frequency of occurrence for hazards and initiating events are appropriate; and (iv) human errors that may lead to radiological consequences are adequately quantified, and adequate human reliability analyses were performed. Therefore, the NRC staff finds that the applicant's assessment of the frequency of occurrence of internal initiating events is acceptable.

2.1.1.3.3.2.3 Quantification of Initiating Event Frequency for Subsurface Operations

The applicant provided information and analysis on hazards and initiating events at the subsurface facilities in SAR Section 1.6.3.1 and BSC (2008bk, Section 6.2). The applicant described the operations within the subsurface facilities that were used in the MLD and HAZOP analyses in BSC (2008bj, Section 6.1.2) and discussed the results in BSC (2008bj, Section 6.1.3). The applicant identified 29 initiating events at the subsurface facilities developed from these analyses, as outlined in BSC (2008bj, Table 11) and SAR Table 1.6-3. Additional information and analyses of these initiating events were provided in BSC (2008bk) and the applicant's responses to the NRC staff's RAIs (DOE, 2009dy,ey). The applicant used this assessment of initiating events as the basis for evaluating whether any of these initiating events would develop into an event sequence. The applicant further detailed the results of the MLD and HAZOP processes in BSC (2008bj, Attachments D and E). The applicant aggregated the individual initiating events into event sequence diagrams (ESDs) and described the results of this aggregation in BSC (2008bj, Section 6.2), where each resulting ESD was discussed. The applicant performed a screening analysis on the aggregated ESDs, as described in SAR Section 1.7.1.2 and in BSC (2008bk, Section 6.0). To quantify the probability of occurrence of the event sequence, the applicant used fault tree analyses, as described in BSC (2008bk, Sections 4.3.2 and 6.2.2 and Attachment B). The applicant also considered the human error and used the results of passive reliability analysis for equipment failure, as given in BSC (2008bk, Attachment D) in developing the initiating event analysis, as detailed in BSC (2008bk, Attachment E).

The applicant identified and quantified the initiating events of the subsurface operations following the approach used in operation of the surface facilities (reviewed in SER Sections 2.1.1.3.3.2.1 and 2.1.1.3.3.2.2). Therefore, the NRC staff's review for quantification of initiating event frequency for subsurface operations focused on how those methodologies were implemented to quantify the frequency of the initiating events for operations in the subsurface facilities. Consistent with the review approach discussed in SER Section 2.1.1.3.3, the NRC staff selected a subset of the initiating events for detailed review. These initiating events were selected based on their risk potential and uniqueness of waste package transportation from surface facilities to the emplacement areas in the subsurface using the transport and emplacement vehicle (TEV). These initiating events included “TEV Impact During Transit,” and “TEV Stops for an Extended Period of Time.” For these initiating events, the NRC staff examined the operational descriptions at the subsurface facility, design of the systems and components, and the scenario description, as given in BSC (2008bj,bk), to determine whether identification, characterization, and screening of the initiating events were conducted appropriately considering site-specific and facility information. The NRC staff’s review is presented below in terms of (i) use of system information, methodology, and data use; (ii) estimation of annual frequency; and (iii) technical basis for screening.

**Use of System Information and Methodology**

For the initiating event “TEV Impact During Transit,” the applicant indicated that collision with another object could take place if the TEV is a runaway while traversing the North Ramp, leading to a derailment and impact with the tunnel wall, or if the TEV collides with an object along the rail line, as outlined in the ESD SSO–ESD–02 in BSC (2008bj, Table 11) and BSC (2008bk, Section B1.4.4). Using this facility-specific information, the applicant developed the fault tree model for the initiating event. The applicant identified three potential failure modes: (i) another vehicle being driven into the TEV on the surface, (ii) uncontrolled descent of the TEV down the North Ramp resulting in an impact with the tunnel wall, and (iii) TEV impact with another object along the rail line due to either spurious signal from the drive controllers or failure of the manual control switch (BSC, 2008bk). The fault tree comprised three subfault trees, one for each failure mode.

The applicant modeled the initiating event “TEV Stops for an Extended Period of Time” in a fault tree, SHIELD–STOP, as described in BSC (2008bk, Section B1.4.5.4) due to motive failure resulting in temperature rise leading to TEV shielding degradation. Failure modes, represented in the fault tree, were (i) loss of offsite power, (ii) a local failure of the third rail power system, (iii) failure of the TEV onboard programmable controllers, and (iv) failure of the TEV motor’s speed sensor. Speed sensor failure was modeled as an OR gate of eight basic events, representing the speed sensor of each motor. The applicant used the alpha-factor method (Mosleh, et al., 1998aa) to analyze common-cause failure of the speed sensor of the motors. The alpha-factor method quantifies the failure rate of a system with several redundant components from a single fault, such as environmental, age-related failure event (common-cause failure).

In conducting the fault tree analysis for the initiating event, labeled as TRANSIT–IMPACT (BSC, 2008bk), the applicant used the reliability data for component failure from NPRD–95 (Denson, et al., 1994aa).

**NRC Staff’s Evaluation**

The NRC staff reviewed the applicant’s assessment of initiating events of “TEV Impact During Transit” and “TEV Stops for an Extended Period of Time” provided in SAR Table 1.6-3 and BSC (2008bk). The NRC staff finds that the fault tree model used by the applicant for assessing the “TEV Impact During Transit” initiating event is acceptable because it modeled the scenarios likely to be encountered based on the system and event descriptions. Additionally, the NRC staff finds that the applicant appropriately used the system information of the TEV and the information on the operating environment to construct the fault tree model for the initiating event "TEV Stops for an Extended Period of Time." The NRC staff also finds that the applicant's use of the alpha-factor method to analyze the common-cause failure scenarios is acceptable because this method is used extensively in reliability analysis to assess the common-cause failures (Mosleh, et al., 1998aa).

Additionally, the NRC staff finds that the information the applicant used on mechanical component reliability is acceptable because this information is from a reference (Denson, et al., 1994aa) widely used in reliability analysis. In addition, the NRC staff finds that the reliability data, including uncertainties, were selected appropriately for the components based on their functionality. As all possible failure modes for TRANSIT–IMPACT were considered in the analysis, the NRC staff finds that the applicant used appropriate information and data to identify initiating events. Therefore, the NRC staff finds that the applicant used the TEV system information and appropriate reliability data from the fault tree analysis to quantify the initiating event frequencies for subsurface operations.

**Estimation of Annual Frequency**

The applicant estimated the frequency of TEV impact during transit in the fault tree "TEV Impact During Transit" to be $3.03 \times 10^{-4}$ per year. The applicant stated that this frequency will be dominated by (99 percent of the contribution) an operator driving another vehicle into the TEV (the operator fails to yield to the TEV at a crossing on the surface). There are eight electric motors to prevent the TEV from overspeeding. To reduce the probability of TEV impact during transit, the applicant's design includes special crossing barricades and signals at all surface intersections. The applicant will restrict all traffic from the area of a loaded TEV in the subsurface facilities. Additionally, the TEV would travel slowly {roughly 3.2 km/hr [2 mph]}, and an operator will watch via camera, as outlined in Table E6.2-2 (BSC, 2008bk).

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant used to determine the annual frequency of occurrence of the initiating event "TEV Impact During Transit," provided in SAR Table 1.6-3 and BSC (2008bk). The associated fault tree model is provided in BSC (2008bk). The applicant used the alpha-factor method (Mosleh, et al., 1998aa) to model the common-cause failure of the speed sensors of the motors. The NRC staff finds that the applicant's conclusion that the frequency of TEV impact during transit will be primarily controlled by an operator driving another vehicle into it, which is approximately 99 percent of the contribution, is acceptable because (i) the human-induced hazard was appropriately identified and (ii) mechanical or electrical failure has a low probability of occurrence, as shown in the fault tree for the initiating event "TEV Impact During Transit" (BSC, 2008bk). Furthermore, the NRC staff finds that the applicant's proposed actions to reduce the probability of TEV impact during transit are appropriate because these actions include installation of special crossing barricades and signals at all surface intersections, restriction of all traffic within the same area of a loaded TEV in the subsurface facilities, the TEV travel speed limit {roughly 3.2 km/hr [2 mph]}, and activity monitoring by an operator via camera, as described in BSC (2008bk, Table E6.2-2). Therefore, the NRC staff finds that the applicant's estimated annual frequencies for the initiating events are acceptable because they were calculated using appropriate methods and reliability information.

**Technical Basis for Screening**

The applicant estimated the annual frequency of Transport and Emplacement Vehicle (TEV) impact during transit to be $3.03 \times 10^{-4}$, on the basis of a point estimate, as detailed in BSC (2008bk, Figure B1.4-7). The applicant also considered the uncertainties associated with the parameters. The applicant (BSC, 2008bk) assumed a lognormal distribution representing the uncertainties of the parameters. The estimated mean and standard deviation of the annual frequency of the initiating event were $2.94 \times 10^{-4}$ and $7.36 \times 10^{-4}$, respectively, as shown in BSC (2008bk, Figure B1.4-7). Consequently, this initiating event was included for event sequence analysis (SAR Table 1.6-3).

The applicant estimated approximately 8.5 occurrences of extended TEV stoppage during the preclosure period, as outlined in the Event Tree SSO–ESD–04 (BSC, 2008bk), but excluded SSO–ESD–04 on the basis of a zero probability for loss of shielding. This event was excluded because the applicant established a requirement that the shielding be designed to sustain the thermal loading for all waste package loadings for 30 days, without significant degradation of the shielding function (DOE, 2009ey). Additionally, the applicant stated that at the limiting waste package power output for emplacement [as per SAR Section 1.3.1.2.5 and Table 5.10-3, 18 kW per waste package for commercial spent nuclear fuel (CSNF) or 11.8 kW per waste package for naval spent nuclear fuel (SNF)], the probability of thermally induced shielding failure is negligible, as the calculated temperature at the steady state is less than the maximum operating temperature of the shielding materials to be used in the TEV. The applicant stated that it would include a layer of synthetic polymer (NS–4–FR, a solid borated hydrogenous synthetic polymer with neutron absorption capability similar to that of borated water) with a maximum continuous operating temperature of 150 °C [302 °F] in the TEV shielded enclosure, as discussed in BSC (2008bk, Section B1.2.4). Because the neutron shielding material NS–4–FR would degrade over time (DOE, 2009ey), the applicant stated that it would implement a preventive maintenance program, routinely assess the effectiveness of the shielding materials, and replace them as necessary.

**NRC Staff's Evaluation**

The NRC staff reviewed SAR Table 1.6-3 and BSC (2008bk) to determine whether the applicant provided an adequate technical basis for excluding initiating events "TEV Impact During Transit," and "TEV Stops for an Extended Period of Time." The NRC staff independently verified that the mean (i.e., the estimated annual frequency of TEV impact during transit) would be the same as the point estimate $3.03 \times 10^{-4}$ in BSC (2008bk, Figure B.14-7) if a sufficient number of Monte Carlo samples were taken.

The NRC staff finds that the applicant acceptably showed that extended TEV stoppage would not initiate an event sequence during the preclosure period, because the applicant would include a layer of synthetic polymer (NS–4–FR) with a maximum continuous operating temperature of 150 °C [302 °F] in the TEV shielded enclosure for shielding purposes. The NRC staff finds the choice of the synthetic polymer to coat the TEV enclosure is acceptable because this polymer can withstand high operating temperatures up to 150 °C [302 °F]. The NRC staff has issued a certificate of compliance (No. 9235) to the NAC-STC package equipped with the same polymeric material for shielding (Denson, et al., 1994aa). The NRC staff further notes that the applicant stated that the effectiveness of this polymeric material would be routinely assessed through the applicant's preventive maintenance program. The NRC staff's evaluation of the applicant's maintenance program is documented in SER Volume 4, Section 2.5.6.4, where the NRC staff found that the applicant has adequately described plans to conduct maintenance, surveillance, and periodic testing activities that would be implemented prior to receiving and possessing HLW at the GROA. Therefore, the NRC staff finds that the applicant acceptably characterized the potential hazards and initiating events at the subsurface facilities during operations.

In summary, the NRC staff finds that the applicant adequately used the site-specific and facility information (including the operating environment) in the analysis to identify and quantify the initiating event frequencies for subsurface operations. Additionally, the NRC staff finds that the technical bases the applicant used were acceptable because they were based on standard methods (e.g., fault tree, alpha-factor) for identifying and screening the initiating events. Therefore, the NRC staff finds that the applicant's screening process for subsurface initiating events is acceptable.

2.1.1.3.3.2.4 Quantification of Initiating Event Frequency for Fire Hazards

The applicant presented information with respect to potential fire hazards from operations at the repository facilities during the preclosure period in SAR Sections 1.6 and 1.7 and Attachment F in BSC (2008ac,as,au,be,bk,bq). SAR Table 1.6-3 identified fire as a potential initiating event in the repository facilities during the preclosure period. The NRC staff's review focus was to determine whether the applicant (i) used appropriate methodologies to assess operation-based fire hazards; (ii) applied the methodologies correctly, consistent with the site-specific data and system information; and (iii) quantified fire-initiating event frequencies appropriately.
The NRC staff's review followed guidance provided in YMRP Section 2.1.1.3.
SER Section 2.1.1.3.3.1.3.5 addresses the NRC staff's review of potential hazards from external fires (wildfires and vegetation).

**Methodologies Used by DOE**

The applicant identified fire as a potential initiating event and employed the probabilistic risk assessment methodology outlined by Science Applications International Corporation (SAIC, 2002aa) to assess the fire potential at the repository facilities. The applicant described the methodology in Attachment F of BSC (2008ac,as,au,be,bk,bq). In applying the methodology, the applicant determined (i) an overall ignition frequency for a particular facility or area, (ii) the likelihood that an ignition event would develop into definitive fire event sequences, and (iii) the likelihood that a developed fire would propagate from the area of origin to other areas of the facility.

The overall building or facility ignition frequency was derived using two different approaches and data sets, based on the facility. The overall ignition frequency within surface facilities (e.g., CRCF, IHF, RF, WHF, and LLWF) was derived on the basis of historical fire data from comparable industrial facilities, as described by Tillander (2004aa). This methodology was used by the applicant because buildings had discernible ignition sources and a distribution of those ignition sources within rooms or areas. The applicant used a scoring methodology, described in EPRI (2005aa), to distribute the facility ignition frequency to individual rooms, based on their content and operations. The product was an ignition frequency per room or area that could then be used to assign fire exposure probabilities to waste forms as they reside in specific rooms or as they are moved through a facility.

The ignition frequencies for the intrasite operations and waste storage areas (e.g., subsurface areas and aging pads) were determined by the applicant on a per-facility basis, using historical data from the U.S. Census Bureau (2000ab). This site-wide or building-wide ignition frequency was adopted for the intrasite and subsurface areas because the ignition sources that exist outside of the buildings are generic across the entire industrial facility, and, according to the applicant, a global ignition frequency was all that was needed to populate the event sequences.

**NRC Staff's Evaluation**

The NRC staff reviewed information the applicant provided in SAR Sections 1.2.1.2, 1.2.1.3, 1.6, and 1.7; BSC (2008ac,as,au,be,bk,bq); and references therein on the methodologies used to assess the fire potential at repository facilities. Specifically, the NRC staff reviewed SAIC (2002aa), EPRI (2005aa), and Tillander (2004aa) in the development of ignition frequencies at these facilities.

The NRC staff evaluated the appropriateness of the methods used to develop initiating events arising from fires that could affect operations at the waste handling facilities, intrasite operations, the Low Level Waste Facility (LLWF), and subsurface operations. The NRC staff examined the applicant's overall approach and subsequently selected specific facilities for detailed review. The fire analysis for the CRCF, as described in BSC (2008ac), was selected by the NRC staff for detailed review because (i) the activities at this facility adequately represented the activities at other waste handling facilities and (ii) the methodology used to derive fire-related initiating event frequencies was similar to the methodology used for the other waste handling facilities (i.e., IHF, LLWF, RF, and WHF). The NRC staff also selected the fire analysis for the Intrasite operations (BSC, 2008au) and subsurface facilities (BSC, 2008bk) for detailed review, as these two fire analyses represent a different analysis methodology, where the potential fire events could take place in distributed areas or no distinct activities could be attributed to initiate a fire event. As appropriate, the NRC staff reviewed the consistency among facility layouts described in SAR Section 1.2.1.2, operations described in SAR Section 1.2.1.3, and fire ignition frequency estimations described by BSC (2008ac,as,au,be,bk,bq).

The NRC staff finds the applicant's selected methodologies in SAIC (2002aa), EPRI (2005aa), and Tillander (2004aa) appropriate because they are based on a probabilistic risk assessment informed by historical fire data, and they produce ignition frequencies on a "per-room" basis. This is the most useful format for derivation of ignition frequencies in surface facilities because these frequencies can then be coupled with propagation probabilities to determine overall exposure frequencies in the event sequences. The NRC staff finds that the applicant showed that similarities exist in the process (both handling hazardous materials), operational characteristics, and fire vulnerabilities between the GROA and facilities handling hazardous chemicals. Therefore, the NRC staff finds it appropriate for the applicant to use the methodologies in SAIC (2002aa), EPRI (2005aa), and Tillander (2004aa) to assess fire-related hazards at the waste handling facilities.

The NRC staff also finds that the methodology to evaluate fire frequencies in the intrasite and subsurface areas is also appropriate because the methodology applies a uniform frequency, based on the overall operations taking place in these areas. Based on its review of the applicant's characterization of activities at the GROA, the NRC staff finds acceptable the applicant's conclusion that the operations within the intrasite and subsurface areas are typical of industrial operations (surface) and mining activities (subsurface), and the industrial nature of the operations is essentially the same over the entire area. The NRC staff finds that it is appropriate to treat the ignition frequency on a "per-site" basis rather than a "per-room" or "per-area" basis because there are no distinct hazardous activities that can be identified for the intrasite or subsurface operations (e.g., welding, grinding, unit heaters). The selected methodology produces ignition frequencies that can be suitably applied to the whole intrasite and subsurface event trees. Therefore, based on the preceding evaluation, the NRC staff finds that the methods selected by the applicant are appropriate for estimating the fire ignition frequencies at the surface waste handling facilities, and also at the subsurface and the Intrasite operations.

**Application of Data and Methodology**

The applicant estimated the number of expected fires annually in each of these facilities on a per-unit floor area (fires/unit area/year) basis using the data derived from industrial buildings having floor areas larger than 1,000 $m^2$ [10,764 $ft^2$], as Tillander (2004aa) reported. The applicant also estimated the confidence limits, as provided in Table F.III-2 of BSC (2008ac,as,au,be,bk,bq). The applicant multiplied the fire ignition frequency with the floor area and the assumed 50-year operating life of these facilities to obtain the overall ignition frequency of each facility over the preclosure period.

To quantify the annual frequency of fire ignitions that would result in an exposure event within the surface waste handling facilities (CRCF, IHF, RF, and WHF), the applicant distributed the overall facility ignition frequency to each room, on the basis of the number and types of ignition sources that would be present in the room. The applicant relied on ignition source data from Ahrens (2007aa), which described the likelihood that a fire would originate from a particular class of equipment (e.g., welders, motors, internal combustion engines), and a scoring methodology, described in EPRI (2005aa), to determine the ignition frequency on a per-room basis.

The fire propagation to adjacent rooms or areas was determined based on historical data from NFPA (2007ag). Room fire propagation probabilities that describe fire spread throughout similar radioactive material handling facilities and nuclear energy plants were used to determine whether ignition sources could result in fires that could challenge fire barriers or waste packages. The propagation probability was assigned to each potential ignition source. The propagation probability for that fire to spread to the waste form was evaluated based on whether the ignition began “at, “near,” or “away” from the waste form. Intuitively, robust structures with multiple passive boundaries would present a substantial decrease in propagation probability for any fire that starts farther from a waste form.

Large fire propagation was calculated by the applicant using a more simplified approach. This methodology assumed that a certain percentage of all ignitions would result in fires (roughly 17 percent) that are capable of spreading throughout the entire facility and exposing a waste form. The assumed ignition frequency for the facility was multiplied by the probability that a fire would propagate throughout the facility, and then multiplied by the target exposure time fraction representing the time the waste package was present in any part of the facility.

The applicant derived the ignition frequency for the intrasite and subsurface facilities using historical data the U.S. Census Bureau (2000ab) provided. The applicant divided the number of reported fires at industrial and chemical facilities from Ahrens (2000aa) by the estimated number of industrial and chemical facilities in operation, as reported by the U.S. Census Bureau (2000ab). This generated a probability of fire per facility. The applicant considered Intrasite operations and subsurface operations as separate facilities when determining the ignition frequency for each. The applicant then apportioned the overall frequency of fire to seven different categories (e.g., storage areas, trash areas, vehicle fires) on the basis of data from Ahrens (2000aa). To estimate the probability of fire affecting a waste form, the applicant assumed that three of the seven categories of fires (vehicle fires, fires in receiving areas, and fires in storage areas) could expose waste forms within the Intrasite operations area and only vehicle fires (e.g., fires originating from the TEV) will have sufficient potential to expose waste forms in the subsurface facility.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Sections 1.6 and 1.7, BSC (2008ac,as,au,be,bk,bq), and references therein, on the application of the selected methodologies to assess the fire potential at the repository facilities. The NRC staff reviewed how the three methodologies (i.e., SAIC, 2002aa; EPRI, 2005aa; Tillander, 2004aa) were applied to the CRCF, IHF, RF, WHF, and LLWF. In addition, the NRC staff reviewed information from the U.S. Census Bureau (2000ab) on the number of a given type of facility and from Ahrens (2007aa) on historical incidence of fires in radioactive material handling facilities. The NRC staff also reviewed Ahrens (2000aa) for fire data in industrial chemical, hazardous chemicals, and plastic manufacturing facilities as a corollary to the industrial operations and equipment planned for the GROA surface facilities.

To determine whether the applicant applied the fire hazard assessment methodologies appropriately, the NRC staff assessed whether (i) the overall facility fire frequency was evaluated correctly, (ii) the ignition frequency by ignition category was estimated appropriately, (iii) the ignition sources were appropriately distributed among the rooms within a facility, and (iv) the fire propagation analysis reasonably estimated the probability of fire affecting a waste form at a particular location within the facility.

The NRC staff finds that the applicant's estimation of fire-related facility initiating event frequencies is acceptable because the calculations are consistent with facility-specific information. In addition, the relationship between ignition frequency and facility floor area contained in Tillander (2004aa) is consistent with the types of operations being performed at the GROA. The NRC staff finds that use of Tillander (2004aa) yields a conservative ignition frequency when compared to historical U.S. data. Additionally, the NRC staff finds that the operations of lifting, welding, packaging, and transporting proposed by the applicant for waste handling operations are also common to other industrial facilities (e.g., heavy equipment manufacturing, materials processing) and that the material actually being handled at an industrial facility generally has little effect on the likelihood of an ignition. Therefore, the probability of ignition data from other industrial facilities can be appropriately used as a surrogate for operations and activities in the GROA facilities.

The NRC staff reviewed the applicant's classification of ignition frequency by ignition sources (e.g., number of fires attributed to welding equipment, electrical equipment, vehicles) and finds that the distribution of ignition frequency by equipment involved is acceptable because the distribution was developed based on historic fires observed at radioactive material handling facilities and nuclear power plants from 1980 through 1998, as documented in Ahrens (2007aa, Table 36). The NRC staff also concludes that use of information in Ahrens (2007aa) to distribute ignition frequency is appropriate, as the types of equipment present at the GROA facilities do not differ substantially from the types of equipment found in other radioactive material handling facilities and nuclear energy plants. The NRC staff finds that all of the ignition categories relevant to the GROA facilities were represented.

The NRC staff reviewed how the applicant distributed the overall building ignition frequency to each room of each facility and finds that ignition sources were acceptably distributed among the rooms within the facilities. The NRC staff finds that the methodology used to determine the ignition frequency in surface facilities was appropriate because the applicant (i) used an approach for ignition frequency distribution consistent with the methodologies proposed in NUREG/CR–6850 (EPRI, 2005aa) and SAIC (2002aa), which are industry-accepted methods for fire hazard assessment and (ii) used site-specific inventories of the types of equipment expected in each room to distribute the ignition frequency.

The NRC staff also reviewed how the applicant assessed the propagation probabilities of fire inside a waste handling facility. The NRC staff finds that the applicant estimated reasonably the probability of fire affecting a waste form at a particular location within the facility. The NRC staff's finding is based on the fact that the applicant used conservative approximations to adapt historical information on fire propagation from events at radioactive material handling facilities and nuclear energy plants of noncombustible construction, as documented in Ahrens (2000aa). The historical data presented in Ahrens (2007aa) were based on actual fire propagation, categorized by the maximum extent of flame travel inside a facility. The NRC staff finds that the applicant's analysis for estimating fire propagation probability is acceptable because the applicant used conservative assumptions regarding flame extent, automatic suppression, and passive fire protection. The NFPA data (Ahrens, 2007aa) do not provide information on the intensity of the fire when it reaches its maximum extent of flame travel. The applicant assumed that all fires would grow to sufficient intensity to become an initiating event, which represents an upper bound analysis and is conservative. Therefore, the NRC staff finds this assumption acceptable.

To determine whether the applicant reasonably applied the fire hazard assessment methodology for intrasite and subsurface operations, the NRC staff examined whether the applicant (i) reasonably estimated the frequency of fire-related initiating events and (ii) reasonably distributed the fires to functional areas within the GROA. The NRC staff finds that the applicant estimated the fire ignition frequency in a manner that is consistent with the approach outlined in SAIC (2002aa). The applicant estimated the fire ignition frequency using data of historical fire events found in U.S. Census Bureau (2000ab) data from chemical, plastic, or petroleum products plants (Category Codes 324, 325, and 3261) in conjunction with the SAIC (2002aa) methodology. In response to the NRC staff's RAI, the applicant provided acceptable justification for the use of the information on chemical, plastic, or petroleum products plants for the waste handling facilities (DOE, 2009fj) because the operations outside the waste facilities (in the intrasite and subsurface areas), associated with the use of site vehicles, movers, shipping/receiving, and transient storage at the GROA, would be similar to most industrial facilities. The applicant also estimated fire frequency using data provided in Ahrens (2000aa). The NRC staff finds that the number of fires, derived from Ahrens (2000aa, Section 5, Table 1) and used in the applicant's analyses, is conservative because several of the fires documented in the Ahrens table occurred in areas that would not be present at the GROA (e.g., fires on highways or public streets, incinerator areas, and attic/concealed spaces would not be credible at the GROA).

The NRC staff independently assessed whether the applicant's use of information on chemical, plastics, or petroleum products facilities would be applicable to the subsurface repository facilities. The NRC staff performed an independent analysis using a separate data set for comparison of operations conducted in noncoal mines from the U.S. Census Bureau (1997aa), as follows. The noncoal mine category was selected because the subsurface facilities will have many operations similar to a noncoal mine. On the basis of an analysis of metal/nonmetal mine fires (De Rosa, 2004aa), there were 144 fires from 1991 through 2001 in the United States. This translates to approximately 13 noncoal mine fires annually. According to the U.S. Census Bureau (1997aa), there were 5,849 operating mines in the United States (Codes 2122 Metal

Ore Mining and 2123 Nonmetallic Mineral Mining and Quarrying) in 1997. Therefore, the frequency of potentially significant fires in these facilities would be $2.2 \times 10^{-3}$ fires/facility-yr. Comparing this frequency with the $1.1 \times 10^{-2}$ fires/facility-yr frequency derived in the applicant's analysis (BSC, 2008bk), the NRC staff finds that, by using the data for chemical, plastics, or petroleum products facilities, the applicant's assessment in BSC (2008bk) provided a conservative estimate of annual facility fire frequency for subsurface fire events.

The NRC staff finds that the applicant used site-specific data and system information appropriately in its evaluation of overall ignition frequencies at the intrasite and subsurface areas because the data sets are consistent with the activities expected in the GROA. The NRC staff finds that the applicant's application of the methodologies for assessing the fire potential in the intrasite and subsurface areas is acceptable because the applicant conservatively assumed (i) a waste form was always present in the subsurface facility, on the aging pad, in the buffer area, and/or on a transportation vehicle and (ii) all fires recorded in the historical data set would be large enough to serve as credible initiating events.

**Initiating Event Frequencies**

The applicant used site-specific information regarding the quantity and types of equipment in each room to assign room fire ignition probabilities. After assessing the ignition frequency for each room, the applicant used fire growth data from Ahrens (2007aa) to evaluate the likelihood of fire propagating from the room or area of origin to adjacent rooms or areas. The applicant quantified the initiating event frequencies for waste forms by evaluating the potential location of various waste forms in each building. By using a compilation of ignition and propagation probabilities, the applicant determined which fires had the potential of reaching a waste form and producing an initiating event. The estimation process also considered the residence time of a waste form in each room or area to assess the likelihood that a waste form will be present in various parts of the building during a fire. The applicant did not credit the performance of any passive-fire-resistance-rated wall assemblies within the facility. The applicant also assumed no benefit from automatic suppression systems and applied fire propagation data for fires where no sprinkler system was present or the system failed to operate. The applicant summarized the results of these analyses in SAR Section 1.7.1.2.2.

The applicant developed an uncertainty distribution for the ignition frequency, a conditional probability based on the extent of flame damage, using data from Ahrens (2007aa), and a categorization of ignition sources by types of equipment, using data from Ahrens (2007aa). The selected distribution for ignition frequency was lognormal, and a normal distribution was selected for flame propagation and ignition source type frequencies. Additionally, the applicant conducted Monte Carlo simulations with 10,000 samples for each initiating event to estimate mean, standard deviation, and maximum and minimum values, using the Crystal Ball software.

The applicant dealt with "large fires" in surface facilities as separate events. These large-fire events were based on the Ahrens (2007aa) data that showed in roughly 16.9 percent of the recorded fires in radioactive material handling facilities and nuclear energy plants, flame propagated throughout the entire floor of origin or beyond.

For the intrasite and subsurface facilities, the applicant assumed that the ignition event originating in each of these facilities would be sufficient to serve as an initiating event to expose waste forms in the facility and did not reduce the frequency on the basis of waste form residence times or fire propagation probabilities. These resulting ignition frequencies were directly used by the applicant as the initiating event sequences in the event sequence analysis (reviewed in SER Section 2.1.1.4.3.1.3).

For fires in the intrasite and subsurface areas, the applicant assumed that a waste form is always present in the subsurface facility, on the aging pad, in the buffer area, and on a transportation vehicle. Additionally, all fires recorded in the historical data set were assumed large enough to serve as a credible initiating event. Consequently, there are no propagation or residence time probabilities associated with fires in these areas. Any fire that originates on the aging pad, in the subsurface facility, or on the TEV during use is assumed to affect a waste form nearby and is also assumed to be large enough to be considered an initiating event. The applicant assigned an error factor of 15 for the fire-initiating event frequencies and associated distributions at the intrasite and subsurface areas to account for uncertainties (e.g., under reporting, inaccurate counting) inherent in the Ahrens (2007aa) and the U.S. Census Bureau (2000ab) data sources (BSC, 2008au,bk).

**NRC Staff's Evaluation**

The NRC staff reviewed information provided in SAR Section 1.7.1.2.2, and references therein, on quantification of fire event frequencies. The NRC staff finds that the applicant's estimation of initiating event frequencies for waste forms in various facility locations is acceptable. This finding is based on the applicant's use of facility-specific data that the NRC staff determined is acceptable, as discussed in SER Section 2.1.1.4.3.1.3.

The NRC staff notes that the information provided in Ahrens (2007aa), used by the applicant in the development of large fire-initiating events, did not state whether the fires that propagated throughout the entire floor of origin or beyond would be capable of breaching multiple 3-hour rated fire barriers, nor did the data indicate the level of intensity the fire had during its progression. As a result, the applicant assumed all of these fires were of sufficient intensity to breach barriers and have sufficient intensity to then expose a waste form. Therefore, the NRC staff finds that the applicant's assumption that these fires would have sufficient intensity to affect a waste package in an adjacent fire area is acceptable because this scenario represents a conservative upper bound condition.

The NRC staff finds that the applicant accounted for the large fire contribution to the initiating event frequency for a particular waste form on the basis of the overall building ignition frequency, multiplied by the propagation frequency [16.9 percent found in Ahrens (2007aa)], and then multiplied by a residence time fraction for a particular waste form within the building. The NRC staff finds that this is an acceptable approach to estimate the fire-initiating event frequency because it uses applicable data from other nuclear facilities and power plants (Ahrens, 2007aa) and uses a propagation probability (16.9 percent of fires will grow to a point where they propagate through the entire facility). The applicant's propagation probability value is conservative because it assumes that all propagating fires are sufficiently intense to consume the facility, rather than assuming that some propagating fires will cause minimal damage, such as smoke damage.

Unlike the fault tree analysis approach used for surface facilities, where ignition frequencies were coupled with propagation probabilities and coupled with the residence times of particular waste forms, the applicant's aging pad and subsurface initiating event determinations did not include any throughput or propagation probabilities. The NRC staff finds that these assumptions are conservative because the applicant assumed (i) waste forms would always be present during potential fire conditions and (ii) fires in these areas would always be of sufficient intensity to affect waste forms.

The NRC staff reviewed the statistical distributions for the parameters used for the fire-initiating event frequencies for the intrasite and subsurface operations and reviewed estimates of the error factor for these distributions. The applicant assumed an error factor of 15 for all fire-initiating event frequencies in subsurface and Intrasite operations (BSC, 2008au,bk) because two different databases [i.e., Ahrens (2007aa) and the U.S. Census Bureau (2000ab)] were used to estimate the fire-initiating frequencies. The NRC staff finds this large error factor is acceptable because it accounts for the variation between the two databases. The error factor is defined as the ratio between the 95$^{th}$ percentile to the median (or the ratio of the median to the 5$^{th}$ percentile due to its logarithmic symmetry) for a lognormal distribution. An error factor of 15 means that there is a 225× spread (15 × 15) between the 95$^{th}$ percentile and the 5$^{th}$ percentile for the fire-initiating event frequency distributions. Additionally, the NRC staff finds that the applicant's justification of using an error factor of 2.0 for LLWF (DOE, 2009fj) is acceptable because the data from Tillander (2004aa) for floor areas between 2,500 and 32,000 m$^2$ [26,910 and 34,445 ft$^2$] show an error factor of 1.8. Therefore, the NRC staff finds that the applicant acceptably quantified uncertainties and included appropriate conservatism.

In summary, the NRC staff finds that the applicant used appropriate methods to assess potential initiating events associated with fires at the repository facilities because the applicant (i) quantified the frequencies of fire-related initiating events adequately and (ii) made several conservative assumptions regarding the initiation, the intensity, and the extent of fire propagation. Therefore, the NRC staff finds that the applicant's technical basis for its assessment of fire-related initiating events is acceptable.

2.1.1.3.3.2.5 Screening of Initiating Events Related to Internal Flood Hazards

The applicant identified internal flooding initiating events and provided its technical bases for excluding internal flooding from the PCSA analyses. For the surface facilities, the applicant identified (i) the potential for internal flooding caused by actuation of the fire protection system and piping or valve failure in SAR Table 1.6-3 and (ii) the potential for subsurface flooding due to a construction-related accident (e.g., water supply piping to the Tunnel Boring Machine) in SAR Section 1.6.3.5. For both the surface and subsurface facilities, the applicant excluded internal flooding as an initiating event. However, the applicant stated in SAR Section 1.7.1.2.3 that moderator entering a breached waste container and contributing to the pivotal event of an event sequence was considered. This pivotal event is evaluated in SER Section 2.1.1.4.3.3.2.2 under moderator intrusion control.

The applicant excluded internal flooding in the surface facilities on the basis of design features as follows (SAR Section 1.7.1.2.3; DOE, 2009fn): (i) a waste form container exposed to water will not lose its structural integrity or shielding capability; (ii) ITS components would be located so they would not be wetted or submerged; and (iii) local barriers would be used, and components would be designed and qualified for operation in a wetted or submerged environment if they would be used in such environments. In SAR Section 1.7.1.2.3, the applicant addressed the potential for criticality by identifying that there will be no water sources sufficient for decreasing boron concentration in the WHF pool to a level that criticality would be a concern. The applicant also referred to discussions in SAR Sections 1.14.2.3.3.1.4, 1.14.2.3.2.1.5, 1.14.2.3.2.3.4, and 1.14.2.3.2.3.5 with regard to the criticality potential for a sealed canister surrounded by water.

As discussed in its response to the NRC staff's RAI (DOE, 2009fm) and in BSC (2008bk, Section 6.0.4), the applicant excluded internal flooding for the subsurface operations because there would not be sufficient volume of water to rise to the level of the emplacement drift and to contact a waste package on the TEV if there were an accident involving the water supply to subsurface machinery. Additionally, the applicant excluded internal flooding based on the TEV design because the floodwater would neither adversely affect the waste package, resulting in degradation of TEV shielding, nor result in release of radionuclides.

**NRC Staff's Evaluation**

The NRC staff reviewed the information provided in SAR Sections 1.14.2.3.3.1.4, 1.14.2.3.2.1.5, 1.14.2.3.2.3.4, 1.14.2.3.2.3.5, and 1.7.1.2.3; BSC (2008bk, Section 6.0.4); and responses to the NRC staff's RAIs (DOE, 2009fn) on identification of internal flooding and the technical bases for exclusion as initiators of event sequences in the PCSA. The NRC staff reviewed whether (i) the systems and components in both surface and subsurface facilities that may be affected were identified and (ii) adequate technical bases were provided to exclude the internal flooding-related events from further consideration in the PCSA.

The NRC staff finds that the applicant reasonably excluded internal flooding in the surface facilities because (i) casks, canisters, and waste packages would be sufficiently robust to protect the waste form against exposure to water (DOE, 2009fn); (ii) canister transfer machine ITS interlocks would be designed for environmental conditions involving water spray (DOE, 2009fn) (see additional discussion in SER Section 2.1.1.6.3.1.); and (iii) as determined in SER Section 2.1.1.3.3.2.6, criticality is not a concern for internal flooding.

On the basis of its review of SAR Section 1.7.1.2.3 and the applicant's response to the NRC staff's RAI (DOE, 2009fm), the NRC staff finds that the applicant provided sufficient technical bases to exclude internal flooding in the subsurface facilities for the following reasons: First, the NRC staff determines that the water level would not reach the elevation of a waste package in an emplacement drift based on the amount of water available and the layout of the subsurface facilities. Additionally, the NRC staff finds that the water level would not reach the level of a waste package being transported in the TEV, based on the information and calculations provided in DOE (2009fm). The NRC staff further finds that the floodwater would not degrade TEV shielding, and the TEV has sufficient weight and structural rigidity such that waterborne debris would not affect the TEV (DOE, 2009fm).

2.1.1.3.3.2.6 Screening and Quantification of Initiating Event Frequency for Criticality Hazards

The applicant provided information with respect to criticality hazards at the repository facilities during the preclosure period in SAR Sections 1.6.1.6, 1.7, and 1.14; BSC (2008ba,bq); and responses to the NRC staff's RAIs (DOE, 2009dy,ey). The applicant provided an overview of its criticality safety analysis process in SAR Section 1.6.1.6.

To show that waste forms will remain subcritical during the preclosure period and, thus, exclude all criticality-related initiating events during the preclosure period, the applicant identified seven parameters as important to criticality in SAR Table 1.14-2. For each parameter, the applicant performed criticality sensitivity analyses to evaluate the impact on reactivity caused by variations in those parameters. In SAR Section 1.14, the applicant summarized results from these analyses. These criticality sensitivity analyses showed which parameters will (i) need to be controlled, (ii) not need to be controlled, or (iii) need to be conditionally controlled (i.e., need to be controlled if another parameter is not controlled).

Based on the hazard identification and screening analyses described in SAR Section 1.6 and on the event sequence development and quantification in SAR Section 1.7, the applicant identified, developed, quantified, and categorized event sequences that would impact the criticality control parameters and the parameters that needed to be controlled. These event sequences were referred to as event sequences important to criticality and were summarized in SAR Section 1.7.

Because the PCSA was performed in conjunction with the design process, if an initial criticality calculation resulted in exceeding the upper subcritical limit, the design was modified or procedural safety controls (PSCs) were employed to prevent such event sequences. The applicant stated that potentially critical configurations that could occur without a breach or require the introduction of a moderator were accounted for in the sensitivity calculations and the screening process described in SAR Section 1.14.2.3.2.

To identify the initiating events from GROA facilities, the applicant used the MLD method supplemented by a HAZOP evaluation (e.g., BSC, 2008bo). The included initiating events were listed in SAR Table 1.6-3, and those excluded were listed in SAR Table 1.7-1. The applicant stated in DOE (2009ey, Enclosure 8, Section 1) that there were no Category 1 or Category 2 event sequences that required crediting fixed-neutron absorbers. Therefore, material-selection errors during manufacturing of fixed-neutron absorbers did not need to be considered for preclosure criticality safety.

The applicant excluded neutron interaction between more than two naval canisters in the IHF by crediting the design of the mechanical handling capabilities of the IHF (SAR Table 1.7-1), because the handling equipment in the IHF would prevent a configuration that might result in interaction among more than two naval canisters, as described in DOE (2009ey, Enclosure 9, Section 1.1). The applicant provided the technical basis for excluding the neutronic interaction of more than four DOE SNF casks/canisters by referencing criticality calculations (BSC, 2008cm). These calculations showed that interaction of casks/canisters does not need to be considered, except for a few types of DOE SNF groups for which interaction was excluded by relying on a combination of human actions and design solutions, as shown in SAR Table 1.7-1.

Boron dilution was excluded in the HAZOP evaluation and MLDs as an initiating event, because boron dilution, in the absence of other independent initiating events, will not initiate a sequence of events that could potentially lead to a criticality, as outlined in DOE (2009dy, Enclosure 5, Section 1.2.1). The applicant developed PSC–9 to ensure sufficient concentrations of enriched boron in the pool. The applicant stated that this control will provide the initial conditions (high concentrations of soluble boron) in the WHF pool to ensure that a critical configuration could not be created in the pool (SAR Table 1.9-10). The applicant compared the amount of boron required by PSC-9 with the boron concentration fraction needed to maintain subcriticality during normal conditions and representative accident sequences, and calculated that subcriticality can be maintained for these sequences with 15 to 53 percent (normal and accident conditions, respectively) of the boron in the pool, as discussed in DOE (2009dy, Enclosure 5, Section 1.2.2-5). In order to dilute the boron concentration to levels resulting in a boron concentration fraction less than 15 percent, greater than 7 million gallons of nonborated water would have to mix with the pool borated water. The total volume of nonborated water available to the WHF is less than 1.5 million gallons, as discussed in DOE (2009dy, Enclosure 5, Section 1.2.1). The applicant stated that even if enough unborated water were to fill the pool to the brim, the concentration fraction would remain above 91 percent, as described in DOE (2009dy, Enclosure 5, Section 1.2.2). The applicant stated that the only sources of unborated water large enough to fill the pool to overflowing were the fire suppression system and the potable water system, as described in DOE (2009dy, Enclosure 5, Section 1.1). The applicant stated that neither of these systems will be connected directly to the pool or pool piping, and the only flow path will be through runoff into the pool. Therefore, when the pool is full, water flow will follow the path of least resistance away from the pool. In DOE (2009dy, Enclosure 5, Section 1.3.3), the applicant stated that procured soluble boron would be accompanied by the necessary material data sheets, and each shipment would be tested upon receipt to verify its enrichment.

The applicant excluded boron dilution/moderator introduction in the HAZOP and MLD evaluations of the DPC fill water, because the boron dilution initiating event was excluded (SAR Table 1.7-1). The applicant stated that the DPC fill water will be drawn from the pool water via the borated water treatment system. Other water sources will not be connected to the fill water piping. Therefore, the only source of water available for DPC fill operations will be the pool borated water, as described in DOE (2009dy, Section 1.2.4).

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant provided in SAR Sections 1.6, 1.7, and 1.14; BSC (2008ba,bq); supporting documents (BSC, 2008ai,bj,bo,cm); references therein; and responses to the NRC staff's RAIs (DOE, 2009dy,ey) to assess the criticality-related hazards at the repository facilities during the preclosure period.

The NRC staff evaluated the applicant's technical basis to assess the criticality-related hazards originated outside the GROA. The NRC staff notes that the main type of external event that could impact preclosure criticality would be an error resulting in an absence of neutron absorbers. The NRC staff finds the applicant's technical basis for excluding neutron absorber manufacturing errors acceptable because this exclusion is supported by criticality analyses, which show that lack of neutron absorber plates will not cause a criticality if the pool boron concentration and enrichment is maintained, as detailed in BSC (2008cm, Section 6.3.4).

The NRC staff examined the methodology the applicant used to assess criticality-related initiating events at the IHF and the interaction among more than two naval canisters in the IHF, which was excluded from further consideration in the PCSA by crediting the design of the mechanical handling capabilities of the IHF. The NRC staff finds that this approach is acceptable because the handling equipment within the IHF would physically preclude configurations that would result in interaction among more than two naval canisters and is in accordance with industry practices. The NRC staff reviewed the technical basis for excluding criticality, resulting from the interaction of more than four DOE SNF canisters at the CRCF, as initiating events. The NRC staff finds the applicant's exclusion of criticality as an initiating event acceptable because it is supported by criticality analyses (discussed in SAR Section 1.14.2.3.2.3.4). These analyses show that a criticality event could only occur for a combination of "worst case" configurations (most reactive canisters, most reactive reflectors, and close-packed configuration) that the applicant will prevent from occurring by relying on a combination of human actions and design solutions. The NRC staff finds crediting physical prevention of critical configurations and human actions to be acceptable because it is consistent with standard industry practice (American Nuclear Society, 2007aa).

The NRC staff also examined the rationale the applicant used to exclude the boron-dilution-related criticality initiating events at the WHF. The NRC staff finds the applicant's justification to exclude boron dilution acceptable because the applicant stated that, even if the pool were to be flooded with nonborated water such that it is filled to overflowing, the resulting boron concentration would still be greater than 91 percent, which is higher than the concentration required for normal operation (<15 percent) and accidents (<53 percent for seismic conditions) (DOE, 2009dy). The NRC staff also finds that excluding soluble boron underenrichment through testing the boron when it is received is acceptable because testing for boron provides an independent step to verify that the correct boron enrichment is received.

The NRC staff also finds that the applicant acceptably excluded the introduction of nonborated water into a DPC because (i) the applicant stated that it will use physical controls to prevent the DPC fill water piping from being connected to nonborated water sources and (ii) boron dilution was excluded (DOE, 2009dy, Enclosure 5).

**NRC Staff's Conclusion on Operational (Internal) Hazards and Initiating Events**

On the basis of the NRC staff's evaluation of the applicant's information on identification of operational (internal) hazards and initiating events, as described in this subsection, the NRC staff concludes, with reasonable assurance, that the regulatory requirements of 10 CFR 63.112(b) and 10 CFR 63.112(d), with regard to internal hazards, are satisfied. The applicant has adequately identified potential internal hazards and initiating events for surface and subsurface operations at GROA, consistent with all modes of operations, including potential fire, flooding, and criticality hazards. The associated probabilities of occurrences of internal hazards and initiating events, including adequate consideration of uncertainties, have been provided based on adequate technical bases, using methodologies consistent with standard industry practice, and considering human reliability analyses.

## 2.1.1.3.4 Evaluation Findings

The NRC staff reviewed the applicant's SAR and other information submitted in support of the license application and concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(b) and 10 CFR 63.112(d) are met, subject to the proposed conditions of the construction authorization enumerated below. Specifically, the NRC staff finds that

- The applicant adequately identified and provided systematic analysis of the naturally occurring and human-induced hazards and potential initiating events, including associated probabilities of occurrence.

- The applicant provided adequate technical basis for either inclusion or exclusion of specific naturally occurring or human-induced hazards and initiating events in the PCSA.

**Proposed Condition of Construction Authorization**

DOE shall provide the NRC staff written notification that the agreements for the six flight restrictions and operational constraints that DOE credits in its frequency analysis (SAR Section 1.6.3.4.1) are in place before commencement of construction, to confirm that the technical bases for exclusion of aircraft crash hazards at the GROA from the Preclosure Safety Analysis (PCSA) that DOE provided in accordance with 10 CFR 63.112(d) remain valid. These restrictions and operational constraints are (i) prohibiting fixed-wing flights below 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (ii) 1,000-overflight limit per year for fixed-wing aircraft above 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (iii) overflights are limited to straight and level flights (i.e., maneuvering is not permitted) ; (iv) carrying ordnance is prohibited within 9 km [5.6 mi] of the North Portal; (v) electronic jamming activities are prohibited within 9 km [5.6 mi] of the North Portal; and (vi) helicopters are not permitted within 0.8 km [0.5 mi] of facilities that process, stage, or age nuclear waste forms.

#### 2.1.1.3.5 References

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# CHAPTER 4

## 2.1.1.4 Identification of Event Sequences

### 2.1.1.4.1 Introduction

This Safety Evaluation Report (SER) section contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's ("DOE" or "applicant") information on identification of event sequences for the preclosure safety analysis (PCSA). The objective of the review is to assess DOE's technical bases for developing, quantifying, and categorizing event sequences used in the PCSA. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.7 (DOE, 2008ab), supplemental documents referenced in the SAR, and information the applicant provided in response to the NRC staff's requests for additional information (RAIs) (DOE, 2009dn,dq,dx,dz,ed,ej,fg,fk,fl,fr,ft–fz,ga–gi,gw).

The evaluation presented in this chapter considers information reviewed in other Safety Evaluation Report (SER) sections: (i) site description in SER Section 2.1.1.1; (ii) the description of facility, structures, systems, and components (SSCs), operational process, and throughput analysis in SER Section 2.1.1.2; (iii) identification of hazards and initiating events in SER Section 2.1.1.3; and (iv) design of SSCs important to safety (ITS) in SER Section 2.1.1.7. The output from this chapter includes event sequences and their associated categorizations that will be used in SER Section 2.1.1.5 to assess compliance with preclosure performance objectives and SER Section 2.1.1.6 for identifications of SSCs important to safety (ITS).

### 2.1.1.4.2 Regulatory Requirements

The regulations at 10 CFR 63.112(b) require that the applicant's preclosure safety analysis of the geologic repository operations area include an identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences. An *event sequence*, as defined in 10 CFR 63.2, means a series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of personnel. Those event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Other event sequences that have at least 1 chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences.

The NRC staff reviewed the applicant's identification and categorization of event sequences using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.4 (NRC, 2003aa). The relevant acceptance criteria in YMRP Section 2.1.1.4.3 are as follows:

- Adequate technical basis and justification are provided for the methodology used and assumptions made to identify preclosure safety analysis event sequences.

- Categories 1 and 2 event sequences are adequately identified.

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA. The applicability of such NRC guidance is discussed in greater detail in the sections where they were used as part of the application or the NRC staff's review.

### 2.1.1.4.3 Technical Review

The NRC staff's review focused on evaluating technical bases and justification for methods, assumptions, and site-specific data used by DOE to identify event sequences. The NRC staff assessed whether event sequence development is based on consideration of relevant operational and site-specific natural hazards, reasonable combinations of initiating events, and consistency with the facility description. The NRC staff also evaluated whether the reliability of the SSCs used to prevent or mitigate event sequences is consistent with the design information. In addition, the NRC staff reviewed whether the quantification of probability of occurrences of the event sequences and the categorization of event sequences are reasonable. As described in SER Section 2.1.1.4.2, event sequences are categorized in the PCSA as Category 1 and Category 2 event sequences, according to the likelihood the event will occur before permanent closure. Category 1 event sequences are those event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area (i.e., during the preclosure period). Category 2 event sequences are not expected to occur during the preclosure period but have at least 1 chance in 10,000 of occurring before permanent closure. DOE has expressed the Category 2 limit of a 1 in 10,000 chance of occurring during the preclosure period with annual probabilities and a time period that represents the same likelihood (e.g., an annual frequency of occurrence of $1 \times 10^{-6}$ over a 100-year-preclosure period and an annual frequency of occurrence of $2 \times 10^{-6}$ over a 50-year-exposure time for specific activities during the preclosure period).

The applicant's identification of event sequences stems from the identification of naturally occurring and human-induced internal and external hazards (SAR Figure 1.7-1). The applicant developed a list of internal and external events in SAR Section 1.6. The NRC staff evaluated, in SER Section 2.1.1.3 DOE's identification of hazards and initiating events and the associated frequency of occurrences for event sequence analyses. Evaluation of event sequence development, quantification, and categorization in this chapter relies on the frequency of occurrence of initiating events reviewed in SER Section 2.1.1.3.

On the basis of the initiating events identified for event sequence analysis, the review in SER Section 2.1.1.4 addresses three broad categories of events: (i) internal events caused by operational hazards encompassing random component failure or human error or both, (ii) seismically initiated events, and (iii) fire-initiated events within the GROA. The NRC staff's evaluation is presented in the following four main sections: (i) methodology for identification and categorization (SER Section 2.1.1.4.3.1); (ii) event sequence development (SER Section 2.1.1.4.3.2); (iii) reliability of SSCs (SER Section 2.1.1.4.3.3); and (iv) event sequence quantification and categorization (SER Section 2.1.1.4.3.4).

#### 2.1.1.4.3.1 Methodology for Identification and Categorization of Event Sequences

This section describes the NRC staff's review of the methodology the applicant described in SAR Sections 1.7.1 and 1.7.5 to identify and categorize event sequences for the PCSA. The NRC staff evaluated the applicant's methodology for identification of event sequences using the guidance in YMRP Section 2.1.1.4, as supplemented by High-Level Waste Repository Safety

(HLWRS)–Interim Staff Guidance (ISG), HLWRS–ISG–01 (NRC, 2006ad) and HLWRS–ISG–02 (NRC, 2007ab). The NRC staff reviewed the applicant's event sequence development analyses reports (BSC, 2008ab,ao,at,bd,bj,bo), and the applicant's event sequence reliability and categorization documents (BSC, 2008ac,as,au,be,bk,bq; BSC, 2009ab,ac), which contained supporting information on methodology for event sequence development and quantification of frequencies of occurrence for various GROA facilities.

The NRC staff focused its review to determine if (i) the methodology for developing event sequences is appropriate, (ii) the applicant selected appropriate modeling methods for event sequence development and quantification, and (iii) the methodology for categorization for event sequences is acceptable. The review to determine the acceptability of the overall methodology and selection of appropriate modeling methods is described in the following subsections: (i) internal events (SER Section 2.1.1.4.3.1.1), (ii) seismically initiated events (SER Section 2.1.1.4.3.1.2), and (iii) fire-initiated events within the GROA (SER Section 2.1.1.4.3.1.3). SER Section 2.1.1.4.3.1.4 describes the NRC staff's review of the applicant's event sequence categorization methodology.

2.1.1.4.3.1.1 Internal Events

In SAR Section 1.7.1, the applicant described the methodology for identification and categorization of event sequences for internal (also called operations-related) events initiated by random failure of equipment or human errors during preclosure operations. The applicant used the American Society of Mechanical Engineers standard RA–S–2002 (ASME, 2005ad) for developing the internal event sequences. The applicant's use of event sequence diagrams (ESD) and models for the identification and quantification of event sequences is illustrated in SAR Figures 1.7-2 to 1.7-5. These figures collectively show the applicant's approach for considering operations-related initiating events before permanent closure and their progression leading to potential consequences or end states. The applicant developed ESDs for each internal hazard (e.g., structural, mechanical challenges), showing (i) initiating events or groups of initiating events caused by random failure of equipment or human error and (ii) the sequence of responses to the failure of SSCs providing containment, shielding, confinement, and criticality control functions. The end states of each event sequence in an ESD is associated with potential radiological consequences, such as filtered and unfiltered radiological release to the public, and direct exposure to workers. The applicant developed internal event ESDs specific to each facility and operations (ESDs include initiating events, pivotal events, and end states).

The applicant identified event sequences for each type of waste form configuration in each facility [Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), Wet Handling Facility (WHF), and other operational areas (e.g., subsurface and intrasite)]. The type of waste form configurations, discussed in SAR Section 1.7.1, consists of commercial spent nuclear fuel, DOE SNF, high-level radioactive waste, and naval SNF. The applicant considered waste form configurations in the identification of event sequences for receiving and handling specific types of containers. For example, the commercial spent nuclear fuel is evaluated in TAD canisters, dual-purpose canisters, or uncanistered transportation casks. The DOE SNF, HLW, and naval SNF are evaluated in their respective canisters. The canisters are handled during the preclosure period either by themselves or in a waste package, a transportation cask, or a site transfer cask. In addition, waste form configurations also include SNF assemblies handled in the WHF and low-level waste generated by waste handling activities in the GROA. An event sequence is evaluated by DOE for a given waste form configuration, and the quantification of frequency of occurrence for the event sequence (i.e., number of occurrences) is directly related to the number of waste containers (throughput) associated with the specific waste form configuration. The throughput of containers for each waste form for the relevant operations and operational areas of the GROA are given in SAR Table 1.7-5.

In SAR Section 1.7, the applicant described its use of event trees, including the use of an Initiator Event Tree and System Response Event Tree to quantify the likelihood of the event sequences. The Initiator Event Tree (SAR Figure 1.7-4) consists of multiple initiating events associated with an ESD and accounts for the number of operations associated with each type of waste form container over the preclosure period. The progression of each initiating event in SAR Figure 1.7-4 was delineated through the System Response Event Trees (SAR Figure 1.7-5), which consists of one or more pivotal events. Each pivotal event represents either success or failure of individual SSCs that are relied on to prevent or mitigate event sequences. The applicant modeled the failure and success of the containment, shielding, confinement, and criticality control functions of the SSCs in the pivotal events as a response to the initiating events. Each branch of the system response tree represents an event sequence terminated in an end state. The end state identifies the event consequence resulting from the event sequence, such as direct exposure, or release of radionuclides with or without criticality potential.

For quantification of internal event sequences, the applicant modeled the Initiator Event Trees and System Response Event Trees using the SAPHIRE computer software (Version 7.26). The fault trees modeled initiating events and the pivotal events, and all fault trees were linked to event trees at the pivotal nodes in the SAPHIRE models. For reliability of active systems (e.g., canister transfer machine; heating, ventilation, and cooling system), the applicant incorporated failure rates and uncertainties into the fault tree when modeling component failures and human errors (SAR Figure 1.7-8). For passive systems, engineering calculations were performed to estimate the passive reliability and were used as input to the pivotal events through a fault tree. The applicant propagated the uncertainties in SAPHIRE software using the Monte Carlo simulation technique to estimate event sequence probabilities. For categorization of an event sequence related to a specific type of waste form in an ESD, the applicant aggregated the event sequences that resulted in the same end state from different initiating events. The applicant used the mean value of the sequence probability distribution for categorization of each event sequence. The applicant stated the expected number of occurrences of an event sequence over the preclosure period is compared to the criteria in 10 CFR 63.2 to categorize the event sequence as Category 1, Category 2, or beyond a Category 2 event sequence (SAR page 1.7-7).

**NRC Staff's Evaluation**

The NRC staff reviewed the methodology for identification and categorization of event sequences for internal events using the guidance in YMRP Section 2.1.1.4. The NRC staff finds that the applicant's methodology is appropriate for the identification of internal event sequences because the applicant's methodology (i) uses applicable standards in ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (ASME, 2005ad) for developing the internal event sequences for activities that are similar to activities at nuclear power plants (e.g., handling); (ii) uses ESDs to describe the event sequences in terms of the internal initiating events (i.e., random failure of equipment or human error), the sequence of responses to the failure of SSCs, and the potential consequences (i.e., end states); (iii) includes consideration for each operational area (e.g., CRCF, WHF, subsurface) and operational process; and (iv) considers one or more initiating events and combinations of repository system component failures when developing event sequences, consistent with the event sequence definition in 10 CFR 63.2.

The NRC staff evaluated the applicant's methods for modeling and quantifying internal event sequences. The NRC staff determines that the applicant's methods for modeling and quantifying the event sequences is acceptable because the methods are consistent with NRC guidance and standard industry practices in probabilistic risk assessment (PRA) (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa; NRC, 2007ab; American Society of Mechanical Engineers, 2005ad, American Society of Mechanical Engineers,2008aa) and hazard analysis used by chemical industries and NRC-licensed fuel cycle facilities (American Institute of Chemical Engineers, 1992aa; NRC, 2001ah). The NRC staff finds the applicant's use of the SAPHIRE software for event tree and fault tree analyses and event sequence quantification acceptable because the software is used by the NRC and the nuclear industry as a standard analytical tool for modeling and quantification of event sequences. Thus, the NRC staff finds the applicant's methodology for categorization of internal event sequences is adequate because (i) methods for modeling and quantifying the occurrence of the internal event sequences are acceptable, (ii) the methodology considers the aggregation of event sequences by combining event sequences with the same end state to determine the expected number of occurrences associated with the same end state, and (iii) the categorization of event sequences is determined by comparison of the expected number of occurrences for the internal event sequences with the criteria in 10 CFR 63.2 for Category 1 and 2 event sequences.

On the basis of the foregoing evaluation, the NRC staff finds that the applicant's methodology for the identification and categorization for internal event sequences, including the assumptions and methods for quantifying event sequences, is appropriate because it is consistent with applicable NRC rules and guidance and standard practices.

#### 2.1.1.4.3.1.2 Seismic Events

The applicant identified seismicity as a credible natural hazard for evaluation in the PCSA (see SER Section 2.1.1.3.1.3.1). The applicant's methods used to identify seismically initiated event sequences are given in SAR Sections 1.7.1.4 and 1.7.2.4 and supporting documentation (BSC, 2008bg). The applicant's method used a four-stage approach: (i) development of seismic event sequences, (ii) development of hazard curves, (iii) evaluation of seismic fragility curves for SSCs, and (iv) quantification of event sequences. The applicant evaluated potential seismically induced initiating events and analyzed event progression by assessing the subsequent failure or success of individual SSCs that could lead to radiological dose consequences and criticality end states. The applicant's initiating events were dependent on the responses and the dominant seismic failure modes of the SSCs to the seismic ground motion. The seismically induced event sequences were modeled by taking into account specific dependencies between initiating events and the pivotal events.

The seismic failure probability of SSCs was quantified by convolution of the site-specific mean seismic hazard curve with the fragility curves of SSCs. The applicant developed site-specific seismic hazard curves for the surface and subsurface facilities. The seismic hazard curve shown in SAR Figure 1.7-7 represents the mean annual probability of exceedance associated with horizontal peak ground acceleration (PGA) for the surface facilities. The applicant summarized, in SAR Section 1.7.2.4, the methodology used to develop mean fragility curves, defining the conditional probability of failure versus ground motion level for a specified ground motion quantity, such as the horizontal PGA [shown in SAR Figure 1.7-9 for the canister transfer machine (CTM)]. The mean fragility curves are represented by a lognormal probability distribution function controlled by two parameters: (i) median seismic capacity, $C_{50\%}$, or seismic capacity at 1 percent probability of failure, $C_{1\%}$, and (ii) logarithmic standard deviation, $\beta_c$, as a measure of dispersion or uncertainty. The applicant developed fragility parameters for facility structures, as shown in BSC (2008bg, Table 6.2-1), and mechanical systems and equipment, as shown in BSC (2008bg, Table 6.2-2). According to the applicant, the failure of these SSCs potentially could initiate event sequences. By convolving fragility and seismic hazard curves, the applicant evaluated the mean annual probability of failure of SSCs, which is the initiating event frequency.

The applicant used event tree and fault tree techniques for quantitative analysis of the event sequences. Seismically initiated event sequences were developed for each type of waste form configuration in each of the four types of waste handling facilities, as well as the intrasite and subsurface facilities. The applicant's event sequences applied pivotal events similar to the internal initiating events. The event trees consist of an Initiator Event Tree, which identifies SSCs and their failure mode that could initiate event sequences during a seismic event, and the Seismic Response Tree, which models the containment, shielding, and criticality control functions of the SSCs as preventive and mitigative features in the pivotal events. However, the applicant did not credit confinement of the heating, ventilation, and air conditioning (HVAC) system in the seismic event sequence analysis; thus, the HVAC system is not included as a mitigative feature in the pivotal events. The applicant used the residence time factor or total exposure time of the waste container (expressed in years) and the total number of waste containers containing the same waste form handled during the preclosure period to obtain the expected number of event sequences. The exposure or residence time for waste handling operations is that time associated with the handling of waste containers using specific equipment in a waste handling facility. The applicant stated that the expected number of occurrences of an event sequence over the preclosure period is compared to the criteria in 10 CFR 63.2 to categorize the event sequence as Category 1, Category 2, or beyond a Category 2 event sequence (SAR page 1.7-7).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's methodology for identification and categorization of seismically induced event sequences using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–01 (NRC, 2006ad). The NRC staff finds that the applicant's methodology is appropriate for the identification of seismically induced event sequences because, consistent with the guidance in HLWRS–ISG–01 (NRC, 2006ad), the applicant's methodology includes (i) consideration of potential seismically induced initiating events; (ii) the sequence of responses to the failure of SSCs to the initiating events; and (iii) the potential consequences (i.e., end states) for the event sequences.

The NRC staff evaluated the applicant's methods for modeling and quantifying seismically induced event sequences. The NRC staff finds the applicant's methods for modeling and quantifying seismically induced event sequences are acceptable because (i) estimation of mean annual probability of failure of SSCs by convolving fragility curves and seismic hazard curves is consistent with the guidance in HLWRS–ISG–01 (NRC, 2006ad); (ii) use of event tree and fault tree techniques is a standard industry practice (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa), including use of SAPHIRE software for event tree and fault tree analyses and event sequence quantification; and (iii) the applicant's assumption of lognormal distribution to define the mean fragility curve for SSCs is a standard industry practice in seismic PRA (ANS/IEEE, 1983aa). Additionally, the applicant's use of the exposure time for evaluating the expected number of occurrences is acceptable because exposure time accurately represents the period of time during which the event sequence can occur. Thus, the NRC staff finds the applicant's methodology for categorization of seismically induced event sequences is adequate because (i) methods for modeling and quantifying the occurrence of seismically induced event sequences are acceptable and (ii) the categorization of event sequences is determined by comparison of the expected number of occurrences with the criteria in 10 CFR 63.2 for Category 1 and 2 event sequences.

On the basis of the foregoing evaluation, the NRC staff finds that the applicant's methodology for the identification and categorization for seismically induced event sequences, including the assumptions and methods for quantifying event sequences, is appropriate because it is consistent with NRC guidance and standard practices.

2.1.1.4.3.1.3 Fire Events

The applicant provided fire-initiated event sequences in SAR Section 1.7.1.2.2. The applicant used information from the fire hazards analyses (BSC, 2007ab,aw,bb,bf; BSC,2008ae,ai,ap,bp) and event sequence development documents (BSC, 2008ab,ao,at,bd,bj,bo) with the fire-related reliability analyses to quantify event sequences (BSC, 2008ac,as,au,be,bk,bq).

The applicant considered fire-initiated event sequences for the Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), Wet Handling Facility (WHF), Low-Level Waste (LLW) facility, and Intrasite operations and Balance of Plant Surface facilities on the basis of exposure of each potential waste form in containers (e.g., DPC, TAD, AO) in the respective facility. The applicant identified areas in each facility where fires could play a role in either directly exposing a waste form or affecting an SSC. The applicant developed initiating event probabilities for local fires that could impact particular waste form container(s) while they are located in specific areas of each facility (e.g., a fire originating in a room or within a single fire area of the building). The applicant considered a large-fire scenario to capture an event sequence that assumes a substantial fire propagates through a facility and impacts waste form container(s) in any location within the facility. The initiating event frequencies for fire events were provided on a per-waste form configuration basis, and were evaluated in SER Section 2.1.1.3.3.2.4.

Event sequences were developed around a series of pivotal events that lead to a set of potential exposure consequences. The primary pivotal event in all Event Sequence Response Trees was canister reliability, which considers the potential for a fire exposure to breach a canister. The derivation of this pivotal event probability involved an analysis of canister reliability under fire conditions, as discussed in SAR Section 1.7.2.3.3. DOE included the probability for the loss of shielding pivotal event in all response trees. This probability was derived using basic heat transfer and material properties data of the shielding materials. The moderator pivotal event was common to response trees and was used to discern between radionuclide releases with or without criticality potential. The reliability of surrounding systems to control moderator releases (e.g., sprinkler systems controlling water releases and mechanical systems controlling lubricating oil releases) was used to determine moderator intrusion probability. Moderator intrusion is discussed in BSC (2008ac, Section 6.2.2.9), BSC (2008bq, Section 6.2.2.10) and the fault tree for moderator intrusion is provided in BSC (2008ac, Figure B9.5-1). DOE included an additional confinement probability pivotal event in its event sequence development for event sequences taking place in buildings with HVAC confinement capabilities.

The applicant used the SAPHIRE software to model the event sequences. Event sequence diagrams (ESD) were compiled for each facility and included throughput data, initiating event probability, and pivotal event probability data. The SAPHIRE model for each facility was divided into ESDs pertaining to individual waste form configuration and incorporated the initiating event and pivotal event probabilities for that particular waste form container and facility.

Because the applicant's fire-related event sequence development methodology was based on a per unit probability, the applicant used throughput data in SAR Table 1.7-5, BSC (2008ac,as,au, be,bk,bq), and DOE (2009ga) of waste containers to convert event frequencies into a total number of occurrences related to the waste form configuration for the containers. The applicant quantified each event sequence outcome using the methodology outlined in BSC (2008ac,as,au,be,bk,bq). The applicant stated that the expected number of occurrences of an event sequence over the preclosure period is compared to the criteria in 10 CFR 63.2 to categorize the event sequence as Category 1, Category 2, or beyond a Category 2 event sequence (SAR page 1.7-7).

**NRC Staff's Evaluation**

The NRC staff reviewed the methodology for developing fire-initiated event sequences using the guidance provided in YMRP Section 2.1.1.4. The NRC staff finds that the applicant's methodology is appropriate for the identification of fire-initiated event sequences because the applicant's methodology (i) identifies areas in each facility where fires could play a role in either directly exposing a waste form or affecting an SSC; (ii) includes pivotal events that consider the potential for a fire exposure to breach a canister; (iii) includes moderator pivotal events to identify event sequences with or without a criticality potential; (iv) includes a large-fire scenario to capture an event sequence that assumes a substantial fire propagates through a facility; and (v) uses ESDs to describe the event sequences in terms of the fire-initiating events, the sequence of responses to the failure of SSCs, and the potential consequences (i.e., end states), including the potential for criticality due to moderator intrusion.

The NRC staff evaluated the applicant's methods for modeling and quantifying fire-initiated event sequences. The NRC staff finds the applicant's methods for modeling and quantifying fire-initiated event sequences are acceptable because (i) initiating event frequencies for fire events were provided on a per-waste form configuration basis and were evaluated and found acceptable in SER Section 2.1.1.3.3.2.4; (ii) use of event tree and fault tree techniques is a standard industry practice (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa), including use of SAPHIRE software for event tree and fault tree analyses and event sequence quantification; (iii) the probability for the loss of shielding was derived using basic heat transfer and material properties data of the shielding materials; and (iv) moderator intrusion probability considers the reliability of surrounding systems to control moderator releases (e.g., sprinkler systems controlling water releases and mechanical systems controlling lubricating oil releases). Additionally, the applicant's per-unit probability for the fire-initiated event sequences is acceptable because it is based on the throughput of containers for the repository (SAR Table 1.7-5). Thus, the NRC staff finds the applicant's methodology for categorization of fire-initiated event sequences is adequate because (i) methods for modeling and quantifying the occurrence of the fire-initiated event sequences are acceptable and (ii) the categorization of event sequences is determined by comparison of the expected number of occurrences of fire-initiated event sequences with the criteria in 10 CFR 63.2 for Category 1 and Category 2 event sequences.

On the basis of the foregoing evaluation, the NRC staff finds that the applicant's methodology for the identification and categorization for fire-initiated event sequences, including the assumptions and methods for quantifying fire-initiated event sequences, is appropriate because it is consistent with NRC guidance and standard practices.

**NRC Staff's Conclusion**

Based on the review documented in SER Section 2.1.1.4.3.1, the NRC staff finds that the technical basis and justification the applicant provided for its methodology for identification and categorization of event sequences initiated by internal, seismic, and fire hazards is acceptable because the applicant's methodology (i) considers the relevant initiating event sequences, response of SSCs to the initiating event, and the ends states are consistent with site-specific data and the design and operations of the facilities; (ii) methods for modeling and quantifying the occurrence of event sequences are acceptable; (iii) uses event tree and fault tree techniques, which is a standard industry practice (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa), including use of SAPHIRE software for event tree and fault tree analyses and event sequence quantification; and (iv) categorizes event sequences by comparison of the expected number of occurrences for the event sequences with the criteria in 10 CFR 63.2 for Category 1 and 2 event sequences.

### 2.1.1.4.3.2 Event Sequences Development

The applicant discussed development of event sequences initiated by naturally occurring and human-induced hazards in SAR Section 1.7.1. This section documents the NRC staff's evaluation of the applicant's technical basis for event sequence development. The scope of this section includes a review of the appropriateness of event sequence development for internal, seismic, and fire events at the surface, subsurface, and Intrasite operations and Balance of Plant facilities. The NRC staff's review in this section is divided into three subsections: (i) Section 2.1.1.4.3.2.1 Internal Events; (ii) Section 2.1.1.4.3.2.2 Seismic Events; and (iii) Section 2.1.1.4.3.2.3 Fire Events.

To evaluate whether event sequences were developed appropriately and modeled consistent with the methodology reviewed in SER Section 2.1.1.4.3.1, the NRC staff's review evaluates whether the event sequences that could result in radiation exposure or release of radioactive materials during operations are consistent with the applicant's design and operation of the repository (i.e., design basis and design criteria). The NRC staff considers whether the (i) initiating events were appropriately included in event sequences; (ii) system response of SSCs to provide containment, confinement, shielding, and criticality control functions are consistent with the specific design and operations information and data; (iii) safety functions of the SSCs relied on to prevent or mitigate exposure are clearly identified and consistent with the applicant's design and operations; and (iv) end states are consistent with the success or failure of the SSCs' safety functions.

#### 2.1.1.4.3.2.1 Internal Events

The applicant discussed the event sequences of internal events for the surface and subsurface facilities in SAR Section 1.7.1 and BSC (2008ab,ac,ao,as,at,bd,be,bj,bk,bo,bq). The internal event sequences are initiated by random failure of equipment or human error during waste handling operations. The NRC staff reviewed the information discussed in these documents and reviewed the SAPHIRE models to evaluate the applicant's event sequence development.

The NRC staff's review of internal events is organized into three subsections that discuss similar operations (e.g., handling of canisters) that can occur at different times in different facilities. Therefore, more than one facility might be discussed in a subsection where similar handling operations occur. The review of event sequence development for the canister and cask handling operations in the Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), Wet Handling Facility (WHF), and during Intrasite operations are discussed in SER Section 2.1.1.4.3.2.1.1. The review of event sequence development for wet handling operations is discussed in SER Section 2.1.1.4.3.2.1.2, and handling of waste packages during subsurface operations is evaluated in SER Section 2.1.1.4.3.2.1.3.

2.1.1.4.3.2.1.1 Canister and Cask Handling Operations at Surface Facilities

In SAR Section 1.7.1 and in BSC (2008ab,ao,at,bd,bo), the applicant described development of the event sequences resulting from random equipment failures or human errors during handling of canisters and casks in the CRCF, IHF, RF, and WHF and during Intrasite operations. The event sequence diagram (ESD) that forms the basis for event sequence development are explained in Attachment F of BSC (2008ab,ao,at,bd,bo), while Table G–2 of BSC (2008ab,ao,at,bd,bo) summarized the relationship among the ESDs, the Initiator Event Trees, and the System Response Trees.

For the CRCF, IHF, RF, and WHF, the applicant identified initiating events related to (i) structural challenges (e.g., drop, object drop on, collision, tipover) to various types of waste form containers (e.g., casks, canisters, waste packages) causing radiological consequences to the public and workers and (ii) temporary loss of shielding causing direct exposure to the workers.

At these facilities, the applicant developed event sequences involving structural challenges to the (i) transportation casks, loaded with waste canisters and uncanistered SNF assemblies, during receipt and transfer operations inside the surface facilities; (ii) aging overpack (AO) loaded with the transportation, aging, and disposal (TAD) canister or dual purpose canister (DPC) during closure and transfer activities inside the facilities and exporting to the Aging Facility; (iii) waste canisters [TAD, high-level radioactive waste (HLW), DPC, naval, U.S. Department of Energy (DOE) standardized] during transfer operations using the canister transfer machine (CTM); and (iv) waste packages, loaded with TAD or other canisters, during transfer, closure, and loading onto the transport and emplacement vehicle (TEV). Event sequences associated with loss of shielding, causing direct exposure to workers, involved cask preparation activities and CTM activities inside the canister transfer rooms. The applicant also considered initiating events associated with Intrasite operations that involved structural challenges to the transportation cask, AO, and shielded transfer cask (STC) during transport within the GROA boundary, and placement and retrieval activities of AO at the aging facility.

When developing event sequences, the applicant modeled a group of initiating events in an Initiator Event Tree and the progression of the event sequences by a System Response Tree (BSC, 2008ab). Each group, which represents a specific challenge to a canister or cask, is an aggregation of similar initiating events. The applicant also considered pivotal events and evaluated event sequence frequencies and the associated end states for each type of waste form canister handled in the facility. For structural challenges to the waste canisters, the pivotal events address the success/failure of SSCs that are relied on to provide containment, shielding, confinement, and moderator control functions to prevent or mitigate event sequences, such as (i) the transportation cask provides shielding; (ii) the transportation cask and canisters provide containment; and (iii) the heating, ventilation, and air conditioning (HVAC) system provide confinement. The applicant also developed event sequences, taking into account specific conditions during handling operations (e.g., event sequences for the transportation casks when the cask is unbolted and when the cask is bolted). The applicant did not credit the reliability of canisters inside a bolted transportation cask and relied on the containment function of the transportation cask. However, the loss of containment function is examined for the canisters inside an unbolted transportation cask. For structural challenges to the aging overpack (AO), the waste canisters provide containment and the aging overpacks provide shielding. In the CRCF, the event sequences associated with structural challenges to the canisters prior to closure of the waste package in the canister transfer room at CRCF consider the reliability of the canisters for containment, the shield bell for shielding, the HVAC for confinement, and moderator exclusion for criticality control. The event sequences for structural challenges to the waste package after closure of the waste package consider the containment capability of both the waste package and the canister inside. Response trees in other facilities for structural challenges to casks, canisters, and waste packages are similar. To address the structural challenges to the transportation cask with uncanistered (bare) spent nuclear fuel (SNF) assemblies in the WHF, the applicant credited the containment function to the transportation cask.

The applicant modeled the initiating events with fault trees and linked them to the pivotal events of the initiator event trees. On the basis of the delineation of the pivotal events and the success or failure branch in the response tree, the applicant assessed the outcome of the event sequences for radiological consequences. The resulting end state identifies the radiation exposure type (i.e., direct exposure from degradation or loss of shielding, filtered radiological release, unfiltered radiological release), a potential criticality, or a safe state with no radiological consequence.

**NRC Staff's Evaluation**

The NRC staff reviewed information on event sequence development for internal events during operations with canisters and casks in the CRCF, IHF, RF, and WHF, and Intrasite operations and Balance of Plant facilities, using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab), to determine whether the applicant adequately considered the internal event sequences that could result in radiation exposure, release of radioactive materials, or criticality events during canister and cask operations. The applicant developed the event sequences based on the ESDs, which include initiating events and pivotal events that represent the success/failure of SSCs that are relied on to prevent or mitigate event sequences. The NRC staff verified that the applicant's ESDs in BSC Attachment F (2008ab,ao,at,bd,bo) included the initiating events applicable to canister and cask handling operations for surface facilities.

The NRC staff finds that the pivotal events are adequate because the models used for the system response are consistent with the facility design and operations, as described in SAR Section 1.2, and reviewed by the NRC staff in SER Section 2.1.1.2. Additionally, the NRC staff finds that the safety functions of the SSCs (SAR Tables 1.9-2 to 1.9-7) that are relied on to prevent or mitigate radiological exposure in the pivotal events were appropriately represented in the event sequence development (i.e., safety functions of the SSCs are consistent with the design basis). For example, the applicant relied on the (i) canisters (TAD, HLW, DOE standardized, naval, and waste package) for containment functions (e.g., DOE's design basis specifies the mean conditional probability of breach of a canister resulting from a drop of the canister is to be less than or equal to $1 \times 10^{-5}$ per drop for the TAD canister); (ii) transportation cask, AO, and STC for shielding functions (e.g., DOE's design basis specifies the mean conditional probability of loss of cask gamma shielding resulting from a drop of a cask is to be less than or equal to $1 \times 10^{-5}$ per drop for the transportation cask); (iii) HVAC system for confinement functions (e.g., DOE's design basis specifies the mean probability that the HVAC system becomes unavailable during a 30-day mission time following a radionuclide release is to be less than or equal to $4 \times 10^{-2}$); and (iv) facility systems and components for moderator intrusion control for prevention of criticality (e.g., DOE's design basis specifies the mean probability of inadvertent introduction of fire suppression water into a canister is to be less than or equal to $1 \times 10^{-6}$ over a 720-hour period following a radionuclide release for the fire suppression system). The applicant stated that frequency of occurrence of an event sequence depends on the frequencies of the initiating events and conditional probabilities of the pivotal events based on the design bases.

The NRC staff evaluated whether end states are consistent with the success or failure of the safety functions of the SSCs. The NRC staff finds that the end states of the response trees for the canister and cask handling are acceptable because the end states are consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences. These event sequences include (i) direct exposure to workers, consistent with loss or degraded shielding associated with the canister and cask; (ii) filtered and unfiltered radiological release to the public and workers, consistent with the success or failure of the containment of the canister and cask, and of operation of the HVAC; and (iii) the potential for criticality, consistent with the success or failure of preventing moderator intrusion into the canister and following canister breach.

Based on the NRC staff's evaluation, the NRC staff finds that the event sequences developed for the internal events for canister and cask handling operations at the CRCF, IHF, RF, WHF, and for Intrasite operations are adequate because the event sequence development (i) included appropriate initiating events for canister and cask handling operations (i.e., structural challenges and loss of shielding); (ii) the system response of SSCs to the initiating events for the canister and cask handling operations event sequences at surface facilities are consistent with the facility design and operations; and (iii) the end states for the event sequences are consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences for canister and cask handling operations at surface facilities.

2.1.1.4.3.2.1.2 Wet Handling Operations

The applicant discussed event sequence development for the wet handling operations of the WHF in SAR Section 1.7.5.4, supported by its event sequence development analysis (BSC, 2008bo) and its reliability and event sequence categorization analysis (BSC, 2008bq). Additionally, the applicant included event sequence diagrams (ESD) in BSC (2008bo, Attachment F) and included event trees in BSC (2008bo, Attachment G; BSC, 2008bq, Attachment A). It cross-referenced ESDs to event trees in BSC (2008bo, Table G–1).

The WHF is the only surface facility that handles uncanistered commercial spent nuclear fuel (CSNF). Wet handling operations in this facility involve the transfer of casks containing uncanistered CSNF to and from the WHF pool, the transfer of CSNF assemblies in the pool, transportation cask preparation activities (e.g., sampling, filling), dual purpose canister (DPC) cutting activities, and TAD canister closure activities.

The applicant included the event sequence development for pool activities associated with (i) transfer of fuel assemblies in and above the pool using the spent fuel transfer machine (SFTM), and (ii) handling and moving of casks to and from the pool. For the transfer of fuel assemblies in and above the pool using the SFTM, the applicant included a fuel drop onto a staging rack. The SFTM only handles spent fuel and is the only crane to handle spent fuel over the staging rack; therefore, the drop of a spent fuel assembly is the only drop considered for the staging rack (DOE, 2009gc).

The applicant accounted for the drop of heavy loads onto casks inside and outside the pool during the movement of casks to and from the pool, and the movement of casks between the pool ledge and the bottom of the pool (DOE, 2009gc). The applicant identified heavy objects that could drop on casks in the pool: (i) the cask or DPC lid from the pool handling crane, (ii) the cask handling yoke from the cask handling crane, and (iii) the TAD canister shield plug from the pool handling crane. The applicant also identified heavy objects that can drop on the cask outside the pool: (i) the cutting machine from the jib crane at the DPC cutting station, (ii) the access port cover from the jib crane at the preparation station, and (iii) the STC shield ring drop from the TAD canister closure station from the jib crane. The applicant provided the number of object drops onto the casks associated with these six event sequences in DOE (2009gc).

For event sequences that could result in direct radioactivity exposure due to loss of shielding during pool operations, the applicant included (i) the initiating events of lifting a fuel assembly too high, (ii) exposure from the splash of pool water, and (iii) improper decontamination of empty transportation casks or DPCs [BSC (2008bo, Figure F–30)]. The applicant excluded improper decontamination [BSC (2008bq, Table 6.0-2)] and the splash of pool water (DOE, 2009fk), indicating that they are off-normal events.

For event trees involving structural challenges to casks when transferring them to or from the pool, the applicant used different response trees, depending on whether the event (e.g., drop) occurs over the pool or the floor [BSC (2008bo, Figures G–32 and G–39)], respectively. For a drop over the pool, the applicant considered the end state of unfiltered radionuclide release of gases, if a cask would not remain intact [BSC (2008bo, Figure G-31)]. Additionally, the applicant considered the potential of a criticality event, if a cask would not remain intact and boron concentration control was not maintained. For a drop over the floor, the applicant considered direct exposure and filtered and unfiltered releases, including those important to criticality. Filtered and unfiltered releases pertain to the confinement pivotal event, which relates to the success or failure of the nuclear confinement HVAC system. The applicant identified in BSC (2008bq, Section 6.3.2.5) that, for containers having both containment and shielding functions, containment failure was considered to result in a concurrent loss of shielding. The applicant included direct exposure from the shielding loss end state for a shielded transfer cask (STC) being transferred from the pool [BSC (2008bo, Figure G–9)]. The applicant also included direct exposure from shielding degradation for the events when containment is not lost for a transportation cask being transferred to the pool and an STC being transferred from the pool [BSC (2008bo, Figures G–3 and G–9)], respectively.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's event sequence development for WHF internal events using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab) to determine whether the applicant adequately considered the internal event sequences that could result in radiation exposure, release of radioactive materials, or criticality events during WHF operations.

The NRC staff finds acceptable the applicant’s use of a spent fuel drop onto a staging rack in the pool as the only event sequence associated with drops onto the staging rack for the transfer of fuel assemblies in the WHF because the spent fuel transfer machine (SFTM) only handles CSNF assemblies, and no other heavy object is involved in the transfer over the staging rack in the pool.

The NRC staff finds that the event sequences the applicant includes for drops of heavy loads, not related to drops over the staging rack, appropriately considered both drops of heavy loads in and over the pool and drops outside of the pool, consistent with the design and operations of the WHF, because the applicant described in DOE (2009gc) that the event sequences included (i) drop of the cutting machine from the jib crane onto the shielded transfer cask with a DPC inside the cask (at the DPC cutting station); (ii) drop of the access port cover from the jib crane onto the transportation cask at the preparation station 1; (iii) drops of the cask handling equipment (i.e., cask handling yoke and cask lid) during handling of the transportation casks in the pool; (iv) drops of the cask handling equipment (i.e., cask handling yoke and cask lid) during handling of DPCs in the pool; (v) drop of the TAD canister shield plug from the pool handling crane during installation in the pool; and (vi) drop of the shielded transfer cask (STC) shield ring from the TAD canister closure station jib crane onto a loaded STC with a TAD canister. Additionally, the NRC staff finds that the applicant’s end states for the event sequences are acceptable because the applicant’s end states are consistent with the operations and design of the WHF (e.g., a drop event sequence over the pool considers the end state of unfiltered radionuclide release of gases, if a cask would not remain intact; a drop event sequence over the floor considers direct exposure and filtered and unfiltered releases, if a cask would not remain intact).

Regarding the structural challenge to loaded transportation casks and loaded STCs during wet handling operations, the NRC staff finds that the applicant reasonably included SSCs that are relied on to prevent or mitigate event sequences related to the handling of STC and the transportation casks in the WHF because the applicant (i) included the casks to provide containment for a drop or tipover; and (ii) specified, consistent with the safety controls for design and operation, that the lid will be held in place by a minimum number of installed fasteners on the cask during movement to limit the potential for loss of containment of the cask due to damage to the lid seal (SAR Table 1.9-10). Additionally, the applicant, in its response to the NRC staff’s RAI (DOE, 2009fk), described procedural safety control (PSC)–6, which specifies the minimum number of installed fasteners, to ensure that the lid is held in place in the event of a drop or tipover.

The NRC staff finds that the applicant’s exclusion of the initiating event of personnel exposure from pool water splash and the initiating event of improper decontamination of empty transportation casks or DPCs (for event sequences involving direct radiation exposure due to loss of shielding during pool operations) is acceptable because (i) the applicant’s designation of these two initiating events as off-normal events is consistent with HLWRS–ISG–03 (NRC, 2007ac) guidance regarding off-normal events; and (ii) the radioactivity in the pool water would be kept low by the applicant’s design and operation of the pool water treatment system (SAR Section 1.2.5). The NRC staff’s evaluation of the pool water treatment system is documented in SER Volume 2, Section 2.1.1.2.3.2.8. The NRC staff finds that the applicant appropriately identified the end state for lifting a fuel assembly in accordance with the safety design and function of the spent fuel transfer machine (SFTM) identified in BSC (2008bq, Table 6.9-1).

Based on the NRC staff's evaluation, the NRC staff finds the event sequences developed for internal events at the WHF are adequate because the event sequence development (i) included appropriate initiating events both in and over the pool and outside of the pool for operations at the WHF; (ii) included initiating events for dropping of CSNF on the staging rack in the pool (note: this is the only facility that handles uncanistered spent fuel); (iii) included drops of both crane equipment (e.g., cask handling yoke) and heavy objects being moved by the crane (e.g., shield plug) resulting in structural challenges and loss of shielding; (iv) the system response of SSCs to the initiating events in the WHF event sequences are consistent with the facility design and operations; and (v) the end states for the event sequences are consistent with the success or failure of the safety functions of the SSCs relied on to prevent or mitigate event sequences for the WHF operations.

#### 2.1.1.4.3.2.1.3 Subsurface Operations

The applicant provided information in SAR Section 1.7.5.6 and the supporting document (BSC, 2008bj) regarding the development of potential event sequences that could occur during loading of waste packages onto the transport and emplacement vehicle (TEV) in a surface facility, transport of the waste packages to the subsurface facility, and emplacement of the waste underground, as shown in BSC (2008bj, Figure 6). SAR Table 1.7-17 and BSC (2008bj, Attachment F) summarized the applicant's identification of event sequences that could occur during subsurface operations.

The applicant grouped the event sequences that could (i) challenge the structural integrity of a waste package due to mechanical impact from a collision with a shield door, other structure, or equipment; a drop or dragging of a waste package; or TEV derailment; (ii) result in a potential loss of shielding; and (iii) present a thermal challenge due to fire [BSC (2008bj, Attachment F)]. The applicant considered that event sequences, which could challenge the structural integrity of a waste package, may arise from (i) mechanical impact from a collision with a shield door, other structure, or equipment; (ii) a drop or dragging of a waste package; or (iii) TEV derailment. The applicant also considered event sequences that could result in loss of radiation shielding from (i) a violation of an administrative or physical control (such as inadvertent worker entry into an emplacement drift containing waste packages, proximity to a loaded TEV, or inadvertent opening of a TEV door); or (ii) TEV shielding degradation due to overheating. The applicant stated that the TEV shielding may degrade if a layer of polymer material in the shielding overheats. This could occur if a waste package (radiating heat) remains inside the TEV for an extended period of time because, for example, the TEV is disabled by derailment or loss of power. Event sequences related to the potential loss of shielding result in a direct exposure end state. Event sequences related to structural challenges to the waste package result in a potential loss of containment end-state.

The applicant considered the operations needed to install drip shields over the waste packages in the emplacement drifts, as shown in BSC (2008bj, Figure 16). The applicant described its plans to install the drip shields toward the end of subsurface operations and prior to permanent closure of the repository. The applicant described the subsurface SSCs for the underground openings and the invert structures and rails, power distribution infrastructure, and subsurface ventilation, which functions within the serviceability limits needed for subsurface operations through the preclosure period [BSC (2008bj, Attachments A and B)]. SAR Sections 1.3.3.3.2 and 1.3.4.4.2 stated that the applicant will use monitoring and inspection programs to assess the need for and frequency of maintenance of the subsurface structures and systems. In an RAI, the NRC staff requested the applicant to clarify its approach to preventing or mitigating potential event sequences related to subsurface structures or systems failure, such as (i) failure of the invert structure due to corrosion, thermal expansion, or loss of rock support; (ii) collapse of an emplacement drift, exhaust main, or exhaust shaft; (iii) loss of operating envelope due to wall convergence; (iv) ventilation failure due to blockage of an exhaust conduit, such as ventilation raise or exhaust main or shaft; or (v) rock deformation due to fault displacement or thermal expansion resulting in buckling or misalignment of the third rail used for power supply or a slotted microwave guide system for communications. In its response to the RAI (DOE, 2009ed), the applicant stated that it established design criteria and bases to ensure stability of the subsurface structures and systems, and a monitoring, inspection, and maintenance program will address any deterioration of the structures and systems in a timely manner. The NRC staff's review of the stability of subsurface structures and systems is documented in SER Section 2.1.1.2.3.7.

**NRC Staff's Evaluation**

The NRC staff reviewed the event sequence development for subsurface operations using the guidance provided in YMRP Section 2.1.1.4, to determine whether the applicant adequately considered the internal event sequences that could result in radiation exposure or release of radioactive materials during subsurface operations (e.g., loading, transport, and emplacement of waste packages; drip shield installation). The NRC staff's review of the applicant's event sequence development examined how the initiating events identified in BSC (2008bj, Tables 10 and 11), which were evaluated in SER Section 2.1.1.3.3.2.3, were assigned to event sequences [BSC (2008bj, Attachment F)]. Also, the NRC staff reviewed the event sequences in the context of the subsurface operations, as described in the process flow diagrams in BSC (2008bj, Figure 15), event sequence diagrams (ESD) in BSC (2008bj, Attachment F), and the initiator event tree and response trees provided in BSC (2008bj, Attachment F).

The NRC staff finds that the event sequences developed for the internal events for subsurface operations are adequate because the event sequence development (i) included appropriate initiating events that could challenge the structural integrity of a waste package (e.g., mechanical impact from a collision with a shield door, TEV derailment), result in a potential loss of shielding, or present a thermal challenge due to fire; (ii) included potential violations of an administrative or physical control (such as inadvertent worker entry into an emplacement drift containing waste packages, proximity to a loaded TEV, or inadvertent opening of a TEV door), consistent with the design and operations for the underground facility; (iii) included the response of SSCs to the initiating events in the event sequences for the underground facility (e.g., TEV shielding may degrade if a layer of polymer material in the shielding overheats), consistent with the subsurface facility design and operations; (iv) included the use of monitoring and inspection programs, consistent with the design criteria for the subsurface facilities, which will address any deterioration of SSCs in a timely manner to prevent or mitigate event sequences for the subsurface operations (e.g., timely maintenance and monitoring to limit the potential for collapse of an emplacement drift, exhaust main, or exhaust shaft); and (v) included the end states for subsurface operations represent potential occurrences that could result in radiation exposure or release of radioactive materials during subsurface operations (note: the potential for moderator intrusion that could lead to criticality is not a concern in the subsurface due to the limited water and the temperature of the waste package).

2.1.1.4.3.2.2 Seismic Events

The applicant provided information on the development of seismically induced event sequences for the GROA in BSC (2008bg). The applicant developed seismically initiated event sequences for the Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), Wet Handling Facility (WHF), and Intrasite operations and Balance of Plant facility. The NRC staff reviewed the information provided by the applicant to determine whether the applicant's event sequence development adequately considered the seismic event sequences that could result in direct radiation exposure or release of radionuclides with or without criticality potential.

**Waste Handling Operations in Surface Facilities**

The applicant discussed development of seismically induced event sequences during waste handling operations in surface facilities in BSC (2008bg, Section 6.0). During handling of the waste canisters and casks at the CRCF, IHF, RF, and WHF, the seismic initiator event trees consisted of multiple branches, identifying the potential SSCs and the seismically induced failure modes. The seismic initiator event trees were developed on the basis of the facilities, operations, and type of waste containers present in the facility or involved in the operations. For example, the initiator event tree shown in BSC (2008bg, Figure 8.6-4) depicts that TAD canisters inside AOs are received in CRCF on the site transporter and transferred to the waste package, which is then sealed, loaded onto the TEV, and transported out of the CRCF. The applicant also identified events initiated by (i) seismically induced collapse of the CRCF building, potentially causing a breach in the waste container, resulting in a loss of waste form containment; (ii) the potential collapse of mechanical structures (e.g., entry door, shield door, mobile or cask prep platform, welding robot arm) on the waste containers as a result of a seismic event, even if the building structure remains intact; and (iii) the failure of equipment and systems should a seismic event occur during the handling of waste containers, similar to failures initiated by internal events. The mechanical handling equipment [e.g., cask and waste package handling cranes, cask and waste package transfer trolleys (WPTTs), canister transfer machine (CTM), transport and emplacement vehicle (TEV), and the site transporter] has several seismic failure modes induced by seismic load that can potentially impact the waste containers (e.g., collapse of the equipment on the waste containers, drop of the waste container, object dropped on waste container).

The applicant developed a fault tree model for each seismic initiating event. A typical fault tree in the applicant's analysis consisted of the exposure time factor of a structure or equipment and its potential failure modes that contribute to the failure. For example, the initiating event "CTM seismic failure," as shown in BSC (2008bg, Figure C1.1-7), was initiated by seismic collapse of the CTM, drop of a canister hoisted by the CTM, or significant swing inside or outside the shield bell, as shown in BSC (2008bg, Figure C1.2-4). The failure probability for each failure mode was quantified by convolving the fragility curve defined by the parameters given in BSC (2008bg, Table 6.2.2) and the seismic hazard curve given in BSC (2008bg, Section 6.1). The exposure/residence time factors in the applicant's analysis accounted for the amount of time the waste container is exposed to the seismic hazard.

For event sequence analysis, the initiating events in the seismic initiating event tree provide input to the seismic System Response Trees. Similar to the internal events, a typical seismic response tree, as shown in BSC (2008bg, Figure C1.1-5), consists of pivotal events that examine potential waste container breach, loss of shielding, failure of confinement, and moderator intrusion following the initiating event, and culminating in several possible end states. In general, the seismic failure of equipment can cause (i) drop, lateral impact, or drop of a heavy object; or (ii) collapse onto a waste container resulting in container breach or loss of shielding. The conditional probability of container breach or loss of shielding, given the seismic failure of the equipment, was determined using passive failure analysis of structural challenges from drop or other impacts. The applicant assumed failure of the heating, ventilation, and air conditioning (HVAC) if the seismic event caused breach to the waste canister (BSC, 2008bg); therefore, taking no credit for the “CONFINEMENT” pivotal event. The applicant also considered a criticality end state resulting from a piping system failure and intrusion of moderator into a canister that is breached from a seismic initiating event.

Additionally, at the WHF, where bare fuel assemblies would be handled, the applicant considered events resulting from failure of the WHF pool, collapse of the SNF staging rack, and failure of the HVAC integrity (e.g., contaminated ducts, filters in the WHF) causing potential for unfiltered release. For seismically induced initiating events caused by failure of the spent fuel transfer machine (SFTM); transfer station; cask handling crane; auxiliary pool crane during cask handling, transport, aging, and disposal (TAD) canister; and fuel assemblies in the pool, the applicant relied on the pool integrity to provide shielding and prevent radionuclide particulate release.

**NRC Staff’s Evaluation**

The NRC staff reviewed the seismically induced event sequences related to waste handling in the CRCF, IHF, RF, WHF, and Intrasite operations and Balance of Plant facilities, using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–01 (NRC, 2006ad). The NRC staff finds that the identification of seismically induced initiating events is acceptable because the applicant’s identification of initiating events is consistent with the facility description and operations. The NRC staff also finds that the applicant appropriately considered collapse of surface facility structures as an initiating event because the applicant’s event sequences include consideration of (i) collapse of a surface facility structure with the potential to result in a breach of waste container(s) present at that time in the facility and failure of the HVAC ITS SSC used for filtering radionuclide releases (i.e., confinement safety function) and (ii) the collapse of several nearby mechanical structures on the waste containers based on the description of the facilities, operations, and the designs of SSCs. The NRC staff finds that the applicant adequately considered failure of equipment used during the waste handling operations by including the seismic interaction of mechanical components impacting the waste containers in the initiating event, commonly known as “two-over-one issues” in a seismic probabilistic risk assessment (PRA) for nuclear power plants (ANS/IEEE, 1983aa).

The NRC staff reviewed the applicant’s seismic System Response Trees for all the surface facilities handling waste containers. The applicant identified the seismically initiated events by considering the seismic failure modes of the mechanical systems handling waste containers and failure of mechanical structures onto the waste containers. The System Response Trees examine the success/failure of SSCs providing containment, shielding, and moderator control functions following an initiating event. The seismically initiated events result in structural challenges to the waste containers. The applicant relied on the passive reliability of the waste canisters for containment and transportation and aging casks for shielding. The NRC staff finds that the seismic response trees used in the event sequence analysis are reasonable, and the postulated end states are consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences because the applicant’s evaluations (i) considered the pivotal events consistent with the safety functions of SSCs; (ii) related the end state of the seismic event sequences to potential radiological consequences from loss of containment and direct exposure from shielding loss or degradation; (iii) did not credit the HVAC system containment function, which results in unfiltered release of radionuclides; and (iv) considered safety functions of the WHF pool to mitigate consequences during handling operations (e.g., the spent fuel pool water provides shielding and scrubbing of radioactive releases). Thus, the NRC staff finds that the postulated end states are consistent with the success or failure of the safety functions of the SSCs relied on to prevent or mitigate event sequences.

Based on the NRC staff's evaluation, the NRC staff finds the event sequences developed for seismic event sequences for waste handling operations in surface facilities are adequate because the event sequence development (i) included appropriate initiating events for both the collapse of the facility structure and failure of equipment used during the waste handling operations, even if the building did not collapse; (ii) included failure of the HVAC for when the initiating event causes a collapse of the building; (iii) included the pivotal events representing the system response of SSCs are consistent with the facility design and operations; and (iv) ensures that the end states for the event sequences are consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences in waste handling operations (e.g., the end states are consistent with the success or failure of the surface nuclear confinement HVAC system and the success or failure state of the cask maintaining containment and shielding integrity; consideration for releases that could occur under water in the pool that would mitigate particulate release).

**Intrasite Operations**

The Intrasite operations involve movement and storage of aging overpacks (AOs) containing TAD canisters and horizontal transportation casks containing dual purpose canisters (DPCs) at the Aging Facility, storage of low-level waste (LLW) in the low-level-waste facility (LLWF), and temporary storage of transportation casks on railcars and trucks in the buffer area and movement to surface processing facilities. The seismic event sequences result in the following failures: (i) aging overpack (AO) failure; (ii) horizontal aging module structure failure; (iii) horizontal transporter and site transporter failures associated with railcar and trucks at the yard and during movement; and (iv) low-level waste (LLW) building collapse. These event sequences result in an unfiltered radionuclide release end state.

In assessing seismically induced event sequences related to failure of cut or fill slopes near the aging pads or on transportation routes that link the aging pads to other surface facilities, the applicant stated that failure of an earth slope near the aging pad would not result in a credible event sequence, because a slope failure would have no effect on the aging pad structure due to the distance of the pad from adjacent cut or fill slopes {the applicant depicted in Figures 1 and 2 of Enclosure 2 [DOE, 2009gg] that the aging pad foundation would be located approximately 22.9 m [75 ft] from the edge of adjacent cut or fill slopes}. Regarding the applicant's assessment of the frequency of canister failure, the applicant's assumption that the slope design would be stable under design basis ground motion (DBGM)–2 earthquakes with a mean annual probability of exceedance of $5.0 \times 10^{-4}$ was supported with an analysis provided in DOE (2009ej), which was reviewed by the NRC staff in SER Section 2.1.1.1.3.5.3.2. In addition, the applicant estimated the frequency cut slope failure at ground motion greater than DBGM–2 (DOE, 2009gg), based on the exceedance frequency of DBGM–2 ground motion (i.e., return frequency of 2,000 years); the period of time the aging pad will be in use (i.e., 50 years); and conditional probability of canister failure from ground motion event sequences (e.g., tipover, sliding impact).

**NRC Staff's Evaluation**

The NRC staff reviewed the seismically induced event sequences related to Intrasite operations using the guidance provided in YMRP Section 2.1.1.4. The NRC staff finds the event sequences developed for seismic initiated event sequences for Intrasite operations in surface facilities are adequate because the event sequence development (i) included appropriate initiating events for Intrasite operations, consistent with facility design and operations [e.g., movement and storage of aging overpacks (AOs) containing TAD canisters and horizontal transportation casks containing dual purpose canisters (DPCs) at the aging facility; aging pad location and design]; (ii) considered seismically induced failure of cut or fill slopes near the aging pads or on transportation routes that link the aging pads to other surface facilities; (iii) included the system response of SSCs to the initiating events for event sequences for Intrasite operations, consistent with the facility design and operations; and (iv) included the end states for the event sequences, consistent with the success or failure of the safety functions of the SSCs relied on to prevent or mitigate event sequences (e.g., LLW building collapse can result in unfiltered release).

Further, based on the NRC staff's review of the aging pad layout design (DOE, 2009gg), the NRC staff finds that failure of the cut or fill slopes near the aging pads is not likely to impair performance of the aging pad design, because the 22.9-m [75-ft]-wide gravel pad included in the design is sufficient to protect the aging pad from the effects of such a slope failure. The NRC staff's review of the design of the earth slopes at the surface facilities to ensure stability of the slopes during a design-basis ground motion–2 (DBGM–2) earthquake is documented in SER Section 2.1.1.1.3.5.3.2, which includes cut-and-fill slopes near the aging pads and along transportation routes. SER Section 2.1.1.1.3.5.3.2 documents the NRC staff's review that the applicant's evaluation of stability of cut-and-fill slopes under DBGM–2 seismic ground motion is acceptable. The applicant also evaluated cut slope failure at ground motions greater than DBGM–2 (DOE, 2009gg). The NRC staff finds the applicant's evaluation of slope failure acceptable because the applicant appropriately considered exceedance frequency of DBGM–2 ground motion, the duration of preclosure operations at the GROA aging pad, and conditional probability of canister failure in its assessment. Therefore, the NRC staff finds that the applicant adequately identified seismically induced event sequences for Intrasite operations.

**Subsurface Operations**

The seismic event sequences for subsurface operations were presented in BSC (2008bg, Section 6.9). The subsurface operations involved (i) movement of the transport and emplacement vehicle (TEV) with a waste package from the surface facilities to the subsurface emplacement drift; (ii) emplacement and storage of a waste package; and (iii) installation of the drip shields before permanent closure. The initiating events included TEV derailment, entry door collapse on the TEV, rockfall on a waste package in an emplacement drift, and drift instability burying the waste package under rock rubble. Other initiating events considered were drip shield and gantry failure with impact to the waste package during a seismic event.

Regarding the assessment of rockfall impacts, the applicant stated that the ground support systems, such as the rock bolts and stainless steel sheeting, in emplacement and access drifts are designed to protect against rockfall during the service life and against the effects of design basis earthquakes for both vibratory ground motion and fault displacements and that these designs reduce the occurrence of events, such as massive rockfall due to an earthquake or collapse of ground support systems that could impact one or more waste packages (BCS, 2008bg, Section 6.9). Accordingly, the applicant considers the likelihood of having a radionuclide release event initiated by rockfall during the preclosure period is expected (qualitatively) to be very small.

The applicant also performed a thermal analysis of the waste package due to seismic events resulting from rockfall that could restrict airflow and increase insulation around the waste package, which would cause the waste package temperature to increase with time. The applicant considered other potential impacts, such as the impacts to the waste package due to seismic ground motions, rockfall impacts, and drip shield and gantry failure, and determined that these types of events were beyond Category 2 event sequences, based on the strength of the waste package. Additional information on the strength of the waste package to withstand structural challenges is provided in SER Section 2.2.1.3.2.

The applicant did consider an end-state of direct exposure resulting from failure of the TEV shielding while holding a TAD canister in route to emplacement; however, the canister is not breached as a result of the seismic event.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on the applicant's development of event sequences for the subsurface facilities resulting from seismically initiated events, using the guidance in YMRP Section 2.1.1.4. The NRC staff finds the event sequences developed for seismically initiated event sequences for subsurface facilities are adequate because the event sequence development (i) considered both structural impacts from seismically induced rockfall, as well as thermal effects that could lead to unfiltered radionuclide release; (ii) considered failure of other SSCs consistent with the design and operations (e.g., subsurface ventilation design for emplacement drifts, TEV shielding, waste package); (iii) included the response of SSCs to the subsurface and other initiating events, consistent with the facility design and operations (e.g., containment of the waste package); (iv) included the end state of direct exposure for the event sequences, consistent with the capacity of the waste package to withstand structural challenges and the success or failure of the safety functions of the SSCs relied on to prevent or mitigate event sequences (e.g., TEV shielding mitigates direct exposure).

2.1.1.4.3.2.3 Fire Events

The applicant described the fire-initiated event sequences in SAR Section 1.7 and referenced supporting fire hazards analyses (BSC, 2007ab,aw,bb,bf; BSC, 2008ae,ai,ap,bp) and resulting event sequence development documents (BSC, 2008ab,ao,at,bd,bj,bo). The applicant excluded external fire- and explosion-related events and focused its analysis on internal fire events. The NRC staff reviewed the applicant's bases for excluding external fire- and explosion-related events in SER Section 2.1.1.3.3.1.3.5 and concluded that the applicant's bases are acceptable because the applicant's administrative controls, such as (i) a vegetation-free buffer zone, (ii) controlled vehicle operation and parking, and (iii) safe separation distances to potential explosion sources, would prevent significant SSC damage from fire- and explosion-related event sequences. The applicant stated that the separation distance to a fire or explosion event reduces the impact of incident heat flux (fire) or overpressures (breach) on an SSC.

The applicant propagated the initiating events through response trees to obtain end-state probabilities that considered loss of containment of spent fuel containers as well as containers used for LLW (SAR Section 1.7.2.3.3) and degradation and loss of shielding (SAR Section 1.7.2.3.4). The response tree diagrams used to develop the fire-related event sequences had a similar format to response trees for other internal events. These response trees shared common pivotal events, including containment, shielding, confinement, and moderator control (e.g., sprinkler systems controlling release of water). The pivotal event probabilities assigned to fire-related event sequences were based on individual SSC responses to hypothetical fire events. This determination of SSC reliability under fire conditions and the corresponding pivotal event probability were based on information provided in BSC (2008ac,as,au,be,bk,bq).

The applicant also performed an independent analysis of the reliability of SSCs that play a role in pivotal events (e.g., canister reliability under thermal challenges, shield performance under thermal challenges). The failure probability in the presence of a thermal challenge used for these pivotal events was based on a fault tree analysis. In other cases, the applicant assumed failure of a particular SSC in the presence of a thermal challenge (e.g., loss of low melting temperature shielding material during a fire, loss of non-ITS HVAC confinement during a building-wide fire). For these cases, the failure probability in the presence of a thermal challenge was taken as 1.0, or success probability of 0.0 (worst-case) for these specific pivotal events.

The fire analysis for the low-level waste facility (LLWF) was developed as a single initiating event that involved all combustible waste at the LLWF. There was one response tree for the LLWF because the applicant identified only one initiating event for the entire facility.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on the applicant's development of event sequences resulting from internal fire events using the guidance provided in YMRP Section 2.1.1.4. The NRC staff finds the event sequences developed for fire-initiated event sequences adequate because the event sequence development (i) considered both internal and external fires [external fires were acceptably excluded from further consideration, based on administrative controls (see SER Section 2.1.1.3.3.1.3.5 for further details)]; (ii) included the response of SSCs to fire-initiated event sequences, consistent with the facility design and operations (e.g., loss of low melting temperature shielding material during a fire, loss of non-ITS HVAC confinement during a building-wide fire; sprinkler systems for control of moderator); and (iii) included the end states for the success and failure of the SSCs for each facility and the specific container types handled in each facility (e.g., LLW containers and SNF containers).

**NRC Staff's Conclusion**

Based on the NRC staff's review in SER Section 2.1.1.4.3.2, the NRC staff finds the applicant's development of event sequences is adequate because the applicant (i) adequately considered internal events during handling operations, seismic events, and fire events; (ii) considers system response of SSCs, consistent with the facility design and operations; and (iii) the end states for the event sequences are consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences.

### 2.1.1.4.3.3 Reliability of Structures, Systems, and Components

The applicant addressed passive components (e.g., waste containers) for containment and shielding functions and active systems (e.g., HVAC) for confinement functions. The quantified reliability, or failure probability values, were input to the pivotal events in the response tree models. The applicant described its methodology for estimating the SSC reliability in SAR Section 1.7.2. Additional information on the applicant's approach and evaluation of the reliability of SSCs was addressed in BSC (2008ac,as,au,be,bg,bk,bq). SER Section 2.1.1.4.3.2 documents the NRC staff's review of the applicant's development of event sequences. The focus of the NRC staff's review in this SER (Section 2.1.1.4.3.3) is to assess whether or not the applicant has adequately estimated the reliability of the SSCs that are relied on to prevent or mitigate the consequences for the event sequences developed for use in the PCSA.

The NRC staff's review presented in this SER section is organized by passive systems (SER Section 2.1.1.4.3.3.1) and active systems (SER Section 2.1.1.4.3.3.2). The review of passive systems is subdivided into internal events, seismic events, and fire events to assess reliability under structural, seismic, and thermal challenges.

2.1.1.4.3.3.1 Passive Systems

The applicant's determination of reliability of passive systems can be broadly categorized into two classes: (i) waste containers [transport, aging, and disposal (TAD) canisters, DOE standardized canisters, dual purpose canisters (DPCs), high-level waste (HLW) canisters, waste package, transportation cask, and aging overpacks (AOs)] subjected to structural and thermal challenges and (ii) seismic fragility of facility structures and mechanical systems. Structural challenges to a container can result from drops and impacts, while thermal challenges to a container can arise during fire events. The applicant estimated reliabilities (failure probabilities) of containers to provide containment and shielding functions. These failure probabilities were input to containment and shielding pivotal events in the System Response Event Trees for internal and seismic event sequence quantification. The applicant estimated seismic fragility of surface structures and mechanical equipment and systems to quantify probabilities for seismically initiated events. Seismic fragility of the SSC relates the probability of failure of the SSC with a full range of earthquake magnitudes at various mean annual probabilities of exceedance in a seismic hazard curve.

The NRC staff's review of passive reliability of structures, systems, and components for structural challenges resulting from internal events is presented in SER Section 2.1.1.4.3.3.1.1. The NRC staff's review of the fragility of facility structures, mechanical equipment, and systems from seismically initiated events is described in SER Section 2.1.1.4.3.3.1.2. Reliability of SSCs to thermal challenges resulting from fire-initiated events is described in SER Section 2.1.1.4.3.3.1.3.

2.1.1.4.3.3.1.1 Passive Structures, Systems, and Components Reliability for Internal Events

The applicant provided information on the reliability of passive SSCs in SAR Section 1.7.2.3. The applicant presented the passive equipment failure analyses (PEFA) and summarized the failure probabilities for each container in BSC (2008ac,as,au,be,bq) for the surface facilities [Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), and Wet Handling Facility (WHF)] and the Intrasite operations and the Balance of Plant facilities. Additionally, the applicant presented passive reliability of containers used in seismic event sequences in BSC (2008bg, Table 6.3-2), and reliability analysis was discussed in BSC (2008bg, Section 6.3.3 and Attachment H).

The containers relied on to provide containment were the waste packages, transportation aging and disposal (TAD) canisters, dual purpose canisters (DPCs), high-level waste (HLW) canisters, and transportation casks (when containing bare spent fuel). The containers providing shielding functions included the transportation casks and aging overpacks (AOs). The applicant used two approaches to evaluate the passive reliability of containers: (i) full-scale drop test and (ii) determination of applied load or demand and the capacity of the component. The applicant used the first approach to determine passive reliability of HLW canisters, in which statistical analyses were performed on the drop test results. The second approach was used to determine the probability of loss of containment for TAD canisters, DPCs, DOE standard canisters and transportation casks, and the loss or degradation of shielding for the transportation casks and AOs by computing the demand from a drop or impact on the containers, using the finite-element modeling and evaluating the capacity of the material by experimental testing.

***Loss of Containment***

The NRC staff's review of reliability of containers for loss of containment subjected to structural challenges during preclosure operations is discussed next for high-level waste canisters; waste packages; transportation, aging, and disposal canisters; and DOE standardized canisters, transportation cask, and aging overpack. The NRC staff's review of the reliability of containers for loss of shielding subjected to structural challenges during preclosure operations is discussed after the loss of containment following a similar format.

Structural challenges causing potential loss of containment include drop and slapdown (subsequent impact) of containers, collision of containers with other structures or objects, and drop of objects onto the waste containers. In its event-sequence analysis, the applicant used the probability of loss of containment or failure of canisters under structural challenges as a point estimate in the pivotal event in the response tree.

**High-Level Waste Canisters**

In SAR Section 1.7.2.3.1 and BSC Sections 6.3.2.2 and D1.3 (2008ac), the applicant evaluated the probability of failure of high-level waste (HLW) canisters for drops from operational and beyond operational height. The applicant's methodology for determining the canister reliability is based on full-scale experimental drop tests; reliability was estimated on the basis of the number of canisters breached out of the total number of tests. HLW canisters were dropped from heights of 7 m [23 ft] (considered as operational height; 14 tests) and 9 m [30 ft] (considered as beyond operational height; 13 tests) for three different orientations (vertically on its bottom surface; vertically on its top, head down; and tilted with a corner of the bottom surface striking first). To evaluate the structural integrity of the canister bottom, fill nozzle, and welds, after each drop test, the applicant inspected the canisters using two standard test techniques (helium leak test and liquid dye penetrant test) to detect leaks and cracks. Although in some cases (e.g., around the top fill nozzle) significant plastic deformations were observed, the stainless steel canisters did not show ruptures or surface cracks.

The applicant treated these test results as Bernoulli trials, where the outcome was either breach or no breach. Because there was no breach (failure) from the tests, the applicant used a Bayesian approach to estimate failure probabilities separately for the two drop heights. The applicant based the Bayesian analysis on a beta-binomial conjugate distribution, which led to a beta posterior failure probability distribution. The applicant then used the drop test results to estimate the mean and standard deviation for the beta posterior failure probability distribution. Using this approach, the applicant determined that for the 7- and 9-m [23- and 30-ft] drop heights, the mean failure probability posterior distribution was $3.4 \times 10^{-2}$ and $6.7 \times 10^{-2}$, respectively. The applicant used the mean values as point estimates in the event sequence analysis. The actual HLW canister failure probabilities used in the event sequence analysis were $3 \times 10^{-2}$ for a drop from the operational height and $7 \times 10^{-2}$ for a drop from greater than operational height, as shown in BSC (2008ac, Table 6.3-7).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of the high-level waste (HLW) canisters using the guidance in YMRP Section 2.1.1.4 and finds that the design of the HLW canisters used in the experimental tests is consistent with the design detail descriptions presented by the applicant in SAR Section 1.5 and reviewed by the NRC staff in SER Sections 2.1.1.2.3.4.1 and 2.1.1.7.3.9.3.2.

The NRC staff concludes that using full-scale experimental drop tests is an acceptable approach to assess HLW canister failure because the use of full-scale testing to evaluate the structural integrity of other canisters is well documented in the literature (e.g., Morton, et al., 2006aa). The NRC staff evaluated the drop test results of 27 HLW stainless steel canisters {14 tests from 7 m [23 ft] and 13 tests from 9 m [30 ft]} and finds that the results indicated no failure (BSC, 2007de), as reported by the applicant. The applicant's approach for estimating failure probability using the Bayesian methodology, given there were no canister breaches in the drop test results, is appropriate and acceptable because this is a standard statistical method (Siu and Kelly, 1998aa). In particular, the applicant's estimated mean and standard deviation of the failure probabilities for the HLW canisters dropped from 7 and 9 m [23 and 30 ft] was determined using a beta-binomial conjugate distribution Bayesian analysis.

The NRC staff finds the applicant's failure probability values in BSC (2008ac, Table 6.3-7) for a failure probability of $3 \times 10^{-2}$ in the event sequence analysis for a canister drop from an operational height {7 m [23 ft]} and a failure probability of $7 \times 10^{-2}$ for a canister drop greater than operational height {9 m [30 ft]} are adequate because (i) the applicant used acceptable data from full-scale drop tests, (ii) the analysis is consistent with the design and operation for the HLW canisters, and (iii) the applicant used standard statistical techniques for estimating probabilities.

**Waste Package**

In SAR Section 1.7.2.3 and BSC (2008ac, Sections 6.3.2.2 and D1.4), the applicant discussed the calculation of the waste package passive reliability.

The applicant defined the waste package as a passive component that may fail when it is subjected to loads that exceed its load capacity (i.e., strength). Moreover, the applicant stated that, because the waste package is designed in accordance with the provisions of American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, Division 1, Subsection NC (2001aa), a failure may only occur under loads that are greater than the design load. Although all waste package configurations consist of an Alloy 22 outer corrosion barrier and a 316 stainless steel inner vessel (see SER Section 2.1.1.2.3.5.1 for more details), the applicant based the waste package passive reliability only on the capacity of the Alloy 22 outer corrosion barrier.

The applicant defined one waste package failure mode as a structural challenge causing loss of containment (breach). Structural challenges that may cause a waste package to lose its containment function involved a waste package drop event, collision of the waste package with an object or structure, and drop of an object onto the waste package. The applicant used the explicit, nonlinear finite-element analyses software, LS–DYNA, which the applicant stated has been used in other nuclear and nonnuclear industrial applications, to determine the demand on the waste package when subjected to different structural challenges.

From the finite-element models, the applicant calculated the time histories of the Von Mises effective stress and strain from the initiation of loading to the time of unloading. Following a simplified toughness index equation and using the maximum Von Mises effective stress and strain, the applicant estimated the waste package demand as a wall-averaged expended toughness (BSC, 2007cq).

The applicant modeled the capacity of the waste package Alloy 22 outer corrosion barrier using a material toughness and determined the waste package capacity by calculating the material toughness index (BSC, 2007bi,cq). The applicant stated (BSC, 2007cq) that averaged properties from vendor-published data were used for the mean strength properties of the Alloy 22 outer corrosion barrier in order to account for the variability of the Alloy 22 material properties. Further, the applicant used a bilinear stress-strain curve to approximate the stress-strain behavior of Alloy 22. Additionally, because the elastic strains of Alloy 22 are negligible when compared to the ultimate tensile strain of the material, the applicant used a simplified toughness index equation (BSC, 2007bi,cq) for estimating the toughness index of Alloy 22.

To determine failure of the Alloy 22 outer corrosion barrier, the applicant calculated an expended toughness fraction (ETF), defined as a ratio of the waste package demand (i.e., wall-averaged expended toughness) to the waste package capacity (i.e., material toughness index). The applicant assumed that waste package damage occurs for values of ETF ≥ 1. The applicant used ETF to compute the probability of containment failure using BSC (2008ac, Section D1.4, Equation D–3). The equation is based on a normal distribution assumption for ETF and, for computational purposes, transforming ETF to a standardized normal value.

The applicant provided the waste package failure probability values that were used for event sequence quantifications for different structural challenges in BSC (2008ac, Table 6.3-7). The tables provided waste package failure probabilities for different impact conditions. The applicant reported a failure probability of $10^{-5}$ for the three categories of structural challenges: (i) 1.8-m [6-ft] horizontal drop, (ii) 9,072-kg [10-T] drop on a container, and (iii) end-to-end collisions at 4 and 14.5 km/hour [2.5 and 9 mph]. The applicant also documented in BSC (2008ac, Appendix D) that calculations have shown the failure probabilities of the waste package to a 5-m [16.7-ft] drop and a 3-m [10-ft] drop of an 18,144-kg [20-ton] object onto the waste package were less than $10^{-8}$, but the applicant stated it used the failure probabilities of $10^{-5}$ to introduce a measure of conservatism. Table D3.3-1 in BSC (2008ac) provides equivalent drop heights for impact speeds that report a 10 mph collision is equivalent to a drop height of 1 m [3.3 ft].

For flat side impacts at 4 and 14.5 km/hour [2.5 and 9 mph], the applicant used a failure probability of $10^{-8}$ in the event sequence analysis (BSC, 2008ac, Table 6.3.7). The applicant stated a comparison of the strains induced by drops and slow speed; side impacts indicate significantly lower strains for the low-velocity impacts and, therefore, did not introduce conservatism by using a higher probability failure rate such as $10^{-5}$.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of waste packages using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab). The applicant's description of the waste package configurations shows that all waste packages are constructed of an Alloy 22 outer corrosion barrier and a 316 stainless steel inner vessel. The applicant based the waste package passive reliability on the capacity of the Alloy 22 outer corrosion barrier.  The NRC staff finds this approach conservative and acceptable because it does not take credit for the contribution of the waste package inner vessel providing containment of the waste form if an outer corrosion barrier breach occurs.

The NRC staff reviewed the applicant's approach for calculating the demands on the Alloy 22 outer corrosion barrier due to structural challenges and finds that using the maximum Von Mises strain and stress in the failure calculations is appropriate because it is a commonly used stress/strain measurement for ductile materials (e.g., metals).  The NRC staff also finds that the data utilized in the finite-element analyses (BSC, 2007cn,cq,cr) are consistent with the information on design description, design of the waste package, and its components reviewed in SER Sections 2.1.1.2.3.5.1 and 2.1.1.7.3.9.1.  In addition, the NRC staff finds that in these analyses, the applicant represented the waste package and its component geometries (including geometry simplifications) and loadings due to structural challenges, following established practice for structural modeling using finite-element methods (Bathe, 1996aa).

The NRC staff finds that the expended toughness fraction (ETF) is an appropriate damage measure to model the capacity of the waste package Alloy 22 outer corrosion barrier because it is indicative of the material's ability to deform without fracture and to absorb the impact energy from drops or collision.  The applicant approximated the Alloy 22 material behavior using a bilinear stress-strain curve to determine the material properties necessary for input to the material toughness index expression.  The NRC staff finds that this is acceptable because it follows standard engineering practice (Bathe, 1996aa).

The NRC staff reviewed BSC (2008ac, Section D1.4, p. D–21, Equation D–3) to evaluate the probability of failure of waste packages using ETF.  To calculate the failure probability for Alloy 22, the applicant assumed a normal distribution for ETF for the relative variability of the capacity (i.e., material strength).  The NRC staff finds that the formula for computing the probability given in BSC (2008ac, Section D1.4, p. D–21, Equation D–3) is appropriate because this approach is consistent with basic statistical procedures for computing probabilities for a normal distribution.

The NRC staff finds the applicant's waste package failure probability data presented in BSC (2008ac, Table 6.3-7) adequate because the applicant's analysis (i) is consistent with the repository design and operation information for the waste package, (ii) does not take credit for the contribution of the waste package inner vessel providing containment, (iii) used established practice for structural modeling using finite-element methods.  Additionally, the NRC staff finds that the applicant's use of a higher failure probability value ($10^{-5}$) in the event sequence quantification for the 1.8-m [6-ft] drop, 9,072-kg [10-ton]-object drop, 4 km/hour [2.5 mph] end-to-end collision, and 14.5 km/hour [9 mph] end-to-end collision instead of the calculated failure probability of $10^{-8}$ is conservative.

**Transportation, Aging, and Disposal Canisters, and Dual-Purpose Canisters**

In SAR Section 1.7.2.3.1 and BSC (2008ac, Sections 6.3.2.2 and D1), the applicant discussed the passive reliability of the transportation, aging, and disposal (TAD) canisters, the dual purpose canisters (DPC), and naval SNF canisters.  The applicant provided performance specifications for the TAD canisters (SAR Section 1.5.1.1.1.2.1.3) and provided a representative canister design formulated to meet the performance specifications to evaluate the failure probability.  The applicant defined this representative canister on the basis of the available information on existing SNF canisters (i.e., DPCs, TAD canisters, and naval SNF canisters), as shown in BSC (2008cp, Table 4.3.3-2).  The applicant also stated that the TAD canister specifications are based on specifications for a naval SNF canister (BSC, 2008cp). Structural features the applicant used in the evaluation of the representative canisters are the loaded weight, total length, diameter, and shell and plate thicknesses. The applicant chose these dimensions to be close to the average of different types of DPC canister, TAD canister, and naval canister, as shown in BSC (2008cp, Table 4.3.3-2). The material properties used for the representative canister was 304 stainless steel. The failure probability of the representative canister was used for TAD canisters, DPC canisters, and naval SNF canisters in the PCSA.

The applicant estimated the reliability of the representative canister by establishing the relationship between demand and capacity, defined in terms of strain in the canister material. The applicant calculated the demand in terms of the maximum effective plastic strain from each finite element drop simulation analysis. The applicant determined the structural capacity of the representative canister using data for tensile elongation at failure obtained from canister material tests. These test data were treated statistically to develop a cumulative distribution function, or fragility curve, that related the magnitude of the strain from the tests to the likelihood of material failure. The applicant then calculated the probability of canister breach by relating the strain obtained from a finite-element analysis to the fragility curve.

The applicant's calculations for demand on the representative canister used the nonlinear explicit finite-element software, LS–DYNA, to simulate different structural challenges to the canister: (i) 9.9- and 12-m [32.5- and 40-ft] vertical drops, (ii) 1.5-, 3-, and 7-m [5-, 10-, and 23-ft] drops with a 4° off-vertical orientation, and (iii) a 3-m [10-ft] drop of a 10,000-kg [11.02-ton] load onto the top of the canister. The applicant presented details of the LS–DYNA finite-element models in BSC (2008cp, Section 6.3.3). The applicant modeled the canister shell with multiple solid brick (three-dimensional) finite-elements and performed finite-element mesh sensitivity studies for mesh refinement and contact friction effects. The sensitivity studies showed that the mesh and friction parameters selected for performing the impact analyses converged to a stable solution. The applicant determined the demand due to impact using the maximum effective plastic strain of a single brick element through the thickness of the shell.

The canister capacity (fragility) curve was based on 304 stainless steel, which is the material used for the representative container. The fragility curve, which represents probability of failure as function of true strain, was determined by fitting a probability density function to engineering tensile strain data for the material. A frequency histogram of the tensile elongation failure data, as outlined in BSC (2008cp, Figure 6.3.7-2), was constructed from tensile failure tests of 204 specimens of 304 stainless steel annealed tubing. The applicant stated that the data were not normally distributed, but were reasonably well modeled using a weighted mixture of two normal distributions (BSC, 2008cp). The goodness of fit was assessed using the Kolmogorov–Smirnov one-sample test with a 95 percent confidence level. This probability density function was then converted to a cumulative distribution function, or fragility curve, using integration. As shown in BSC (2008cp, Figure 6.3.7-3), the applicant shifted this original fragility curve by 8.3 percent to lower values of true strains at failure, to account for the difference in the original test-data material (annealed 304 stainless steel tubing) and the material proposed for the canister (un-annealed 304 stainless plate). This resulted in a higher estimate of the failure probability for a given true strain at failure.

The probability of failure of the representative canister was determined by relating the magnitude of maximum effective plastic strains from finite-element analysis for different drop heights to the likelihood of failure of the container in the fragility curve, as shown in BSC (2008cp, Figure 6.3.7-1). The applicant used a canister failure probability of $1 \times 10^{-5}$ for event sequence analyses, related to canister drop from heights of 9.9, 12, and 13.7 m

[32.5, 40, and 45 ft, respectively], and a 3-m [10-ft] drop of a 9,072-kg [10-ton] object on the canister, as shown in BSC (2008ac, Table 6.3-7). The reliability of the representative canister was used for TAD canisters, DPC, and naval canisters in the event sequence analyses, as stated in BSC (2009ab, Attachment D). The applicant also documented in Table D.1.2.3 (BSC, 2008ac) that calculations have shown the failure probabilities of the representative canister to a 10- and 12-m [32.5- and 40-ft]-drop and a 3-m [10-ft] drop of a 9,072-kg [10-ton] object onto the waste package were less than $10^{-8}$, but the applicant stated it used the failure probabilities of $10^{-5}$ to introduce a measure of conservatism.

The applicant included the probability of failure for a 4° off-vertical drop in BSC (2008ac, Table D1.2-3). The applicant stated in SAR Section 1.7.2.3.1 that the TAD canister and DPC could only undergo a flat-bottom drop during transfer operations because the canisters would fit tightly inside the CTM shield bell. In this configuration, a canister guide sleeve would ensure that the canister is oriented vertically, so that only a flat-bottom or near-flat-bottom drop could occur (DOE, 2009fy).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of the TAD canister and DPC using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab), to assess the methodology used to estimate the reliability of the TAD canister, DPC, and naval SNF canister. The NRC staff finds that the applicant's approach for reliability estimates, based on evaluating demand and capacity, is consistent with the methodology in HLWRS–ISG–02 (NRC, 2007ab).

The applicant used the reliability of a representative canister to determine failure probabilities for the TAD canister, the DPC, and the naval SNF canister. The applicant provided key features, including overall dimensions of three types of DPCs and the TAD canister in Table 4.3.3.2 (BSC, 2008cp). The applicant stated in BSC (2008cp) and the NRC staff verified that the specifications for the TAD canister (DOE, 2008ag) are based on the specifications for the naval SNF canister (SAR Section 1.5.1.4.2.1). The NRC staff finds that the thickness of the canister shell and top and bottom plates (BSC 2008cp, Table 4.3.3-2) of the representative canister used in the analysis is, on average, less than that of the corresponding DPC, TAD, and naval SNF canister components. The applicant demonstrated that the maximum effective plastic strain is a function of the shell thickness, as shown in BSC (2008cp, Figure 6.3.3.6-1). The NRC staff concludes that the use of a relatively thinner shell for the representative canister will produce larger calculated effective plastic strains, and is thus conservative. Therefore, the NRC staff finds that the use of a representative canister for the DPC, the TAD canister, and the naval SNF canister is appropriate.

The NRC staff finds the use of the LS–DYNA code for the nonlinear finite-element analysis for estimating demand acceptable because it is used as a standard software in the nuclear industry for highly nonlinear, transient impact analyses (e.g., transportation of SNF in casks licensed under 10 CFR Part 71) (Shah, et al., 2007aa). The canister shell is modeled with multiple solid brick (three-dimensional) finite-elements, which the NRC staff finds appropriate, because it adequately models the gradient of plastic strain through the shell thickness. The demand on the representative canister was measured in terms of the maximum effective plastic strain of a single solid element experienced during impact. The NRC staff finds that the use of the maximum effective plastic strain is appropriate because the highest likelihoods of failure would be located at the points of maximum strain in the material and normally the point of impact in a drop simulation would experience the largest strains in the canister. The use of the effective plastic strain accounts for the multi-axial state of strain in the material. Additionally, defining failure on the basis of maximum effective plastic strain of a single element (e.g., through the thickness of the shell) does not account for the possibility that failure could be arrested due to the ductility of the material. Therefore, the NRC staff concludes that the applicant's use of maximum effective plastic strain is conservative. The NRC staff also finds that appropriate engineering modeling techniques were applied to the finite-element analyses for estimating demand because the applicant (i) studied mesh refinement and demonstrated convergence of the mesh and (ii) performed a sensitivity study and demonstrated that the value of friction used between the canister and impact surface had a negligible effect on the solution.

The NRC staff reviewed the applicant's canister capacity (fragility) curve, as shown in BSC (2008cp, Figure 6.3.7-3), based on material used for the representative container. The NRC staff finds that using the tensile elongation (failure) test data for 204 specimens of 304 stainless steel annealed tubing to construct the fragility curve is reasonable because the behavior of the 304 stainless steel in compression is similar to its behavior in tension. The NRC staff reviewed how the uncertainty in the probability distribution for the failure test data is reflected in the uncertainty in the fragility curve. On the basis of this review, the NRC staff finds that uncertainty in the fragility curve based on the test data is small because the probability distribution goodness-of-fit test showed a 95 percent confidence level. Additionally, the NRC staff finds that adjustment of the fragility curve, based on the 304 stainless steel annealed tubing test data, to obtain the fragility curve for the proposed material (un-annealed 304 stainless steel plate) is reasonable because the adjustment accounts for the specific properties of the proposed materials, and the adjusted or shifted curve results in a more conservative (i.e., higher) estimate of the failure probability. In its response to the NRC staff's RAI (DOE, 2009fv), the applicant stated that the strain rate and thermal effects during a drop event were not included in the fragility curve used in BSC (2008cp). The NRC staff reviewed the applicant's response and finds that the fragility curve, without considering the strain rate and thermal effects, is conservative because the higher strain rate would increase the material strength; and the increased temperatures would result in redistribution of strains, thus lowering the maximum effective plastic strain (EPS). Therefore, the NRC staff concludes that the fragility curve developed by the applicant is reasonable.

The NRC staff reviewed failure probability values for several different cases in BSC (2008ac, Table 6.3-7) for the representative canister. The NRC staff finds that the failure probability values for cases related to canister drops from heights of 9.9 and 12 m [32.5 and 40 ft], and a 3-m [10-ft] drop of a 9,072-kg [10-ton] object on the canister are consistent with results obtained from the applicant's finite-element analysis. The NRC staff finds that the applicant's determination of probability of failure for a 13.7-m [45-ft] canister drop is acceptable because the approach of extrapolation of strains to different drop heights is based on the conservation of energy, in which the impact energy is proportional to drop height.

In response to an RAI (DOE, 2009fy), the applicant stated that it will use a guide sleeve located inside the CTM shield bell during canister transfer operations to ensure that any potential drop of the canister will be vertical. The NRC staff reviewed the guide sleeve information provided by the applicant in its response to the RAI (DOE, 2009fy). The applicant considered the maximum drop angle for a canister with dimensions similar to the TAD canister with a guide sleeve in place to be approximately 0.9° from vertical and estimated the failure probability of the canister for the 0.9° off-vertical drop to be less than1.0 × $10^{-8}$; however, the applicant used a failure probability of 1.0 × $10^{-5}$ for PCSA. The NRC staff finds that the applicant's use of a failure probability of 1.0 × $10^{-5}$ in the PCSA acceptable because this value is conservative relative to the value estimated for a 0.9° off-vertical drop and the safety function of the guide sleeve to restrict the lateral movement of the canister to minimize the drop angle.

On the basis of the NRC staff's evaluation discussed in this section, the NRC staff finds that the applicant's TAD canisters, DPCs, and naval SNF canisters failure probabilities presented in BSC (2008ac, Table 6.3 7) are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the TAD canisters, DPCs, and naval SNF canisters; (ii) the analysis included reasonable test data for material failure to construct the fragility curve; and (iii) the applicant used industry-accepted methodologies to estimate the capacity of the canisters to withstand demands from the drop events and estimated reliability as failure probabilities. Additionally, the NRC staff finds that the applicant's use of a higher probability value ($10^{-5}$) in the event sequence quantification for the failure probabilities of the representative canister to a 10- and 12-m [32.5- and 40-ft] drop and a 3-m [10-ft] drop of a 9,072-kg [10-ton] object onto the waste package instead of the calculated failure probability is conservative.

**DOE Standardized Canisters**

The applicant discussed the determination of failure probabilities of DOE standardized canisters in SAR Section 1.7.1 and BSC (2008ac, Section D1.2). The applicant estimated the reliability of the DOE standardized canister by establishing the relationship between demand and capacity, defined in terms of strain in the canister material. The overall approach was similar to that used for the representative canister, as discussed in the previous section.

As before, the applicant calculated the demand in terms of the maximum effective plastic strain from each finite element drop simulation analysis. In this case, for the DOE standardized canister, a series of finite-element analyses was performed using ABAQUS/Explicit, which is an explicit nonlinear finite-element computer code designed for modeling the highly nonlinear, transient characteristics of drop/impact types of analyses. The applicant described a series of full-scale, experimental drop tests that were performed on 46- and 61-cm [18- and 24-in]-diameter DOE standardized canisters. The purpose of these tests was to validate the finite-element simulations of the corresponding experimental drop tests. The applicant compared the numerical results obtained from the finite-element analyses with experimental observations, which were measured in terms of permanent deformation (SAR Figures 1.5.1-23 through 1.5.1-28). The applicant stated that the results of the nonlinear finite-element analyses are consistent with the actual deformations for the drop tests (SAR Figures 1.5.1-23 through 1.5.1-28).

The applicant determined that the structural capacity of the canister depended on tensile elongation at failure obtained from canister material tests. The applicant used the stainless steel fragility curve outlined in BSC (2008cp, Figure 6.3.7-3) since the DOE standardized canisters are made from stainless steel.

The applicant calculated the probability of the DOE standardized canister breach by relating the maximum effective plastic strain obtained from a finite-element analysis to the fragility curve. The maximum equivalent plastic strains, obtained at select locations in the canister model, were listed in BSC (2008ac, Table D1.2-6) for both the 46- and 61-cm [18- and 24-in]-diameter canisters. Using the canister capacity curve (i.e., fragility for the stainless steel material), the applicant calculated the failure probabilities using the maximum equivalent plastic strains.

The applicant summarized the failure probabilities for the DOE standardized canister in BSC (2008ac, Tables D1.2-7 and 6.3-7). For the case of vertical container drop from normal operating height {7 m [23 ft]}, the applicant stated the failure probability is equal to $1.0 \times 10^{-8}$, as shown in BSC (2008ac, Table D1.2-7). However, the applicant used the failure probability of $1.0 \times 10^{-5}$, as detailed in BSC (2008ac, Table 6.3-7), for event sequence quantification to introduce a measure of conservatism. For the 9-m [30-ft] vertical drop case, the applicant extrapolated the amount of strain from the 7-m [23-ft] drop case following the procedure in BSC (2008ac, Section D1.5) and estimated the failure probability to be $1.0 \times 10^{-5}$. For the cases of 4 and 14.5 km/hour [2.5 and 9 mph] end-to-end collisions, the applicant reported a failure probability of $1.0 \times 10^{-5}$.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of the DOE-standardized canister using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab), to assess the methodology the applicant used. The NRC staff finds that using the ABAQUS/Explicit finite-element code to obtain the demand on the canister resulting from drop (impact)-induced structural challenges is acceptable, because use of the ABAQUS/Explicit finite-element code is a standard industry practice for performing nonlinear, highly transient analyses. The staff finds that the models utilized data that are consistent with the design, as evaluated in SER Sections 2.1.1.2.3.5.2 and 2.1.1.7.3.9.3.1. The NRC staff also finds that the finite-element analysis approach is acceptable because the finite-element models were benchmarked and calculated similar deformations to actual deformations obtained from drop tests. In addition, the NRC staff finds that the finite-element analyses used for the reliability estimates are based on acceptable engineering modeling techniques.

The applicant used the fragility (capacity) curve in BSC (2008cp, Figure 6.3.7-3) for the DOE standardized canister. This fragility curve [BSC (2008cp, Figure 6.3.7-3)] was based on grade 304 stainless steel. Since the DOE standardized canister is fabricated from 316 stainless steel, the applicant accounted for the different steel by using a shifted curve to obtain a more conservative (i.e., higher) estimate of the failure probability, as discussed in the previous section where the representative TAD canister is reviewed. Therefore, the NRC staff finds that the conservatism in the shifted fragility curve for the DOE standardized canister acceptably accounts for the difference in material.

The NRC staff finds that the DOE standardized canisters failure probabilities presented in BSC (2008ac, Table 6.3 7) are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the DOE standardized canisters, (ii) the analysis included reasonable test data for material failure to construct the fragility curve, (iii) finite-element models were benchmarked and calculated similar deformations to actual deformations obtained from drop tests, and (iv) the applicant used industry-accepted methodologies to estimate the capacity of the canisters to withstand demands from the drop events and estimated reliability as failure probabilities. Additionally, the NRC staff finds that the applicant's use of a higher probability value ($10^{-5}$) in the event sequence quantification for the failure probabilities for the DOE standardized canisters for the vertical container drop from operational height {7 m [23 ft]}, beyond operational height {9 m [30 ft] vertical drop, 4 km/hour [2.5 mph] end-to-end collisions, and 14.5-km/hour [9-mph] end-to-end collisions} instead of the calculated failure probability $10^{-8}$ is conservative.

### Transportation Cask

The applicant discussed loss of containment of the transportation cask due to drops and impacts in SAR Section 1.7.2.3 and BSC (2008ac, Section 6.3.2.2). The applicant stated that a representative transportation cask was used to analyze the reliability of transportation casks, site transfer casks (STCs), and horizontal STCs. The applicant's methodology for estimating the transportation cask reliability is the same as that for the representative canister. The applicant established a relationship between demand and capacity, defined in terms of strain in the canister material.

The applicant stated in BSC (2008cp) that the transportation cask is relied on to (i) provide shielding but not containment for event sequences where the canister inside the transportation cask provides the containment safety function and (ii) provide both shielding and containment when the transportation cask contains bare CSNF. The applicant evaluated the probability of failure of containment for the transportation cask with a canister inside and without a canister for event sequences involving bare CSNF inside the transportation cask. When the internal representative canister is relied upon to provide containment, the breach of the container "system" occurs only when the internal representative canister material fails. Thus, the applicant determined the demand on that material by calculating maximum equivalent plastic strains in the internal representative canister (inside the transportation cask) using finite element drop simulations. These maximum equivalent plastic strains of the internal canister were then compared to the fragility curve representing the stainless steel material from which the failure probability is determined. The applicant used the fragility curve developed for the representative canister, as shown in BSC (2008cp, Figure 6.3.7-3).

The applicant presented a number of drop scenarios at different heights and cask orientations. Two of these drop scenarios included the effects of slapdown. For all of the drop scenarios, an explicit finite-element analysis using the LS–DYNA code was performed, as described in BSC (2008cp, Section 6.3.2). BSC (2008cp, Table 4.3.3-1a) listed all of the cases analyzed for the transportation cask. BSC (2008cp, Figure 6.3.2-1) showed the structural components in the finite-element model that were used to perform the drop analyses. The modeled components included the simulated SNF, the basket containing the SNF, a thin-walled representative canister, shielding, and a bolted-lid transportation cask that holds the internal canister. The applicant listed all dimensions of the components in BSC (2008cp, Table 6.3.2-1), and the necessary material property data were given in BSC (2008cp, Table 6.3.2-2). The finite-element model was shown in BSC (2008cp, Figure 6.3.2-2).

The applicant calculated the failure probabilities corresponding to the low-velocity events (collisions) using the principle of energy conservation to convert the low speeds into an equivalent drop height (BSC, 2008ac). The failure probabilities were determined using BSC (2008ac, Section 6.3.2.2, Equation 17), which used the known failure probabilities from the LS–DYNA- (BSC, 2008cp) analyzed drop heights, as shown in BSC (2008ac, Table 6.3-2). BSC (2008ac, Section 6.3.2.2, Equation 17) is based on the concept that the strain is approximately proportional to the impact energy, which directly relates to the drop height.

The applicant discussed the failure probabilities for the transportation cask with bare fuel as well as for a representative canister within a transportation cask in BSC (2008ac, Section 6.3.2). The applicant determined that the failure probabilities for all drop events considered (including slapdown) are less than $1.0 \times 10^{-8}$ in BSC (2008ac, Tables D1.2-4 and D1.2-5); however, the applicant used failure probabilities of $1.0 \times 10^{-5}$ in the event sequence quantifications [BSC (2008ac, Table 6.3-7)], stating that this was used to add conservatism. For the low-velocity impacts, such as collisions (which correspond to very small drop heights), the applicant used a failure probability of $1.0 \times 10^{-8}$, as outlined in BSC (2008ac, Table 6.3-7).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of the transportation cask using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab). The applicant's methodology for estimating the reliability for the transportation cask is the same as that used for the representative canister (i.e., establish a relationship between demand and capacity, defined in terms of strain in the canister material). The NRC staff finds the applicant's methodology used for estimating the reliability of the transportation cask is acceptable because it is a commonly used methodology in the industry to estimate reliability of a mechanical system.

The applicant utilized nonlinear finite-element analysis using a standard industry computer code (LS–DYNA) to determine the demand on the internal representative canister (inside the transportation cask). The NRC staff finds that the applicant's use of the nonlinear finite-element analysis is acceptable because this approach is commonly used in industry for impact analysis, including highly nonlinear, transient impact analyses.

The NRC staff reviewed the applicant's approach to convert travel speeds into equivalent drop heights for calculating the failure probabilities corresponding to the low-velocity events (collisions) reported in BSC (2008ac, Table 6.3-4) using BSC (2008ac, Section 6.3.2.2, Equation 17). The NRC staff finds that this approach is acceptable because (i) the strain is approximately proportional to the impact energy, which directly relates to the drop height and (ii) the approach is based on the fundamental principle of conservation of energy.

The NRC staff finds that the applicant's representative transportation cask failure probabilities with a canister inside the cask and with bare CSNF in the cask presented in BSC (2008ac, Table 6.3 7) are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the transportation casks, STCs, horizontal STCs, and the canisters inside the casks; (ii) the analysis, which uses drop height, is based on the fundamental principle of conservation of energy; (iii) the analysis constructs the fragility curve, consistent with the approach used for the analysis, for the representative canister; and (iv) the applicant used industry-accepted methodologies to estimate the capacity of the canisters to withstand demands from the drop events and estimated reliability as failure probabilities. Additionally, the failure probabilities the applicant discussed for drops and collisions in BSC (2008ac, Section 6.3.2) were determined to be $1.0 \times 10^{-8}$ or lower. The applicant, as outlined in BSC (2008ac, Table 6.3.7), decided to use a higher value of $1.0 \times 10^{-5}$ for the failure probability for events involving drops. The NRC staff finds this acceptable because, consistent with the other failure probabilities discussed in the previous sections, the applicant's use of $1.0 \times 10^{-5}$ adds conservatism. Additionally, the NRC staff finds that the applicant's use of failure probabilities for collisions at $1.0 \times 10^{-8}$ is reasonable because these low velocities correspond to relatively negligible drop heights, as shown in BSC (2008ac, Table 6.3-4).

**Aging Overpack**

The applicant discussed the loss of containment for the aging overpacks (AOs) due to drops and impacts in SAR Section 1.7.2.3 and BSC (2008ac, Section 6.3.2.2). The applicant's methodology for estimating AO reliability is the same as that for the representative canister. The applicant stated in BSC (2008cp) that the AO provides a shielding safety function but not a containment safety function during event sequences, because the canister inside the transportation is relied on to provide the containment safety function. There are no event sequences with bare CSNF inside an AO.

The internal representative canister is relied upon to provide containment, and the breach of the container "system" occurs only when the internal representative canister material fails. The applicant followed the approach given in BSC (2008cp), in which the relationship between demand and capacity is defined in terms of strain in the internal representative canister contained within the AO, as was also done for the evaluation of the transportation casks with the representative canister inside. As discussed in BSC (2008cp, Section 6.3.1), an LS–DYNA finite-element model was made of an AO containing an internal representative canister. The demand is calculated in terms of the maximum effective plastic strain in the internal representative canister from each LS–DYNA drop simulation. In these LS-DYNA analyses, the applicant considered two different loading scenarios for the AO/canister model: (i) a 0.9-m [3-ft] vertical drop (normal operating height) onto a rigid surface and (ii) a slapdown from a vertical orientation while also having a 4 km/hour [2.5 mph] horizontal velocity.

The applicant summarized in BSC (2008ac, Table 6.3-7) the aging overpack (AO) failure probabilities used for event sequence quantifications. For the cases of the 0.9-m [3-ft] vertical drop and the slapdown from a vertical orientation with a 4 km/hour [2.5 mph] horizontal velocity, the applicant calculated the probability of AO containment failure as $1.0 \times 10^{-5}$. The applicant also documented in BSC (2008ac, Table D1.2.2) that calculations have shown the failure probabilities of the aging overpack for the 0.9-m [3-ft] vertical drop and the slapdown from a vertical orientation with a 4 km/hour [2.5 mph] horizontal velocity were less than $10^{-8}$, but the applicant stated it used the failure probabilities of $10^{-5}$ to introduce a measure of conservatism. Additionally, the applicant specified a failure probability of $1.0 \times 10^{-8}$ for the cases of low velocity impact/collisions.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of loss of containment for the aging overpack (AO) using the guidance provided in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab). The NRC staff reviewed the LS–DYNA finite-element analyses the applicant performed to determine the structural demand for the cases of a 0.9-m [3-ft] vertical drop and the slapdown from a vertical orientation with a 4 km/hour [2.5 mph] horizontal velocity. The NRC staff finds that the use of the LS–DYNA code for the finite-element analyses is appropriate because it is a commonly used analytical code in the industry for highly nonlinear, transient impact analyses.

The NRC staff finds that the applicant's AO failure probabilities with a canister inside the cask presented in BSC (2008ac, Table 6.3 7) are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the AO and the canisters inside an AO; (ii) the analysis, which uses drop height, is based on the fundamental principle of conservation of energy; (iii) the analysis constructs the fragility curve consistent with the approach used for the analysis for the representative canister; and (iv) the applicant used industry-accepted methodologies to estimate the capacity of the canisters to withstand demands from the drop events and estimated reliability as failure probabilities. Additionally, the failure probabilities the applicant discussed for drops and collision in BSC (2008ac, Section 6.3.2) were determined to be $1.0 \times 10^{-8}$ or lower. The applicant, as outlined in BSC (2008ac, Table 6.3.7), used a higher value of $1.0 \times 10^{-5}$ for the failure probability for events involving drops. The NRC staff finds this acceptable because, consistent with the other failure probabilities discussed in the previous sections, the applicant's use of $1.0 \times 10^{-5}$ adds conservatism. Additionally, the NRC staff finds that the applicant's use of failure probabilities for collisions at $1.0 \times 10^{-8}$ is reasonable because these low velocities correspond to relatively negligible drop heights, as shown in BSC (2008ac, Table 6.3-4).

***Loss of Shielding***

The NRC staff's review of reliability of aging overpack (AO) and transportation cask for loss of shielding subjected to structural challenges during preclosure operations is discussed in the following two subsections.

In SAR Section 1.7.2.3.2, the applicant explained that a loss of shielding occurs when an AO or transportation cask fails in a manner that leaves a direct path for radiation to stream, for example as a result of a breach. Degradation of shielding occurs when the shielding is not breached but its shielding function is degraded (e.g., lead slumping after an impact). For containers that have both a containment and shielding function, the PCSA considers a probability of containment failure (which is considered to result in a concurrent loss of shielding) and also a probability of shielding degradation (which is associated with those structural challenges that are not sufficiently severe to cause loss of containment). A transportation cask that handles bare CSNF has both containment and shielding functions; whereas, the transportation cask only provides a shielding function in the PCSA when a canister is inside the cask, which provides the containment function.

**Aging Overpack**

The applicant discussed the probabilities of the loss of shielding function of aging overpacks (AOs) in SAR Section 1.7.2.3.2 and BSC (2008ac, Section D3.4). The AO is used during the transport of a canister from the Canister Receipt and Closure Facility (CRCF) to the aging pad. The overpack transport vehicle is specified to have a maximum speed of 4 km/hour [2.5 mph], and the maximum vertical lift height for the AO is 0.9 m [3 ft] from the ground.

The applicant's approach to determine the probability of shielding failure is based on equating the overall probability of canister success within an AO to conditional probabilities of canister success, given that the AO shielding does not fail, and the conditional probabilities of canister success, given that the AO shielding fails, as provided in BSC (2008ac, Equation D–26). Therefore, the applicant expresses the probability of the AO shielding failure as a function of the internal canister failure.

To calculate the demand on the internal representative canister contained within the AO, the applicant followed the methodology in BSC (2008cp). As in the containment analyses discussed previously, the applicant established the relationship between demand and capacity, defined in terms of strain in the internal representative canister material. The demand is calculated in terms of the maximum effective plastic strain in the internal representative canister from each finite-element analysis of a drop simulation. The applicant performed the explicit finite-element analyses using the computer code LS–DYNA to calculate the demand on the internal representative canister, as discussed in BSC (2008cp, Section 6.3.1). The finite-element model consisted of an AO and the internal representative canister contained within the AO. The applicant stated that the internal spent nuclear fuel (SNF) canister was the same as that used for the representative canister.

The applicant considered two different loading scenarios in the analyses: a 0.9-m [3-ft] vertical drop (normal operating height) onto a rigid surface and a slapdown from a vertical orientation with a 4 km/hour [2.5 mph] horizontal velocity.

The applicant summarized the AO failure probabilities used for event sequence quantifications in BSC (2008ac, Table 6.3-7). For the 0.9-m [3-ft] vertical drop, the applicant calculated the AO shielding failure probability as $5.0 \times 10^{-6}$. Additionally, the applicant specified a shielding failure probability of $1.0 \times 10^{-5}$ for the cases of collisions (low-velocity impact) and slapdown.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of degradation of loss of shielding for the aging overpack (AO) using the guidance provided in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab). The NRC staff reviewed the applicant's methodology for determining the AO shielding failure due to a structural challenge and finds that the applicant's approach, which expresses the probability of AO shielding failure as a function of the internal canister failure, is acceptable because the AO design is robust against impact loads (i.e., the likelihood of AO breach is lower than the likelihood of AO success for values exceeding the drop and impact speed conditions used for the PCSA).

The NRC staff reviewed the finite-element analyses the applicant performed to determine the structural demand and finds that the analyses are acceptable because (i) the applicant followed the same methodology for determining structural capacity as was used for the representative canister (NRC staff review of the representative canister is documented earlier in this SER section under transportation, aging and disposal canisters, and dual purpose canisters); (ii) use of a representative canister within the AO is consistent with repository design and operations; and (iii) LS–DYNA finite-element analysis code is a standard software in the nuclear industry for highly nonlinear, transient impact analyses (e.g., transportation of SNF in casks licensed under 10 CFR Part 71) (Shah, et al., 2007aa).

Based on the NRC staff's evaluation, the NRC staff finds the applicant's failure probabilities listed in BSC (2008ac, Table 6.3-7) for the AO degradation of loss of shielding are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the AO and the canisters inside an AO; (ii) the analysis, which uses drop height, is based on the fundamental principle of conservation of energy; (iii) the analysis constructs the fragility curve consistent with the approach used for the analysis for the representative canister; and (iv) the applicant used industry-accepted methodologies to estimate the capacity of the canisters to withstand demands from the drop events and estimated reliability as failure probabilities.

**Transportation Cask**

The applicant discussed the degradation of shielding for a transportation cask when subjected to a structural challenge due to impact in SAR Section 1.7.2.3.2 and BSC (2008ac, Section D3).

The applicant's methodology for estimating the failure probability used finite-element analysis to determine the structural demand on the representative transportation cask (i.e., transportation casks, STCs, and horizontal STCs) subjected to transportation-accident impacts. The applicant estimated the amount of damage by determining the amount of plastic strain in the transportation cask's inner shell as a function of the impact speed. On the basis of the impact speed, equivalent drop heights were calculated relating the maximum plastic strain as a function of drop height.

The applicant cited the finite-element analyses used to assess transportation cask performance during impacts, presented in NUREG/CR–6672, Section 5 (Sprung, et al., 2000aa), to estimate structural demand on the transportation cask during impacts. The applicant stated in BSC (2008ac, Section D3) that, on the basis of the finite-element analyses results reported in NUREG/CR–6672 (Sprung, et al., 2000aa), the monolithic steel rail casks and the steel/depleted uranium truck casks exhibited no loss of shielding. Therefore, only the steel/lead/steel rail and truck casks show loss of shielding due to lead slumping. Specifically, the applicant stated that lead slump occurs mainly for the end-impact orientations and, to a lesser extent, for corner impacts. For side impacts, the applicant stated that there is no significant reduction in shielding. Thus, the applicant's analysis focused only on the steel/lead/steel casks with the primary orientation being the end-impact condition. The applicant listed various impact speeds and the resulting maximum plastic strains for impacts onto an unyielding surface, as outlined in BSC (2008ac, Table D3.2-1). The applicant also listed the equivalent speeds from impacts onto real surfaces, such as soil and concrete, as shown in BSC (2008ac, Table D3.3-2), and established a damage threshold for lead slumping, as described in BSC (2008ac, Sections D3.1 and D3.2). The applicant stated that for maximum effective plastic strain levels exceeding 2 percent, lead slumping is likely. Using BSC (2008ac, Figures D3.2-2 and D3.2-3), the applicant estimated the threshold velocities in which the loss of shielding (from lead slumping) would occur. The applicant further stated that the 2 percent maximum plastic strain threshold for a truck cask would correspond to a 101-km/hour [63-mph] impact on a concrete surface, which translates into an equivalent drop height of 41 m [133 ft], as outlined in BSC (2008ac, Table D3.3-1). Thus, the applicant used an estimate of a median threshold for the failure drop height as 41 m [133 ft] (i.e., 2 percent plastic strain).

Additionally, the applicant assumed the strain and probability of failure are normally distributed. The applicant also assumed the drop height was normally distributed because the strain is proportional to the drop height, as described in BSC (2008ac, Section 6.3.2.2). The applicant presented the failure probabilities for different drop heights and collisions in BSC (2008ac, Table 6.3-7). The previous discussion focused on the steel/lead/steel sandwich-type transportation cask. The applicant stated that for all other casks, the only loss of shielding mechanism was by radiation streaming. Thus, the loss of shielding was equated to the probability of rupture of the cask due to closure failure, as outlined in BSC (2008ac, Section D3.4).

The applicant determined the failure probabilities for the loss of shielding due to lead slumping from a 4.6- and 9.1-m [15- and 30-ft]-vertical drop are less than $1.0 \times 10^{-8}$; however, the applicant used failure probabilities of $1.0 \times 10^{-5}$ in the event sequence quantifications, stating that this was used to add conservatism (BSC, 2008ac; Section D3.4).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's reliability analysis of degradation of loss of shielding for the representative transportation cask (i.e., transportation casks and STCs) using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab). The NRC staff determines that the 2 percent maximum plastic strain threshold the applicant established would correspond to a 101-km/hour [63-mph] impact on a concrete surface for a truck cask using data from BSC (2008ac, Figure D3.2-3) and linear interpolation. The NRC staff also determines that the speed of 101 km/hour [63 mph] translates into an equivalent drop height of 41 m [133 ft], using the data from BSC (2008ac, Table D3.3-1). Therefore, the NRC staff finds that the threshold for lead slumping is correct for a 41-m [133-ft] drop.

The NRC staff finds that the applicant's transportation cask failure probabilities for degradation of loss of shielding presented in BSC (2008ac, Table 6.3 7) are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the transportation casks and the canisters inside transportation casks; (ii) the analysis, which uses drop height, is based on the fundamental principle of conservation of energy; (iii) the analysis constructs the fragility curve consistent with the approach used for the analysis for the representative canister; and (iv) the applicant used industry-accepted methodologies to estimate the capacity of the canisters to withstand demands from the drop events and estimated reliability as failure probabilities. Additionally, the failure probabilities the applicant discussed for drops and collisions in BSC (2008ac, Section 6.3.2) were determined to be $1.0 \times 10^{-8}$ or lower. The applicant, as outlined in BSC (2008ac, Table 6.3.7), used a higher value of $1.0 \times 10^{-5}$ for the failure probability for events involving drops. The NRC staff finds this acceptable because, consistent with the other failure probabilities discussed in previous SER sections, the applicant's use of the higher value of $1.0 \times 10^{-5}$ for the failure probability increases the likelihood of failure, which is a conservative approach. Additionally, the NRC staff finds that the applicant's use of failure probabilities for collisions at $1.0 \times 10^{-8}$ is reasonable because the low velocities considered for the GROA {i.e., site transporter has a maximum speed of 4 km/hr [2.5 mph]} correspond to relatively negligible drop heights {e.g., less than 10 cm [4 in]}, as shown in BSC (2008ac, Table 6.3-4).

**NRC Staff's Conclusion**

Based on the NRC staff's review in SER Section 2.1.1.4.3.3.3.1, the NRC staff finds that the applicant provided acceptable information for passive reliability of the high-level waste containers, waste packages, TAD canister, dual-purpose canister, DOE standardized canisters, transportation cask, aging overpack, and shielded transfer cask (STC) for internal events that potentially lead to loss of containment or loss of shielding. In particular, the applicant's information is acceptable because (i) standard engineering practices were used appropriately to estimate reliability of the SSCs, (ii) modeling techniques were used appropriately to estimate reliability, and (iii) the analysis appropriately considered uncertainty in test data for material behavior in the drop and collision tests used to estimate reliability.

2.1.1.4.3.3.1.2 Passive Structures, Systems, and Components Reliability for Seismic Events

This SER section provides the NRC staff's review of the reliability of the passive important-to-safety (ITS) structures, systems, and components (SSC) for structural challenges resulting from seismic events. The NRC staff's review is presented in SER Section 2.1.1.4.3.3.1.2.1 for surface civil structures and in SER Section 2.1.1.4.3.3.1.2.2 for mechanical equipment and systems.

2.1.1.4.3.3.1.2.1 Surface Facilities Buildings

The applicant provided information on seismic performance of surface facilities buildings in SAR Sections 1.7.1.4 and 1.7.2.4 and BSC (2008bg). The applicant described the methodology for seismic performance evaluation of surface facilities buildings in SAR Section 1.7.1.4. The methodology to develop the fragility of surface facilities buildings was summarized in SAR Section 1.7.2.4. The applicant's approach was to demonstrate that the probability of failure of a building structure is a beyond Category 2 event sequence, which therefore precludes the need for a radiological dose computation. The applicant presented the parameters used to develop fragility curves (i.e., median capacity and dispersion) and estimates of the annual failure probability for the surface facilities in BSC (2008bg, Table 6.2-1).

The ITS surface facilities reviewed in this section are the Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), and Wet Handling Facility (WHF). The NRC staff reviewed the adequacy of the information presented in the SAR with respect to (i) mean annual probability of failure, (ii) the approach used to generate fragility curves for ITS structural facilities, and (iii) appropriateness of the capacity and uncertainty data used to develop fragility curves for surface facilities.

For the ITS surface facilities, the applicant stated that seismic loading controls the structural performance (SER Section 2.1.1.7.3.1.1) and evaluated the structural performance at the Limit State A [American Society of Civil Engineers (2005aa, Table 1-1)], which is defined as Large Permanent Distortion (short of collapse, but structurally stable). The applicant defined four stages to conduct seismically initiated event sequences: (i) development of seismic event sequences, (ii) development of seismic hazard curves for surface facilities, (iii) evaluation of seismic fragilities, and (iv) quantification of event sequences. To evaluate event sequences associated with structure failure, the applicant assumed that unfiltered radionuclide release will occur after the structure collapse (e.g., BSC, 2008bg). Therefore, the NRC staff's review evaluated the seismic fragility curves for the Limit State A for the ITS surface facilities and the calculation of the mean probabilities of failure of the ITS surface facilities.

**Mean Annual Probability of Failure**

The applicant assessed the structural performance of ITS surface facilities by computing the mean annual probability of failure of the ITS surface facilities. To obtain the mean annual probability of failure of a surface facility, the applicant convolved the seismic hazard curve with the seismic fragility curve. The seismic hazard curve plots the mean annual probability of exceedance of a given earthquake peak ground acceleration. The seismic hazard curve for the surface facilities at the site is provided by the applicant in SAR Section 1.1.5 and reviewed by the NRC staff in SER Section 2.1.1.1. The mean annual probability of failure for each facility was then compared to the Category 2 event sequence probability of occurrence of 1 in 10,000 over the preclosure period (note: the applicant has expressed the same occurrence frequency using an annual frequency of occurrence of $1 \times 10^{-6}$ over a 100-year-preclosure period and an annual frequency of occurrence of $2 \times 10^{-6}$ over a 50-year-exposure time for specific activities during the preclosure period).

The mean annual probability of failure for each of the ITS buildings was presented in BSC (2008bg, Table 6.2-1). The applicant also provided a table summarizing the computation of the mean annual probability of failure, $P_f$, for the Canister Receipt and Closure Facility (CRCF) in DOE (2009dz, Enclosure 3), which the applicant considered representative of the ITS buildings. The annual probability of failure of the ITS surface facility buildings vary from $3.8 \times 10^{-7}$ to $8.7 \times 10^{-7}$.

**NRC Staff's Evaluation**

The NRC staff reviewed the methodology the applicant provided for computing the probability of failure using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–01 (NRC, 2006ad). The NRC staff finds that the applicant's methodology for determining seismic performance of the ITS surface facility buildings as the annual probability of failure, using the convolution of the seismic hazard curve and the seismic fragility curve, is acceptable because it is consistent with standard engineering practice (American Society of Civil Engineers, 2005aa) and the NRC guidance in HLWRS–ISG–01 (NRC, 2006ad). The applicant's use of the probability threshold of $2 \times 10^{-6}$ per year for Category 2 event sequences for surface facilities, based on the exposure time during waste emplacement (the first 50 years of the preclosure period), is reviewed by the NRC staff in SER Sections 2.1.1.3.3.1.3.2 and 2.1.1.3.3.1.3.3 and found to be appropriate.

**Methodology for Generation of Seismic Fragility Curves**

The applicant used a simplified methodology for developing seismic fragility curves for ITS surface facilities in SAR Section 1.7.2.4 and DOE (2007ab) using the conservative deterministic failure margin (CDFM) method developed by the Electric Power Research Institute (EPRI, 1994aa) to assess the capacity of a structure for seismic events beyond design basis ground motion. In this methodology, the applicant used a lognormal distribution to represent the mean fragility curve of a surface facility as a function of the horizontal peak ground acceleration. The applicant used two parameters to develop the fragility curve: (i) the peak ground acceleration at which the probability of failure of the structure is 1 percent ($C_{1\%}$) and (ii) a composite logarithmic standard deviation ($\beta_c$) assumed based on engineering judgment and described in DOE (2007ab, Sections 4.2 and 4.4.2). Assuming a lognormal distribution, DOE generated the fragility data by anchoring the curve at $C_{1\%}$ and extrapolating the rest of the fragility curve based on $\beta_c$. The only fragility curve parameter directly obtained from structural analysis of the building is $C_{1\%}$, which was obtained from a simplified elastic model, Tier #1 (see SER Section 2.1.1.7.3.1.1 for a detailed discussion). Additionally, in its response to the NRC staff's RAI (DOE, 2009dz), DOE provided the fragility calculation for the CRCF (BSC, 2007df).

**NRC Staff's Evaluation**

The NRC staff reviewed the methodology the applicant provided to generate the fragility curves for the ITS surface using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–01 (NRC, 2006ad). The NRC staff determines that the CDFM method the applicant used to generate the fragility curves for the ITS surface facilities included (i) computation of the capacity at 1 percent conditional probability of failure; $C_{1\%}$; (ii) assumption of a lognormal distribution; and (iii) assumed composite logarithmic standard deviation, $\beta_c$, based on engineering judgment. The NRC staff finds that use of the CDFM method is acceptable because it is consistent with the standard industry practices for nuclear facility structures (American Society of Civil Engineers, 2005aa) and nuclear power plant structures, which are similar to GROA structures (EPRI, 1994aa) and NRC staff guidance in HLWRS–ISG–01 (NRC, 2006ad).

**Computation of $C_{1\%}$ Capacity**

To obtain $C_{1\%}$, the applicant performed elastic analyses at beyond design basis ground motion (BDBGM) seismic levels to compute the high confidence of low probability of failure (HCLPF) capacity of the system, which was considered a reasonable approximation of $C_{1\%}$. The applicant computed the HCLPF capacity on the basis of the conservative deterministic failure margin method.

The applicant designed the ITS surface facilities to withstand DBGM–2 seismic levels, which correspond to a mean annual probability of exceedance (MAPE) of $5.0 \times 10^{-4}$, as detailed in DOE (2007ab, Section 3.1.1), and a horizontal peak ground acceleration (PGA) of 0.45g (BSC, 2007ba). The applicant based the design of surface facilities on simplified linear elastic analyses (Tier #1 models) for CRCF, WHF, and RF facilities (SAR Table 1.2.3.2-2) that the NRC staff reviewed in SER Section 2.1.1.7.3.1.1 and found acceptable. The applicant based Tier #1 analyses on lumped-mass, multiple-stick models, in which floors were considered rigid slabs and soil–structure interaction was approximated using equivalent linear soil springs. For the IHF, the applicant used a finite-element model for the superstructure on a fixed-base support, based on the guidance in ASCE 4–98 (American Society of Civil Engineers, 2000aa).

On the basis of the structural configuration obtained from the design for DBGM–2 events, the applicant performed linear elastic analyses based on Tier #1 models using BDBGM seismic events with a MAPE of $1.0 \times 10^{-4}$ and horizontal PGA of 0.91g, as detailed in BSC (2007ba, Section B.4.2). The applicant used the results from analyses for BDBGM events to compute $C_{1\%}$ of the fragility curves. The applicant used the BDBGM at the MAPE of $1.0 \times 10^{-4}$ (PGA of 0.91g) for the seismic fragility evaluation of SSCs ITS, to ensure that SSCs ITS will remain within acceptable inelastic limits for twice the DBGM horizontal PGA of 0.45g.

In its response to an NRC staff's request for additional information on using the Tier #1 models for seismic analyses of the Canister Receipt and Closure Facility (CRCF), Wet Handling Facility, and Receipt Facility (RF) buildings, the applicant stated that the Tier #1 models provided realistic forces and moments that were sufficient for the initial design and the seismic performance evaluation for use in the PCSA. The applicant explained that the Tier #1 analyses results were conservative because (i) the out-of-plane resistance of the cross-walls, which would increase the stiffness of the structure, is not considered and (ii) the fundamental system frequencies used in the analysis are close to the frequencies at which the peak acceleration occurs in the seismic ground design response spectra (DOE, 2009gh, Enclosure 1).

In its response to the NRC staff's RAI in DOE (DOE, 2009dn, Enclosure 8) regarding not performing nonlinear structural analyses to generate fragility curves, the applicant stated that the inelastic energy absorption factor, $F_{\mu}$, values (Electric Power Research Institute, 1991aa) accounted for the inelastic behavior adequately. The applicant also stated that the nonlinear time history analyses performed for the Diablo Canyon Turbine Building (Kennedy, et al., 1988aa) showed excellent agreement between the $F_{\mu}$ values derived from the nonlinear analyses and those obtained from a simplified approach (Electric Power Research Institute, 1991aa). The applicant added that because the Diablo Canyon structure exhibited structural irregularities (asymmetric structural properties for the plan and vertical configurations), the applicant's approach for estimating $F_{\mu}$ values is representative for structures with structural irregularities similar to the Yucca Mountain surface facilities buildings (DOE, 2009dn).

To determine seismic fragility of a surface facility building, the applicant considered various failure modes of the building elements, including in-plane shear, out-of-plane shear and bending failures of walls and floor slabs, and axial force in combination with in-plane bending of walls. The applicant chose two failure modes to determine the $C_{1\%}$ or $C_{HCLPF}$ capacity: (i) in-plane shear for walls and (ii) out-of-plane bending and shear of floor slabs, based on nuclear industry experience in performing seismic probabilistic risk assessment (PRA) of similar facilities at nuclear power plants. The applicant stated in BSC (2007df) that the $C_{1\%}$ or $C_{HCLPF}$ capacity of a surface facility building is governed by the lowest in-plane shear $C_{1\%}$ or $C_{HCLPF}$ capacity of the walls. For low-rise shear walls, the applicant computed the HCLPF capacity on the basis of

American Society of Civil Engineers (2005aa, Equation 4-3) and BSC (2007ba, Section B4.3, Step 3).

**NRC Staff's Evaluation**

The NRC staff reviewed the information the applicant provided for the computation of the $C_{1\%}$ or the high-confidence-low-probability of failure, $C_{HCLPF}$, capacity for generation of a surface facility building seismic fragility curve, using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–01 (NRC, 2006ad).

On the basis of the review of the applicant's calculations in SER Section 2.1.1.7.3.1.1 for the WHF and IHF buildings, the NRC staff finds that the maximum soil foundation bearing pressures for the WHF and IHF buildings would lead to the maximum soil demand-to-capacity ratio of 0.88[1] under DBGM–2 seismic events for a horizontal peak ground acceleration (PGA) of about 0.45g. The NRC staff notes that the elastic analysis results for the DBGM–2 indicate that soil may experience inelastic behavior during the seismic events beyond DBGM–2. The NRC staff determines that potential inelastic behavior of soil during a seismic event beyond DBGM–2 would decrease the soil-structure system frequency and reduce the seismic effects on the surface facilities buildings; and thus, the elastic seismic analyses the applicant used is conservative.

Therefore, the NRC staff finds that the applicant's use of the Tier #1 linear-elastic structural analysis modeling technique, using the lumped-mass, multi-stick models, is acceptable because (i) structural models similar to the Tier #1 models have been used by the nuclear industry for seismic analyses of similar surface buildings for nuclear power plants and approved by the NRC (NUREG–0800, Section 3.7.2; ASCE, 2000aa); (ii) inelastic behavior of structures at seismic ground motions greater than BDBGM is accounted for by considering the energy absorption factor, $F_\mu$; (iii) the effects of structural irregularities causing increase of shear forces in the walls is captured in the lumped-mass multi-stick models because each wall mass and its rigidity are modeled; and (iv) the Tier #1 analysis is conservative (i.e., the fundamental system frequencies used in the analysis are close to the frequencies at which the peak acceleration occurs in the seismic ground design response spectra).

The NRC staff finds that the applicant's use of the in-plane, shear-mode failure of the walls as representing the seismic fragility of the building is acceptable because the lateral seismic forces are transmitted primarily through the concrete shear-walls to the foundation in the CRCF, RF, and WHF. Therefore, the in-plane shear mode failure of the wall with the lowest capacity would yield the lowest seismic fragility values and the highest annual probability of failure of the surface building. The approach of using the shear-mode failure of the concrete walls as representing the seismic fragility of the concrete shear-wall buildings is consistent with the standard industry practices for nuclear facility structures, similar to the structures at the GROA (American Society of Civil Engineers, 2005aa). The NRC staff also finds that the applicant used the capacity equation for low-rise rectangular concrete shear walls, consistent with Section 4.2.3 of the American Society of Civil Engineers, ASCE 43–05 (2005aa) for shear walls with boundary elements[2] or end walls, for all cases at the Canister Receipt and Closure Facility (CRCF) (BSC, 2007df). The ASCE equation, detailed in American Society of Civil Engineers (2005aa, Section 4.2.3), is only applicable to shear walls with boundary elements or end walls

[1]The soil demand-to-capacity ratio is based on the soil allowable bearing capacity of 2,394 kPa [50 ksf] (SER Section 2.1.1.1.3.5.4).
[2]Shear walls with boundary elements have flanges or perpendicular walls at both ends. Rectangular shear walls have no flanges or walls at the ends. Thus, the horizontal cross section of the wall is rectangular.

and overestimates the capacity of shear walls without boundary elements (Hwang, et al., 2001aa; Gulec, et al., 2008aa). Reduction of shear strength of shear walls leads to a higher probability of building failure. In response to the NRC staff's RAI in DOE (2009dz, Enclosure 4), the applicant stated that, where the design does not include end or cross walls (e.g., a pier between openings), the vertical reinforcement displaced by the opening is placed as additional confinement reinforcement on the two sides of the opening, in accordance with ASCE 43–05 code (Section 4.2.3) for use of the shear-wall capacity equation. The NRC staff finds the applicant's use of additional reinforcement at the openings in walls to strengthen the area, in accordance with the ASCE 43–05 (2005aa) standard for boundary elements, is acceptable because it is consistent with the surface facilities design code (SAR 1.2.2.1.8) requirements in Chapter 21—Provisions for Seismic Design (American Concrete Institute, 2001aa).

Based on the foregoing evaluation, the NRC staff finds that the applicant's methodology for estimation of $C_{1\%}$ for development of the seismic fragility curves for the preclosure safety analysis of the Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Initial Handling Facility (IHF), and Receipt Facility (RF), is appropriate, because the methodology is consistent with standard industry practices (e.g., ASCE, ACI Standards).

**Estimation of $\beta_c$**

To develop the lognormally distributed seismic fragility curves for the surface facilities buildings, the applicant assumed the composite logarithmic standard deviation, $\beta_c$, equal to 0.4, the midpoint from the range of 0.3 to 0.5 recommended for structures and equipment mounted at ground level, as representative of the mean value for the seismic event sequence quantification, as described in DOE (2007ab, Section 4.4.2) and American Society of Civil Engineers (2005aa, Section C2.2.1.2). In response to the NRC staff's RAI in DOE (2009ge, Enclosure 8), the applicant indicated that $\beta_c$ estimates higher than 0.52 were not credible and that $\beta_c$ estimates exceeding 0.4 were the result of simplified evaluations. The applicant also stated that the use of the lower $\beta_c$ value would be conservative and would yield higher annual probability of failure of the surface facilities. The applicant stated, based on the example study of the CRCF building, that, using $\beta_c$ equal to 0.3 instead of 0.4, would increase the CRCF annual probability of failure from $8.1 \times 10^{-7}$ to $1.2 \times 10^{-6}$, which results in a Category 2 event sequence over the preclosure period that is less than 1 chance in 10,000 over the preclosure period, as described in DOE (2009ge, Enclosure 8).

**NRC Staff's Evaluation**

The NRC reviewed the applicant's basis for composite logarithmic standard deviation, $\beta_{c,}$ using the guidance in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–01 (NRC, 2006ad). The applicant used $\beta_c$ equal to 0.4, the midpoint between the recommended range of 0.3 to 0.5 in ASCE 43–05 (American Society of Civil Engineers, 2005aa). The NRC staff finds that the use of $\beta_c$ equal to 0.4 is reasonable for use in the preclosure safety analysis of a surface facility building because it represents a mean value of the range of 0.3 to 0.5, derived from a number of probabilistic risk studies of similar facilities at nuclear power plants (ASCE, 2005aa). The NRC staff finds that the results of the applicant's study of the CRCF building on sensitivity of the annual probability of failure to $\beta_c$ are acceptable because lowering the $\beta_c$ from 0.4 to 0.3 resulted in an increase of the annual probability of failure by a factor of approximately 1.5, which is consistent with other industry studies (American Society of Civil Engineers, 2005aa, Kennedy, 1999aa).

2.1.1.4.3.3.1.2.2 Mechanical Equipment and Systems

The applicant provided information on seismic fragilities of the mechanical equipment in SAR Section 1.7 and in BSC (2008bg, Section 6.2). The applicant then used this information to determine the reliability of mechanical equipment during seismically initiated event sequences. Seismic fragilities are defined as the conditional probability of equipment to perform its function at different values of a selected seismic ground motion. DOE used PGA to define the fragility curve.

The applicant developed fragility parameters for mechanical structures (e.g., cask preparation platform, mobile platform, shield door, entry door) and handling equipment [e.g., crane, canister transfer machine (CTM), transfer trolley, and transport and emplacement vehicle (TEV), as shown in BSC (2008bg, Table 6.2.2)]. The applicant identified the failure modes of the equipment under seismic loads and provided median capacity, composite uncertainty, and annual probability of failure associated with the failure modes, as outlined in BSC (2008bg, Table 6.2.2). The annual probability of failure was calculated by convolving the fragility curve and the seismic hazard curve.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on the applicant's approach for assessing seismic fragility of mechanical equipment and systems using the guidance provided in YMRP Section 2.1.1.4, as supplemented by interim staff guidance for seismically initiated event sequences (HLWRS–ISG–01; NRC, 2006ad) and reliability estimation (HLWRS–ISG–02; NRC, 2007ab). In its response to the NRC staff's RAI (DOE, 2009gd), the applicant provided supporting calculations (BSC, 2008co) for equipment fragilities listed in BSC (2008bg, Table 6.2-2). The NRC staff reviewed the fragility calculations for the cask handling crane (CHC) because the CHC is used to handle casks in all surface facilities (CRCF, RF, WHF and IHF), and thus is a risk-significant piece of equipment. The NRC staff reviewed the applicant's information on mechanical systems to verify that (i) the applicant's methodology to develop the seismic fragility curve for the CHC is consistent with accepted engineering practices, (ii) the methodology was applied appropriately, and (iii) the resulting fragility parameters are reasonable. The applicant stated that the CHC will be designed in accordance with the ASME NOG–1–2004 standard for a Type I overhead bridge crane. The applicant estimated the seismic fragility parameters of the CHC girder structural failure as (i) median capacity of 2.79g; (ii) high-confidence-low-probability failure (HCLPF) or capacity at 1 percent failure probability of 0.98g; and (iii) $\beta_c$ of 0.45. The applicant's estimate accounted for the conservatism in code-strength equations, in-structure response spectra, and nonlinear behavior at higher seismic ground motion level. The NRC staff finds that the applicant's methodology and the assumptions made by the applicant to account for the design conservatism and develop seismic fragility parameters for the CHC are appropriate, because they are consistent with the standard nuclear industry practices for seismic fragility analysis of nuclear power plants SSCs (Kennedy, et al., 1980aa; Kennedy and Ravindra, 1984aa; Electric Power Research Institute, 1994aa). The methodology for evaluating seismic performance of mechanical handling equipment in nuclear power plants provided by Kennedy, et al., 1980aa; Kennedy and Ravindra, 1984aa; and Electric Power Research Institute, 1994aa is applicable to mechanical handling equipment at the GROA because (i) the equipment and radioactive materials handling operations at the GROA are similar to those in a nuclear power plant and (ii) the methodology is independent of the facility type and location. Additionally, the NRC staff verified the applicant's determination of the CHC fragility parameters (HCLPF = 0.98g, $\beta_c$ = 0.45) and finds that the CHC fragility parameters are reasonable, because the HCLPF capacity of 0.98g accounts appropriately for the design margins in the ASME NOG–1–2004 code (ASME, 2005aa) and the inelastic material behavior.

The NRC staff finds that the applicant's failure probabilities of mechanical equipment during seismically initiated event sequences, as shown in BSC (2008bg, Table 6.2.2), are adequate because (i) the applicant's analysis is consistent with the repository design and operation information for the mechanical equipment and systems and (ii) the analysis constructs the fragility curves consistent with the methodology used for similar mechanical handling equipment and systems used for nuclear power plants (Kennedy, et al., 1980aa; Kennedy and Ravindra, 1984aa; Electric Power Research Institute, 1994aa) and the NRC staff guidance in HLWRS–ISG–01 (NRC, 2006ad).

**NRC Staff's Conclusion**

Based on the NRC staff's review in SER Section 2.1.1.4.3.3.1.2, the NRC staff finds that the applicant provided acceptable information on passive reliability of surface facilities buildings and mechanical systems and equipment for seismic events because (i) standard engineering practices were used appropriately to estimate reliability of the SSCs; (ii) modeling techniques were used appropriately to estimate reliability; (iii) the applicant considered uncertainties in the supporting numerical models, structural system parameters, and demands consistent with NRC staff guidance in HLWRS–ISG–02 (NRC, 2007ab); and (iv) standard industry practices were used appropriately to develop the seismic fragility curve, consistent with the NRC staff's guidance in HLWRS–ISG–01 (NRC, 2006ad).

2.1.1.4.3.3.1.2.3 Passive Reliability for Structural Challenges Resulting From Fire Events

The applicant provided information on the development of passive reliability probabilities for canister shielding and canister containment during fire-induced thermal challenges in SAR Sections 1.7.2.3.3 and 1.7.2.3.4 and BSC (2007ab,aw,bb,bf; BSC, 2008ap,bp).

The applicant developed passive reliability of SSCs subjected to fire challenges on the basis of either probabilistic analysis (e.g., probability of canister failure due to fire exposure) or basic design assumptions (e.g., assessment of concrete spalling under thermal challenges or the performance of low melting temperature shielding materials in a fire).

Dominant pivotal events in fire-related event sequences were the probability of canisters maintaining containment and the shielding safety function. The applicant estimated the probabilities of these pivotal events on the basis of an assessment of potential thermal challenges to various canisters and shielding configurations, and their predicted response to those exposures. SAR Section 1.7.2.3.3 summarized information on potential loss of containment or breach under thermal challenges, and SAR Section 1.7.2.3.4 summarized information on loss of shielding under thermal challenges.

The exposure profile selected by the applicant was based on fuel sources for fires found in handling and storage facilities that could ignite and burn in proximity to waste packages. The sources included cable trays, electrical cabinets, and liquid hydrocarbon fuels.

### Thermal Challenges and Loss of Containment

The applicant characterized the thermal demands on a canister as the canister wall temperature resulting from a fire exposure of a certain temperature and duration. To quantify the demands, the applicant postulated fire exposure conditions and calculated the expected canister wall temperatures that could result from those exposures. The applicant used the fire data from large-scale tests conducted by different laboratories to develop a reasonable distribution of fire durations. The assessment assumes that the automatic sprinkler system was not classified as important to safety (ITS); therefore, the analysis assumed a fire duration in the absence of any automatic fire protection. The applicant stated that, in a building without sprinklers, 10 percent of fires would have a duration of 10 minutes or less, and 90 percent of fires would have a duration of 60 minutes or less. This was based on a series of tests performed by multiple laboratories using fuel sources that are typical of those found in handling and storage facilities. During the tests, packages were ignited and the fire duration in the absence of manual intervention or the intervention of an automatic sprinkler system was recorded (BSC, 2008ac,as,au,be,bk,bq). Based on these test data, a lognormal distribution was assigned to the fire duration.

The applicant also quantified the intensity of the fire upon arrival at a cask containing a waste form, so that the reliability or fragility of the cask under a set of thermal exposure conditions could be evaluated (BSC, 2008ac,as,au,be,bk,bq). The applicant took the fire temperature as the temperature of the burning material, assuming the burning material is a uniform source emitting energy as efficiently as possible (known as a blackbody). This blackbody temperature, which is typically higher than the actual measured flame temperature, is an estimate of the actual flame temperature and is commonly published in literature in reference material for solid (e.g., wood, paper, or plastic) and liquid (e.g., hydrocarbon) fuels (Society of Fire Protection Engineers, 2002aa). The applicant also used flammable liquid fire data obtained from large-scale hydrocarbon fires involving railcars (Birk A.M. Engineering, 2005aa). The applicant reported effective temperatures ranging from 400 to 1,200 °C [752 to 2,192 °F] in fires involving solid fuel materials and temperatures from 927 to 1,327 °C [1,701 to 2,421 °F] in flammable-liquid pool fires. The applicant used these data to develop a range of potential fire temperatures, which was represented by a normal distribution having a mean of 799 °C [1,470 °F] and a coefficient of variation of 16 percent. On the basis of the normal distribution of fire temperature (BSC, 2008ac,as,au,be,bk,bq), the applicant indicated that 99.9 percent of all fires would have a temperature lower than 1,330 °C [2,426 °F].

The applicant calculated the heat transfer to bare fuels and canisters inside casks using standard heat transfer models that were validated using finite-element analyses. The ultimate wall temperature of a canister exposed to a fire is a function of fire duration, exposure temperature, and the physical properties of the container. The applicant used Monte Carlo simulations of fire temperature and duration, coupled with the heat transfer models described previously, to generate a distribution of potential canister wall temperatures. The canister wall temperatures were calculated for various canister types (e.g., thick-walled and thin-walled canisters) in various configurations (e.g., in waste packages, transportation casks, a shield bell). The applicant excluded the failure of canisters outside of a cask, waste package, or other enclosing structure during transfer operations because of the very low probability that a canister would be in this state during a fire event.

The applicant evaluated the canister response to a thermal challenge on the basis of the canister's ability to withstand stresses induced by the elevated temperature. The applicant stated that creep-induced failure and limit load failure were two possible failure modes

(BSC, 2008ac,as,au,be,bk,bq) and described the canister temperatures that could result in failures. The applicant evaluated both failure modes independently.

When the range of thermal demands and responses were identified, the applicant developed a demand curve using the range of canister wall temperatures resulting from a distribution of fire exposure temperatures and durations, and a corresponding response curve based on load limit and creep-induced failure probability as a function of canister temperatures. The superposition of the demand curve and the response curve yielded the number of expected failures that would occur in a given number of trials. The number of "observed" failures divided by the number of trials was taken as the failure probability.

Demand curves (showing resulting canister wall temperatures) were based on 100,000 to 1,000,000 Monte Carlo trials of exposure temperature and duration. The goal of the analysis was to select a sufficiently large number of trials that would generate a sufficient number of failures. The applicant assigned a failure probability of less than $10^{-6}$ for cases with no observed failures after 1,000,000 trials.

The applicant calculated failure probabilities of six canister configurations and also calculated the failure probability of bare fuel in a standard transportation cask (BSC, 2008ac,as,au,be,bk,bq). The resulting failure probabilities for canisters in different configurations ranged from $1.0 \times 10^{-6}$ to $3.2 \times 10^{-4}$, as shown in BSC Table 6.3-5 (2008ac,as,au,be,bk,bq). The failure probability of $5.4 \times 10^{-4}$ for the bare fuel in a transportation cask was estimated on the basis of a failure temperature of 700° C [1,292° F], as shown in BSC (2008ac,as,au,be,bk,bq, Table D2.1-10).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's passive reliability of canisters under thermal challenges using the guidance provided in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab). The NRC staff finds that the applicant appropriately excluded the structural response of bare canisters to fire events because bare canisters outside of a waste package, cask, CTM, or shield bell are not credible, given the operational sequences of the facilities. The remainder of the NRC staff's evaluation is for bare fuel and canisters inside casks.

The NRC staff finds that the applicant's estimate of fire duration and the distribution of fire temperature are adequate because (i) the analyses are consistent with the design and operation of the GROA facilities (e.g., fuel density of commodities expected in the GROA facilities for demand calculations); (ii) the applicant considered National Fire Protection Association historical data (Ahrens, 2007aa) for different sizes of fires and their ability to propagate over time through a facility (see SER Section 2.1.1.3.3.2.3.4.2 for further details); (iii) the applicant used test data based on fires where sprinklers were not assumed to operate, as shown in BSC (2008ac,as,au,be,bk,bq, Table F.II-2); (iv) the fire temperature rises to a peak and decreases when the combustible materials are consumed in the fire; however, the applicant modeled a more conservative "steady-state" exposure (i.e., a constant fire temperature); and (v) the fire is assumed to be engulfing and to occur next to the waste containers.

The NRC staff finds the applicant's thermal demands are acceptable because (i) the applicant's mean fire temperature of 799 °C [1,470 °F] and standard deviation of 172 °C [342 °F] (using a normal distribution) is consistent with the fire exposure temperature for hypothetical accident conditions outlined in the Standard Review Plan for Transportation Packages for Spent fuel (NRC, 2000aj, NUREG–1617); (ii) the applicant's use of published blackbody temperatures of standard fuels is a conservative approach because the blackbody temperature assumes emission of thermal energy at a much higher efficiency than occurs during actual fuel combustion; (iii) the assumed standard deviation for the temperature encompasses the expected temperatures from a range of ordinary combustibles (e.g., paper, wood, and plastic), as well as concentrated amounts of plastics or flammable liquids capable of producing fires at higher temperatures; and (iv) the applicant's thermal demands are consistent with exposures derived from large-scale testing and industry-accepted data (e.g., Society of Fire Protection Engineers, 2002aa).

The NRC staff finds that the overall methods and data the applicant used to determine canister response to thermal events are appropriate to calculate response parameters, such as creep and limit load, because the calculations were based on fundamental structural mechanics and utilized readily available material property data.

Based on the NRC staff's evaluation, the applicant's failure probabilities for loss of containment during a fire, as reported in Tables 6.3-2 and D2.1-10 in BSC (2008ac) are adequate because (i) the applicant's estimate of fire duration and the distribution of fire temperature are adequate; (ii) the applicant's analyses were based on fundamental structural mechanics and utilized readily available material property data consistent with NRC staff guidance for reliability estimation at HLWRS–ISG–02 (NRC, 2007ab).

**Thermal Challenges and Loss of Shielding**

The applicant stated that thermal challenges to transportation casks may degrade the cask shielding by impacting any low–melting-temperature shield materials present in the cask. In contrast, the canister transfer machine (CTM) shielding is not assumed to be affected by thermal challenges due to the high melting temperature of the uranium shield material and the absence of combustible materials in proximity to the CTM or shield bell. The applicant further stated that, for concrete shielding in aging overpacks (AOs), the shielding degradation mechanisms include spalling, cracking, or other physical damage to the concrete encasement.

The applicant described the loss of shielding in transportation casks due to a thermal challenge in BSC (2008ac,as,au,be,bk,bq, Section D2.2.1). The applicant indicated that all transportation casks have separate gamma and neutron shields. Because the neutron shield is typically fabricated from a low-melting-temperature polymer, the applicant stated that its shielding function would be quickly lost when subjected to a thermal challenge; therefore, the neutron shield would govern the loss-of-shielding event in the majority of transportation casks. Gamma shielding may also be present and can take a number of different forms, depending on the cask design; however, the steel/lead/steel design was identified as the design most likely to result in loss of shielding due to fire. The applicant stated that the mode of failure in this design would be melting and subsequent discharge of molten lead as a result of long-term heating and some form of physical damage. Because molten lead behavior could not be fully characterized, the applicant assigned a probability of failure of 1.0 for loss of transportation cask shielding due to a thermal challenge. The applicant also conservatively applied a failure probability of 1.0 to all transportation casks that use other forms of shielding, as shown in BSC (2008ac,as,au,be,bk,bq, Table D3.4-1).

The applicant described the loss of shielding in aging overpacks (AOs) due to a thermal challenge in BSC (2008ac,as,au,be,bk,bq, Section D2.2.3). Because concrete thickness {roughly 860 mm [34 in]} provides AO shielding, the primary concern is loss of concrete thickness due to spalling. The applicant discussed the predicted dose as a function of concrete loss due to spalling and demonstrated that up to 20 percent {345 mm [13.6 in.]} of the AO concrete thickness could be lost due to spalling without resulting in an unacceptable worker dose. The corresponding thermal exposure study and AO configurations described in BSC (2008ac,as,au,be,bk,bq, Section D2.2.3.2) showed that this amount of spalling would not be reached during the postulated thermal challenges, because the high permeability concrete used in the design of the AO is robust to spallation from thermal challenges. As a result, the applicant assumed a loss-of-shielding probability of 0.0 for AOs.

The applicant also considered loss of shielding from thermal challenges during transfer operations using the canister transfer machine (CTM) and indicated that loss of shielding in the CTM was based on the failure of a shield bell that encompasses the waste container. The applicant assumed that loss of CTM shielding probability was 0.0 because the shield bell components (primarily depleted uranium) have very high melting temperatures {3,400 °C [6,152 °F]}. Additionally, the applicant stated that absence of combustible materials in proximity to the CTM made it highly unlikely that these melting temperatures would ever be achieved.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's evaluation of the reliability of passive shielding systems under thermal challenges using the guidance provided in YMRP Section 2.1.1.4, as supplemented by HLWRS–ISG–02 (NRC, 2007ab).

The NRC staff finds that the applicant's reliability probabilities for the shield materials were based on the configuration of various shielding systems (polymers, lead, concrete, depleted uranium), coupled with material properties for each shield material consistent with design and operations of shielding material at the GROA. Material properties that drive the loss of shielding parameter are (i) the melting temperature of lead or polymers (transportation cask); (ii) the thermal performance of uranium (shield bell); and (iii) the thermal performance of concrete (aging overpack). The NRC staff verified that the material properties for the shielding in BSC (2008ac,as,au,be,bk,bq) were consistent with published data for these materials. The applicant's analysis of transportation casks and the shield bell shielding was focused on the melting temperature of shielding components, and the applicant's analysis of AO shielding, which is made of concrete, was based on concrete spalling characteristics and the shielding capacity of any remaining concrete.

The applicant assigned a failure probability of 1.0 for loss of transportation cask shielding because of the low melting temperature of shielding materials used in transportation casks (i.e., polymers and lead). The applicant's design for the canister transfer machine (CTM) is for the shielding material to be comprised of depleted uranium and did not assume failure of the shielding due to a thermal challenge. The NRC staff finds that the applicant made reasonable assumptions regarding the canister transfer machine (CTM) shield bell performance based on its material properties (high melting temperature of depleted uranium), and the lack of credible fire scenarios in proximity to the canister transfer operations.

The NRC staff finds the applicant's evaluation of spalling of the aging overpack (AO) due to fire is adequate because (i) significant spalling would need to occur to significantly reduce the effectiveness of the concrete AO; (ii) the AO concrete is classified as normal strength concrete, which is not prone to spalling from thermal challenges due to the high permeability; (iii) the AO design is with an outer steel liner that would diminish the effects of concrete spalling.

Based on the NRC staff's evaluation, the applicant's failure probabilities for degradation or loss of shielding during a fire, as reported in D3.4-1 (BSC, 2008ac) are adequate because (i) the applicant's analysis considered the design and operations for shielding; (ii) the applicant's analysis considered uncertainty in material properties in estimating reliabilities (e.g., low melting point for lead and uncertainty with molten lead); (ii) as documented in the above NRC staff evaluation, the applicant's estimates of fire duration and the distribution of fire temperature are adequate; and (iii) the applicant's analyses were based on fundamental structural mechanics and utilized readily available material property data consistent with NRC staff guidance for reliability estimation at HLWRS–ISG–02 (NRC, 2007ab).

**NRC Staff's Conclusion**

Based on the NRC staff's review in SER Section 2.1.1.4.3.3.1.3, the NRC staff finds that the applicant provided acceptable information on passive reliability of canister shielding and containment functions during fire-induced thermal challenges because the applicant's information is consistent with NRC staff guidance at HLRS–ISG–02 (NRC, 2007ab) and is based on: (i) characteristics for fires at the GROA facility and the design and operation of shielding that were considered in the applicant's analyses; (ii) standard engineering practices that were used appropriately to estimate reliability; (iii) the application of standard engineering practices, consistent with the design methodologies; (iv) modeling techniques that were used appropriately to estimate reliability; and (v) analysis that appropriately considered uncertainty in the reliability estimates.

2.1.1.4.3.3.2 Active Systems

The NRC staff's review of reliability of active systems includes the heating, ventilation, and air conditioning (HVAC) system and moderator intrusion control. The probability of ITS HVAC system failure was used as input to the "Confinement" pivotal event, and the loss of moderator control was used as input to the "Moderator" pivotal event in the System Response Trees.

2.1.1.4.3.3.2.1 Heating, Ventilating, and Air Conditioning Systems

The applicant discussed heating, ventilation, and air conditioning (HVAC) system reliability in SAR Sections 1.2.2.3, 1.2.4.4, and 1.2.5.5 and corresponding sections of its updated SAR (DOE, 2009av). The applicant included the confinement pivotal event in event sequences leading to a filtered radionuclide release end-state for the surface nuclear confinement ITS HVAC systems in the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF). The applicant developed fault trees to quantify the failure to maintain confinement, which it characterized in these fault trees as a loss of differential pressure in the CRCF and WHF. The applicant used the results from its fault tree analyses to specify the controlling parameters for its nuclear safety design bases.

The applicant provided design information for the surface facilities HVAC systems in SAR Section 1.2.2.3. It described the surface nuclear confinement HVAC system specific to the CRCF in SAR Section 1.2.4.4 and BSC (2008ac) and specific to the WHF in SAR Section 1.2.5.5 and BSC (2008bq). Additionally, in the event of loss of offsite power, the emergency diesel generators supply power to the surface nuclear confinement ITS HVAC exhaust fans in the CRCF and WHF. The applicant described the surface non-confinement ITS HVAC system for the EDGF in SAR Section 1.2.8.3.

The surface nuclear confinement ITS HVAC system in the CRCF has one ITS subsystem that provides filtration following a potential radionuclide release and another subsystem that provides cooling to the ITS electrical equipment and battery rooms. The ITS subsystem providing filtration is referred to in this section as the ITS HVAC exhaust subsystem. The ITS HVAC exhaust subsystem is a two-train subsystem, in which one train is normally operating, and, upon failure of this operating train, the standby train will automatically start. The applicant provided a ventilation and instrumentation diagram in SAR Figure 1.2.4-101 (DOE, 2009av) for the operating train, which it refers to as Train A. This diagram showed one exhaust fan with an adjustable-speed drive and an interlock that connects to the standby train (i.e., Train B). SAR Figure 1.2.4-101 shows (i) three exhaust high efficiency particulate air (HEPA) filter plenums; (ii) various dampers (a tornado damper, manual isolation dampers, and a backdraft damper); (iii) differential pressure switches across the HEPA filter plenums and exhaust fan; and (iv) flow instrumentation and a radioactivity monitor. The diagram specifically designated the differential pressure switches across the exhaust fan and HEPA filter plenums as ITS and the flow instrumentation as ITS.

The applicant provided a fault tree model for loss of differential pressure in the CRCF in BSC (2009ab, Section B7.4). The applicant calculated a point estimate failure probability of $4.0 \times 10^{-2}$ and mean failure probability of $4.5 \times 10^{-2}$ for failure to maintain differential pressure in the CRCF, as outlined in BSC (2009ab, Figure B7.4-1). The associated controlling parameter in SAR Table 1.9-3 (DOE, 2009av) is $4.0 \times 10^{-2}$.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's HVAC system failure probability quantification using the guidance provided in YMRP Section 2.1.1.4. The NRC staff determines that the failure probability the applicant quantified for the surface nuclear confinement ITS HVAC system is acceptable because this failure probability is consistent with (i) the applicant's nuclear safety design basis controlling parameter of $4.0 \times 10^{-2}$ shown in SAR Tables 1.9-3 and 1.9-4 and (ii) the system-level data in IAEA–TECDOC–478 (IAEA, 1988aa). The NRC staff notes that IAEA–TECDOC–478 contains a database of component reliability data, including HVAC systems collected from publicly available literature. This database includes (i) derived reliability data that are used by nuclear power plants for probabilistic risk assessments; (ii) nuclear power plant-specific component failure data, including those reported to the NRC by the U.S. nuclear power plant operators; and (iii) data developed based on expert opinion using the nuclear and non-nuclear experiences (IAEA, 1988aa). This database lists mean failure probability of HVAC systems for the auxiliary buildings derived from reported plant-specific failure incidences: $6.1 \times 10^{-5}$/hour. The data designated as QVXRH in IAEA–TECDOC–478 (IAEA, 1988aa) refer to data for HVAC systems operating under normal environment and in alternating operating modes in auxiliary buildings at nuclear power plants. The IAEA failure rate data was compiled by updating the generic data with plant-specific data, which were obtained from plant operating records. The IAEA failure probability is equivalent to the applicant's mean probabilities of $4.4 \times 10^{-2}$ for a nuclear confinement HVAC system to become unavailable over the 30-day mission time the applicant provided as part of the HVAC design basis following a radionuclide release (SAR Table 1.9-3). The NRC staff finds that the failure probability of $4.0 \times 10^{-2}$ that the applicant quantified for the surface nuclear confinement ITS HVAC system is acceptable because (i) the applicant's evaluation considered the design and operations of the HVAC; and (ii) the applicant's failure probabilities are consistent with failure probabilities reported in IAEA–TECDOC–478 (IAEA, 1988aa) for similar HVAC systems.

2.1.1.4.3.3.2.2 Moderator Intrusion Control

The applicant identified the Design Criteria and Design Bases for the fire protection system in areas where there is the potential for breach of a canister in SAR Table 1.4.3-2 and discussed moderator intrusion in BSC (2008ac, Section 6.2.2.9), BSC (2008bq, Section 6.2.2.10), and similar sections in other reliability and event sequence categorization documents. The fault tree for moderator intrusion was provided in reliability and event sequence categorization analysis documents [e.g., BSC (2008ac, Figure B9.5-1)]. The applicant identified water from fire suppression systems, water from building service piping, and lubricating oil from overhead hydraulic equipment as potential moderator sources that could result in criticality concerns as an end state.

The applicant included specialized automatic fire suppression systems in the moderator control areas to reduce the potential for inadvertent water discharge due to spurious activation. The selected system is a double-interlock preaction (DIPA) system. The applicant identified these systems as important to safety (ITS) because of their role in preventing accidental moderator intrusion. A complete evaluation of the DIPA system is provided in SER Section 2.1.1.7.3.8. The applicant described the potential failure modes of this system in a fault tree (DOE, 2009fr). The fault tree included probabilities for inadvertent water introduction into sprinkler piping by either valve failure or human error and then evaluated the subsequent probability of spurious operation of the system, allowing trapped water to be released into the moderator control area.

For other water sources, such as domestic water pipes, the applicant used historical data (expressed as a failure rate per unit length of pipe) to derive a probability for other water sources introducing a moderator. These other potential sources were included in the moderator fault tree. The applicant also described the fault tree associated with the probability of lubricating oil leakage and the potential leakage path for lubricating oil to contact a breached canister. The data sources used in this fault tree were based on failure rates of a gearbox, which would introduce lubricating oil into the drip pan. This probability was superimposed with the probability that the drip pan will fail while the gearbox leak is undetected and oil is present in the pan. This potential condition of an undetected leak resulting in oil in the pan was assumed to persist for no more than 30 days, since a failure of the gearbox that results in leaking oil would terminate operation of a crane (BSC, 2008ac,as,au,be,bk,bq) and, thus, would be detected quickly. As part of the design basis for the handling equipment, the applicant specified the mean probability of inadvertent introduction of an oil moderator into a canister shall be less than or equal to $9 \times 10^{-5}$ over a 30-day period following a radioactive release (SAR Table 1.9-3).

The probability of moderator intrusion is a pivotal event in sequences that have potential moderator sources following a canister breach. The presence of a moderator determines whether an important-to-criticality end-state is realized. The applicant derived the probability of moderator introduction using a simple fault tree that combined the failure probability of water and lubricating oil moderator sources (e.g., overhead sprinkler system, building service piping, or crane gearbox and containment pan) and evaluated their potential intrusion into a breached canister. The applicant provided the nuclear safety design basis for the prevention of DIPA system failure in SAR Table 1.4.3-2. The probabilities of inadvertent operation and water intrusion ranged from $5.0 \times 10^{-7}$ to $1.0 \times 10^{-6}$. These probabilities were facility-dependent and primarily based on the number of sprinkler heads present in moderator-controlled areas.

For moderator intrusion to occur, a canister must have been breached before it is capable of accepting a moderator. The applicant assumed that breached canisters will be susceptible to moderator intrusion for a maximum of 30 days after a breach because a 30-day period would be sufficient time to allow canister containment to be reestablished or other mitigation procedures for moderator control to be put in place.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's evaluation of reliability of the moderator control system using the guidance provided in YMRP Section 2.1.1.4. The applicant calculated the actual failure probability of the DIPA system (including mechanical and human-induced failures) to be $2.0 \times 10^{-7}$ (DOE, 2009fr); however, the applicant specified higher failure probabilities of $1.0 \times 10^{-6}$ to $5.0 \times 10^{-7}$ as the design bases for the DIPA system, which is a conservative approach (i.e., higher likelihood of failure). The NRC staff finds that the applicant's calculated failure probability of the DIPA system is adequate because this estimate is in agreement with actual performance data on reliability of DIPA sprinkler systems (DOE, 2009fg), when installed, inspected, and tested in accordance with NFPA–13 (NFPA, 2007ab).

The NRC staff finds the applicant's failure probability for the introduction of lubricating oil as a moderator is adequate because the applicant (i) included the failure rates of gearboxes based on applicable industry data (Denson, et al., 1995aa), subsequent failure rates of the oil drip pan, and a maximum time between leak initiation and detection (30 days), which provides a sufficient time for detection of any significant oil leak; (ii) appropriately considered the design and operation of handling equipment (i.e., a breach of the crane gearbox, a concurrent leak of the containment pan, and the presence of a breached canister in a position where lubricating oil could intrude from above); and (iii) considered the sequence and combinations of basic events leading to the introduction of lubricating oil.

**NRC Staff's Conclusion**

Based on the NRC staff's review in SER Section 2.1.1.4.3.3.2, the NRC staff finds that the applicant provided acceptable information on reliability of active systems (HVAC systems and moderator intrusion control) because, consistent with staff guidance at HLRS–ISG–02 (NRC, 2007ab) (i) standard engineering practices were used appropriately to estimate reliability of the SSCs; (ii) modeling techniques were used appropriately to estimate reliability for moderator intrusion control; (iii) the applicant's failure probabilities for the HVAC system are consistent with failure probabilities reported in IAEA–TECDOC–478 (IAEA, 1988aa) for similar HVAC systems; and (iv) uncertainty in the reliability estimates were appropriately considered in the analysis, which included the use of industry data to support failure probabilities.

### 2.1.1.4.3.4 Event Sequence Quantification and Categorization

This section of the SER (Section 2.1.1.4.3.4 Event Sequence Quantification and Categorization) is the last of four primary sections that document the NRC staff's technical review of the identification of event sequences. The previous three sections document the NRC staff's review regarding the (i) Methodology for Identification and Categorization of Event Sequences (SER Section 2.1.14.3.1); (ii) Event Sequence Development (SER Section 2.1.14.3.2); and (iii) Reliability of Structures, Systems, and Components (SER Section 2.1.14.3.3). This section describes the NRC staff's review of the applicant's quantification and categorization of event sequences for its preclosure operations. The NRC staff's review relies on the following NRC staff evaluations provided earlier in SER Sections 2.1.1.4.3.1, 2.1.1.4.3.2, and 2.1.1.4.3.3: (i) the NRC staff's evaluation of the applicant's methodology, event sequences, and the associated reliability estimates that evaluate the basis for the quantification of the occurrence of event sequences and (ii) the NRC staff's evaluation of the applicant's categorization of the event sequences.

Consistent with the organization of other sections in this chapter, the remainder of this section contains three subsections: (i) Internal Events (SER Section 2.1.1.4.3.4.1), (ii) Seismic Events (SER Section 2.1.1.4.3.4.2), and (iii) Fire Events (SER Section 2.1.1.4.3.4.3).

2.1.1.4.3.4.1 Internal Events

The NRC staff's review of the applicant's event sequence quantification and categorization for internal events is described in the following three sections for the three major handling operations: (i) Canister and Cask Handling Operations, (ii) Wet Handling Operations, and (iii) Subsurface Operations. The NRC staff's review of internal event sequences does not consider those events initiated by either seismic events or fires, which are documented in SER Sections 2.1.1.4.3.4.2 and 2.1.1.4.3.4.3, respectively.

2.1.1.4.3.4.1.1 Canister and Cask Handling Operations

The applicant listed the internal event sequences for canister and cask handling operations in the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and the Intrasite operations and Balance of Plant Facility in SAR Tables 1.7-7, 1.7-9, 1.7-11, 1.7-13, and 1.7-15, respectively. These SAR tables include all Category 1, Category 2, and beyond Category 2 event sequences, listed in descending order of the expected number of occurrences during the operational period. As discussed in SER Section 2.1.1.4.3.1.4, the applicant's quantification and categorization of event sequences is based on the expected number of occurrences (i.e., the frequency of event sequence occurrence during the preclosure period). The applicant categorized as beyond Category 2 those event sequences with less than a 1 in 10,000 chance of occurring during the preclosure period. SAR Section 1.7.5 summarized the applicant's categorization analysis.

The applicant identified no Category 1 event sequences that could lead to exposure of individuals to radiation. Event sequences associated with direct exposure and filtered and unfiltered release of radionuclides were identified by the applicant as Category 2 event sequences.

Direct Exposure

The applicant identified event sequence frequencies for direct exposure during various waste handling operations at the Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), Wet Handling Facility (WHF), and Intrasite operations. For workers, categorization of event sequences in the applicant's analysis of this potential end-state results in direct exposure from the loss or degradation of shielding. The applicant concluded that the event sequence frequencies that could potentially cause radiological exposure to workers (SAR Tables 1.7-7, 9, 11, 13, and 15) are Category 2 or beyond Category 2.

Radionuclide Release

The applicant's results for the event sequences that may result in the end state of radionuclide release are depicted in SAR Table 1.7-11 for the Canister Receipt and Closure Facility (CRCF). For this facility, the applicant's results identified three Category 2 event sequences, with the other event sequences categorized as beyond Category 2. The Category 2 event sequences identified by the applicant were associated with structural challenges to HLW and TAD canisters during transfer by the canister transfer machine (CTM). Event sequences that involve breach of two sealed HLW canisters result in end states of filtered and unfiltered radionuclide releases (ESD09–HLW–SEQ5–RRU). An event sequence with the breach of one TAD canister results in an end state of filtered release. Similarly, for the Initial Handling Facility (IHF), one Category 2 event sequence was identified, involving an HLW canister (SAR Table 1.7-7). This event sequence was caused by structural challenges during transfer by the CTM. The remaining event sequences from structural challenges at this facility were categorized as beyond Category 2. At the Receipt Facility (RF) (SAR Table 1.7-9) and during Intrasite operations (SAR Table 1.7-15), all event sequences involving spent fuel and high-level waste were categorized as beyond Category 2. For the Wet Handling Facility (WHF), the applicant identified eleven Category 2 event sequences involving structural challenges, including potential releases from transportation casks with uncanistered SNF, from DPCs during preparation activities and cutting operations, and from TAD canisters in shielded transfer casks (STC) during transfer from the pool and drying of the canister (SAR Table 1.7-13). The applicant evaluated dose consequences for the Category 2 event sequences in accordance with the end states for filtered and unfiltered radionuclide releases, and assigned dose consequence designators.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's quantification and categorization for internal event sequences for canister and cask handling operations in SAR Section 1.7 using the guidance provided in YMRP Section 2.1.1.4. Based on the NRC staff's review, the NRC staff finds that the applicant provided adequate information on the quantification and categorization of internal event sequences for canister and cask handling operations because

(i) As documented in SER Section 2.1.1.4.3.1.1, the technical basis and justification the applicant provided for its methodology for the identification and categorization for internal event sequences, including the assumptions and methods for quantifying event sequences, is acceptable because it is consistent with applicable NRC guidance and standard practices.

(ii) As documented in SER 2.1.1.4.3.2.1.2, the event sequences developed for the internal events for canister and cask handling operations at the CRCF, IHF, RF, WHF, and for Intrasite operations (1) included appropriate initiating events for canister and cask handling operations (i.e., structural challenges and loss of shielding); (2) included the system response of SSCs to the initiating events for the canister and cask handling operations event sequences at surface facilities, consistent with the facility design and operations; (3) included direct exposure to workers, consistent with loss or degraded shielding associated with the canister and cask; (4) included filtered and unfiltered radiological release to the public and workers, consistent with the success or failure of the containment of the canister and cask, and of operation of the HVAC; and (5) included the potential for criticality, consistent with the success or failure of preventing moderator intrusion into the canister and following canister breach.

(iii) As documented in SER Section 2.1.1.4.3.3.1.1, the applicant's information for passive reliability of the high-level waste containers, TAD canister, dual-purpose canister (DPC), DOE standardized canisters, transportation cask, aging overpack, and shielded transfer cask (STC) for internal events that potentially lead to loss of containment or loss of shielding was based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs, (2) modeling techniques that were used appropriately to estimate reliability, and (3) analysis that appropriately considered uncertainty in test data for material behavior in the drop and collision tests used to estimate reliability.

(iv) As documented in SER Section 2.1.1.4.3.3.2, the applicant's information on reliability of active systems (HVAC systems and moderator intrusion control) is consistent with NRC staff guidance in HLRS–ISG–02 (NRC, 2007ab) and is based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs; (2) modeling techniques that were used appropriately to estimate reliability for moderator intrusion control; (3) the applicant's failure probabilities for the HVAC system that are consistent with failure probabilities reported in IAEA–TECDOC–478 (IAEA, 1988aa) for similar HVAC systems; and (4) analysis that appropriately considered uncertainty in the reliability estimates, which included the use of industry data to support failure probabilities.

(v) The applicant's categorization of the internal event sequences associated direct exposure and radionuclide release during cask and canister handling operations in the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Intrasite operations and Balance of Plant Facility as Category 2 and beyond Category 2 event sequences (there were no Category 1 event sequences identified) is consistent with the applicant's quantification for the event sequences and the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

In summary, the applicant's categorization of no Category 1 event sequences, Category 2 and beyond Category 2 event sequences associated with direct exposure and radionuclide releases from the internal event sequences associated cask and canister handling operations in the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Intrasite operations and Balance of Plant Facility is acceptable because the applicant has appropriately (i) identified the event sequences; (ii) quantified the probability of the event sequences consistent with, and supported by, facility description and site-specific data; and (iii) categorized the event sequences, consistent with the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

#### 2.1.1.4.3.4.1.2 Wet Handling Operations

Wet handling operations involve uncanistered spent nuclear fuel (SNF), such as the transfer of fuel assemblies in the Wet Handling Facility (WHF) pool. This SER section provides the NRC staff's evaluation of event sequence quantification and categorization associated with uncanistered SNF handling in the WHF. The NRC staff reviewed event sequences associated with (i) the transfer of fuel assemblies in the pool; (ii) the transfer of casks to and from the pool; and (iii) cask preparation activities.

The applicant described the WHF event sequence analysis in SAR Section 1.7.5.4. Additionally, the applicant included the event sequence diagram (ESD) in BSC (2008bo, Attachment F) and event trees in BSC (2008bo,bq, Attachments G and A), respectively. The Safety Analysis Report (SAR) Table 1.7-13 showed a total of 43 internal event sequences for the WHF. Of these, the applicant identified no Category 1 event sequences, 17 Category 2 event sequences; and 26 beyond Category 2 event sequences.

For the transfer of fuel assemblies to a TAD canister, the applicant categorized the event sequences from operations of the pool (i.e., lifting assemblies out of the water, and pool water splash); cask preparation activities (e.g., DPC cutting); and structural challenges to SNF assemblies, transportation casks, TADs, and DPCs. Event sequences associated with direct exposure from cask preparation activities, pool operations, and structural challenges to the transportation casks and TADs were identified as Category 2 events. Event sequences associated with radionuclide release from cask preparation activities, structural challenges to SNF assemblies, transportation casks, DPCs, and TADs were also identified as Category 2 events.

The applicant also identified beyond Category 2 event sequences for pool operations (i.e., transfer of LLW from the pool), and structural challenges to the transportation cask, DPCs and TADs associated with direct exposure, and radionuclide release. For event sequences involving the handling operations, the applicant separated the ESD and the event trees to account for whether the initiating event (e.g., drop or impact) occurs over the pool or over the floor outside of the pool. For those initiating events occurring over the floor, failure of the cask to remain intact, failure of the shielding on the cask to remain intact, and failure to maintain confinement (i.e., HVAC failure) were included as possible end states that result in radionuclide release and direct exposure.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's Wet Handling Facility (WHF) event sequence quantification and categorization involving uncanistered fuel using the guidance provided in YMRP Section 2.1.1.4. Based on the NRC staff's review, the NRC staff finds that the applicant provided adequate information on the quantification and categorization of internal event sequences for the WHF because

(i) As documented in SER Section 2.1.1.4.3.1.1, the technical basis and justification the applicant provided for its methodology for the identification and categorization for internal event sequences, including the assumptions and methods for quantifying event sequences, is acceptable because it is consistent with applicable NRC guidance and standard practices.

(ii) As documented in SER Section 2.1.1.4.3.2.1.2, the event sequences developed for internal events at the WHF (1) included appropriate initiating events both in and over the pool and outside of the pool for operations at the WHF; (2) included initiating events for dropping of CSNF on the staging rack in the pool (note: this is the only facility that handles uncanistered spent fuel); (3) included drops of both crane equipment (e.g., cask handling yoke) and heavy objects being moved by the crane (e.g., shield plug), resulting in structural challenges and loss of shielding; (4) the system response of SSCs to the initiating events in the WHF event sequences are consistent with the facility design and operations (including design of casks); and (5) the end states for the event sequences are consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences for the WHF operations.

(iii) As documented in SER Section 2.1.1.4.3.3.1.1, the applicant's information for passive reliability of the TAD canister, dual-purpose canister (DPC), transportation cask, aging overpack, and shielded transfer cask (STC) for internal events that potentially lead to loss of containment or loss of shielding was based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs, (2) modeling techniques that were used appropriately to estimate reliability, and (3) analysis that appropriately considered uncertainty in test data for material behavior in the drop and collision tests used to estimate reliability.

(iv) As documented in SER Section 2.1.1.4.3.3.2, the applicant's information on reliability of active systems (HVAC systems and moderator intrusion control) is consistent with staff guidance in HLRS–ISG–02 (NRC, 2007ab) and is based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs; (2) modeling techniques that were used appropriately to estimate reliability for moderator intrusion control; (3) the applicant's failure probabilities for the HVAC system are consistent with failure probabilities reported in IAEA–TECDOC–478 (IAEA, 1988aa) for similar HVAC systems; and (4) analysis that appropriately considered uncertainty in the reliability estimates, which included the use of industry data to support failure probabilities.

(v) The applicant's categorization of the event sequences associated with direct exposure and radionuclide release during SNF handling in the WHF (i.e., the transfer of fuel assemblies in the pool, the transfer of casks to and from the pool, and cask preparation activities) as Category 2 and beyond Category 2 event sequences (there were no Category 1 event sequences identified) is consistent with the applicant's quantification for the event sequences and the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

In summary, the applicant's categorization of no Category 1 event sequences, Category 2 event sequences associated with direct exposure and radionuclide releases, and beyond Category 2 event sequences associated with direct exposure and radionuclide releases from SNF handling at the WHF is acceptable because the applicant has appropriately (i) identified the event sequences for the wet handling operations; (ii) quantified the probability of the event sequences for the wet handling operations consistent with, and supported by, facility description and site-specific data; and (iii) categorized the event sequences consistent with the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

2.1.1.4.3.4.1.3 Subsurface Operations

The applicant provided information in SAR Section 1.7.5.6 and BSC (2008bk, Table G–1) regarding the quantification and categorization of potential event sequences associated with subsurface operations. The applicant did not identify any Category 1 event sequences regarding subsurface operations. The applicant identified one Category 2 event sequence and two beyond Category 2 event sequences for internal events (SAR Tables 1.7-17). The Category 2 event sequence is associated with an inadvertent opening of a loaded transport and emplacement vehicle (TEV) door or prolonged immobilization of the TEV in a heated drift resulting in a loss of shielding. The beyond Category 2 event sequences are associated with potential radionuclide releases due to structural challenges during TEV operations at the loadout area at the CRCF or the IHF.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's quantification and categorization of the event sequence frequencies for subsurface operations, using the guidance provided in YMRP Section 2.1.1.4. Based on the NRC staff's review, the NRC staff finds that the applicant provided adequate information on the quantification and categorization of internal event sequences for the subsurface operations because

(i) As documented in SER Section 2.1.1.4.3.1.1, the technical basis and justification the applicant provided for its methodology for the identification and categorization for internal event sequences, including the assumptions and methods for quantifying event sequences, is acceptable because it is consistent with applicable NRC guidance and standard practices.

(ii) As documented in SER 2.1.1.4.3.2.1.3, the event sequences developed for the internal events for subsurface operations (1) included appropriate initiating events that could challenge the structural integrity of a waste package (e.g., mechanical impact from a collision with a shield door, TEV derailment), result in a potential loss of shielding, present a thermal challenge due to fire, and affect drip shield emplacement; (2) included potential violations of an administrative or physical control (such as inadvertent worker entry into an emplacement drift containing waste packages, proximity to a loaded TEV, or inadvertent opening of a TEV door), consistent with the design and operations for the underground facility; (3) included the response of SSCs to the initiating events in the event sequences for the underground facility (e.g., TEV shielding may degrade if a layer of polymer material in the shielding overheats) is consistent with the subsurface facility design and operations; (4) included the use of monitoring and inspection programs, consistent with the design criteria for the subsurface facilities, that will address deterioration of SSCs in a timely manner to prevent or mitigate event sequences for the subsurface operations (e.g., timely maintenance and monitoring to limit the potential for collapse of an emplacement drift, exhaust main, or exhaust shaft); and (5) included the end states for subsurface operations that represent potential occurrences that could result in radiation exposure or release of radioactive materials during subsurface operations;

(iii) As documented in SER Section 2.1.1.4.3.3.1.1, the applicant's information for passive reliability of the high-level waste containers, waste packages, TAD canisters, dual-purpose canisters (DPC), DOE standardized canisters, and shielded transfer casks (STC) for internal events that potentially lead to loss of containment or loss of shielding was based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs, (2) modeling techniques that were used appropriately to estimate reliability, and (3) analysis that appropriately considered uncertainty in test data for material behavior in the drop and collision tests used to estimate reliability.

(iv) The applicant's categorization of the internal event sequences associated with direct exposure and radionuclide release during subsurface operations into Category 2 and beyond Category 2 event sequences (there were no Category 1 event sequences identified) is consistent with the applicant's quantification for the event sequences and the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

In summary, the applicant's categorization of no Category 1 event sequences, a Category 2 event sequence associated with direct exposure, and beyond Category 2 event sequences associated with radionuclide releases from the internal event sequences associated with the subsurface operations is acceptable because the applicant has appropriately (i) identified the event sequences; (ii) quantified the probability of the event sequences consistent with, and supported by, facility description and site-specific data; and (iii) categorized the event sequences, consistent with the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

2.1.1.4.3.4.2 Seismic Events

The applicant addressed categorization of the seismic event sequences for the geologic repository operations area (GROA) facilities in SAR Section 1.7.5, with the seismic event sequence probability and category presented in SAR Table 1.7-8 for the Initial Handling Facility (IHF), SAR Table 1.7-10 for the Receipt Facility (RF), SAR Table 1.7-12 for the Canister Receipt and Closure Facility (CRCF), SAR Table 1.7-14 for the Wet Handling Facility (WHF), SAR Table 1.7-16 for the Intrasite operations and Balance of Plant Facilities, including the low-level waste (LLW) Facility, and SAR Table 1.7-18 for the Subsurface Facilities. The applicant described the seismic event sequence analyses in BSC (2008bg).

The applicant's analysis of seismic event sequences can be broadly divided into three groups: (i) large permanent distortion (short of collapse) of surface facility buildings, (ii) failure of equipment or mechanical components, and (iii) tipover and sliding of transporters and transfer trolleys. Additionally, the applicant considered rockfall into the emplacement drift initiated by seismic events. The applicant's tables in SAR Section 1.7 identified 115 seismic event sequences, which included no Category 1 event sequences and eight Category 2 sequences [two in the Canister Receipt and Closure Facility (CRCF), four in the Wet Handling Facility (WHF), one for Intrasite operations (i.e., the Low-Level Waste facility), and one for the Subsurface operations]. Six of the identified event sequences end in unfiltered radionuclide releases, which includes consideration of the potential radiation exposure. The applicant identified two Category 2 event sequences that involved direct exposure due to loss of transport and emplacement vehicle (TEV) shielding by seismic failure (i.e., one event sequence in the CRCF and one event sequence for subsurface operations).

**Structural Collapse of Facilities**

The applicant evaluated the collapse, defined as Limit State A (Large Permanent Distortion—Short of Collapse, in American Society of Civil Engineers, 2005aa) of the ITS Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), and Wet Handling Facility (WHF) structures as potential seismically initiated events. The applicant presented the mean annual probability of failure or collapse of the surface facility structures in BSC (2008bg, Table 6.2-1). The mean annual probabilities of failure/collapse of the structures were calculated to be between $3.8 \times 10^{-7}$ and $8.7 \times 10^{-7}$. The applicant categorized the collapse of all surface facility structures involved with handling HLW and SNF as beyond Category 2 event sequences.

**Failure of Equipment and Mechanical Systems**

The seismic event sequences initiated by the seismic failure of facility equipment and mechanical systems are discussed in SAR Section 1.7. The applicant presented the results of the event sequence analysis in SAR Table 1.7-8 for the Initial Handling Facility (IHF), SAR Table 1.7-10 for the Receipt Facility (RF), SAR Table 1.7-12 for the Canister Receipt and Closure Facility (CRCF), and SAR Table 1.7-14 for the Wet Handling Facility (WHF). The applicant did not identify any Category 1 event sequences and did not identify any Category 2 event sequences for the IHF and RF. The applicant did identify two Category 2 event sequences for the CRCF (i.e., seismic failure of the TEV shielding resulting in a direct exposure; seismic failure of CTM breaching an HLW canister resulting in radionuclide release). The applicant did identify the following four Category 2 event sequences for the WHF that result in radionuclide release: (i) seismic failure of the HVAC and (ii) three seismic event sequences associated with tipovers of the transportation cask, spilling SNF assemblies in the pool, resulting in radionuclide release.

**Subsurface Operations**

The applicant addressed the subsurface event sequences associated with the seismic event in SAR Section 1.7.5.6 and BSC (2008bk, Table G-1). The applicant did not identify any Category 1 event sequences (SAR Table 1.7-18). The applicant identified one Category 2 event sequence associated with seismic failure of TEV shielding with a TAD canister in route to emplacement (no breach of the TAD canister), resulting in direct exposure to workers.

**Intrasite Operations**

The applicant considered the potential quantification and categorization of event sequences during a seismic event for the aging overpack (AO) on an aging pad and the LLW building. The applicant considered in BSC (2008bg, Attachment E) the failure modes of AO tipover, sliding of AO and impact with another AO, and aging pad displacement and tipover of AO. The applicant did not identify any seismically initiated Category 1 or Category 2 event sequences for the AO on the aging pad. The applicant did categorize seismic collapse of the non-ITS low-level waste (LLW) building breaching multiple LLW containers to be a Category 2 event sequence.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's event sequence quantification and categorization for seismically initiated event sequences using the guidance provided in YMRP Section 2.1.1.4. Based on the NRC staff's review, the NRC staff finds that the applicant provided adequate information on the quantification and categorization of seismically initiated event sequences because

(i) As documented in SER Section 2.1.1.4.3.1.2, the technical basis and justification the applicant provided for its methodology for the identification and categorization for seismically initiated event sequences, including the assumptions and methods for quantifying event sequences, is acceptable because it is consistent with applicable NRC guidance and standard practices.

(ii) As documented in SER 2.1.1.4.3.2.2, the event sequences developed for seismically initiated event sequences for waste handling operations in surface facilities (1) included appropriate initiating events for both the collapse of the facility structure and failure of equipment used during the waste handling operations, even if the building did not collapse; (2) included failure of the HVAC for when the initiating event causes a collapse of the building; (3) included the system response of SSCs, consistent with the facility design and operations; and (4) included the end states for the event sequences, consistent with the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences for waste handling operations (e.g., the end states are consistent with the success or failure of the surface nuclear confinement HVAC system and the success or failure state of the cask maintaining containment and shielding integrity; consideration for releases that could occur under water in the pool that would mitigate particulate release);

(iii) As documented in SER 2.1.1.4.3.2.2, the event sequences developed for seismic initiated event sequences for Intrasite operations in surface facilities: (1) included appropriate initiating events for Intrasite operations consistent with facility design and operations [e.g., movement and storage of aging overpacks (AOs) containing TAD canisters and horizontal transportation casks containing dual purpose canisters (DPCs) at the aging facility; aging pad location and design]; (2) considered seismically induced failure of cut or fill slopes near the aging pads or on transportation routes that link the aging pads to other surface facilities; (3) the system response of SSCs to the initiating events for event sequences for Intrasite operations are consistent with the facility design and operations; and (4) the end states for the event sequences are consistent with the success or failure of the safety functions of the SSCs relied on to prevent or mitigate event sequences (e.g., LLW building collapse can result in unfiltered radionuclide release);

(iv) As documented in SER Section 2.1.1.4.3.2.2, the event sequences developed for seismically initiated event sequences for subsurface operations (1) considered both structural impacts from seismically induced rockfall as well as thermal effects that could lead to unfiltered radionuclide release; (2) considered failure of other SSCs, consistent with the design and operations (e.g., subsurface ventilation design for emplacement drifts, TEV shielding, waste package); (3) included the system response of SSCs to the subsurface and other initiating events, consistent with the facility design and operations (e.g., containment of the waste package); (4) included the end state of direct exposure for the event sequences, consistent with the capacity of the waste package to withstand structural challenges and the success or failure of the safety functions of the SSCs that are relied on to prevent or mitigate event sequences (e.g., TEV shielding mitigates direct exposure).

(v) As documented in SER Section 2.1.1.4.3.3.1.1, the applicant's information for passive reliability of the high-level waste containers, waste packages, waste packages, TAD canisters, dual-purpose canisters (DPC), DOE standardized canisters, transportation casks, aging overpacks, and shielded transfer casks (STC) for seismically initiated event sequences that potentially lead to loss of containment or loss of shielding was based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs; (2) modeling techniques that were used appropriately to estimate reliability; and (3) analysis that appropriately considered uncertainty in test data for material behavior in the drop and collision tests used to estimate reliability.

(vi) As documented in SER Section 2.1.1.4.3.3.1.2, the applicant's information for the passive reliability of surface facilities buildings and mechanical systems and equipment for seismically initiated event sequences was based on (1) standard engineering practices and modeling techniques appropriate for estimating reliability of the SSCs; (2) consideration of uncertainties in the supporting numerical models, structural system parameters, and structural challenges, consistent with NRC staff guidance in HLWRS–ISG–02 (NRC, 2007ab); and (3) standard industry practices appropriate for development of the seismic fragility curve, consistent with NRC staff guidance in HLWRS–ISG–01 (NRC, 2006ad).

(vii) The applicant's categorization of the seismically initiated event sequences associated with direct exposure and radionuclide release at the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Intrasite operations, and the subsurface operations into Category 2 and beyond Category 2 event sequences (there were no Category 1 event sequences identified) is consistent with the applicant's quantification for the event sequences and the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

In summary, the applicant's categorization of no Category 1 event sequences, and Category 2 and beyond Category 2 event sequences associated with direct exposure and radionuclide releases, based on the seismically initiated event sequences for the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Intrasite operations, and subsurface operations is acceptable because the applicant has appropriately (i) identified the event sequences; (ii) quantified the probability of the event sequences, consistent with and supported by facility description and site-specific data; and (iii) categorized the event sequences, consistent with the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

2.1.1.4.3.4.3 Fire Events

The applicant quantified and categorized event sequences initiated by fires. The applicant listed the fire-related event sequences for the geologic repository operations area (GROA) in SAR Tables 1.7-7, 1.7-9, 1.7-11, 1.7-13, 1.7-15, and 1.7-17. SAR Section 1.7.1.2.2 referred to BSC (2008ac,as,au,be,bk,bq) for fire event sequence quantification and categorization. The applicant developed a number of fire-related event sequences for each particular waste form container, depending on the processes surrounding that waste container. All direct exposure and radionuclide release event sequences relied on canister reliability in a fire to reduce the overall likelihood of the event. Other SSCs are assumed to fail depending on specific aspects of the fire event. For example, the probability for HVAC confinement failures is $3.47 \times 10^{-2}$ for fires confined to a single fire zone, whereas, the HVAC confinement is assumed to fail (failure probability of 1.0) for the large fire event sequences (i.e., fires affecting the entire facility).

The applicant did not identify any Category 1 fire-initiated event sequences. The applicant categorized fire-initiated event sequences associated with loss of shielding events and radionuclides release events as Category 2 events. The applicant also categorized other fire-initiated event sequences associated with radionuclides release events as beyond Category 2.

The applicant identified Category 2 event sequences associated with direct exposure due to loss of shielding at the Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), Wet Handling Facility (WHF), Subsurface Facility, and Intrasite operations. For workers, categorization of event sequences in the applicant's analysis of this potential end-state results in direct exposure from the loss of shielding (SAR Tables 1.7-7, 9, 11, 13, 15, and 17).

The applicant identified Category 2 fire-initiated event sequences that may result in the end state of radionuclide release for the WHT (SAR Table 1.7-13), due to a thermal challenge to uncanistered spent nuclear fuel and for Intrasite operations (SAR Table 1.7-15) due to a fire at the low-level waste facility. Other Category 2 fire-initiated event sequences that may result in the end state of radionuclide release were identified for the (i) IHF (SAR Table 1.7-7) from a thermal challenge to an HLW canister and naval SNF canister; (ii) CRCF (SAR Table 1.7-11) from a thermal challenge to an HLW canister, TAD canister, and DOE standardized; and (iii) subsurface operations from a thermal challenge to the waste package.

The applicant identified beyond Category 2 fire-initiated event sequences that may result in end states important to criticality, particularly intrusion of moderator into a loaded canister for the IHF (SAR Table 1.7-7) due to a thermal challenge that could result in moderator intrusion after breach of a naval SNF canister; and for Intrasite operations (SAR Table 1.7-15) from a thermal challenge that could result in moderator intrusion after breach of a TAD or DPC inside a transportation cask or aging overpack; and breach of a horizontal DPC in a transportation cask, aging module, or STC.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's event sequence quantification and categorization for fire event sequences using the guidance provided in YMRP Section 2.1.1.4. Based on the NRC staff's review, the NRC staff finds that the applicant provided adequate information on the quantification and categorization of fire-initiated event sequences because

(i) As documented in SER Section 2.1.1.4.3.1.3, the technical basis and justification the applicant provided for its methodology for the identification and categorization for fire-initiated event sequences, including the assumptions and methods for quantifying event sequences, is acceptable because it is consistent with applicable NRC guidance and standard practices.

(ii) As documented in SER Section 2.1.1.4.3.2.3, the event sequences developed for fire-initiated event sequences (1) considered both internal and external fires [external fires were acceptably excluded from further consideration based on administrative controls (see SER Section 2.1.1.3.3.1.3.5 for further details)]; (2) included the system response of SSCs to fire-initiated event sequences, consistent with the facility design and operations (e.g., loss of low melting temperature shielding material during a fire, loss of non-ITS HVAC confinement during a building-wide fire; sprinkler systems for control of moderator); and (3) included the end states for the SSCs for each facility and the specific container types handled in each facility (e.g., LLW containers and SNF containers);

(iii) As documented in SER Section 2.1.1.4.3.3.1.1, the applicant's information for passive reliability of the high-level waste containers, waste packages, TAD canisters, dual-purpose canisters (DPC), DOE standardized canisters, transportation casks, aging overpacks, and shielded transfer casks (STC) for fire-initiated event sequences that potentially lead to loss of containment or loss of shielding was based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs; (2) modeling techniques that were used appropriately to estimate reliability; (3) analysis that appropriately considered uncertainty in test data for material behavior in the drop and collision tests used to estimate reliability.

(iv) As documented in SER Section 2.1.1.4.3.3.2, the applicant's information on reliability of active systems (HVAC systems and moderator intrusion control) is consistent with staff guidance in HLRS–ISG–02 (NRC, 2007ab) and is based on (1) standard engineering practices that were used appropriately to estimate reliability of the SSCs; (2) modeling techniques that were used appropriately to estimate reliability for moderator intrusion control; (3) the applicant's failure probabilities for the HVAC system are consistent with failure probabilities reported in IAEA–TECDOC–478 (IAEA, 1988aa) for similar HVAC systems; and (4) analysis that appropriately considered uncertainty in the reliability estimates, which included the use of industry data to support failure probabilities.

(v) The applicant's categorization of the fire-initiated event sequences associated with direct exposure and radionuclide release at the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Intrasite Operations, and the subsurface operations into Category 2 and beyond Category 2 event sequences (there were no Category 1 event sequences identified) is consistent with the applicant's quantification for the event sequences and the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

In summary, the applicant's categorization of no Category 1 event sequences and Category 2 and beyond Category 2 event sequences associated with direct exposure and radionuclide releases based on the fire-initiated event sequences for the Initial Handling Facility (IHF), Receipt Facility (RF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Intrasite operations, and subsurface operations is acceptable because the applicant has appropriately (i) identified the event sequences; (ii) quantified the probability of the event sequences, consistent with and supported by, facility description and site-specific data; and (iii) categorized the event sequences, consistent with the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

**NRC Staff's Conclusion**

Based on the NRC staff's review in SER Section 2.1.1.4.3.4, the NRC staff finds that the applicant provided acceptable information on the quantification and categorization of event sequences initiated by internal, seismic, and fire hazards in the preclosure safety analysis because (i) the applicant's methodology for event sequence identification and categorization is consistent with NRC guidance and standard industry practices; (ii) the quantification of the probability of the event sequences, including the assumptions made in identifying event sequences, is consistent with and supported by facility description and site-specific data; and (iii) the applicant's categorization of event sequences is consistent with the definition of Category 1 and Category 2 event sequences in 10 CFR 63.2.

## 2.1.1.4.4 Evaluation Findings

The U.S. Nuclear Regulatory Commission staff has reviewed the Safety Analysis Report and other information submitted in support of the license application and finds, with reasonable assurance, that the requirements of 10 CFR 63.112(b) are satisfied regarding the identification and categorization of event sequences for naturally occurring and human-induced hazards and initiating events at the geologic repository operations area.

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# CHAPTER 5

## 2.1.1.5 Consequence Analysis

### 2.1.1.5.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.5 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's ("DOE" or "applicant") consequence analysis methodology and demonstration that the repository design meets 10 CFR Parts 20 and 63 radiation protection requirements. The applicant's consequence analysis was used to support its preclosure safety analysis (PCSA). The NRC staff evaluated the information in the Safety Analysis Report (SAR) (DOE, 2008ab; DOE, 2009av, and SAR Section 1.8), and the applicant's responses to the NRC staff's requests for additional information (RAIs) (DOE, 2009ek–eq). In addition, the NRC staff used the information in SAR Section 1.5.1 in which the applicant described the characteristics of spent nuclear fuel (SNF) and high-level radiological waste (HLW) to evaluate the applicant's source term calculations.

In SAR Section 1.8, the applicant described the dose calculation methodology, potential releases of radioactive material, potential doses from normal operations, Category 1 and Category 2 event sequences, and uncertainty and sensitivity analyses. Normal operations are considered those planned, routine activities by the applicant in which monitored exposures are expected from the HLW processing at the geologic repository operations area (GROA). As defined in 10 CFR 63.2, Category 1 event sequences include one or more initiating events and associated combinations of repository structure, system, or component failures that could potentially lead to radiation exposure and are expected to occur at least one or more times before permanent closure of the GROA. Category 2 event sequences are the events other than Category 1 that could potentially lead to radiation exposure and have at least 1 chance in 10,000 of occurring before permanent closure. The applicant did not identify any Category 1 event sequence in the GROA (SAR Section 1.8.5).

### 2.1.1.5.2 Regulatory Requirements

The regulatory requirements applicable to this section are in 10 CFR 63.21(c)(5); 10 CFR 63.111(a), (b), and (c); and 10 CFR 63.204. The applicant is required by 10 CFR 63.21(c)(5) to include in its SAR a PCSA of the GROA for the period before permanent closure to ensure compliance with 63.111(a), as required by 63.111(c). The requirements in 10 CFR 63.111(a), (b), and (c) specify the preclosure performance objectives for normal operations and Category 1 and Category 2 event sequences. Specifically:

- 10 CFR 63.111(a)(1) provides the preclosure performance objectives for the GROA to not exceed the radiation exposure limits in 10 CFR Part 20.

- 10 CFR 63.111(a)(2) provides the preclosure performance objectives for the GROA, during normal operations and for Category 1 event sequences, in that the annual dose to any real member of the public, located beyond the boundary of the site, may not exceed 0.15 mSv [15 mrem].

- 10 CFR 63.111(b)(1) requires that the GROA be designed so that, taking into consideration Category 1 event sequences and until permanent closure has been completed, aggregate radiation exposures and radiation levels in restricted and unrestricted areas, and aggregate releases of radioactive materials to unrestricted areas, will be maintained within the limits of 10 CFR 63.111(a).

- 10 CFR 63.111(b)(2) requires that the GROA be designed such that taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on or beyond any point on the site boundary will receive, as a result of the single Category 2 event sequence, the more limiting of a total effective dose equivalent (TEDE) of 0.05 Sv [5 rem] or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent will not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin will not exceed 0.5 Sv [50 rem].

- 10 CFR 63.111(c)(1) requires the PCSA to meet the requirements specified in 10 CFR 63.112 and demonstrate that the radiation protection limits of 10 CFR Part 20 will be met.

- 10 CFR 63.111(c)(2) requires the PCSA to meet the requirements specified in 10 CFR 63.112 and demonstrate that the numerical guides for design objectives will be met for GROA normal operations and Category 1 and Category 2 event sequences.

The applicant is also required by 10 CFR 63.204 to ensure that no member of the public in the general environment, as defined in 10 CFR 63.202, will receive an annual dose exceeding 0.15 mSv [15 mrem] from combined management and storage (defined in 40 CFR 191.2) and storage (defined in 10 CFR 63.202) of radioactive material inside the Yucca Mountain repository.

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa, Section 2.1.1.5). The acceptance criteria are as follows:

- The applicant's consequence analyses adequately assess normal operations and Category 1 event sequences, as well as factors that allow an event sequence to propagate within the GROA.

- Consequence calculations by the applicant adequately assess the consequences to workers and members of the public from normal operations and Category 1 event sequences.

- The dose to workers and members of the public from normal operations and Category 1 event sequences is within the limits specified in 10 CFR 63.111(a).

- Consequence analyses by the applicant include Category 2 event sequences, as well as factors that allow an event sequence to propagate within the GROA.

- Consequence calculations by the applicant adequately assess the consequences to members of the public from Category 2 event sequences.

- The dose to hypothetical members of the public from Category 2 event sequences is within the limits specified in 10 CFR 63.111(b)(2).

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA. The applicability of such NRC guidance is discussed in greater detail in the sections where they were used as part of the application or the NRC staff's review.

### 2.1.1.5.3 Technical Review

The NRC staff's review of SAR Section 1.8 focused on (i) the methodology and input parameters used for the dose calculation, (ii) the consistency of source terms used in the dose calculation with those described in SAR Section 1.5, (iii) the methodology for the worker and public dose determination, and (iv) demonstration of compliance with 10 CFR 63.111(a), 63.111(b), 63.111(c), and 63.204.

In SAR Section 1.8.1, the applicant defined two categories of individuals that it considered for the application of performance objectives and operational dose limits: (1) individuals receiving occupational doses and (2) members of the public. As defined by the applicant, individuals receiving occupational doses are personnel, designated as *radiation workers*, who are assigned duties at the repository that involve exposure to radiation and/or radioactive material. The applicant defined the *public* as any individual not receiving an occupational dose. Additionally, within the preclosure controlled area, referred to as the onsite areas, the applicant defined an *onsite member of the public* (SAR Section 1.8.1) as any individual not receiving an occupational dose in performing duties. This included construction workers, delivery personnel, and public visitors within the preclosure controlled area. Onsite members of the public are subject to the dose limits in 10 CFR 20.1301. The Cind-R-Lite mining lease is located southwest of the surface facility GROA, within the preclosure controlled area, but near the site boundary. The mining personnel who periodically access this area are considered to be onsite members of the public.

The applicant defined *offsite members of the public* as individuals located at or beyond the site boundary of the preclosure controlled area (SAR Figure 1.8-2). The *general environment*, as defined in 10 CFR 63.202, is the area outside the Yucca Mountain site, the Nellis Air Force Range (Nevada Test and Training Range), and the Nevada Test Site (NTS). Therefore, members of the public in the general environment, located south and west of the site boundary of the preclosure controlled area, are considered *offsite members of the public in the general environment*. According to DOE, this area is accessible to the public and is subject to the 0.15 mSv/year [15 mrem/year] limit in 10 CFR 63.111(a)(2) for normal operations and for Category 1 event sequences and 0.05 Sv [5 rem] exposure for any single Category 2 event sequence, as specified in 10 CFR 63.111(b)(2) (SAR Figure 1.8-2). For the areas north and east of the site boundary, which are areas controlled by the NTS and the Nevada Test and Training Range, access by general members of the public is restricted. These areas are referred to as "*offsite but not in the general environment*." Individuals who are *offsite members of the public not in the general environment* are subject to the 1 mSv/year [100 mrem/year] public dose limit in 10 CFR 20.1301 for normal operations and 0.05 Sv [5 rem] for any single Category 2 event sequence, as required in 10 CFR 63.111(b)(2).

#### 2.1.1.5.3.1 Dose Calculation Methodology and Input Parameter Selection

In SAR Section 1.8, the applicant discussed the methodology used to calculate dose consequences to site workers and members of the public, both onsite and offsite. The applicant considered the radiological doses to workers and the public for normal operations, off-normal events, and Category 2 event sequences for the GROA activities. The applicant did not identify any Category 1 event sequence that required evaluation of dose consequences to workers or the public.

The NRC staff's review and evaluation of the applicant's dose calculation methodology and input parameter selection included: (i) dose calculation methodology, (ii) atmospheric dispersion determination, and (iii) assumptions and input parameter selection, as discussed in the following sections.

**Dose Calculation Methodology**

The applicant described the methodology for estimating doses to workers and the public as well as the various activities and events that could lead to worker and public dose in SAR Sections 1.8.1 and 1.8.2. The doses calculated for onsite personnel—radiation workers and the onsite public—consisted of contributions from direct radiation, inhalation, and submersion doses (SAR Section 1.8.4). The applicant estimated direct radiation dose rates at various distances from the GROA facilities, including the rail and truck casks at the buffer areas, using the MCNP computer program (Briesmeister, 1997aa). MCNP is an industry standard Monte-Carlo transport computer code that simulates particle transport through a three-dimensional modeling of the nuclear system. Direct radiation dose rates from the aging pads were calculated using the MCNP and the SCALE (Oak Ridge National Laboratory, 2000aa) computer codes. SCALE is an industry standard modular code system developed for NRC for performing standardized computer analyses for licensing evaluation. The applicant used SCALE to calculate doses from the aging pad assuming transportation, aging, and disposal (TAD) canisters were loaded with design basis fuel out to a distance of 1 km [0.6 mi]. Both MCNP and SCALE considers primary gammas, neutrons, and photons generated by neutron interactions.

Direct radiation doses for radiation workers were based on the estimated dose rates and time-motion inputs for specific operational tasks or assumed continuous occupancy. The applicant considered inhalation and air submersion exposure pathways for atmospheric releases of radioactive material and resuspension of surface contamination. For onsite public locations, the applicant indicated that direct exposure from the waste handling facilities was minor compared to the applicant's 2.5 μSv/hour [0.25 mrem/hour] shielding design limit. The onsite public doses were dominated by the transportation cask rail and truck buffer areas and the aging overpacks stored at the aging pads. The applicant indicated in its response to the NRC staff's RAI in DOE (2009eq, Section 1.3) that transient sources, such as a single transportation cask, the transportation and emplacement vehicle, and site transporter, were minor sources of exposure to the onsite public and were not included in the projected doses presented in SAR Section 1.8. Additionally, the applicant stated in its response to the NRC staff's RAI in DOE (2009eq, Section 1.3) that the impact of these transient sources on the estimated annual onsite public dose is small relative to the stationary sources located in the aging pad and railcar and truck buffer areas.

The applicant calculated the inhalation dose resulting from the normal operational releases by multiplying the radionuclide concentration to which the individual was exposed during the

2,000-hour work year by a dose conversion factor for inhalation from International Commission on Radiological Protection (ICRP)–68 (1995aa) and the breathing rate in 10 CFR Part 20, Appendix B. The release of radioactive material from the facility was assumed to be over a 1-year period. For submersion, the applicant calculated the external dose by multiplying the radionuclide concentration to which the individual is exposed by a dose conversion factor for submersion from the U.S. Environmental Protection Agency (EPA) (1993aa) and the exposure time. The applicant calculated the TEDE for workers and the onsite public by summing (i) the external dose due to the direct radiation pathway, (ii) the external dose due to the submersion pathway, and (iii) the internal dose due to the inhalation pathway. The applicant stated that there were no agricultural activities in the onsite area and, thus, the applicant did not include any dose due to ingestion of foodstuffs in the calculations.

The applicant used the GENII Gaussian statistical model, Version 2.05, to calculate the airborne exposures to the offsite public (Napier, 2007aa). GENII Version 2.05 is an industry standard computer code used for estimating the consequences of radionuclides released into the environment. In GENII Version 2.05, radionuclide air transport options include both plume and puff models. The applicant assumed that releases were at ground level for the dose calculations at the site boundary.

For assessment of internal exposures, the applicant used the methods proposed in EPA guidance (1988aa; EPA 1993aa; EPA 1999aa). The applicant used the GENII Gaussian statistical model, Version 2.05, which implemented dosimetry models recommended by the International Commission on Radiological Protection (International Commission on Radiological Protection, 1991aa) and related guidance in EPA (1988aa; EPA 1993aa; EPA 1999aa). The applicant applied both deterministic and stochastic approaches to model the impacts of GROA operations. A majority of the applicant's calculations used a combination of deterministic bounding values coupled with stochastic parameters characterized by a mean value and distribution. The applicant's deterministic dose calculation methods used receptor characteristics that, according to the applicant, bound those characteristics for any real member of the public or a worker. The applicant's stochastic dose calculation methods used mean values and distributions for parameters including receptor-related parameters. Using sensitivity and uncertainty analysis techniques, the applicant determined a dose distribution. The applicant then determined the dose to a real member of the public by using the maximum value obtained from the calculated dose distributions.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's dose calculation methodology using the guidance in YMRP Section 2.1.1.5. The NRC staff finds that the applicant's use of (i) MCNP to estimate direct radiation dose rates at various distances from the GROA facilities, including the rail and truck casks at the buffer areas; (ii) MCNP and SCALE to estimate direct radiation dose rates from the aging pads; and (iii) SCALE to calculate doses from the aging pad are acceptable because the computer codes used by the applicant are industry standards and have been previously used by the NRC staff in licensing activities for nuclear power plants and independent spent fuel storage installations.

The NRC staff also finds that use of the GENII Gaussian statistical model, Version 2.05, which implemented dosimetry models recommended by the International Commission on Radiological Protection (International Commission on Radiological Protection, 1991aa) and related guidance in EPA (1988aa; EPA 1993aa; EPA 1999aa) to assess internal exposures, is acceptable because the dosimetry model recommended by the International Commission on Radiological

Protection (1991aa) and risk models used by GENII Version 2.05 are considered state-of-the art by the international radiation protection community and have been adopted by national and international organizations as the standard dosimetry methodology.

For calculating offsite doses, the NRC staff finds the applicant's assumption of modeling airborne radionuclide releases as ground level releases at the site boundary to be acceptable because it conservatively simulates maximum radionuclide exposure to an individual at the site boundary. The NRC staff also finds the applicant's assumption of not including any dose due to ingestion of foodstuffs from the onsite area acceptable because there are no agricultural activities in the onsite area.

For onsite public doses, the NRC staff finds the applicant's assumption of excluding transient sources (i.e., a single transportation cask, a transportation and emplacement vehicle, or site transporter) from projected doses of the onsite members of the public acceptable because the potential direct radiation contribution is low due to shielding and the brief period of exposure to transient sources. The NRC staff also finds that the applicant's use of shielding from external sources in unrestricted areas is acceptable because the shielding would reduce the potential direct radiation contribution below the dose limits for members of the public as specified in 10 CFR 20.1301.

**Atmospheric Dispersion Determination**

The applicant estimated airborne doses using the annual average onsite atmospheric dispersion coefficients (X/Q). To calculate the annual average onsite X/Q, the applicant used the guidance in Regulatory Guide 1.194 (NRC, 2003ah) and performed calculations using the ARCON96 (NRC, 1977ab) computer code. The applicant estimated airborne doses onsite and normal exposure to the offsite public using the annual average onsite X/Q values from a meteorological monitoring station located approximately 1 km [0.6 mi] south-southwest of the North Portal. Data used in the dose calculations were based on the period from January 1, 2001, through December 31, 2005. The applicant calculated X/Q values for the 16 meteorological sectors based on Regulatory Guide 1.111 (NRC, 1977ac) for annual releases and Regulatory Guide 1.145 (NRC, 1982aa) for hourly X/Q values for use with the Category 2 event sequences. The annual average and 95$^{th}$ percentile X/Q values were presented in SAR Table 1.8-12. The 95$^{th}$ percentile values were used for the Category 2 event sequence calculations. To estimate airborne exposures to the offsite public, GENII Version 2.05 accounts for radioactive material falling out and depositing on the ground and vegetation as the plume travels from the release point. The contaminated air concentration decreases as the material depletes out by deposition. The depleted X/Q values (SAR Table 1.8-12) were used for the dose from the volatile radionuclides and particulates. The deposition rates (SAR Table 1.8-12) were used for groundshine, soil contamination, and radionuclide uptake by vegetation. The undepleted X/Q values were used for the dose from the gaseous radionuclides.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's atmospheric dispersion determination using the guidance in YMRP Section 2.1.1.5 and finds that the applicant's use of ARCON96 to calculate the annual average onsite X/Q is acceptable for the following reasons: (i) ARCON96 is a peer reviewed, industry standard atmospheric dispersion computer code developed for the NRC for use in licensing activities; (ii) ARCON96 is designed to analyze release and receptor points in close proximity, which would be the case for the waste handling facilities and the onsite locations where workers and the onsite public could be exposed to a radioactive plume; and (iii) ARCON96 accounts for building wake factors over short distances.

**Assumptions and Input Parameter Selection**

The applicant provided information concerning the assumptions and basis for the selection of models, source terms, exposure pathways, and dose coefficients used in calculating the radiological exposures (SAR Sections 1.8.1 and 1.8.2).

To estimate dose contributions from surface contamination, the applicant assumed that the entire external surface area of each transportation cask was contaminated at the regulatory limit [49 CFR 173.443(a), Table 9]. The applicant assumed that the contamination levels on the casks are 4 $Bq/cm^2$ [$10^{-4}$ $\mu Ci/cm^2$] beta/gamma and low toxicity alpha and 0.4 $Bq/cm^2$ [$10^{-5}$ $\mu Ci/cm^2$] for all other alpha, using 49 CFR 173.443(a) Table 9, which provides non-fixed external radioactive contamination limits for packages. For direct radiation exposures to individuals outside of a facility, the applicant established a limit of 2.5 μSv/hour [0.25 mrem/hour] as a shielding design limit for the various waste handling facilities. The applicant indicated that the calculation results showed that resuspension of surface contamination was an insignificant contributor to the calculated total annual radiation worker dose. Time-motion calculations versus expected dose rates during the work activities were predicted and summed to generate an overall worker direct radiation dose. The applicant added to this dose the estimated airborne exposures from inhalation and submersion that could occur during work activities and from normal airborne releases from nearby facilities and the subsurface exhaust shafts. These included exposures to loose contamination on the casks and normal releases that could occur from failed fuel and crud during the handling of bare fuel in the Wet Handling Facility.

The applicant used the breathing rates for calculating doses from normal operations provided in Regulatory Guide 1.109 (NRC, 1977ad), and rates recommended in Regulatory Guide 1.183 (NRC, 2000ag) for accidental releases from Category 2 event sequences. The applicant based its dose conversion factors on information presented in EPA (1988aa; EPA, 1993aa; EPA, 1999aa). As discussed previously, the applicant performed the deterministic dose calculation using the receptor characteristics that bound those characteristics of any real member of the public or a worker. The applicant selected individual parameter values at the $95^{th}$ percentile level for receptor-related parameters including food consumption rates and periods, as well as external and inhalation exposure times. The applicant indicated that the use of $95^{th}$ percentile input values for each receptor and related pathway parameters provides a conservative dose because it represents a maximized combination of receptor characteristics.

When using GENII, Version 2.05 (Napier, 2007aa) as the biosphere model to calculate doses to the public resulting from inhalation, ingestion, and external exposure pathways, the applicant used model inputs that were representative of Amargosa Valley. Site-specific parameters chosen and assumptions employed were discussed in SAR Section 1.8.1.4.4 and more fully in BSC (2007cm). Site-specific parameters included time typically spent outdoors versus indoors, local weather, soil parameters, use of local feed stock for livestock and poultry, and consideration of human consumption of local foods. Consumption rates incorporated data collected during a 1977 survey of Amargosa Valley residents. Contingent average daily intake values by gender and food group from national data, coupled with an estimate of days per year when locally produced food is consumed, provided another estimate for site-specific consumption rates of potentially contaminated foods. To assign daily exposure times, the applicant used four population groups: nonworkers, commuters, and local indoor and outdoor workers. The percentages of people assigned to each group were derived from the 2000 census data.

**NRC Staff's Evaluation**

The NRC staff reviewed SAR Sections 1.8.1 and 1.8.2 and supporting documentation using the guidance in YMRP Section 2.1.1.5. The NRC staff finds that the applicant's assumptions and input parameter selections are acceptable because (i) the applicant used the applicable NRC guidance (i.e., NRC, 2000ag; NRC, 1977ac) for selection of dose modeling assumptions and input parameters and (ii) the applicant's assumptions and selected input parameters from the applicable NRC guidance (i.e., NRC, 2000ag; NRC, 1977ac) are acceptable because the assumptions and parameter values represent a maximized combination of receptor characteristics, which yield a conservative dose calculation. The applicant's determination that the local outdoor worker group is conservative for evaluating exposures from airborne releases is acceptable because this group spends the most time outside and, therefore, has the highest exposure potential to airborne releases.

**NRC Staff's Conclusion**

The NRC staff concludes, with reasonable assurance, that the applicant's dose calculation methodology and input parameter selection used in meeting the preclosure performance requirements of 10 CFR 63.111(a) and (b) are acceptable because (i) the models are commonly used by the radiation protection industry and the NRC for its licensing activities, and adopted by national and international organizations; and (ii) the model assumptions and input parameter selections are consistent with NRC guidance (mentioned in the previous sections) and are conservative when compared to industry standards.

### 2.1.1.5.3.2 Source Term Evaluation

The applicant described the kind, amount, and specifications of the radioactive material that could be received and possessed at the GROA as part of the applicant's development of the source term in SAR Section 1.5. For conducting its PCSA, the applicant assumed that the GROA operations would be carried out at the maximum capacity and rate of receipt of radioactive waste (SAR Sections 1.5, 1.8.1, 1.8.2, and 1.10). Source terms analyzed by the applicant for normal operations included commercial spent nuclear fuel (CSNF), naval SNF, DOE SNF (including a small amount of CSNF in the applicant's possession), and vitrified DOE HLW. The waste stream scenarios for CSNF were assumed to be 5 years old with an upper limit of 25 kW heat load. The applicant indicated that this assumption for the CSNF, when based on the earliest projected fuel receipt date (2017), is conservative when considering the current industry inventory of CSNF that will be available for disposal.

In SAR Section 1.8.1, the applicant discussed the source term released inputs, the material at risk, the damage ratio for fuel releases, the release and respirable fractions, and the leak path factors (LPFs).

The NRC staff's review and evaluation of the applicant's source term identification include (i) source term for dose calculation and (ii) cladding damage and leak path factor assumptions, as discussed in the following sections.

### Source Term for Dose Calculation

In SAR Section 1.8.2, the applicant provided potential releases and direct radiation source terms during normal surface and subsurface operations and Category 1 and Category 2 event sequences that could lead to radiological consequences. In SAR Section 1.10, the applicant discussed gamma and neutron sources for CSNF, naval SNF, DOE SNF, and DOE HLW, including gamma and neutron energy spectra. SAR Section 1.8.2.2 discussed the potential surface and subsurface operations that could lead to radiological doses to the public and radiation workers. The discussion identified the types of exposure that could be expected from the various facilities. A broad range of operational activities were evaluated, including potential radiological exposures during cask handling, repackaging of CSNF, receipt and transfer operations, storage of the casks at the aging pads, and storage of the waste packages in the emplacement drifts. The applicant stated that credit was taken, as appropriate, for ventilation system filters, shielding of facilities, shielding of transportation and storage casks, and the depth of the pool in the Wet Handling Facility that provided for retention of certain radionuclides in the pool water. Source terms included radioactive gases, volatile species, and particulates from the surface facilities; direct radiation from contained sources; resuspension of radioactive contamination on external surfaces of the casks; and activation products from the emplaced waste packages in the drifts.

The applicant assumed that DOE SNF (including naval SNF), HLW, and approximately 90 percent of the CSNF are received in sealed canisters inside transportation casks. The remaining 10 percent of the CSNF would be received in either dual-purpose canisters or as bare, intact assemblies in rail or truck transportation casks. The various waste forms are removed from the transportation vehicles and handled in the Initial Handling Facility, Canister Receipt and Closure Facility, Wet Handling Facility, and Receipt Facility, depending on their waste form. SAR Figure 1.2.1-3 provided an overview of the various pathways that the different types of waste forms will take.

The applicant evaluated the potential releases from normal operations using representative pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assembly radionuclide inventories (SAR Section 1.8.1.3). For releases from Category 1 and Category 2 event sequences, the applicant used maximum assembly inventories discussed in SAR Section 1.8.1.3. When developing the source terms, the applicant evaluated the onsite, ongoing work activity to determine the maximum available radioactive material that could contribute to the worker and public dose. For example, exposures from the casks temporarily located at the rail and truck buffer areas considered the maximum number of casks (5 trucks and 25 rails) that would be present at any time. For the aging pad, the applicant's calculations assumed a full capacity of $2.1 \times 10^7$ kg [21,000 MTHM] of CSNF. For the SNF, the applicant provided representative and maximum inventory values for both PWR and BWR SNF (SAR Tables 1.8-2 and 1.8-3). The applicant used representative values for normal operations calculations. The maximum inventory values were used for Category 2 calculations.

The applicant used SCALE/ORIGEN-S to estimate radionuclide inventories and neutron/gamma source terms for various burnups and initial enrichments of CSNF (SAR Section 1.5). In a response to an NRC staff RAI (DOE, 2009ep), the applicant compared the SCALE/ORIGEN-S-calculated concentrations of the dose-significant radionuclides in high-burnup PWR and BWR SNF and in event sequence consequence analyses to the experimental data presented in published papers and NUREG/CR–6798 (Sanders and Gauld, 2003aa). The applicant discussed the effect of the conservatism of the parameters of maximum (bounding) CSNF (SAR Section 1.5) on gamma and neutron sources for shielding analyses.

**NRC Staff's Evaluation**

The NRC staff reviewed the calculations of the source term using the guidance in YMRP Section 2.1.1.5. The NRC staff finds the characteristics of the HLW used in the source term calculations (e.g., enrichment, burnup, and decay time) reasonably represent or bound the range of characteristics of waste that could be handled at the GROA because to predict these characteristics, the applicant used representative PWR and BWR SNF, naval spent fuel, and defense waste radionuclide inventories to estimate the source terms for each waste type. The staff, however, notes that this evaluation of the applicant's bounding source term analysis does not assess whether the applicant could actually receive and possess certain fuel types at this time. For additional discussion and evaluation of the types and quantities of fuel the applicant proposes to handle at the GROA, see SER Section 2.1.1.2.3.6.1.

The NRC staff also finds the applicant's identification of the dose-significant radionuclides reasonable for the following reasons. The NRC staff evaluated the radionuclide concentrations the applicant calculated using SCALE/ORIGEN-S and the data presented in published papers and NUREG/CR–6798 (Sanders and Gauld, 2003aa) and finds the applicant's calculations and data are generally consistent with applicable guidance, and that the minor differences in radionuclide concentrations would not affect dose calculation results. In addition, the NRC staff finds that the SCALE/ORIGEN-S software is adequate for calculating radionuclide concentrations in the representative CSNF because it is a standard software widely used in the nuclear industry and the gamma and neutron source term calculated using the SCALE/ORIGEN-S accurately reflect the dependency on the two parameters, burnup and initial enrichment, consistent with NUREG/CR–6700 (Gauld and Ryman, 2001aa, Section 5) and NUREG/CR–6701 (Gauld and Parks, 2001aa, Section 4.2.2).

Furthermore, the NRC staff reviewed the applicant's analysis and finds the applicant's calculated isotopic compositions of high-burnup CSNF are consistent with published papers and NUREG/CR–6798 (Sanders and Gauld, 2003aa). The NRC staff concludes that the applicant's assumptions on the parameters of the CSNF (e.g., uranium mass, initial enrichment, burnup, cooling time, cobalt impurity contents) are conservative compared to the average SNF in the existing and projected waste streams described in SAR Section 1.5, thus overpredicting doses and exceeding potential differences between calculated and measured values associated with calculated isotope concentrations in high-burnup SNF using SCALE/ORIGEN-S. Therefore, on the basis of these considerations, the NRC staff finds that the applicant's use of SCALE/ORIGEN-S to calculate source terms for maximum SNF used in the PCSA for event sequences is acceptable.

**Cladding Damage and Leak Path Factor Assumptions**

In assessing dose consequences, the applicant made the following assumptions on cladding damage and LPF: (i) a damage ratio of 1.0 for Category 1 and Category 2 event sequences for CSNF and HLW and Category 2 seismic and fire event sequences; (ii) a damage ratio of 0.01 for normal operations and Category 1 and Category 2 event sequences involving CSNF but not resulting in cladding damage; (iii) an LPF of 0.0 for transportation casks and canisters designed and tested to be leak tight (SAR Section 1.8.1.3.6); (iv) an LPF of 0.1 for waste packages (SAR Section 1.8.1.3.6); (v) an LPF of 1.0 with no credit taken for depletion of particulates released inside the buildings; (vi) an LPF of 0.01 per stage, which resulted in a $10^{-4}$ two-stage combined high efficiency particulate air (HEPA) LPF; (vii) a 10-μm [3.9 × $10^{-5}$ in] aerodynamic equivalent diameter waste form respirable fraction; (viii) an LPF of 1.0 (i.e., no filtration) (SAR Section 1.8.1.3) for HEPA filtration for Category 1 and Category 2 event sequences when HEPA filters are unavailable; and (ix) release fraction and respirable fraction of 0.01 and 1.0, respectively, corresponding to unenclosed filter media, which are higher than values for closed filter media. In addition, the applicant discussed its assumptions on (i) cladding burst release fractions and respirable fractions and oxidation release fractions and respirable fractions from CSNF during normal operations or a Category 1 or Category 2 event sequence for SNF in a dry environment, (ii) fuel fines and volatiles, and (iii) low-burnup and high-burnup SNF.

The applicant provided CSNF in-pool release fractions for drop or impact events in the Wet Handling Facility SNF pool; pool decontamination factors; and LPFs for Wet Handling Facility SNF pool for noble gases, halogens, and alkali metals (SAR Section 1.8.1.3.6 and Table 1.8-9). In SAR Section 1.2.5.3.2.2, the applicant stated that Regulatory Guide 1.183 recommendations in NRC (2000ag, Appendix B) apply because the depth of water above the damaged fuel is at least 7.0 m [23 ft].

The applicant described the airborne release fraction and respirable fraction for the radioactivity from the combustible portion of the low-level waste facility inventory as the source for release for a fire event (SAR Section 1.8.1.3.5). The applicant used specific release fractions and respirable fractions for the dry active waste in drums, Wet Handling Facility pool filter and spent resins in high-integrity containers, burning uncontained combustible dry active waste, and heat-induced damage to a HEPA filter, respectively (SAR Section 1.8.1.3.5). In lieu of airborne material size distribution, the applicant assumed a respirable fraction of 1.0. The applicant used bounding values of measured respirable fractions and airborne release fractions for uncontained waste.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the waste form characteristics and evaluation of the potential releases from normal operations and Category 1 and Category 2 event sequences using the guidance in YMRP Section 2.1.1.5. The NRC staff finds the type, quality, and concentration of airborne radionuclides released during normal operations and Category 1 and Category 2 event sequences are supported by appropriate data, or are in accordance with appropriate NRC guidance documents, as explained below.

The NRC staff finds that the applicant's assumptions of the fraction of the material at risk, release fractions, respirable fractions, and LPF values are acceptable because the assumptions are in accordance with NUREG/CR–6410 (Science Applications International Corporation, 1998aa), SFST–ISG–5 (NRC, 2000af), NUREG/CR–6672 (Sprung, et al., 2000aa), ANSI N14.5–1997 (American National Standards Institute, 1997aa), and ANSI/ANS–5.1–1998 (American Nuclear Society, 2006ab).

More specifically, the NRC staff finds that the applicant's assumption regarding the damage ratio of 1.0 for Category 1 and Category 2 event sequences for CSNF and HLW and Category 2 seismic and fire event sequences is consistent with SFST–ISG–5 (NRC, 2000af). The NRC staff also finds that

- The 0.01 damage ratio assumption for CSNF not involving cladding damage during normal operations and Category 1 and Category 2 event sequences is consistent with SFST–ISG–5 (NRC, 2000af)

- The assumption of the waste form respirable fraction of 10-μm [$3.9 \times 10^{-5}$-in] aerodynamic equivalent diameter is within the cutoff limit –presented in Appendix B of ANSI/ANS–5.1–1998 (American Nuclear Society, 2006ab)

- Assumptions on cladding burst respirable fraction and oxidation respirable fraction for fuel fines and volatiles and low- and high-burnup SNF are consistent with SFST–ISG–5 (NRC, 2000af) and the published test results in SAR Section 1.8.1.3.3

- The LPF assumption for transportation casks and canisters is consistent with ANSI N14.5–1997 (American National Standards Institute, 1997aa) and SFST–ISG–5 (NRC, 2000af)

- The 0.0 LPF assumption for transportation casks and canisters designed and tested to be leak tight is acceptable because it is consistent with NUREG/CR–6672 (Sprung, et al., 2000aa) and SFST–ISG–5 (NRC, 2000af)

- The 0.1 LPF assumption for waste packages is acceptable because it is consistent with the recommendations in SFST–ISG–5 (NRC, 2000af)

- The LPF assumption of 1.0 for the buildings is conservative because no credit is taken for depletion of particulates released inside the buildings

- The assumption of an LPF of 0.01 per stage, which gives a $10^{-4}$ two-stage combined HEPA LPF is consistent with DOE (2003ae) and NRC Section F.2.1.3 (Science Applications International Corporation, 1998aa)

- Use of an LPF of 1.0 (i.e., no filtration) is appropriate for HEPA filtration for Category 1 and Category 2 event sequences when HEPA filters are not available because no credit is taken for HEPA filtration

- The selection of 0.01 and 1.0 for the release fraction and respirable fraction for unenclosed filter media during a seismic event sequence are conservative because these values are consistent with SFST–ISG–5 (NRC, 2000af)

In addition, the NRC staff finds that the applicant's assumptions for release fractions; pool decontamination factors; and LPF for the Wet Handling Facility pool for noble gases, halogens, and alkali metals are conservative because these fractions are consistent with the release fractions in Regulatory Guide 1.183 (NRC, 2000ag).

**NRC Staff's Conclusion**

The NRC staff finds that for the purposes of the PCSA, the applicant's assumption that the GROA operations would be carried out at the maximum capacity and rate of receipt of radioactive waste is acceptable because it is in accordance with the 10 CFR 63.21(c)(5) requirement. The NRC staff also finds that the characteristics of the HLW used in the applicant's source term calculations (e.g., enrichment, burnup, and decay time) reasonably represent or bound the range of characteristics of waste that could be handled at the GROA. The NRC staff finds that the type, quality, and concentration of airborne radionuclides released during normal operations and Category 1 and Category 2 event sequences are supported by appropriate data or are in accordance with appropriate NRC guidance documents. Therefore, the NRC staff finds, with reasonable assurance, that the requirements in 10 CFR 63.21 (c)(5) and 10 CFR 63.111(a) are satisfied.

#### 2.1.1.5.3.3 Public Dose Calculation

The applicant performed calculations for members of the public for both the onsite and offsite area. These calculations included both normal operations and Category 2 event sequences. As stated in SAR Section 1.8.3.2 1.7.5, no Category 1 event sequences were identified that required analysis for public dose. The applicant identified several areas, both onsite and offsite, for determining the public dose.

Public exposure may occur from either direct radiation or from airborne releases. Exposure sources included the release of radioactive gases; volatile species and particulates from surface and subsurface facility operations; and direct exposure from contained radioactive sources within transportation casks, aging overpacks, and surface facilities and buildings. Radiological exposures from background radiation and offsite transportation were not included in the public dose calculations.

The NRC staff's review and evaluation of the applicant's public dose calculation includes (i) features limiting onsite public exposures, (ii) onsite members of the public dose calculation, (iii) features limiting offsite public exposures, and (iv) offsite members of the public dose calculation, as presented in the following sections.

**Features Limiting Onsite Public Exposures**

The applicant determined that potential public exposures within the surface facility GROA due to waste handling activities could occur at several locations due to both direct radiation and airborne radiation. To limit the general public's exposure to direct radiation while onsite at the GROA, the applicant established a restricted area within the surface facility GROA where radioactive material is handled and stored. The restricted area includes the fenced protected area that encompasses the truck and train buffer area (Areas 33A and 33B), the waste handling facilities, the aging pad, and the North Portal entrance. Casks arriving onsite are moved into the restricted area and temporarily stored at the rail and truck buffer areas. From there, the casks are moved to the waste handling buildings to be placed in canister configurations for storage. The applicant stated in some cases, this requires cutting the cask open and repackaging the fuel. The fuel in the waste package would then be moved into its assigned emplacement drift for disposal or is temporarily placed in suitable casks at the aging pad for aging. The applicant stated throughout this process that exposures will occur that could affect the onsite public; and in particular, the construction workers completing work on other portions of the site. The applicant assumed that these construction workers were onsite 2,000 hours/year as opposed to the transient public, such as delivery personnel.

To reduce the exposure to the onsite public while the casks are inside the waste handling facilities, the applicant established a maximum 0.0025 mSv/hour [0.25 mrem/hour] dose limit for the exterior of the buildings at the personnel level as specified in SAR Table 1.10-2. The applicant stated that public exposures from SNF and HLW being processed inside the waste handling facilities will be minimized on the basis of the shielding design of the facilities and the use of remote operations. Design criteria for areas where canisters are handled or spent fuel is repackaged into TAD canisters include thick concrete walls, floors, and ceilings; shielded viewing windows; shielded doors; slide gates in concrete floors; shielded canister transfer machines; shielded waste package trolleys; and specially designed penetrations through walls and floors to provide shielding for piping; heating, ventilation, and air conditioning ducts; and electrical raceways. Large concrete shield walls surrounding facility work areas allow routine occupancy in repository open areas. Provisions for shielding in the waste handling buildings and the transportation and emplacement vehicle were described in SAR Tables 1.10-35 through 1.10-46. The applicant established shielding requirements using the point of maximum or peak radiation dose. Therefore, the applicant stated that the overall general area radiation levels would be less than this maximum calculated dose. The applicant performed dose calculations for the onsite handling facilities using the MCNP transport computer code (Briesmeister, 1997aa).

The applicant's shielding calculations were presented in a number of SAR sections, including Sections 1.10.3 and 1.8.4.1.3. The applicant stated that concrete required for shielding of personnel associated with the waste handling facilities will be designed to American Concrete Institute code requirements and site seismic criteria. The applicant also stated that the Wet Handling Facility, where fuel assemblies will be transferred into TAD canisters, is designed with an in-ground steel-lined concrete pool. SAR Section 1.7.2.3 discussed degradation or loss of shielding for several types of failures. In addition to the structural aspects designed into the buildings, should any shielding or protective systems be lost during an event, the applicant's emergency plan includes provisions for warning site personnel and evacuating personnel to safe areas (SAR Section 5.7.2.2.3). The applicant stated that distance attenuation between the waste handling facilities and the various onsite public areas further reduces the dose rates. The applicant indicated that the restricted area fence is more than 200 m [656 ft] from the waste handling facilities resulting in more than 3,000 times reduction in the dose rate due to distance. By providing shielding and establishing a dose rate limit at the exterior of the waste handling facilities, according to the applicant, the dose contribution to the onsite public becomes negligible due to work activities underway within these facilities.

**NRC Staff Evaluation**

The NRC staff reviewed the facility features used to limit onsite public dose using the guidance in YMRP Section 2.1.1.5. The NRC staff finds the applicant's selection of features limiting onsite public exposure acceptable because the applicant followed common, industry standard approaches to incorporate the principles of time, distance, and shielding to limit radiological exposure.

**Onsite Members of the Public Dose Calculation**

For the onsite areas, the Yucca Mountain site consists of a restricted area, protected area, GROA, and controlled area. SAR Section 1.1.1.1 described these areas with their visual representation in SAR Figures 1.1-1, 1.1-2, 1.2.1-2, 1.8-2, and 5.8-2.

SAR Tables 1.8-1 and 1.8-36 and Figure 1.8-2 listed radiation dose limits that apply to the public in the areas within and outside the preclosure controlled area. As discussed previously, the onsite members of the public included construction workers, delivery personnel, public visitors, and mining personnel of the Cind-R-Lite mine within the preclosure controlled area. These onsite members of the public are assumed to be present 2,000 hours per year.

According to the applicant, radiological exposures to these members of the public would be bounded by the calculated doses to the construction workers in the surface facility GROA.

For direct radiation during normal operations, the applicant used design limits and regulatory limits for the source terms to assess public doses. This included a 0.0025 mSv/hour [0.25 mrem/hour] exterior building design limit, a 0.4 mSv/hour [40 mrem/hour] design limit on the surface of the aging cask, and the 10 CFR 71.47 dose rate limits for transportation casks. On the basis of these design and regulatory limits, the applicant indicated that the direct radiation dose to the onsite public from the waste handling facilities becomes negligible compared to the direct radiation dose from the aging pad and the rail and truck buffer area.

As discussed in SAR Section 1.8.3.1.3, the applicant assumed that the dose rates for the transportation casks were bounded by the limits in 10 CFR 71.47. These limits are 2 mSv/hour [200 mrem/hour] at any point on the cask exterior surface and 0.1 mSv/hour [10 mrem/hour] at 2 m [6.6 ft] from the cask surface. The applicant used cask dose rates based on calculations from cask safety analysis reports submitted to NRC and calculations the applicant performed to develop dose rate values versus distance that would be below the 10 CFR 71.47 limits. The Transnuclear TN-32 cask and the British Nuclear Fuels TS-125 cask with a W21 canister were used as models. The TN-32 cask holds 32 PWR and the TS-125/W21 cask holds 21 PWR fuel assemblies. The applicant performed dose calculations at various distances with 25-rail and 5-truck casks parked in the buffer areas to develop the annual doses contribution to the onsite public areas from the rail and cask buffer areas (SAR Table 1.8-28).

According to the applicant, airborne releases may occur from casks that require opening. The applicant stated that this will involve only the CSNF handled in the Wet Handling Facility. As shown in SAR Table 1.8-28, dose contributions to the onsite public from airborne releases during normal operations are small when compared to direct radiation exposures and the 1.0 mSv/year [100 mrem/year] limits of 10 CFR 20.1301, even when combining the airborne source terms from the handling facilities and the subsurface exhaust shafts.

The direct exposures resulting from the casks at the Aging Facility, according to the applicant, included both direct exposure and skyshine. The casks consist of an inner 8.18-cm [1.25-in] stainless steel basket that is placed in a 95.3-cm [37.5-in] concrete overpack or in concrete horizontal modules with thick concrete end walls to provide shielding. The aging pad is located separately from the surface facility GROA facilities to reduce worker exposure. According to the applicant, the highest estimated dose to the public {0.098 mSv/year [9.8 mrem/year]} at the nearest location to the aging pad is at the North Perimeter Security Area.

The applicant considered normal subsurface radiological releases in the public dose calculations. Because the canisters placed inside the drifts are sealed, the applicant assumed that only contamination on the outside of the canister was available for release. In addition, the applicant stated that neutron emission from the canisters would activate dust and air in the drifts that would also be released from the subsurface facility shafts, which are not filtered. Dose calculations for the subsurface facilities were discussed in SAR Section 1.8.2.2.2. The predominant isotopes modeled for the subsurface facility shaft releases were provided in SAR Table 1.8-24.

The highest estimated doses to the public, according to the applicant, were determined to be at the lower muck yard (Area 780) and the warehouse and nonnuclear Receipt Facility (Area 230) due to the casks that will be temporarily stored at the rail and truck buffer areas (Areas 33A and 33B). The applicant stated that the maximum dose rate limits for each cask were the shipping limits specified in 10 CFR 71.47. These limits were discussed in SAR Section 1.8.3.1.3. When considering the maximum number of trucks and rail casks that could be present at the buffer area, the applicant indicated that the dose rates at this location become the predominant dose contributor to the onsite public. The applicant determined that the maximum calculated dose to the public was 0.78 mSv/year [78 mrem/year] at the lower muck yard and 0.76 mSv/year [76 mrem/year] at the nonnuclear Receipt Facility on the basis of an occupancy time of 2,000 hours/year. As discussed in SAR Section 1.1.9.3.2.12, the applicant stated that it would use the lower muck yard for parking and equipment storage, public outreach, and the test coordination office, and as a maintenance and repair area. The basis for the muck yard dose projection was further discussed in BSC (2007am, Sections 6.1.2 and 6.1.3). The technical basis for the warehouse and nonnuclear Receipt Facility dose projection was discussed in the applicant's response to the NRC staff's RAI (DOE, 2009eq).

**NRC Staff's Evaluation**

The NRC staff reviewed SAR Section 1.8.3 and supporting documentation including the applicant's response to the NRC staff's RAI using the guidance in YMRP Section 2.1.1.5. The NRC staff concludes that the applicant provided sufficient information for onsite public dose assessment to satisfy 10 CFR 63.111 and 10 CFR Part 20 for the following reasons. The NRC staff finds that the applicant's approach to calculate dose to onsite members of the public is acceptable because DOE accounted for the appropriate exposure pathways (i.e., direct radiation, submersion, and inhalation), accounted for public exposure using industry standard computer codes (i.e., MCNP transport computer code) and calculation methods, and followed applicable NRC guidance for making these calculations. The NRC staff also finds that the applicant accurately applied the dose limit for each of these areas, including the limits specified in the performance objectives of 10 CFR 63.111(a), 63.111(b), and the limits for the members of the public defined in 10 CFR 63.204, 10 CFR 20.1101(d), and 10 CFR 20.1301.

**Features Limiting Offsite Public Exposures**

According to the applicant, buildings that may handle fuel assemblies or are involved with the cutting open of the canisters have HEPA filtration systems to reduce radioactive particulate releases. In addition, the applicant incorporated the operational constraint in 10 CFR 20.1101(d) for air emissions of 0.1 mSv/year [10 mrem/year] to any individual member of the public into its Operational Radiation Protection Program.

The Aging Facility incorporates features to reduce exposures to workers and the public (SAR Section 1.2.7.6.5), which include installing shield walls on the horizontal storage modules, locating the aging pads away from other facilities, establishing a posted restricted area around the aging pads to warn personnel of radiation, and controlling access onto the aging pads by use of a security fence.

The applicant stated that because the surface facility GROA is isolated on a very large controlled area away from the site boundaries, the public is restricted from establishing a close, permanent residence. Three sides of the preclosure controlled area are bordered by federally controlled lands. The nearest location where the public could establish permanent residence is approximately 18.5 km [11.5 mi] south of the site boundary. The applicant stated that the nearest member of the public lives 22.4 km [13.9 mi] from the surface facility GROA boundary. At these distances, the applicant indicated that radiation from normal site operations would not be distinguishable from normal background levels, even with the facilities operating at maximum capacity. For residents in the Amargosa Valley, average annual dose from cosmic, cosmogenic, and terrestrial radiation is 0.96 mSv/year [96 mrem/year], as outlined in DOE (2002aa, Section 3.1.8.2). Adding radon and internal radioactivity naturally results in an average annual dose to an Amargosa Valley resident of 3.4 mSv/year [340 mrem/year]. This is slightly higher than the U.S. average of 3.0 mSv/year [300 mrem/year], as shown in DOE (2002aa, Table 3-30). For Category 2 event sequences, even the worst case event results, according to the applicant, is only 0.1 mSv [10 mrem] at the site boundary within the general environment.

**NRC Staff's Evaluation**

The NRC staff reviewed the facility features used to limit offsite public dose using the guidance in YMRP Section 2.1.1.5. The NRC staff finds the applicant's selection of the features limiting offsite public exposures to be acceptable because the applicant followed common, industry standard approaches to incorporate the principles of time, distance, and shielding to limit radiological exposure.

**Offsite Members of the Public Dose Calculation**

The applicant performed a series of calculations to determine distances from the surface facility GROA and the subsurface exhaust shafts to the site boundary. The applicant stated that the closest real member of the public to the surface facility GROA is located at the intersection of U.S. Route 95 and Nevada State Route 373, 22.3 km [13.9 mi] toward the south wind sector. From the closest subsurface exhaust shaft, Exhaust Shaft 3, the nearest real member of the public is 21.5 km [13.3 mi] toward the south-southeast wind sector (BSC, 2007bp).

The applicant did not use the distance to the real member of the public for the dose calculations, but instead determined the dose to a hypothetical member of the public located at the site boundary, closer than real members of the public. To determine X/Q values from the surface facility GROA boundary, the applicant took 8 of the 16 wind sectors that impacted the south and the east site boundaries and calculated the X/Q values. The distance used in both the south wind sector (from north) and the south-southeast wind sector (from north-northwest) was 18,500 m [11.5 mi]. For the direction to the west (from the east) the applicant used 11,000 m [6.8 mi]. The applicant's calculations for the eight sectors determined that due to the predominant wind patterns for the site, the south-southeast wind direction resulted in the highest X/Q values. Wind patterns for the site were shown in SAR Figures 1.1-14 through 1.1-51. The applicant selected the south-southeast X/Q value for the offsite public in the general environment (SAR Table 1.8-12) for the dose calculations for the maximum exposure to a hypothetical member of the public.

For the distances from the subsurface exhaust shafts to the site boundary within the general environment, the applicant used the closest exhaust shaft to perform the X/Q calculations. The most conservative X/Q value was in the southeast wind direction (from northwest). The subsurface exhaust shaft X/Q values are similar in magnitude to the surface facility GROA values.

For the site boundary in the north and east direction toward the NTS and the Nevada Test and Training Range, SAR Table 1.8-10 listed the distances used to perform X/Q calculations. The applicant stated that the highest X/Q value was in the southeast wind direction. The southeast wind direction intersects the southernmost corner of the NTS and was used for calculating doses to the NTS and the Nevada Testing and Training Range for the offsite public not in the general environment. Distances for the subsurface exhaust shafts and the resulting X/Q values were shown in SAR Tables 1.8-11 and 1.8-12.

For the offsite public (i.e., outside the preclosure controlled area), the radiological source term was discussed in BSC (2008ay, Section 6.7) and provided in SAR Table 1.8-29. Source terms that the applicant considered for normal operations included (i) fission product gases, volatile species, and fuel fines and crud particulates released from the waste handling facility, such as during opening and handling of a canister, that are not removed by the HEPA filters; (ii) neutron activation of the air and silica dust inside the emplacement drifts that could become airborne; and (iii) resuspension of radioactive contamination on the canisters contained in the aging overpacks. The applicant added the calculated doses from these three source terms to produce the values in SAR Table 1.8-29. Offsite doses to the general public were calculated as 0.0005 mSv/year [0.05 mrem/year]. Offsite doses at the site boundary with the NTS and the Nevada Test and Training Range were calculated as 0.0011 mSv/year [0.11 mrem/year]. The applicant indicated that these values were well below the dose limits of 0.15 mSv/year and 1.0 mSv/year [15 mrem/year and 100 mrem/year] that apply to these offsite areas.

The applicant stated the property north and east of the Yucca Mountain site boundary controlled by the NTS and the Nevada Test and Training Range was evaluated using the 10 CFR 20.1301 limits for individual members of the public because these are U.S Government-controlled areas that restrict the presence of the general public. The applicant also stated that it evaluated the area south and west of the Yucca Mountain site boundary using the limits in 10 CFR 63.111(a)(2) and 63.204 because this area is open to the general public. The applicant determined that the direct radiation levels from source terms associated with waste handling operations at the surface facility GROA decreased by a factor of more than 13 orders of magnitude because of large distances from the offsite public to the surface facility GROA, resulting in insignificant offsite public dose from direct radiation and skyshine from normal operations.

For airborne dose calculations and determination of the X/Q values for the eight sectors, the applicant used the minimum distances from the surface facility GROA boundary to the site boundary, as shown in BSC (2007bp, Tables 9 and 10) and SAR Table 1.8-10, to calculate the X/Q values for the offsite public not in the general environment. For the general environment calculation, the applicant used the distance values in BSC (2007bp, Table 18) to determine X/Q values. BSC (2007bp, Table 34) provided the X/Q values shown in SAR Table 1.8-12 and referenced Table 18 in BSC (2007bp) as its source.

The applicant presented airborne exposures from normal operations in SAR Section 1.8.3.1.2. The applicant evaluated potential airborne release doses from inhalation, ingestion, resuspension inhalation, air submersion, and groundshine pathways as a continuous release throughout the year. The applicant modeled the airborne releases as ground-level releases. Offsite public dose values presented in SAR Tables 1.8-28 and 1.8-32 included the sum of the releases from the Wet Handling Facility, the aging pads, and the subsurface exhaust shafts. For the offsite public in the general environment where food ingestion doses were evaluated, the applicant calculated internal doses using a 50-year dose commitment period. Ground contamination and subsequent food pathway exposures included the buildup of contamination for the entire operational period of 50 years.

SAR Table 1.8-29 provided the estimated public dose during normal operations. The applicant calculated these doses on the basis of airborne releases. The applicant stated that it did not include dose contributions from offsite transportation, because these dose contributions are not required per 40 CFR 191.01(a).

For Category 2 event sequences, only offsite doses to the public are required to be analyzed by 10 CFR 63.111(b)(2). The applicant modeled the dose calculations on the basis of airborne releases from both the surface facilities and the subsurface exhaust shafts. The airborne releases resulted in an acute individual exposure during the transient release and a chronic individual exposure to ground contamination and contaminated food after plume passage. The applicant assumed ground exposure and food consumption by the offsite public was 30 days. Because of the large distances to the site boundary, the applicant found that direct radiation resulting from a Category 2 event sequence was insignificant.

The applicant provided the resulting doses for Category 2 event sequences in SAR Tables 1.8-30 and 1.8-31. For the offsite public in the general environment, the highest whole body dose was for Event Sequence 2-01 (Seismic event resulting in Low-Level Waste Facility collapse and failure of HEPA filters and ductwork in other facilities) involving a seismic event resulting in low-level waste facility collapse and failure of the HEPA filters and ductwork in the other facilities. The resulting dose was 0.1 mSv [10 mrem] TEDE. For the offsite dose not within the general environment, the highest whole body dose was associated with Event Sequence 2-03 (Breach of a sealed HLW canister in an unsealed waste package) as well as Event Sequence 2-01. The resulting dose was 0.3 mSv [30 mrem] TEDE. Of the two event sequences, according to the applicant, 2-03 yielded the highest organ dose of 6.8 mSv [680 mrem] to the bone surface and the highest lens of eye dose of 1.0 mSv [100 mrem] and skin dose of 0.9 mSv [90 mrem].

**NRC Staff's Evaluation**

The NRC staff reviewed SAR Section 1.8.3 and supporting documentation using the guidance in YMRP Section 2.1.1.5 and concludes that the applicant provided sufficient information for offsite public dose assessment to satisfy 10 CFR 63.111(a), (b), and (c) and 10 CFR Part 20 for the following reasons. The NRC staff finds that the applicant's approach to calculate dose for offsite members of the public is acceptable because the applicant accounted for the appropriate exposure pathways (i.e., inhalation, ingestion, resuspension inhalation, submersion, and groundshine); accounted for public exposure using industry standard computer codes (i.e., GENII Version 2.05) and calculation methods; and as described further below, followed the applicable NRC guidance for making these calculations. The applicant's definition of the two areas where the offsite public may be located is consistent with the definition of the general environment in 10 CFR 63.202. The NRC staff also finds that the applicant accurately applied the dose limit for each of these areas, including the limits specified in the performance objectives of 10 CFR 63.111(a), (b), and (c) and the limits for the members of the public defined in 10 CFR Part 20.

The NRC staff finds that the applicant's use of consumption rates for Amargosa Valley residents is consistent with the Regulatory Guide 1.109 (NRC, 1977ad) values when the parameter values are adjusted for site-specific data.

The NRC staff finds that the applicant-established 0.4 mSv/hour [40 mrem/hour] combined neutron and gamma dose design rate limit for the casks at the aging pad is acceptable because it is consistent with shielding design criteria in NUREG–1536 [NRC, 1997ae, Section 5.0 (V)(1)(a)], which provides an acceptable range of 0.2 to 4.0 mSv/hour [20 to 400 mrem/hour].

The NRC staff finds the use of a 50-year commitment period for calculating internal dose acceptable because it is consistent with the definition of committed dose equivalent in 10 CFR 20.1003.

In addition, the NRC staff finds that the applicant's estimation of the 0.0011 mSv/year [0.11 mrem/year] TEDE for the offsite public not within the general environment is well below the 1.0 mSv/year [100 mrem/year] limit in 10 CFR 20.1301(a)(1). The 0.0005 mSv/year [0.05 mrem/year] TEDE for the offsite public in the general environment is below the 0.15 mSv/year [15 mrem/year] limit in 10 CFR 63.204. These doses are also below the operational dose constraint for radioactive air emissions of 0.1 mSv/year [10 mrem/year] specified in 10 CFR 20.1101(d).

**NRC Staff's Conclusion**

The NRC staff concludes, with reasonable assurance, that the applicant satisfies the requirements of 10 CFR 63.111(a), 10 CFR 63.111(b), 10 CFR 63.204, 10 CFR 20.1101(d), and 10 CFR 20.1301 because the applicant's calculations accounted for the appropriate exposure pathways and public exposure using standard codes and standard calculation methods. Specifically, the NRC staff finds that the consequence analyses and associated input parameters for members of the public are in compliance with the dose performance objectives of 10 CFR 63.111(a), 10 CFR 63.111(b), 10 CFR 63.111(c), 10 CFR 20.1101(d), and 10 CFR 20.1301. The NRC staff also finds that the applicant satisfies the dose performance objectives of 10 CFR 63.204 for offsite members of the public located in the general environment.

**2.1.1.5.3.4 Worker Dose Calculation**

The applicant calculated radiological doses to radiation workers as part of PCSA (SAR Section 1.8.4). The radiation worker dose is assessed for (i) compliance determinations with the dose limits of 10 CFR Parts 63 and 20 and (ii) important to safety structures, systems, and components determinations. As specified in 10 CFR 63.111(a)(1) and 10 CFR 63.111(b)(1), 10 CFR Part 20 dose limits for radiation workers apply during normal operations and Category 1 event sequences. Radiation worker safety assessments are not required for Category 2 event sequences. Because the applicant's PCSA indicated that there are no Category 1 event sequences (SAR Section 1.8.6), radiation worker safety assessments and mitigation of worker doses provided by the structures, systems, and components do not factor into the applicant's important-to-safety structures, systems, and components determination. The applicant credited the structures, systems, and components to prevent Category 1 event sequences. SAR Table 1.8-36 indicated that annual doses to radiation workers were estimated to be 26 percent or less {0.013 Sv [1.3 rem] or less} of the 0.05-Sv [5-rem] annual limit for the TEDE [annual dose limit specified in 10 CFR Part 20.1201(a)(1)]. SAR Table 1.8-25 demonstrated that the direct external exposure to radiation emitted from radioactive waste in sealed containers during normal operations dominates worker TEDE. According to the applicant, worker external dose from emitted radiation provides the greatest contribution to dose when compared to other exposure pathways for workers (e.g., inhalation of airborne radioactive materials) with respect to preclosure safety.

The applicant based total annual doses to radiation workers on four major sources: (i) direct radiation from normal operations within the facility, (ii) direct radiation from sealed sources located outside the facility, (iii) airborne releases of radioactive material from normal operations at surface and subsurface facilities, and (iv) Category 1 event sequences. The maximum potential total annual dose of 0.013 Sv [1.3 rem] the applicant calculated corresponds to a radiation worker in the Receipt Facility (SAR Table 1.8-25). This calculated dose was dominated by direct radiation from normal operations in the Receipt Facility for an annual process rate of 273 casks.

The applicant calculated radiation worker exposure for the following pathways: (i) direct irradiation inside facilities by contained sources therein; (ii) direct irradiation at outside receptor locations by casks in buffer or aging areas; and (iii) inhalation and air submersion at outside receptor locations due to estimated airborne releases from surface facilities, aging pads, and subsurface emplacement drifts. As shown in BSC (2008al, Tables 3, 6, and 7), the estimated radiation worker doses at different facilities indicated that direct irradiation during normal operations inside facilities represented the greatest contribution to dose for radiation workers.

The NRC staff's review and evaluation of the applicant's worker dose calculation includes (i) direct radiation calculation, (ii) airborne releases of radioactive material, and (iii) aggregation of worker doses, as presented in the following sections.

**Direct Radiation Calculation**

The applicant's estimated radiation worker doses are dominated by direct external exposure to radiation emitted from radioactive waste in sealed containers during normal operations, as shown in BSC (2008al, Table 7). The applicant's worker dose assessments for individual facilities culminated in the results in SAR Table 1.8-25. Facility throughput (i.e., amount of radioactive waste processed per year), number of work crews, time spent performing operational tasks, and dose rates from the radiation field at different work locations were factored into these external dose calculations (SAR Section 1.8.4.1.3). The applicant stated that estimated dose rates depend on the radiation emission rates from radioactive waste, direct radiation scaling factors the applicant used, credit taken for shielding materials and distances of workers to direct radiation source terms, and flux-to-dose conversion factors. The applicant assumed five work crews will be available to staff the three shifts of operations, as detailed in SAR Equation 1.8-26 and in BSC (2008bw, Section 3.2.4).

The NRC staff organized its evaluation of the applicant's direct radiation calculations in the following four sections: (i) radiation emission rates, (ii) direct radiation scaling factors, (iii) credit for shielding materials and worker distances, and (iv) flux-to-dose conversion factors.

**Radiation Emission Rates**

The applicant performed shielding calculations with the MCNP computer program (SAR Section 1.8.4.1.3). The applicant used a maximum source term for establishing shielding design parameters and selected a design basis source term for calculating worker doses, as discussed in SER Section 2.1.1.5.3.2 (BSC, 2008cc). The applicant specified PWR fuel with a burnup of 60 GW-day per metric ton and cooling time of 10 years as the design basis source term for its worker dose assessments (SAR Table 1.10-19).

**NRC Staff's Evaluation**

The NRC staff reviewed the information on radiation emission rates using the guidance in YMRP Section 2.1.1.5. The NRC staff finds that the source terms and assumptions used by the applicant are conservative for external dose assessments and adequately accounts for the preferential loading of SNF with a range of burnups because PWR fuel represents a greater source term for penetrating radiation (gamma ray and neutron) than other waste forms and the design basis characteristics of PWR fuel are expected to overestimate the radiation emission rates compared to the average SNF assembly handled at the GROA. The NRC staff provided a detailed review on the applicant's source terms in SER Section 2.1.1.5.3.2.

**Direct Radiation Scaling Factors**

The applicant applied scaling factors for facility throughput and direct radiation source term to its worker dose calculations, as described in its response to the NRC staff's RAI, DOE (2009eo, Enclosure 1). The applicant applied the source-term scaling factor to adjust annual worker doses, initially calculated for irradiation by a maximum source term for the entire year, to an annual dose from direct radiation that is more representative of full-scale operations. Because the applicant used a design basis source term instead of a maximum source term for calculating annual worker doses, the applicant applied a dose reduction factor of 2.7, as detailed in its response to the NRC staff's RAI in DOE (2009eo, Enclosure 1).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on the scaling factor for the direct radiation source term, outlined in the applicant's response to the NRC staff's RAI, DOE (2009eo, Enclosure 1), using the guidance in YMRP Section 2.1.1.5 and finds that the scaling factor applied to the worker dose results is reasonable because individual facilities will receive CSNF with a range of characteristics (e.g., burnup and cooling time) during a single year of operation. The NRC staff also finds that the factor of 2.7 reduction in dose is acceptable because the design basis source term represents an upper bound for at least 95 percent of direct radiation sources expected to be received at the repository, as detailed in DOE (2009eo, Enclosure 1). The NRC staff concludes that the applicant's scaled doses for radiation workers are conservative because doses due to an average source term are expected to be significantly lower than doses for a design basis source term [i.e., lower by more than a factor of 4.8, as described in DOE (2009eo, Enclosure 1). For these reasons, the NRC staff finds that the applicant's consequence analysis for radiation workers is acceptable for demonstrating compliance with the preclosure performance objectives and the occupational dose limits, which are annual limits that cannot be exceeded during any year of operation.

**Credit for Shielding Materials and Worker Distances**

The applicant used dose rate profiles for a TS125 rail transportation cask in the Canister Receipt and Closure Facility, as shown in BSC (2007cl, Table 3), although, according to the applicant, additional credit could have been taken for shielding materials in aging overpacks and shielded transfer casks. The applicant stated that dose rates from TS125 casks are bounding because of the higher dose rates compared to other cask configurations, as detailed in BSC (2007cl, Section 3.2.1). The applicant used a similar approach for estimating external doses to workers in other facilities, as shown in BSC (2008bw, Table 1) and in BSC (2007ck, Table 1).

In its calculations for estimating shielding design requirements, the applicant assumed that each transportation cask received in the Receipt Facility and Canister Receipt and Closure Facility will contain one canister of any type, as outlined in the applicant's response to the NRC staff's RAI in DOE (2009en, Section 3.1.7). The applicant clarified in its response to the NRC staff's RAI, in DOE (2009ek, Enclosure 5) that dose estimates for radiation workers were not affected by this assumption, because operations involving handling of more than one canister within a transportation cask would be performed remotely in rooms where other workers would not be present, as described in DOE (2009ek, Enclosure 5, Section 1). The applicant's dose calculations for radiation workers depended on annual throughput estimates, which were based on five DOE HLW canisters and nine DOE SNF canisters per transportation cask, as detailed in DOE (2009ek, Enclosure 5, Section 1).

**NRC Staff's Evaluation**

The NRC staff reviewed the technical bases and input data used in the applicant's worker dose analyses including the applicant's response to the NRC staff's RAI using the guidance in YMRP Section 2.1.1.5 and finds the credit the applicant assumed for shielding materials and worker distances in worker dose calculations acceptable to control and limit worker exposure during waste handling and transfer operations because remote operations would limit time workers spend in elevated radiation areas. The NRC staff performed a detailed review on the applicant's source terms in SER Section 2.1.1.5.3.2 and found that the applicant's approach of using dose rate profiles for a rail transport cask is conservative and adequately accounts for uncertainty, including any degradation of shielding materials during normal operations.

**Flux-to-Dose Conversion Factors**

The applicant used the flux-to-dose conversion factors, provided in ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa), for converting neutron and gamma fluxes to dose rates, as described in BSC (2008cc, Section 6.1.2). The SCALE computer code uses ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa) for the dose calculation of the casks at the aging pad inside their concrete storage modules. The MCNP computer code uses the conversion factors in ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa). ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa) includes internationally accepted data the ICRP proposed.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on flux-to-dose conversion factors using the guidance in YMRP Section 2.1.1.5. The NRC staff notes that the 1977 version of the ANSI/ANS-6.1.1-1977 (American Nuclear Society, 1977aa) standard was superseded when the latest version of the standard, ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa), was issued in 1991. The 1991 version is consistent with the effective dose equivalent, summation of weighted organ dose equivalents, and organ weighting factors in International Commission on Radiological Protection–26 (1977aa) and 10 CFR 20.1003. In the 10 CFR Part 63 final rule, NRC stated that "… use of external dosimetry methods in existing federal radiation guidance, Federal Guidance Report No. 12 (EPA, 1993aa), in combination with the more current internal dosimetry methods consistent with 40 CFR Part 197, Appendix A, is an acceptable approach for calculating TEDE." EPA (1993aa) is consistent with the ICRP–26 dosimetry. The NRC staff compared the 1977 and 1991 versions of the flux-to-dose rate conversion factors. Because the use of the 1977 conversion factors would not lead to an underestimation of dose, the NRC staff finds the applicant's selection of flux-to-dose rate conversion factors to be acceptable. The NRC staff also reviewed the dose conversion factors the applicant used in the dose assessments for airborne releases of radioactive material, as outlined in SAR Section 1.8.1.4.1 and BSC (2008al, Section 6.1.2). The NRC staff finds the dose conversion factors the applicant used acceptable because the dose coefficients are consistent with the definition and implementation of TEDE in 10 CFR 63.2 and 10 CFR 63.102(o).

The NRC staff also finds that it is acceptable for the applicant to use the flux-to-dose conversion factors provided in ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa) for converting neutron and gamma fluxes to dose rates instead of using those provided in ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa) for the dose calculation of the casks at the aging pad inside their concrete storage modules because these casks can be treated as a highly shielded source. As noted in the Appendix of ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa), there is no significant difference between the 1991 and 1977 version of ANSI/ANS-6.1.1 in neutron fluence-to-dose conversion for calibration of detector systems and conversion factors for exposure to gamma rays, except in the case of low-energy bremsstralung or x-ray sources. However, a significant difference between the two versions in calculated values occurs for neutron energies in the range of 10 keV to 1 MeV because the 1991 version adopted the ICRP-defined effective dose equivalent for personnel exposures. The Appendix to ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa) notes that for exposure to a highly shielded source having an average energy in the hundreds of keV range, the result obtained using the effective dose equivalent of the 1991 standard is approximately a factor of 2.7 less than the value obtained using the 1977 conversion factors. Therefore, NRC staff finds that the applicant's use of ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa) is conservative and acceptable because it results in a higher estimate of personnel exposures than would be calculated from ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa).

**Airborne Releases of Radioactive Material**

The applicant stated that airborne releases from handling individual assemblies of CSNF with cladding damage with pinhole leaks or hairline cracks in the Wet Handling Facility represented the largest airborne release source term from surface facilities during normal operations, as described in BSC (2008al, Section 6.1.2). SAR Section 1.2.1 stated that individual fuel assemblies will be transferred underwater in the pool. SAR Section 1.8.1.3.6 discussed pool LPFs (SAR Table 1.8-9) for evaluating consequences from potential fuel handling accidents in the pool. The applicant clarified in its response to the NRC staff's RAI (DOE, 2009en, Enclosure 1, Section 1) that worker doses from handling damaged fuel assemblies were included in its PCSA for normal operations. The applicant described in its RAI response in DOE (2009en, Enclosure 1, Section 1) that potential airborne radioactive material from damaged fuel assemblies would be confined and routed through the heating, ventilation, and air conditioning system. The applicant's dose assessment for workers operating the spent fuel transfer machine accounted for the presence of radioactive particulates in the pool water from damaged fuel assemblies. The applicant used industry data from operational experience to support its conclusion that releases of radioactive gases from pinhole leaks and hairline cracks from cladding would be insignificant during handling operations in the Wet Handling Facility, as outlined in DOE (2009en, Enclosure 1, Section 1). The applicant clarified in DOE (2009ek, Enclosure 4) that workers would access the pool room in the Wet Handling Facility during normal operations and indicated that the potential Category 2 event sequence for direct exposure of radiation workers from an assembly being lifted too high during transfer operations in the pool was related to a maximum lift height of approximately 3 m [10 ft] below the pool surface. Because this event sequence would not result in damage to the assembly from a drop or collision, the applicant determined in DOE (2009ek, Enclosure 4) that there would not be any additional radionuclide releases. Onsite ground contamination from estimated releases during normal operations was excluded in the applicant's consequence analysis for radiation workers and onsite members of the public, as described in BSC (2008ak, Section 3.2.8). The applicant indicated that such exclusion is justifiable because the applicant would control onsite areas and monitor them for radionuclide contamination so that remedial actions could be taken. However, in BSC (2008ay, Appendix IV), the applicant considered ground surface irradiation in the consequence analysis to constitute radiation worker exposure to an off-normal event, liquid low-level waste spill, because it represented a significant pathway in that calculation.

The applicant expects airborne releases of radionuclides during normal operations in the Wet Handling Facility (SAR Section 1.8.2.2.1). The applicant performed these atmospheric release and dispersion calculations for several outdoor locations (SAR Tables 1.8-13 and 1.8-14). Among the sources of airborne releases during normal operations, SAR Table 1.8-32 indicated that radiation workers located at the Wet Handling Facility would receive the greatest doses. SAR Section 1.8.2.2.1 described the applicant's calculation of airborne releases in the Wet Handling Facility assuming 1 percent cladding damage to individual fuel assemblies. HEPA filters mitigate normal operation releases from the Wet Handling Facility, as described in BSC (2007al, Section 6.2.1). With regard to onsite workers located outside, the applicant stated that this information provides a basis for consequence analyses from airborne radioactive material that originates inside the Wet Handling Facility and is subsequently released to the atmosphere during normal operations. The applicant indicated that gaseous releases from cooling or flushing a transportation cask or dual-purpose canister would be routed through HEPA filters before being discharged to the atmosphere. The applicant stated in its response to the NRC staff's RAI in DOE (2009em, Enclosure 1) that no unfiltered releases would occur directly into interior rooms of the Wet Handling Facility or to the atmosphere during normal operations.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on airborne releases of radioactive material using the guidance in YMRP Section 2.1.1.5. The NRC staff finds that the applicant's accounting of radioactive particulates in the pool water from damaged fuel assemblies is acceptable because radionuclide concentrations in the pool water will be controlled by the pool water treatment subsystem as described in SAR Section 1.2.5.3.2. The NRC staff also finds the applicant's characterization of radioactive gas releases from pinhole leaks and hairline cracks to be acceptable. The NRC staff performed an independent scoping calculation of atmospheric releases of radioactive material from CSNF handling and determined that the exposure from ground surface contamination provided a negligible contribution compared to other exposure pathways (Benke and Waters, 2006aa). The NRC staff finds that the applicant's approach to account for airborne releases of radioactive material is acceptable because it accounts for the significant radiological pathways and accounts for radiation worker exposure both inside and outside of operational facilities. The NRC staff finds the applicant's classification of off-normal events is consistent with HLWRS–ISG–03 (NRC, 2007ac, Footnote 1) and finds that these off-normal event dose contributions do not represent significant elevations in worker exposure compared to normal operations.

**Aggregation of Worker Doses**

In SAR Section 1.8.4.2 and Table 1.8-25, the applicant aggregated the estimated dose contributions for the four major sources discussed previously to individual radiation workers: (i) direct radiation from normal operations within the facility, (ii) direct radiation from sealed sources located outside the facility, (iii) airborne releases of radioactive material from normal operations at surface and subsurface facilities, and (iv) Category 1 event sequences. The applicant calculated doses for each contribution by assuming worker exposure for 2,000 hours/year (100 percent occupancy). Because the applicant's PCSA identified no Category 1 event sequences, contributions for Category 1 event sequences are zero

(SAR Table 1.8-25). The applicant also assessed worker doses from off-normal events in BSC (2008ay, Appendices IV and V) and determined that these events did not provide significant contributions to the worker TEDE (SAR Section 1.8.4.2). Nevertheless, these off-normal doses were still factored into the applicant's dose aggregation.

**NRC Staff's Evaluation**

The NRC staff reviewed the information on aggregation of worker doses using the guidance in YMRP Section 2.1.1.5. The NRC staff finds that the applicant's dose aggregation approach is consistent with the applicant's methodology for estimating doses to radiation workers and acceptable because it accounts for the sources of radiological exposure and does not underestimate the annual dose to an individual worker. The NRC staff concludes that radiation worker exposure during waste handling and transfer operations, including the control and limitation of exposure duration, were adequately considered in the applicant's consequence analyses because the applicant stated that it will use remote operations to limit the time spent by workers in elevated radiation fields. On the basis of these considerations, the NRC staff finds that the applicant's technical bases and input data for worker dose analyses are acceptable.

**NRC Staff's Conclusion**

The NRC staff concludes that the applicant provided sufficient information on worker dose assessment to satisfy 10 CFR 63.111(a)(1) and (b)(1). In particular, the NRC staff concludes that (i) performance objectives for the GROA before permanent closure have been met with reasonable assurance that the dose limits for radiation workers in 10 CFR Part 20 will not be exceeded, in compliance with 10 CFR 63.111(a)(1); and (ii) the aggregate radiation exposure to workers, aggregate radiation levels accessed by workers, and aggregate releases of radioactive materials to workers for normal operations and Category 1 event sequences in the GROA will be maintained, in compliance with 10 CFR 63.111(b)(1) requirements.

#### 2.1.1.5.3.5 Dose Compliance

The applicant discussed potential public and worker dose consequences and compliance confirmation (SAR Section 1.8.6).

The NRC staff organized its evaluation of the applicant's dose compliance information in the following five sections.

1. Interactions between hazard, event sequence, and consequence analyses
2. Facility throughput
3. Aggregation of annual doses
4. Compliance with dose limits for normal operations and Category 1 event sequences
5. Compliance with dose limits for Category 2 event sequences

**Interactions among Hazard, Event Sequence, and Consequence Analyses**

According to the applicant, because there are no Category 1 event sequences to aggregate, the applicant evaluated only the doses from normal operations for compliance with the preclosure performance objectives. SAR Section 1.8.6 summarized the applicant's analysis of potential public and worker dose consequences for normal operations and Category 2 event sequences. In SAR Table 1.8-36, the applicant listed the results of the public and worker dose consequences in comparison to the preclosure dose performance objectives of 10 CFR 63.111(b)(1) for offsite public exposure, onsite public exposure, and radiation worker exposure. SAR Tables 1.7-7 to 1.7-18 listed event sequences at various facilities for which, according to the applicant, consequence analyses were either performed by the applicant or not needed. The applicant also considered internal events that were not propagated into event sequences (SAR Table 1.7-1).

**NRC Staff's Evaluation**

The NRC staff reviewed the dose compliance information using the guidance in YMRP Section 2.1.1.5. Because the applicant provided analysis of the interaction between hazards, Category 2 event sequences, and calculated the resulting consequences, the NRC staff finds that the applicant appropriately treated interactions of the hazard on credited structures, systems, and components in the set of Category 2 consequence analyses, as detailed in BSC (2008ay, Table 49). The NRC staff also finds that SAR Table 1.7-13 did not associate event sequences with two Category 2 event sequences, numbered 2-02 and 2-12. Because each of these event sequences are bounded by other Category 2 event sequences, 2-03 and 2-11, respectively, the NRC staff finds that the set of consequence analyses is acceptable with respect to the results presented for the applicant's event sequence analysis. Detailed NRC staff technical reviews of the applicant's consequence analyses for members of the public and radiation workers are documented in SER Sections 2.1.1.5.3.3 and 2.1.1.5.3.4, where the NRC staff concluded that the applicant's consequence analyses for members of the public and radiation workers are acceptable.

In the detailed NRC staff technical reviews of the applicant's consequence analyses for members of the public and radiation workers documented in SER Sections 2.1.1.5.3.3 and 2.1.1.5.3.4, the NRC staff reviewed the credit taken for particulate filtration in the applicant's consequence analysis of three Category 2 event sequences that involved structural challenges to a transportation cask with uncanistered SNF, a dual-purpose canister, or a TAD canister. The NRC staff finds that mitigation of atmospheric releases by HEPA filtration is appropriate for these three Category 2 event sequences because nonseismic, internal structural challenges associated with the handling and transfer of the various containers would not affect the filtration of air in the facility prior to its atmospheric release.

**Facility Throughput**

SAR Section 1.8.5.1.1 stated average and maximum annual rates of receipt of $3 \times 10^6$ and $3.6 \times 10^6$ kg [3,000 and 3,600 MTHM] per year, respectively, at the GROA. As stated in BSC (2008al, Assumption 3.1.1 and Table 3), the nominal worker doses, used for comparison to the regulatory limits, were based on expected nominal facility throughput of $3 \times 10^6$ kg [3,000 MTHM] (500 casks) of CSNF annually. The applicant clarified in its response to the NRC staff's RAI in DOE (2009el, Enclosure 3, Section 1) that this throughput of $3 \times 10^6$ kg [3,000 MTHM] (500 casks) of CSNF reflects the repository maximum capacity and rate of receipt.

**NRC Staff's Evaluation**

The NRC staff reviewed the throughput assumptions the applicant used for calculating consequences to radiation workers including the applicant's response to NRC staff's RAI using the guidance in YMRP Section 2.1.1.5 and finds that use of the maximum rate of receipt in the applicant atmospheric release calculations is acceptable because the maximum annual rate of receipt provides a bounding nominal worker dose calculation. The NRC staff also finds the applicant's throughput assumptions for calculating doses to radiation workers is acceptable because the sum of maximum annual throughputs for individual facilities (1,055 casks) was shown to exceed the repository maximum annual rate of receipt stated in SAR 1.2.1.1.2 and outlined in DOE (2009el, Enclosure 3, Section 1).

**Aggregation of Annual Doses**

The applicant aggregated doses for members of the public and radiation workers (SAR Section 1.8.1.2) by including four major contributions discussed in SER Section 2.1.1.5.3.4. Because the applicant did not identify any Category 1 event sequences (SAR Section 1.8.6), contributions from Category 1 event sequences to aggregated doses are zero. In addition to doses for radiation workers, the applicant calculated doses from normal operations for different representative members of the public, including individuals such as onsite construction workers, other onsite persons, offsite persons in the general environment, and offsite persons not in the general environment. Compared to offsite persons, SAR Table 1.8-36 showed higher TEDE estimates for onsite persons. For onsite members of the public, direct radiation doses provided the greatest contribution to aggregated annual doses for normal operation (SAR Table 1.8-28). These direct radiation doses could be received at onsite locations outside of the main operational facilities for waste handling and aging. The applicant calculated doses at these onsite locations by assuming exposure duration of 2,000 hours/year for direct radiation (SAR Table 1.8-28) and approximately 2,000 hours/year for exposure to airborne radioactive material releases (SAR Tables 1.8-16 and 1.8-19). For the greater source-to-receptor distances at offsite locations, contributions from direct radiation emitted from sealed containers at the GROA become negligible. In light of no Category 1 event sequences, according to the applicant, aggregation for offsite public doses can be reduced to contributions from airborne releases of radioactive material from normal operations at surface and subsurface facilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's methodology to aggregate annual doses for normal operations and doses from Category 1 event sequences using the guidance in YMRP Section 2.1.1.5. Because onsite locations are nonresidential, the NRC staff finds the exposure times the applicant used for onsite members of the public conservative. The NRC staff also finds the aggregation approach that the applicant used for onsite members of the public acceptable because it is consistent with the summation of doses aggregation approach commonly used in the nuclear industry. SER Section 2.1.1.5.3.4 provides the NRC staff evaluation of the dose aggregation approach for radiation workers, which was found to be acceptable because it accounts for the main sources of radiological exposure and does not underestimate the annual dose to an individual worker.

The NRC staff compared the aggregated offsite doses in SAR Table 1.8-36 to those doses in the supporting documentation in BSC (2008ay, Tables 43–46), where the applicant showed that the aggregated offsite TEDE results equaled the sum of the highest TEDE estimates for airborne releases from the Wet Handling Facility, Aging Facility, and subsurface facility. The NRC staff finds this offsite public aggregation acceptable because it accounts for the sources of radiological dose and does not underestimate annual doses to offsite persons during normal operations.

**Compliance with Dose Limits for Normal Operations and Category 1 Event Sequences**

The applicant selected the largest aggregated annual dose for demonstrating compliance with the dose limits (SAR Table 1.8-36). Because no Category 1 event sequences were identified in the applicant's PCSA, there were no dose contributions from Category 1 event sequences. Onsite public exposures during normal operations are attributed to direct radiation and skyshine. The applicant assumed an annual exposure duration of 2,000 hours for an onsite member of the public. The applicant determined that the contribution from airborne releases was insignificant. For workers who are members of the public having access to the restricted area, like construction workers, the applicant estimated a maximum dose of 0.098 mSv/year [9.8 mrem/year]. For public located outside the restricted area but still onsite, results were presented in SAR Table 1.8-28. The highest dose rates (SAR Table 1.8-36) were near the truck and rail buffer areas at the lower muck yard {0.78 mSv/year [78 mrem/year]} and the nonnuclear Receipt Facility {0.76 mSv/year [76 mrem/year]}.

For the offsite public not in the general environment where access is controlled, the applicant determined dose rates to be 0.0011 mSv/year [0.11 mrem/year] TEDE. For the offsite public in the general environment, the applicant determined that, due to the long distance to the closest location in the general environment, direct radiation and skyshine contributions to dose were negligible. The applicant discussed airborne exposures that result from normal operations in SAR Section 1.8.3.1.2. The applicant indicated that the estimated TEDE was a factor of 900 below the applicable regulatory limit in 10 CFR 63.111(a). Internal doses were calculated on the basis of a 50-year dose commitment period. SAR Table 1.8-29 provided estimates of dose for members of the public in the general environment {0.0005 mSv/year [0.05 mrem/year] TEDE}. SAR Table 1.8-36 showed that the estimated TEDE was a factor of 300 below the regulatory dose limit in 10 CFR 63.111(a). SER Section 2.1.1.5.3.3 describes the detailed technical review of the applicant's public dose assessments.

In SAR Table 1.8-36, the applicant presented consequence analysis results for radiation workers of 0.013 Sv/year [1.3 rem/year] TEDE. The applicant reported a maximum TEDE of 0.014 Sv/year [1.4 rem/year] and maximum shallow dose equivalent to the skin of 0.001 Sv/year [0.1 rem/year] for radiation workers in DOE (2009el, Enclosure 2, Table 4). The applicant committed to update the license application (SAR Tables 1.8-25 and 1.8-36) with changes to the results for radiation workers, which is presented in its response to the NRC staff's RAI (DOE, 2009el, Enclosures 1 and 2, Sections 2 and 3). The applicant reported a maximum lens dose equivalent of 0.013 Sv/year [1.3 rem/year] in DOE (2009el, Enclosure 2, Table 4).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's demonstration of compliance with the preclosure performance objectives and regulatory dose limits for workers and members of the public from normal operations and Category 1 event sequences using the guidance in YMRP Section 2.1.1.5. In its evaluation, the NRC staff considered onsite members of the public, radiation workers, offsite members of the public within the general environment, and offsite members of the public not within the general environment public. For onsite and offsite members of the public, the NRC staff finds that the applicant has demonstrated compliance because calculated doses for normal operations and Category 1 event sequences are below regulatory limits in 10 CFR 63.111(a).

The NRC staff reviewed the applicant's approach for calculating TEDE, including the approach for calculating the lens dose equivalent, the dose equivalent to the maximally exposed organ and the shallow dose equivalents to the skin. The applicant's approach for calculating the lens dose equivalent, as described in SAR Equation 1.8-11, is acceptable because it is consistent with NRC regulatory guidance (see SER Section 2.1.1.5.3.1 for additional information). However, for a given calculated result of TEDE, the NRC staff finds that the applicant's approach in SAR Equation 1.8-7 and in its response to the NRC staff's RAI in DOE (2009el, Enclosure 2, Table 4, Notes on Formula for Columns B and C) could underestimate the dose equivalent to the maximally exposed organ and shallow dose equivalent to the skin. Because organ dose equivalents can exceed the effective dose equivalent due to external irradiation, the NRC staff conducted an independent confirmatory calculation evaluating the potential uncertainty associated with the applicant's dose equivalent calculations for the maximally exposed organ and skin from both gamma rays and neutrons, primary components of the direct radiation source term. This independent confirmatory calculation, which is described in the following three paragraphs, confirms that the applicant's organ and skin dose calculations are well below regulatory limits.

For external irradiations by gamma rays, the organ dose equivalent to the bone surface typically exceeds the effective dose equivalent, as detailed in EPA (1993aa, Table III.3) and in International Commission on Radiological Protection (1996aa, Tables A.2 to A.20). The NRC staff compared the maximum organ dose equivalent to the effective dose equivalent for irradiation by gamma rays of different energies and geometries to quantify uncertainties for the maximally exposed organ dose equivalent. The NRC staff identified specific radionuclides—Co-57, Ba-137m (Cs-137), Co-60, and Na-24—as proxy sources emitting gamma rays with lower to higher energies, respectively. The NRC staff investigated data from air submersion and ground surface contamination exposure geometries in EPA (1993aa, Tables III.1 and III.3) for these proxy sources to approximate the maximum uncertainty in organ dose equivalents from direct exposure. The NRC staff finds that the maximally exposed organ dose equivalent exceeded the effective dose equivalent by less than a factor of three. An upper-bound uncertainty of a factor of three is also supported by information on effective dose by the International Commission on Radiological Protection (1996aa, Tables A.2 to A.20) that reports external dose coefficients for irradiation geometries that are representative of the direct exposure pathway.

For external irradiation by neutrons, doses to the bone surface and skin are commonly less than the dose to other organs, as outlined in International Commission on Radiological Protection (1996aa, Tables A.26 to A.20). Using the energy-dependent radiation weighting factors for neutrons from International Commission on Radiological Protection (1996aa, Table 2), the NRC staff also compared the maximum organ doses to effective doses at several neutron energies between 0.001 eV and 10 MeV. The NRC staff finds that the maximum organ doses exceeded the effective dose by less than a factor of two for neutron irradiation, which is bounded by the factor-of-three uncertainty previously determined for irradiation by gamma rays. For external human exposure to gamma rays and neutrons emitted over a broad range of energies from contained and shielded sources at the operational facilities (SAR Section 1.10.3.4), the NRC staff concludes that the dose equivalent to the maximally exposed organ would not exceed the effective dose equivalent by more than a factor of three.

Combining the maximum TEDE to a radiation worker of 0.013 Sv [1.3 rem] in the Receipt Facility (SAR Table 1.8-36) with the bounding uncertainty for a maximally exposed organ of a factor of three for direct irradiation, the NRC staff determines that the maximum organ dose equivalent would not exceed 0.04 Sv [4 rem], which is more than an order of magnitude less than the 0.5 Sv [50 rem] limit to an individual organ (other than the lens of the eye) and skin in 10 CFR 20.1201. Therefore, based on the evaluation and independent confirmatory calculations described previously, the NRC staff finds that doses to radiation workers for normal operations and Category 1 event sequences are well below the 10 CFR part 20 limits; and therefore, the applicant complies with 10 CFR 63.111(a). The applicant stated that it will update SAR Tables 1.8-25 and 1.8-36 with changes to the results for radiation workers, which is presented in its response to the NRC staff's RAI (DOE, 2009el, Enclosures 1 and 2, Sections 2 and 3). The NRC staff notes that its evaluation of the applicant's radiation protection program, including procedures and controls to assure onsite activities are conducted in a safe manner, would be one element of an NRC review of an application to receive and possess waste.

**Compliance with Dose Limits for Category 2 Event Sequences**

The applicant performed public consequence analyses for Category 2 event sequences to demonstrate compliance with the preclosure requirements specified in 10 CFR 63.111(b)(2). As required in10 CFR 63.111(b)(2), the GROA must be designed so that no individual located on, or beyond, any point on the boundary of the site should receive, as a result of the single Category 2 event sequence, the more limiting of the following: (i) a TEDE of 0.05 Sv [5 rem] or (ii) the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent may not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin may not exceed 0.5 Sv [50 rem]. For external exposure, consistent with the revised definition of TEDE in 10 CFR Part 20 (72 FR 68043), it is permissible to use the effective dose equivalent in place of the deep dose equivalent.

Because of the distances involved, the applicant determined that direct radiation from a Category 2 event sequence was negligible. Fourteen bounding Category 2 event sequences were analyzed (SAR Tables 1.8-30 and 1.8-31) for airborne releases. The applicant calculated the highest TEDE for the offsite public in the general environment to be 0.0001 Sv [0.01 rem] and demonstrated that organ, lens of the eye, and skin doses were all well below limits. For offsite exposure of the public not in the general environment, the highest TEDE was 0.0003 Sv [0.03 rem] for the organ dose; the applicant also demonstrated that lens of the eye and skin doses were all well below limits. To demonstrate compliance with 10 CFR 63.111(b)(2), the applicant selected the highest calculated doses for TEDE, individual organ, lens of the eye, and skin from the set of Category 2 consequence analyses, as described in BSC (2008ay, Tables 67 and 68).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's demonstration of compliance for members of the public from a single Category 2 event sequence and concludes that the Category 2 consequence analyses do not underestimate dose using the guidance in YMRP Section 2.1.1.5. The NRC staff finds that the applicant demonstrated compliance with 10 CFR 63.111(b)(2) because SAR Table 1.8-36 showed that estimated consequences to offsite members of the public from a single Category 2 event sequence are more than 70 times below regulatory limits.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant demonstrated compliance with 10 CFR 63.111(a), 10 CFR 63.111(b), and 10 CFR 63.111(c) because the doses the applicant estimated are below regulatory limits. More specifically, the NRC staff finds that the consequence analyses for radiation workers are in compliance with 10 CFR 63.111(a)(1)and (b)(1), and 10 CFR 20.1201(a). The NRC staff also finds that the applicant is in compliance with 10 CFR 63.111(a) and 10 CFR 63.111(b) for members of the public. The NRC staff finds that the applicant satisfies 10 CFR 63.204 for offsite members of the public located in the general environment. Facility throughput assumptions the applicant used in calculating radiological consequences were supported and, therefore, are acceptable. The NRC staff also concludes that the applicant has demonstrated compliance with dose limits in 10 CFR 20.1101(d) and 10 CFR 20.1301 for members of the public.

#### 2.1.1.5.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application and finds, with reasonable assurance, that

- The requirements of 10 CFR 63.111(a)(1) are satisfied. Performance objectives for the geologic repository operations area, up to the time of permanent closure, have been met in that the radiation exposure limits in 10 CFR Part 20 will not be exceeded.

- The requirements of 10 CFR 63.111(a)(2) are satisfied. Performance objectives for the geologic repository operations area up to the time of permanent closure have been met in that, during normal operations and for Category 1 event sequences, the annual dose to any real member of the public, located beyond the boundary of the site, will not exceed 0.15 mSv [15 mrem] in 10 CFR 63.204.

- The requirements of 10 CFR 63.111(b)(1) are satisfied. The geologic repository operations area has been designed such that, taking into consideration normal operation and Category 1 event sequences, radiation exposures, radiation levels, and releases of radioactive materials will be maintained, within the limits of 10 CFR 63.111(a).

- The requirements of 10 CFR 63.111(c)(1) are satisfied. The preclosure safety analysis performed in accordance with 10 CFR 63.112, demonstrates that the radiation protection limits of 10 CFR Part 20 will be met. During normal operations and Category 1 event sequences, the annual dose to any real member of the public, located beyond the boundary of the site, will not exceed 0.15 mSv [15 mrem].

- The requirements of 10 CFR 63.111(c)(2) are satisfied. The preclosure safety analysis performed in accordance with 10 CFR 63.112 demonstrates that the preclosure numerical radiation protection requirements will be met for geologic repository operations area normal operations and Category 1 event sequences.

- The requirements of 10 CFR 63.111(b)(2) are satisfied. The design of the geologic repository operations area is such that, taking into consideration Category 2 event sequences, no individual located on, or beyond, any point on the boundary of the site will receive, as a result of the single Category 2 event sequence, the more limiting of a total effective dose equivalent of 0.05 Sv [5 rem] or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent will not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin will not exceed 0.5 Sv [50 rem].

- The requirements of 10 CFR 63.111(c) are satisfied. The preclosure safety analysis performed in accordance with 10 CFR 63.112 demonstrates that the numerical guides for design objectives for Category 2 events in 10 CFR 63.111(b)(2) are met.

In summary, on the basis of review of information in the SAR, the applicant's response to the NRC staff RAIs and supporting documents, the NRC staff concludes, with reasonable assurance, that the information on dose consequences for the GROA for construction authorization is adequate and satisfies 10 CFR 63.111(a), (b), and (c); 10 CFR 63.204; 10 CFR 20.1101(d); 10 CFR 20.1201(a); and 10 CFR 20.1301.

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# CHAPTER 6

## 2.1.1.6 Identification of Structures, Systems, and Components Important to Safety, Safety Controls, and Measures to Ensure Availability of the Safety Systems

### 2.1.1.6.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.6 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's ("DOE" or "applicant") identification of important to safety (ITS) structures, systems, and components (SSCs), safety controls, and measures to ensure availability and reliability of the safety systems. SSCs, identified as ITS, are relied upon to provide reasonable assurance that high-level radioactive waste (HLW) can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the dose limits specified at 10 CFR 63.111(b)(1) for Category 1 event sequences or to prevent or mitigate the Category 2 event sequences that could result in radiological exposures exceeding the values specified at 10 CFR 63.111(b)(2). Category 1 event sequences are those expected to occur one or more times before permanent closure of the geological repository operations area (GROA). Category 2 event sequences are other event sequences having at least 1 chance in 10,000 of occurring before permanent closure of the GROA. The NRC staff evaluated the information provided in the Safety Analysis Report (SAR) Section 1.9 (DOE, 2008ab), supporting documents, and the applicant's responses to the NRC staff's requests for additional information (RAIs). In this chapter, the NRC staff documented its risk-informed review of ITS SSCs using the results from its review of hazards and event sequences documented in SER Chapters 2.1.1.3 and 2.1.1.4.

To identify SSCs as ITS, the applicant performed a preclosure safety analysis (PCSA). The PCSA represents a systematic analysis of the safety of the preclosure design and operations of the repository. The PCSA includes consideration and evaluation of safety design, operational procedures, initiating events (both natural and human induced), fault tree analyses to determine system reliability, and accident progression and mitigation strategies. The NRC staff evaluated the analyses and considerations conducted as part of the PCSA to determine whether DOE properly identified ITS SSCs. The applicant provided the criteria used in ITS SSC identification and listed the identified ITS SSCs and associated nuclear safety design bases in SAR Section 1.9.1 and Tables 1.9-1 through 1.9-7. The applicant addressed regulatory requirements in 10 CFR 63.112(e)(1) through 10 CFR 63.112(e)(13) in SAR Sections 1.9.1.1 through 1.9.1.13. SAR Section 1.9.1.14 summarized the applicant's consideration of interactions between ITS SSCs and not important to safety (non-ITS) SSCs. SAR Sections 1.9.3 and 1.9.4 described procedural safety controls (PSCs) and ITS SSC risk significance categorization. The applicant's ITS SSCs include (i) surface facilities; (ii) mechanical systems, including heating, ventilation, and air-conditioning (HVAC) systems; (iii) transportation systems; (iv) electrical components, normal and emergency power systems; (v) fire protection systems; (vi) transportation, aging, and disposal (TAD) canisters; and (vii) waste packages and overpacks.

### 2.1.1.6.2 Regulatory Requirements

As stated in 10 CFR 63.112(e), the applicant is required to conduct a preclosure safety analysis that includes an analysis of the performance of the SSCs to (i) identify those SSCs that are important to safety, (ii) identify and describe the controls relied on to limit or prevent potential event sequences or mitigate their consequences, and (iii) identify measures taken to ensure the availability of safety systems. This analysis must include, but is not necessarily limited to, consideration of the following:

(1) Means to limit concentration of radioactive material in air

(2) Means to limit the time required to perform work in the vicinity of radioactive materials

(3) Suitable shielding

(4) Means to monitor and control the dispersal of radioactive contamination

(5) Means to control access to high radiation areas or airborne radioactivity areas

(6) Means to prevent and control criticality

(7) Radiation alarm system to warn of significant increases of radiation levels, concentrations of radioactive material in air, and increased radioactivity in effluents

(8) Ability of structures, systems, and components to perform their intended safety functions, assuming the occurrence of event sequences

(9) Explosion and fire detection systems and appropriate suppression systems

(10) Means to control radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency

(11) Means to provide reliable and timely emergency power to instruments, utility services systems, and operating systems important to safety if there is a loss of primary electric power

(12) Means to provide redundant systems necessary to maintain, with adequate capacity, the ability of utility services important to safety

(13) Means to inspect, test, and maintain structures, systems, and components important to safety, as necessary, to ensure their continued functioning and readiness

Using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.6 (NRC, 2003aa), the NRC staff reviewed the applicant's (i) analyses supporting classification of structures, systems, and components as important to safety; (ii) safety controls; and (iii) measures to ensure availability of the safety systems. In particular, the focus of the review was to verify that the applicant appropriately used the results of the preclosure safety analysis, reviewed in SER Sections 2.1.1.3, 2.1.1.4, and 2.1.1.5, to identify the ITS SSCs, safety controls, and measures to ensure the availability of safety systems. As part of the design process, the designation of individual components as ITS and the design of subsystems could be different in the final design. Volume 5 of this SER includes the NRC staff's evaluation of DOE's proposed change process.

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA (e.g., handling of spent nuclear fuel, criticality controls during storage of spent nuclear fuel, shield doors and interlocks for worker safety from direct radiation of spent nuclear fuel). The applicability of such NRC guidance is discussed in greater detail in the sections where they were used as part of the application or the NRC staff's review.

### 2.1.1.6.3 Technical Review

The NRC staff's technical review is provided in three subsections: (1) Section 2.1.1.6.3.1 documents the NRC staff's review of the applicant's identification of ITS SSCs; (2) Section 2.1.1.6.3.2 documents the NRC staff's review of the applicant's analysis of the items in 10 CFR 63.112(e); and (3) Section 2.1.1.6.3.3 documents the NRC staff's review of the applicant's administrative or procedural safety controls in SAR Section 1.9.3. The PSCs are relied upon by DOE to reduce the likelihood of an initiating event or an event sequence or to mitigate the consequences of an event sequence.

The NRC staff's review of the applicant's identification of measures to ensure availability of safety systems is documented in SER Section 2.1.1.6.3.2, which addresses the requirements in 10 CFR 63.112(e)(1) through 10 CFR 63.112(e)(13).

The NRC staff's review of the measures the applicant would use for explosion protection [10 CFR 63.112(e)(9)] is provided in SER Sections 2.1.1.7.3.3, 2.1.1.7.3.5.2, 2.1.1.7.3.5.3, and 2.1.1.7.3.5.4 for hydrogen buildup in the battery rooms of the ITS HVAC systems, fuel tanks for site transporters, cask tractor and cask transfer trailers, and site prime movers, respectively.

#### 2.1.1.6.3.1 List of Important to Safety Structures, Systems, and Components and Nuclear Safety Design Bases

DOE presented the list of ITS SSCs and associated nuclear safety design bases in SAR Section 1.9.1 and Tables 1.9-1 through 1.9-7. The NRC staff's review of this information is presented in this SER Section.

**ITS SSC Identification**

The applicant performed a preclosure safety analysis (PCSA) that was used to (i) assess the potential natural and operational hazards for the preclosure period; (ii) assess potential initiating events and event sequences and their consequences; and (iii) identify the SSCs and procedural safety controls intended to prevent or reduce the probability of an event sequence or mitigate the consequences of an event sequence, should it occur. The PCSA was conducted using site-specific information (external hazards, including both natural and human-induced) and facility-specific operational processes.

The applicant developed and used four criteria, as listed in SAR Section 1.9.1, to identify SSCs that are ITS, to ensure that the limits set forth in 10 CFR 63.111(b)(1) and 10 CFR 63.111(b)(2) would not be exceeded. The applicant classified an SSC as ITS if it appears in an event sequence and at least one of the following criteria apply: (i) the SSC is relied upon to reduce the frequency of an event sequence from Category 1 to Category 2, (ii) the SSC is relied upon to reduce the frequency of an event sequence from Category 2 to beyond Category 2, (iii) the SSC is relied upon to reduce the aggregated dose of Category 1 event sequences by reducing the event sequence mean frequency, or (iv) the SSC is relied upon to perform a dose mitigation or criticality control function. The ITS SSCs identified by the applicant were listed in SAR Table 1.9-1.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.6 to review the applicant's identification of ITS SSCs. The NRC staff reviewed the applicant's safety classification of the SSCs listed in SAR Table 1.9-1 to evaluate whether the applicant's designation of the SSCs as ITS is based on the applicant's four criteria for identifying ITS SSC. The NRC staff determines that all SSCs the applicant identified as ITS in SAR Table 1.9-1 were relied upon to meet one or more of the applicant's four ITS SSC criteria. For example, the applicant classified the actual building structure as ITS for the canister receipt and closure facility (CRCF), initial handling facility (IHF), and wet handling facility (WHF) because the building structures of these facilities would be relied on to protect ITS SSCs that are located within these facilities {e.g., Waste Packages (WPs), WP Handling Crane}, or from external event hazards such as winds and tornadoes (SAR Tables 1.9-3, 1.9-2, and 1.9-4). In particular, the building structures associated with the CRCF, IHF, and WHF facilities will be relied on to reduce the frequency of an event sequence to a beyond Category 2 event sequence. The applicant classified canisters and casks as ITS based on a safety function to provide containment of radioactive materials (e.g., TAD canisters, HLW canisters, transportation casks and DOE standardized canisters). The applicant also classified aging overpacks and casks (e.g., transportation casks and shielded transfer casks) as ITS based on a safety function to protect against direct exposure to workers. The NRC staff finds that the applicant's identification of ITS SSCs in Table 1.9-1 is adequate because (i) the identified ITS SSCs are relied on to meet one or more of the applicant's four criteria for identifying SSCs as ITS and (ii) the SSCs identified as ITS are consistent with the NRC staff's review of the applicant's identification of hazards and initiating events (SER Section 2.1.1.3), identification of event sequences (SER Section 2.1.1.4), and consequences of event sequences (SER Section 2.1.1.5).

To develop further understanding of the design of and safety function of SSCs that were identified as ITS by DOE, the NRC staff also reviewed specific details of the design for the Initial Handling Facility (SAR Table 1.9-2), Canister Receipt and Closure Facility (SAR Table 1.9-3), Wet Handling Facility (SAR Table 1.9-4), Receipt Facility (SAR Table 1.9-5), Intrasite operations (SAR Table 1.9-6), and subsurface operations (SAR Table 1.9-7). Based on the NRC staff review of event sequences documented in SER Chapters 2.1.1.3 and 2.1.1.4, the NRC staff selected specific aspects of the design for review based on its significance in preventing or mitigating the consequences from an event sequence. Specifically, the applicant provided information on ITS components in the process and instrumentation diagrams (P&IDs) in SAR Sections 1.2 and 1.4 and in the applicant's response to the NRC staff's RAI (DOE, 2009dq). The NRC staff reviewed the P&ID, the components (the SSCs identified as ITS by DOE are made up of a number of components and subsystems, some of which are ITS and others that are non-ITS) on the diagram designated ITS, and the tables/notes on the diagram for these ITS components to identify the association between an ITS component and its safety function. The NRC staff evaluated this information to determine whether the function of the individual components that were designated ITS on the P&ID diagram are consistent with the overall safety function for the system they support. For example, the NRC staff determined that the applicant's ITS-identified components for a canister transfer machine (CTM) grapple, such as the grapple actuator and related switches (SAR Figure 1.2.4-64), are those components associated with the safety function of preventing a grapple from dropping a load (i.e., preventing an event sequence).

**Nuclear Safety Design Bases**

The applicant developed nuclear safety design bases for the important to safety structures, systems, and components (ITS SSCs) from the PCSA event sequence analyses. The applicant's design bases are provided for 34 systems, 67 subsystems, and 161 components (SAR Tables 1.9-2 through 1.9-7). The design bases included the required safety functions of the ITS SSCs and the associated controlling parameters and values. As described in SAR Tables 1.9-2 through 1.9-7, these nuclear safety design bases would ensure that the ITS SSCs would be reliable and available to perform their intended safety functions. The applicant also provided safety functions for ITS controls, in response to an NRC staff RAI (DOE, 2009dk).

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.6 to review the applicant's nuclear safety design bases for ensuring that the ITS SSCs will be able to perform their intended safety functions. The NRC staff reviewed SAR Tables 1.9-2 through 1.9-7, which provide design bases for facilities involved with SNF handling operations (i.e., Initial Handling Facility–SAR Table 1.9-2, Canister Receipt and Closure Facility–SAR Table 1.9-3, Wet Handling Facility–SAR Table 1.9-4, and Receipt Facility–SAR Table 1.9-5), Intrasite operations (e.g., operations involving aging overpacks, shielded transfer casks, and transportation casks) (SAR Table 1.9-6), and subsurface operations (SAR Table 1.9-7). The NRC staff finds that the controlling parameters and values of the design bases for the ITS SSCs are consistent with the safety functions assigned to the SSCs. For example, the safety function of the shield door to protect against direct exposure of personnel is supported by a controlling value that equipment shield doors have a mean probability of inadvertent opening of less than or equal to $1 \times 10^{-7}$ per waste container handled (SAR Table 1.9-5).

The NRC staff also performed a detailed review of the design bases for ITS SSCs and the PCSA results associated with the handling operations of casks, canisters, and SNF, based on the risk-significance and frequency for these handling operations. The evaluation included review of the SAR and several supporting documents related to reliability and event sequence categorization analyses (BSC, 2008ac,as,be), including the review of a representative event sequence for each design basis reviewed. On the basis of this review, the NRC staff identified features or passive controls that were required to perform safety functions associated with ITS SSCs listed in the Tables 1.9.2 through 1.9.7. These design features or passive controls included (i) guide sleeves to preclude non-flat-bottom drop of canisters in the CTM; (ii) divider plates in unsealed waste packages to prevent drop of heavy loads onto a DOE or HLW canister [SAR Table 1.7-1; Table 6.0-2 (BSC, 2008ac)]; (iii) the CTM slide gate and supporting structures (shield bell, shield bell trolley, and bridge) to withstand a drop of the heaviest canister amongst those that would be disposed of in the repository; and (iv) CTM interlocks to prevent workers from direct radiation exposure or shield doors from inadvertent opening due to spurious signals caused by water spray. The following provides the NRC staff's review of the design basis information for these four features and passive controls. These design features and passive controls have also been evaluated by the NRC staff in SER Sections 2.1.1.3.3.2.2.1, 2.1.1.3.3.2.5, and 2.1.1.4.3.4.1.1.

The applicant stated that it will rely on guide sleeves' features to preclude non-flat-bottom drops of the naval, DPC, and TAD canisters, when handled by the CTM (e.g., SAR Tables 1.2.3-3 and 1.2.4-4). In response to an NRC staff RAI, the applicant described the guide sleeve design and its functions (DOE, 2009dy). In assessing event sequences related to canister drop, the applicant stated that the "inside diameter of the guide sleeves will be based on the maximum outside diameter of the canister and will account for the tolerances, design, thermal expansion, straightness, seismic, and out-of-roundness (ovality) of the canister" (DOE, 2009fy). Based on its evaluation of the two RAI responses (DOE, 2009dy,fy), the NRC staff finds that the applicant's design basis information for the guide sleeves is adequate because the applicant (i) illustrates how the guide sleeves will perform their intended safety function of precluding non-flat-bottom canister drops when the canisters are handled by the CTM, (ii) provided the dimensions for the guide sleeves, and (iii) explained how the guide sleeve design will account for variations in canister diameters from such effects as thermal expansion and ovality.

The applicant stated that it will rely on the divider plates in unsealed waste packages during loading operations to prevent drop of heavy loads, such as a lid, onto a DOE or HLW canister [SAR Table 1.7-1; Table 6.0-2 (BSC, 2008ac)]. The applicant stated that the divider plates in a waste package would extend higher than the canisters inside the waste package; and thus, the divider plates protect the canisters from the potential drop of a lid (SAR Figures 1.5.2-4, 1.5.2-5). Based on its evaluation of this information, the NRC staff finds that the applicant's design of the divider plates, which would extend higher than the canisters and preclude lid contact with the canisters following a potential drop of a lid, provides an adequate basis for screening out a lid drop onto a canister as an initiating event

In response to the NRC staff's RAI regarding whether the CTM slide gate and supporting structures (shield bell, shield bell trolley, and bridge) would be able to maintain their integrity and support a canister that is dropped within the shield bell, the applicant stated that the slide gate will be designed to withstand a 30.5 cm [12 in] vertical drop of the heaviest canister and remain intact (DOE, 2009dy). In the same RAI response, the applicant provided the results of its analysis stating, "The stiffness of the canister transfer machine slide gate, when subject to the impact of a dropped canister, is equivalent to that of a 254-cm [100-in] square, 25.4-cm [10-in] thick, solid carbon-steel plate that is supported at an opening with a diameter of 223.5-cm [88 in] (DOE, 2009dy). The applicant also stated that the canister transfer machine structural and shielding components, including the slide gate and supporting structures (e.g., shield bell, shield bell trolley, bridge), will be designed to withstand the drop impact and remain intact. The applicant further stated that ITS mechanical handling equipment (including CTM) would be designed for applicable loads to prevent or mitigate Category 1 and Category 2 event sequences (DOE, 2009dy; SAR Section 1.2.2.2.9). The NRC staff's evaluation of the design of the mechanical handling transfer systems (including CTM) is documented in SER Section 2.1.1.7.3.2. Based on the evaluation of the information provided by the applicant in SAR Section 1.2.2.2.9 and the response to the NRC staff's RAI (DOE, 2009dy) on the design of the mechanical handling transfer systems, the NRC staff finds that the applicant provided an adequate nuclear safety design basis for the CTM slide gate and supporting structures to withstand the drop impact of the heaviest canister and remain intact because the applicant provided (i) the drop height and load considered for the design of the slide gate to withstand the impact of a dropped canister and (ii) the design loads for the CTM, including the supporting structures (e.g., shield bell, shield bell trolley, and bridge).

In response to the NRC staff's RAI regarding susceptibility of the CTM interlocks to failure due to internal flooding and water impingement, the applicant stated that CTM interlocks will be designed and qualified for the environmental conditions (DOE, 2009fn). The applicant further stated in the same RAI response that (i) spray guards and waterproof encasements would be used to protect the ITS interlocks on an as-needed basis and (ii) the environmental conditions will include water spray (e.g., due to fire suppression system sprinkler operation or other piping failures). Based on the evaluation of the information provided by the applicant in response to the NRC staff's RAI (DOE, 2009fn) on the design of the CTM interlocks, the NRC staff finds that the applicant provided adequate design basis information for the CTM interlocks, as part of the ITS CTM design, to prevent the shield doors from inadvertently opening from failure of the interlock due to environmental conditions, such as water spray, because the applicant's design includes spray guards and waterproof encasements to protect the ITS interlocks from these conditions. Further information on the NRC staff's review of the design of the instrumentation and controls, including the CTM ITS interlocks, is provided in SER Section 2.1.1.7.3.7.

On the basis of its evaluation discussed above, the NRC staff finds that the applicant provided adequate design bases for the ITS SSCs, consistent with the safety functions required to prevent or mitigate the consequences of event sequences, in the PCSA.

**NRC Staff's Conclusion**

Based on the NRC staff evaluations in SER Section 2.1.1.6.3.1, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e) are satisfied, with respect to identification of ITS SSCs and nuclear safety design bases, because the applicant identified ITS SSCs, safety controls, and the related nuclear safety design bases, using a systematic preclosure safety analysis for natural and human-induced hazards, consistent with the GROA design and operational information in the application.

### 2.1.1.6.3.2 Important to Safety Structures, Systems, and Components and Safety Controls

The NRC staff's review of the applicant's process for identifying important to safety structures, systems, and components, safety controls, and measures to ensure the availability and reliability of the safety systems is provided in this SER section. This review includes the NRC staff's evaluation of the following: (i) the applicant's means to limit radioactive material concentration in the air; (ii) the applicant's means to limit time required to perform work in the vicinity of radioactive materials; (iii) the applicant's ability to provide suitable shielding; (iv) the applicant's means to monitor and control dispersal of radioactive contamination; (v) the applicant's means to control access to high radiation or airborne radioactivity areas; (vi) the applicant's means to prevent or control criticality; (vii) a radiation alarm system designed to warn of significant increases in radiation levels, concentrations of radionuclides in air, and increased radioactivity in effluents; (viii) the ability of structures, systems, and components to perform their intended safety functions, assuming the occurrence of event sequences; (ix) fire detection systems and appropriate suppression systems; (x) the applicant's means to control radioactive waste and radioactive effluents and to permit prompt termination of operations and evacuation of personnel during an emergency; (xi) the applicant's means to provide timely and reliable emergency power to instruments, utility service systems, and operating systems important to safety; (xii) the applicant's means to provide reliable and timely redundant systems necessary to maintain, with adequate capacity, the capability of utility services important to safety; and (xiii) the applicant's means to inspect, test, and maintain ITS SSCs, as necessary, to ensure their continued function and readiness.

#### 2.1.1.6.3.2.1 Limiting Concentration of Radioactive Material in Air

For events leading to releases of radioactive material, the applicant discussed, in SAR Section 1.9.1.1, equipment and facility designs that it would use to limit the concentration of radioactive material in air. The applicant stated that it will rely on heating, ventilation, and air-conditioning (HVAC) systems as the primary means to limit airborne radioactive contamination by controlling airflow from areas of low contamination potential to areas with higher contamination potential. The applicant further stated that the HVAC systems' design is consistent with Regulatory Guide 8.8, Regulatory Position C.2.d (NRC, 1978ab), which provides guidance on maintaining airflows from areas of low potential airborne contamination to areas of higher potential contamination. The applicant stated that all surface facilities will be equipped with HVAC systems. The applicant further stated that the portions of HVAC systems in the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF) that filter air in the confinement areas (SAR Sections 1.2.4.4 and 1.2.4.5) and exhaust air from areas with a potential breach of the waste container are designated as ITS.

The applicant stated that other potential sources of release of radioactive material into air are subsurface releases from radioactive sources such as re-suspension of external surface contamination from the waste packages and neutron activation of air and dust. The applicant stated that its analyses showed that the airborne concentrations from these releases are below regulatory limits.

According to SAR Section 1.9.1.4, the radiation-monitoring and alarm systems will be used for area radiation, continuous air, and airborne radioactivity effluent monitoring. Alarms will be triggered by high radiation levels and will be located at potential release points, at the Central Control Center and on appropriate consoles in the facility operations room, to alert operators of radiological releases or extreme radiation conditions. The applicant stated that airborne radioactivity effluent monitors in designated release points in surface facilities will be used to routinely monitor sampled air. Radiation alarm systems are further discussed in SER Section 2.1.1.6.3.2.7.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.6 and Regulatory Guide 8.8 (NRC, 1978ab) to review the applicant's means to limit the concentration of radioactive material in air. The NRC staff finds that the use of HVAC systems to limit the concentration of radioactive material in air is acceptable because the design of the ITS HVAC systems is consistent with Regulatory Guide 8.8, Regulatory Position C.2.d (NRC, 1978ab), which provides guidance for maintaining airflows from areas of low potential airborne contamination to areas of higher potential contamination. The NRC staff also finds that the applicant's plan to use radiation monitoring and alarm systems to notify operators of radiological releases, by monitoring radiation at the source point and manning operating stations and other locations in the surface facilities, is acceptable because the use of radiation monitoring systems is consistent with Regulatory Guide 8.8, Regulatory Position C.2.g (NRC, 1978ab). The NRC staff further finds the applicant's ITS designation of the HVAC systems in the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF) acceptable because the applicant relied on these HVAC systems in the PCSA to mitigate radioactive releases. Lastly, the NRC staff finds that subsurface releases are not a significant risk, and that ITS HVAC systems are not required for their mitigation due to the limited surface contamination associated with the sealed waste packages. The NRC staff's evaluation of the HVAC systems proposed by the applicant to limit concentration of radioactive material in air is provided in SER Section 2.1.1.6.3.2.8.2.2.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(1) are satisfied because the applicant's preclosure safety analysis included adequate consideration of methods to limit the concentration of radioactive material in air.

2.1.1.6.3.2.2 Limiting Worker Exposure Time When Performing Work

The applicant discussed the equipment and facility designs that it would use to limit the exposure time for workers when performing work in the vicinity of radioactive materials in SAR Section 1.9.1.2. The applicant stated that there are no Category 1 event sequences that lead to exposure of individuals to radiation and limit the time required for workers to perform activities in radiation areas during normal operations that will be part of the applicant's as low as is reasonably achievable (ALARA) principles and program (NRC evaluation of event sequence categorization is documented in SER Section 2.1.1.4.3). The applicant designated these design features as non-ITS. The applicant stated that the ALARA principles would be incorporated into the design of SSCs to (i) accommodate remote and semi-remote operations and maintenance, (ii) reduce radiation and contamination levels so that operations and maintenance can be performed in lower radiation environments, and (iii) reduce the time spent in radiation environments during operations and maintenance (SAR Section 1.10.2). SER Section 2.1.1.8 provides the NRC staff's review of the applicant's ALARA program.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.1.6 to determine whether the applicant adequately considered in its PCSA the means to limit time required to perform work in the vicinity of radioactive materials. The NRC staff finds that the applicant's statement that the ALARA program is part of normal operations is acceptable because there are no design features associated with limiting worker exposure time that the applicant will rely on to prevent or mitigate a Category 1 or Category 2 event sequence. The applicant's consideration of the design of SSCs used for limiting worker exposure time when performing work in the vicinity of radioactive materials as part of normal operations is reviewed by the NRC staff under the ALARA program in SER Section 2.1.1.8.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(2) are satisfied because the applicant's PCSA adequately evaluated equipment and facility designs to limit the time required to perform work in the vicinity of radioactive materials that would be relied upon to prevent or mitigate a Category 1 or Category 2 event sequence. The applicant's consideration of the design of SSCs used for limiting worker exposure time when performing work in the vicinity of radioactive materials as part of normal operations is reviewed by the NRC staff under the ALARA program in SER Section 2.1.1.8.

2.1.1.6.3.2.3 Suitable Shielding

The applicant discussed equipment and facility designs related to shielding protection in SAR Section 1.9.1.3. The applicant's objective for radiation shielding is to reduce worker dose, in conjunction with a program of controlled personnel access to, and occupancy of, restricted areas, to levels that are ALARA within the dose standards of 10 CFR Part 20 (SAR Section 1.10.3). The applicant's shielding design considers normal operations and Category 1 and Category 2 event sequences. The shielding design considers both 'fixed' shielding (e.g., concrete walls, viewing windows) and 'movable' shielding (e.g., shield doors, slide gates in concrete floors) that involves some operator action to ensure the safety function is available. The applicant stated that facility shielding will include concrete walls, floors, and ceilings; shielded viewing windows; shield doors; slide gates in concrete floors; canister transfer machines; waste package trolleys; and penetration designs (SAR Section 1.10.3). The applicant designated shielding features that would reduce dose to the workers during normal operations as non-ITS because these shielding features will be used exclusively to comply with 10 CFR Part 20 during normal operations and event sequences are not part of normal operations. The shielding features credited in the PCSA for reducing the mean frequency of inadvertent exposure of personnel to below the mean frequency of the Category 1 event sequence are designated as ITS. These ITS shielding features include shield doors and slide gates in the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Receipt Facility (RF), and Wet Handling Facility (WHF). For example, the applicant specified a controlling value for the inadvertent opening of a slide that affects the frequency of a potential exposure of personnel (i.e., the mean probability of inadvertent opening of a slide gate shall be less than or equal to $1 \times 10^{-9}$ per transfer–SAR Table 1.9-2).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's shielding protection information using the guidance in YMRP Section 2.1.1.6. The purpose of the NRC staff's evaluation was to determine whether the applicant adequately considered the need for shielding to limit worker exposure in its PCSA. The NRC staff finds that the applicant's designation of the shielding features for 10 CFR Part 20 compliance during normal operations as non-ITS is acceptable because these non-ITS shielding features (e.g., concrete walls, floors and ceilings) are not relied upon in mitigating the frequency or consequences of Category 1 or 2 event sequences in the PCSA. In addition, offsite doses resulting from direct radiation are negligible due to the distance between the radiation source and potential offsite exposure locations. The NRC staff also finds the applicant's designation of those ITS shielding features (the shield doors and slide gates in IHF, CRCF, RF and WHF) relied on to reduce the likelihood of an event sequence to below Category 1 is acceptable because (i) this designation is based on the PCSA results and (ii) the use of these shielding features is a standard practice at nuclear facilities.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(3) are satisfied because the applicant's PCSA included adequate consideration of suitable shielding. In particular, the applicant identified (i) ITS shielding features relied on to reduce the likelihood of an event sequence to below Category 1 and (ii) shielding features used to protect worker radiological safety during normal operations (classified as non-ITS).

2.1.1.6.3.2.4 Radioactive Contamination Dispersal Monitoring and Control

The applicant discussed equipment and facility designs it will use to monitor and control dispersal of radioactive contamination in SAR Section 1.9.1.4. The applicant described how it will continuously monitor release points of radioactive contaminants in surface process facilities. The monitors will sample the effluent streams for airborne radioactivity particulates and gases and alert operators to off-normal conditions such as radiological releases or high levels of radiation. The applicant designated the radioactive contamination dispersal monitoring and control system as non-ITS because the monitoring and control system will not initiate automatic actions to reduce the event sequence frequency or mitigate the consequences of an event sequence. The applicant described the capability to monitor radioactive effluents in SAR Section 1.4.2.2.

The applicant stated that the heating, ventilation, and air-conditioning (HVAC) systems will be designed to minimize the spread of radioactive contamination by controlling air flows from areas of low potential contamination to areas of higher potential contamination. The applicant designated the portions of the surface confinement HVAC system that will exhaust air from areas with a potential for breach inside the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF) as ITS. The applicant classified the HVAC systems in the Initial Handling Facility (IHF), the Receipt Facility (RF), and the subsurface ventilation systems as non-ITS because these HVAC systems do not exhaust air from areas with a potential for a breach of a waste package (SAR Table 1.9-1). The applicant classified the subsurface ventilation system as non-ITS because all event sequences involving a waste package breach were classified as beyond Category 2. The evaluation of the event sequences involving waste package breach in the subsurface is evaluated in SER Section 2.1.1.4.3.4.2. The applicant described the surface facility HVAC systems in SAR Sections 1.2.3.4, 1.2.4.4, 1.2.5.5, and 1.2.6.4. The applicant discussed its means to monitor and control dispersal of radioactive contamination in SAR Section 1.4.2.2.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's radioactive contamination dispersal monitoring and control information using the guidance provided in YMRP Section 2.1.1.6. The NRC staff's evaluation focused on determining whether the applicant adequately considered radioactive contamination dispersal monitoring and control in its PCSA. The NRC staff determines that the applicant's designation of the radioactive contamination dispersal monitoring system as non-ITS is appropriate because the monitoring system is not used to initiate actions that reduce the frequency of an event sequence or to mitigate the consequences of an event sequence. The NRC staff also finds that the applicant's designation of those portions of the HVAC system exhausting air from areas with a potential for a canister breach in the CRCF and WHF as ITS acceptable because (i) these portions of the HVAC mitigate the effects of a Category 1 or 2 event sequence by removing radioactive materials through air filtration and (ii) this designation is based on the results of the PCSA. The NRC staff evaluates the adequacy of the HVAC systems controlling the dispersal of radioactive contamination in the surface facilities in SER Section 2.1.1.6.3.2.8.2.2, where the NRC staff finds that the applicant adequately addressed the ability of the HVAC ITS SSCs to perform the intended safety functions.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(4) are satisfied because the applicant's PCSA included adequate consideration of means to monitor and control dispersal of radioactive contamination.

2.1.1.6.3.2.5 Access Control to High Radiation Areas and Airborne Radioactivity Areas

The applicant discussed equipment and facility designs that it will use to provide access control to high radiation, very high radiation and airborne radioactivity areas in SAR Section 1.9.1.5. The applicant stated that controlling personnel access to normally unoccupied high radiation areas, very high radiation areas, or airborne radioactivity areas will be part of normal operations and will not be relied on for prevention or mitigation of Category 1 or Category 2 event sequences. Therefore, the applicant designated these design features as non-ITS. For those areas requiring periodic personnel access for waste-handling operations, and where the radiation levels may change as a result of Category 1 or Category 2 event sequences, the applicant identified procedural safety controls (PSCs) or ITS SSCs to provide access controls.

As described in SAR Section 5.11.3.2.1, the applicant stated that the access controls to high radiation and very high radiation areas would be consistent with the guidance in Regulatory Guide 8.38 (NRC, 2006ac). The applicant also stated, in SAR Section 5.11.3.6, that its respiratory protection program would follow the applicable regulations in 10 CFR 20.1701 and 10 CFR 20.1705 and the guidance in Regulatory Guide 8.15 (NRC, 1999aa). The applicant further stated that if other methods of respiratory protection against airborne radioactivity, such as the use of process or engineering controls, are not practical, then additional monitoring and limiting intakes will be provided through the use of access controls, limiting exposure times, and using respiratory protection equipment.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on access control to high radiation areas and airborne radioactivity areas using the guidance in YMRP Section 2.1.1.6 and Regulatory Guides 8.15 and 8.38 (NRC, 1999aa; NRC, 2006ac). The purpose of the NRC staff's evaluation was to determine whether the applicant adequately considered access control in its PCSA. The NRC staff finds that the applicant's designation of access control to high radiation areas or airborne radioactivity areas, as part of normal operations, as non-ITS is acceptable because the applicant will not rely on this access control to prevent or mitigate Category 1 or Category 2 event sequences in the PCSA. Additionally, the applicant's access control programs (SAR Section 5.11.3.2.1), including the respiratory protection program (SAR Section 5.11.3.6), is acceptable because it is consistent with Regulatory Guides 8.15 and 8.38 (NRC 1999aa; NRC, 2006ac;). Regulatory Guide 8.15 also states that methods of protection against airborne radioactive material, such as the use of process or other engineering controls, and limitation of exposure times, should be considered before the use of respirators.

The applicant also considered access controls for radiation levels associated with Category 1 or Category 2 event sequences. The NRC staff finds that the access-control-related procedural safety controls (PSCs) the applicant identified for areas requiring periodic personnel access for waste handling operations are reasonable because they would provide controls to personnel access in those areas where radiation levels may change as a result of Category 1 or Category 2 event sequences.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(5) are satisfied because the applicant's PCSA included adequate consideration of the means to control access to high radiation areas, very high radiation areas, and airborne radioactivity areas.

2.1.1.6.3.2.6 Criticality Control and Prevention and Ability to Perform Safety Functions

The applicant described its criticality safety program in SAR Section 1.14 and BSC (2008ba) and listed the ITS SSCs and procedural safety controls for criticality safety in SAR Tables 1.9-2 through 1.9-7 and Table 1.9-10. The applicant stated that the goal of the program is to prevent criticality during the preclosure period for normal conditions and Category 1 and Category 2 event sequences. The applicant described criticality sensitivity calculations that will be performed to evaluate the impact on system reactivity from the variation of parameters important to criticality (e.g., waste form characteristics, composition of borated water, and geometric rearrangement of SNF in the fuel basket). These criticality calculations assist the identification of hazards and event sequence development and quantification.

The applicant provided additional information in its response to the NRC staff's request for additional information (RAIs) (DOE, 2009az) asking about the organization of the applicant's proposed criticality safety program. The applicant also described the use of its organizational structure to implement the program in SAR Section 5.3. In its response to the NRC staff's RAI (DOE, 2009az), the applicant stated that it will revise SAR Figure 5.3-1 to show that the radiation protection and criticality safety manager will report directly to the site operations manager and, therefore, will be administratively independent of operations. Under the revised organizational chart, both the radiation protection and criticality safety manager and the operations manager are at an equivalent level and report to the site operations manager and chief nuclear officer (DOE, 2009az). The applicant further stated that the waste handling manager will ensure that the GROA design bases, including the means to prevent and control criticality, are maintained. The waste handling manager is also responsible for implementing the criticality safety measures (SAR Section 5.3.1.2.9).

The applicant stated that it will follow the applicable portions of Regulatory Guide 3.71 (NRC, 2005ac), which endorses certain criticality-related industry standards. To demonstrate that the program is robust and is based on accepted industry standard practices, the applicant listed the applicable American National Standards Institute/ American Nuclear Society (ANSI/ANS) standards used in the design and operations in SAR Section 1.14.3.1. For example, the applicant's criticality safety training will be developed in accordance with ANSI/ANS–8.20–1991 (ANS, 1991ab). The applicant stated that criticality safety practices and procedures, including practices and procedures for criticality safety audits and assessments, will be developed. The applicant further stated that criticality safety audits and assessments will be performed in accordance with ANSI/ANS–8.19–2005 (ANS, 2005aa). As described in SAR Section 5.1, the applicant stated that quality assurance requirements for operational activities will be developed and implemented prior to operations. The NRC staff's evaluation of the applicant's description of its quality assurance program is in SER Section 2.5.1. The applicant also stated that it will use other applicable standards (SAR Section 1.14.3.1) related to moderator control, use of neutron absorbers, and validation of criticality safety analysis to prevent criticality.

In BSC (2008ba), the applicant presented a detailed preclosure criticality safety evaluation, which demonstrates how the applicant intends to prevent criticality events. The applicant evaluated the following seven parameters to determine whether these parameters will need to be controlled to prevent criticality events during the preclosure period: (i) waste form characteristics, (ii) moderation, (iii) fixed neutron absorbers, (iv) soluble neutron absorbers, (v) geometry, (vi) interaction, and (vii) reflection. For each parameter, the applicant performed criticality sensitivity calculations for the different fuel types and conditions. The applicant summarized its sensitivity studies in SAR Section 1.14.2.4.1.7.

Using bounding waste form characteristics, such as modeling 5 wt.% enriched fresh fuel and bounding reflection based on its sensitivity analyses, the applicant stated that moderation will be the primary criticality control parameter for commercial spent nuclear fuel (SNF), and soluble boron concentration control would be the primary control parameter for the WHF pool. The applicant determined that the event sequences related to the potential for moderator to come in contact with fissile materials outside the pool were beyond Category 2 (SAR Tables 1.7-7 through 1.7-18). For SNF inside the pool, the applicant stated that the boron concentration will be maintained to prevent criticality (SAR Table 1.7-13); therefore, the applicant determined that the related event sequences were beyond Category 2. The NRC staff's evaluation of the applicant's designation of these beyond Category 2 event sequences is documented in SER Section 2.1.1.4.3.4.1.

The applicant identified the SSCs relied on to maintain subcriticality as ITS in the PCSA; these SSCs also prevent the moderator from coming into contact with fissile material. SAR Tables 1.9-2 through 1.9-7 listed the following three ITS systems that will provide containment throughout the geologic repository operations area (GROA) facilities: (i) the U.S. Department of Energy (DOE) and commercial waste package system, (ii) the naval SNF waste package system, and (iii) the mechanical handling system. In addition, SAR Tables 1.9-3 and 1.9-4 listed two ITS systems [the mechanical handling system and the fire protection system, which includes the use of double-interlock preaction (DIPA) sprinklers] in both the Canister Receipt and Closure Facility (CRCF) and the Wet Handling Facility (WHF) that will have a moderator control safety function. The applicant will also rely on PSC–9 to maintain the concentration of boron (enriched to 90 percent of isotope boron-10) as a neutron absorber to above 2,500 mg/L [0.02 lb/gal] in the WHF pool and cask/canister.

The applicant identified two other SSCs (DOE canister staging racks in the CRCF and staging racks in the WHF pool) as important to safety (ITS) to prevent criticality. These SSCs rely on controlling spacing between fuel rods to perform their ITS functions. The staging racks in the WHF pool will also contain fixed non-ITS neutron absorbers that will be used to provide extra margin.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on its criticality control and prevention methodology using the guidance in YMRP Section 2.1.1.6 to determine whether the applicant adequately considered criticality control and prevention in its PCSA and whether the related ITS SSCs could perform their intended safety functions. In addition, the NRC staff used the ANSI/ANS-8 standards endorsed in Regulatory Guide 3.71 (NRC, 2005ac) to review the applicant's design. The applicant relied on ITS SSCs to prevent criticality and did not credit mitigating systems or actions, such as the presence of shielding or evacuation alarms, to reduce mean frequency or consequence of a criticality event.

The NRC staff finds the applicant's use of the ANSI/ANS–8 standards listed in SAR Section 1.14.3 acceptable because the NRC endorsed these standards in Regulatory Guide 3.71 (NRC, 2005ac). Therefore, the NRC staff finds that the applicant's criticality safety program is consistent with this NRC guidance. The NRC staff reviewed the applicant's organizational structure and responsibilities to ensure the prevention of criticality events. The NRC staff finds the applicant's criticality safety program acceptable because, as stated in the applicant's RAI response (DOE, 2009az), this program will be administratively independent of operations, which is consistent with ANSI/ANS–8.19–2005 (ANS, 2005aa). In addition, the functions and responsibilities of this criticality protection program are consistent with ANSI/ANS–8.19–2005 (ANS, 2005aa). The NRC staff also finds that the applicant's statement in SAR Section 1.14.1 that training will be developed in accordance with ANSI/ANS–8.20–1991 (ANS, 1991ab) makes crediting PSC–9 acceptable because personnel will be trained and familiar with the importance of the soluble neutron absorber to criticality safety through the training developed by the applicant's criticality safety staff.

The NRC staff finds the applicant's approach of controlling moderators through the use of ITS systems, such as the double-interlock preaction (DIPA) sprinklers in the fire protection system, acceptable because the use of DIPA sprinklers reduces the likelihood of inadvertent introduction of sprinkler water into a breached cask. This approach is consistent with ANSI/ANS–8.22–1997 (ANS, 1997ac), a national standard for limiting and controlling moderators to achieve criticality safety in operations with fissile materials in a moderator control area. In addition, the NRC staff reviewed the applicant's technical basis for identifying the fixed neutron absorbers in the WHF pool staging racks as non-ITS and finds the technical basis acceptable because the use of fixed neutron absorbers to provide an additional margin of safety is a standard engineering practice for criticality control.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(6) and 10 CFR 63.112(e)(8) are satisfied because the applicant's PCSA adequately considered the means to control criticality and the ability of the ITS SSCs for criticality control to perform their intended safety functions during the occurrence of event sequences, including (i) not crediting mitigating systems or actions, such as the presence of shielding or evacuation alarms, to reduce mean frequency or consequence of a criticality event; (ii) the applicant's criticality safety program is consistent with ANSI/ANS–8.19–2005; and (iii) controls to prevent a moderator (e.g., water) from entering a breached waste package [e.g., use of double-interlock preaction (DIPA) sprinklers in the fire protection system].

2.1.1.6.3.2.7 Radiation Alarm System

The applicant discussed the radiation alarm system in SAR Section 1.9.1.7 and the radiation monitoring system in SAR Section 1.4.2.2. The applicant stated that these systems would monitor the gamma radiation levels throughout the surface and subsurface areas and effluents from the GROA release points to alert plant personnel via visible and audible alarms when a threshold radiation level has been exceeded. The radiation monitors operate continuously providing local indication of radiation levels, as well as remote data transmission to the Central Control Center for tracking and trending.

In SAR Section 1.9.1.7, the applicant identified the radiation monitoring system as non-ITS because this system will not be relied upon to alert the operator to take manual actions in response to an event sequence nor will the monitoring system initiate automatic actions to prevent or mitigate an event sequence.

SAR Section 1.4.2.2 describes three major components of its radiation monitoring system: (i) area radiation monitors, (ii) continuous air monitors, and (iii) airborne radioactivity effluent monitors. The system will sound an alarm when a threshold radiation level is reached. The applicant stated that an uninterruptible power supply (UPS) will power the radiation monitoring system.

The applicant stated that it will use the methods and practices of ANSI/ANS–HPSSC–6.8.1–1981 (ANS, 1981aa) in designing and locating the area radiation monitors. Additionally, in SAR Section 1.10.4.1.7, the applicant stated that it will perform radiation surveys to determine beta, gamma, and neutron radiation levels.

The continuous air monitors will be located throughout the surface facilities, the subsurface facilities, and the access shafts to measure inhalation dose by determining the radioactivity present in the air and the concentration of airborne radioactive particulates. The applicant stated that the performance for the continuous air monitors will be based upon the methods and practices of ANSI N42.17B–1989 (ANSI, 1989aa), which are described in SAR Section 1.4.2.2.1.

The airborne radioactivity effluent monitors will measure any airborne effluent releases. The sampled air will be continuously monitored for radioactivity by monitors located in designated exhaust stacks in the handling facilities. Air from the subsurface exhaust will be continuously sampled on filters and periodically measured for possible radioactive releases. The airborne radioactivity effluent monitors sample the surface effluent stream for airborne radioactivity particulate and gases. The applicant stated that air sampling will be performed using the methods and practices of ANSI/HPS N13.1-1999 (ANSI, 1999ab) and the performance requirements for the airborne effluent monitors will use the methods and practices of ANSI N42.18-2004 (ANSI, 2004aa).

**NRC Staff’s Evaluation**

The NRC staff reviewed the information on the radiation alarm system in SAR Sections 1.4.2.2 and 1.9.1.7 using the guidance in YMRP Section 2.1.1.6 and NUREG-0800 (NRC, 2013aa,ab) with the focus on determining whether the applicant adequately considered the radiation alarm system in its PCSA. On the basis of this evaluation, the NRC staff determines that the design codes and standards selected by the applicant (SAR Section 1.4.2.2.2) are applicable to the radiation monitoring system design because they are consistent with standard engineering practices at nuclear reactor facilities, and applicable NRC guidance in NUREG–0800 (NRC, 2013aa), for the design of radiation alarm systems.

The NRC staff finds the applicant’s information on area monitors acceptable because the applicant stated that their location, design, and use will be consistent with the methods and practices of ANSI/ANS-HPSSC-6.8.1-1981 (ANSI, 1981aa). These practices have been accepted by the NRC for use in a nuclear material processing facility similar to the proposed Yucca Mountain facility and commercial nuclear power plants NUREG–0800 (NRC, 2013aa), and the NRC staff finds their use, likewise, appropriate here. The NRC staff also finds that the applicant’s use of the ANSI N42.17B–1989 (ANSI, 1989aa) standard for performance requirements of continuous air monitors acceptable because the standard provides industry practices for radiation protection instruments, as referenced in NUREG–0800 (NRC, 2013ab).

The NRC staff finds the airborne radioactivity effluent monitors acceptable because the applicant stated its sampling and performance criteria for the monitors will conform to ANSI/HPS N13.1-1999 (ANSI, 1999ab) and ANSI N42.18-2004 (ANSI, 2004aa) standards.
These standards have been accepted for use in nuclear power plants, which also have the potential for airborne release of radioactive material from spent fuel that require radiation monitoring and controls NUREG–0800 (NRC, 2013aa).

The NRC staff also finds that the applicant’s use of an uninterruptable power supply (UPS) to the radiation monitoring system is acceptable because the UPS will ensure that the radiation monitoring system will have the power needed to perform its intended safety functions.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(7) are satisfied because the applicant's PCSA adequately considered a radiation alarm system designed to warn of significant increases of radiation levels, concentrations of radioactive material in air, and increased radioactivity in effluents.

2.1.1.6.3.2.8 Ability of Structures, Systems, and Components Important to Safety to Perform Their Intended Safety Functions

The applicant discussed the reliability of the ITS SSCs to perform their intended safety functions based on the reliability assessments described in SAR Section 1.7 and procedural safety controls (PSCs). To provide additional assurance that the ITS SSCs will perform their safety functions to an appropriate level of reliability, the applicant described its equipment qualification program for these ITS SSCs for the range of environmental conditions anticipated at the time of functional demand. One of the objectives of this qualification program will be to ensure the ability of ITS SSCs to perform their intended safety functions under applicable environmental, seismic, and event sequence conditions (SAR Section 1.13.1). The applicant stated that this program "will be implemented prior to initiating procurement of ITS SSCs to ensure that the design of ITS SSCs will adequately incorporate qualification requirements before fabrication, construction, or installation into a repository facility" (SAR Section 1.13.2). Additionally, the applicant stated that it will implement a monitoring program (SAR Sections 5.6.4.5 and 5.6.5) to (i) detect operational deviations (indicative of a degraded state of reliability) of ITS SSCs and (ii) initiate appropriate corrective actions (SAR Section 1.9.1.8). The applicant stated that it will also develop reliability-centered maintenance, inspection, and testing programs for the ITS SSCs, as necessary, to ensure their continued functioning and readiness (SAR Section 1.13).

In the following subsections, the NRC staff reviewed the ITS SSCs identified by the applicant and listed in SAR Tables 1.9-2 through 1.9-7 and 1.9-10 to determine whether the ITS SSCs the applicant identified through its PCSA will perform their intended safety functions. The NRC staff's review is presented under the following ITS SSCs groupings: (i) surface facilities; (ii) mechanical systems; (iii) transportation systems; (iv) electrical components and emergency power systems (EPS); (v) fire protection systems; (vi) transportation, aging, and disposal (TAD) canisters; and (vii) waste packages.

2.1.1.6.3.2.8.1 Surface Facilities Important to Safety

The applicant provided descriptions and design information for the building structure for each of the surface facilities in SAR Section 1.2 and discussions on reliability used in event sequence categorization in SAR Section 1.7. The applicant also addressed the ability of ITS surface facility structures to perform their intended safety functions in SAR Section 1.9.1.8.

The applicant identified four surface facility building structures as ITS because the building structure protects ITS mechanical handling equipment operating inside the building (e.g., equipment for unloading transportation casks and loading waste packages and waste containers) from external hazards (e.g., seismic and tornadoes). The four ITS surface facility structures identified are the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Receipt Facility (RF). The applicant identified nuclear safety design bases and criteria for the ITS surface facility structures in SAR Tables 1.9-2 through 1.9-6. The applicant also stated that the determination of design bases was based on the PCSA, as shown in SAR Figure 1.7-1. The applicant further stated that it will rely on the structural integrity of the surface facilities to (i) protect ITS SSCs inside the building from wind and volcanic ash and (ii) prevent building collapse onto waste containers under seismic event sequences.

The applicant stated that the mean annual probability of building collapse will not exceed $10^{-6}$/year for the wind and volcanic ash fall loads (SAR Tables 1.9-2 through 5). The applicant stated that the maximum design tornado wind speed for ITS structures is 304 km/h [189 mph] (SAR Sections 1.2.2.1.6.1.2 and 1.6.3.4.4 and Table 1.2.2-1) and the tornado wind speeds expected at the repository site would not exceed 267 km/h [166 mph] (SAR Section 1.6.3.4.4). The applicant estimated the straight-line wind hazard corresponding to a probability of $10^{-6}$/year to be 193 km/h [120 mph] (SAR Section 1.6.3.4.4). The applicant determined that the design basis volcanic ash load on the roof to be 1.0 kPa [21 lb/ft$^2$] (SAR Table 1.2.2-1), corresponding to a mean annual probability of $6.4 \times 10^{-8}$. Thus, the applicant stated that the surface facilities are designed to a volcanic ash roof load that is less than the probability threshold for Category 2 event sequences (BSC, 2008ai, pg. 41). Additionally, since the largest expected tornado wind speed of 267 km/h [166 mph] and the straight-line wind speed {193 km/h [120 mph]} estimated at the probability threshold for Category 2 event sequences at the site are smaller than the design basis tornado wind speed of 304 km/h [189 mph] (SAR Sections 1.2.2.1.6.1.2 and 1.6.3.4.4 and Table 1.2.2-1), the applicant stated that the straight-line wind and tornado wind would not be expected to initiate event sequences. The NRC staff's evaluations of straight-line wind and tornado wind are documented in SER Section 2.1.1.3.3.1.3.2.

The applicant stated that the design basis for a building collapse due to a spectrum of seismic events is less than or equal to a mean annual probability of $2 \times 10^{-6}$ (SAR Tables 1.9-2 through 5). The NRC staff's evaluation of building collapse event sequences is documented in SER Section 2.1.1.4.3.4.2. This threshold value is based on event sequences that have at least 1 chance in 10,000 of occurrence over the 50-year waste emplacement period. The applicant assumed a preclosure period of 100 years, of which 50 years would be an operational period for the surface facilities involving SNF and HLW handling. The applicant addressed the design basis ground motion for all the facility structures and the structural design in SAR Section 1.2.2.1.6.3. For each facility, the applicant determined the seismic fragility or mean probability of unacceptable performance as a function of ground motion. The fragility for the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Receipt Facility (RF) was calculated for imminent collapse or Limit State A—large permanent distortion short of collapse (significant damage) (American Society of Civil Engineers, 2005aa), and the fragility parameters were shown in Table 6.2-1 (BSC, 2008bg). The applicant calculated the structural performance or annual probability of failure by convolving the fragility curves and the site-specific seismic hazard curve. As shown in Table 6.2-1 (BSC, 2008bg), the applicant estimated the annual frequency of failure to be $3.8 \times 10^{-7}$, $4.1 \times 10^{-7}$, $7.8 \times 10^{-7}$, and $8.7 \times 10^{-7}$ for the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Receipt Facility (RF), and Wet Handling Facility (WHF), respectively (BSC, 2008bg). The applicant stated that the annual probability of failure for all the surface facility structures is less than the design basis threshold of $2 \times 10^{-6}$ (SAR Table 1.2.4-4) during the proposed emplacement period of 50 years.

**NRC Staff's Evaluation**

The NRC staff reviewed the safety functions and basis for controlling parameters and their relations to the PCSA using the guidance in YMRP Section 2.1.1.6. The NRC staff's assessment of the ability of the facility structures to perform the intended safety functions and meet the nuclear safety design bases and criteria for wind, ash fall, and seismic events is based on the evaluation of the structural design evaluated by the NRC staff in SER Section 2.1.1.7 and the NRC staff's evaluation of structural performance in SER Section 2.1.1.4. The NRC staff's evaluation of the probability of tornado effect on structures is given in SER Section 2.1.1.3.3.1.3.2, where the NRC staff finds that the applicant's estimated probability of structural damage caused by tornado wind speed is acceptable. The applicant's justification for the design ash load is evaluated in SER Section 2.1.1.1.3.6, where the NRC staff finds that the applicant's estimation of ash fall load is acceptable.

The NRC staff's evaluation of the ITS facility structural design is provided in SER Section 2.1.1.7.3.1.1. In that section, the NRC staff finds that the applicant's seismic analysis, methodology, use of codes and standards, and design parameters (e.g., load combinations; material properties; and seismic design of shear wall, diaphragm slabs, and foundations) are adequate. The NRC staff's evaluation of the applicant's determination of seismic performance (probability of failure to support the structure) and applicant's facility structure design and fragility parameters is provided in SER Section 2.1.1.4.3.3.1.2.1. In that section, the NRC staff finds that the mean annual probability of unacceptable performance of structural collapse for all surface facility structures is less than the threshold value of $2 \times 10^{-6}$ (SAR Table 1.2.4-4) during the proposed emplacement period of 50 years.

On the basis of the NRC staff's evaluations in the above SER sections, the NRC staff finds that the applicant's analysis of the ITS building structures' ability to perform their intended safety functions during wind, ash, or seismic events is acceptable because (i) the applicant developed the related design bases using site-specific information, and (ii) the design and construction of these ITS structures will follow the codes and standards consistent with standard engineering practices for facilities performing similar waste-handling operations. The design of ITS structures is evaluated in SER Section 2.1.1.7.3.1.1.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(8) are satisfied because the applicant's PCSA adequately considered the surface facility structures' ability to perform their intended safety functions

2.1.1.6.3.2.8.2 Mechanical Systems Important to Safety

2.1.1.6.3.2.8.2.1 Mechanical Handling Equipment Important to Safety

The applicant discussed the ability of ITS mechanical handling systems to perform their intended safety functions in SAR Sections 1.2.3 (Initial Handling Facility), 1.2.4 (Canister Receipt and Closure Facility), 1.2.5 (Wet Handling Facility), and 1.2.6 (Receipt Facility). Start up, maintenance, inspection, and testing of the ITS mechanical systems were described in SAR Sections 5.5 and 5.6. The NRC staff review of ITS mechanical handling systems is grouped according to (i) canister transfer machine (CTM), (ii) waste package transfer trolley (WPTT), (iii) cask-handling crane (CHC), (iv) spent fuel transfer machine (SFTM), and (v) cask transfer trolley (CTT). These five groups are ITS mechanical handling systems that will be used in the GROA facilities to handle nuclear wastes.

**Mechanical Handling Systems**

The applicant stated that the five ITS mechanical handling systems prevent or mitigate event sequences related to handling of commercial SNF assemblies, canisters, and casks. In general, all five systems will limit movement speed.  The applicant stated that the CTM, WPTT, and CHC will also prevent spurious movement.  The applicant also stated that the CTM, CHC, and SFTM are designed to prevent collapse during a seismic event.  The ITS mechanical handling systems are also designed to prevent a load drop.  In addition, the applicant stated that the WPTT design prevents tipover or rocking during a seismic event.  Furthermore, the applicant stated the WPTT design prevents rapid tilt-down.  According to the applicant, the CTM design includes the capability to protect personnel from direct exposure from an inadvertent opening of the CTM slide gate, the inadvertent raising of the CTM shield skirt, or an inadvertent motion of the CTM away from an open port.  The applicant stated the SFTM design precludes a lift of the commercial SNF assembly beyond a safe limit.  These safety functions (listed in SAR Tables 1.9-2 through 1.9-5) are evaluated in SER Section 2.1.1.7.3.2.

The applicant stated that it will use the American Society of Mechanical Engineers (ASME) NOG–1–2004, Type I (ASME, 2005aa) in the design of all five ITS mechanical handling systems reviewed in this SER section.  The applicant took additional measures to ensure safety of the ITS mechanical handling systems.  For example, the applicant stated that  the design of the CTM, CHC, and SFTM includes (i) integrated overspeed switches to limit trolley/bridge overspeeding; (ii) rope mis-spool sensors, broken-rope sensors, hoist-dynamic-braking-resistor temperature monitors, and motor-winding-resistance temperature detectors to safeguard against a load drop; (iii) circuit breakers for speed drives of bridges and trolleys to protect against spurious movement; and (iv) interlocks and anti-collision sensors to prevent collision during CTM operation (SAR Section 1.2.4.2.2.1).  The WPTT will be equipped with two redundant drive trains to rotate the shielded enclosure, either of which can support the enclosure.  For the cask transfer trolley (CTT), the applicant will use redundant air pressure regulators to control the air-bearing pressure so that the loss of one regulator will not cause lack of air supply to the entire CTT.  The applicant stated that the CHC will be equipped with redundant lower and upper limit switches to ensure that the grapple cannot be raised or lowered beyond the safe limits.  The applicant also stated that it will use various interlocks to ensure safe operations of the ITS mechanical handling systems (see section “Important to Safety Controls Linking the Mechanical Handling Systems and Shield Doors” in this SER section for more discussion).

In addition to applying safe engineering practices and adhering to the safe margins of design in ASME NOG–1–2004, Type I (ASME, 2005aa), the applicant will use procedural safety controls (PSCs) for the ITS mechanical handling systems to prevent event sequences or mitigate their effects.  The applicant defined a PSC for the WPTT, which verifies that personnel are outside the waste package positioning room and load-out area before the WPTT will move.  To limit spurious CTT movement, operators will need to independently verify that the CTT is on the floor and the pneumatic systems are inactive while a cask is loaded onto the CTT.  To ensure seismic stability, the applicant stated that it will develop an operational procedure to ensure that the cask remains attached to the CHC until the cask is placed onto the CTT and the seismic restraints are properly engaged.

In addition, the applicant described its equipment qualification program that will be prepared and implemented to (i) ensure the ability of active mechanical and electrical ITS SSCs to perform their intended safety functions under applicable environmental, seismic, and event sequence conditions; (ii) ensure the availability, reliability, and component-aging management of ITS SSCs; (iii) ensure the materials, parts, and equipment used as ITS SSCs are suitable for the application; and (iv) verify the adequacy of the design through qualification testing or analysis, including a corrective action program to document and evaluate equipment failures (SAR Section 1.13).  Additionally, the applicant described its plans and procedures for conducting preventive and corrective maintenance, surveillance, and periodic testing of ITS mechanical handling systems, including instrumentation and controls, using a reliability-centered maintenance methodology (SAR Sections 5.6 and 5.10).

**NRC Staff's Evaluation**

The NRC staff reviewed the ability of the five mechanical handling systems to perform their intended safety functions using the guidance in YMRP Section 2.1.1.6. The NRC staff finds that the codes and standards [e.g., ASME NOG–1–2004, Type I (ASME, 2005aa)] for the applicant's design information for the ITS mechanical handling systems are appropriate because of their applicability to the proposed GROA activities and because these codes and standards have been used for similar activities at other NRC-licensed facilities. The NRC staff also finds that the applicant's procedural safety controls (PSCs) for the mechanical handling systems are adequate because these PSCs would ensure safe operations of these systems either by reducing the likelihood of an event sequence or mitigating its effect. On the basis of its review, the NRC staff finds that the applicant's design information for the mechanical handling systems is adequate to demonstrate that these systems will perform their intended safety functions.

Additionally, the applicant described its equipment qualification program and its plan to conduct preventive and corrective maintenance, surveillance, and periodic testing of ITS mechanical handling systems to ensure that the ITS SSCs of the mechanical handling systems will be available to perform their safety functions. The applicant's description of reliability-centered maintenance programs for ITS SSCs is evaluated in SER Section 2.1.1.6.3.2.10. Additional descriptions of the applicant's plans for inspection, testing, and maintenance of the SSCs are provided in SER Section 2.5.6.

**Important to Safety Controls Linking the Mechanical Handling Systems and Shield Doors**

The applicant identified 29 key groups of ITS SSCs (SAR Table 1.4.2-1) that rely on ITS controls to accomplish their safety functions. Included in the 29 key groups are safety controls (interlock subsystems) for slide gates, shield doors, TEVs, and other SSCs that are external to the mechanical handling systems and that interact with them to protect personnel from inadvertent direct exposure to radiation.

In SAR Section 1.4.2, the applicant stated that all ITS controls will be made up of individual hardwired devices, instead of being driven by software or programmable devices. The applicant further indicated that the hardwired ITS controls will be designed to prevent their safety functions from being overridden by the non-ITS controls. To facilitate maintenance and surveillance activities or to facilitate recovery from a spurious actuation of an ITS control function, key-locked switch bypasses will be used under administrative controls to override an ITS control function.

The applicant stated that it will use ITS interlock controls for the interactions between the mechanical handling systems and other ITS SSCs that are external, such as the slide gates, shield bell skirts, shield doors, and other exposure-protection components. In response to an NRC staff RAI (DOE, 2009do), the applicant cited the following codes and standards for these ITS interlock controls: IEEE–308, IEEE–379, IEEE–603, IEEE–384 (Institute of Electrical and Electronics Engineers, 2001aa,ab; Institute of Electrical and Electronics Engineers, 1998ab,aa). However, the applicant stated it may take exceptions to the specific design features at the detailed design stage (DOE, 2009dl, Tables 3 through 6 for the specific sections that are applicable to the ITS SSCs). These exceptions are related to design criteria for fail-safe safety, fault tolerance, redundancy, high availability and diversity and defense in depth, and protection against single point of failure.

**NRC Staff's Evaluation**

The NRC staff reviewed the design of the ITS safety controls (interlock subsystems) that link the mechanical handling systems and other ITS SSCs external to these safety controls, such as the shield doors and other exposure protection devices, using the guidance provided in YMRP Section 2.1.1.6. The NRC staff finds that the IEEE codes and standards the applicant identified for the ITS safety interlock subsystems design are appropriate because these codes and standards represent standard nuclear industry practice for safety interlock subsystem design. However, the applicant stated that exceptions to the IEEE standards used for interlock subsystem design may be taken. As detailed design progresses, exceptions the applicant identified that it may take are related to design criteria for fail-safe safety, fault tolerance, redundancy, high availability and diversity and defense in depth, and protection against single point of failure associated with IEEE Standards 308-2001, 384-1992, 379-2000, and 603-1998 (DOE, 2009dl). The applicant did not provide the design basis for the use of exceptions to these IEEE Standards (DOE, 2009dl, Tables 3 through 6). Further, the use of exceptions to these IEEE standards is not consistent with nuclear industry practice for safety interlock subsystems. Worker safety is assured through design criteria in these IEEE standards, which include redundancy, independence, and the single-failure criterion to achieve reliability of ITS interlock subsystem design and to reduce uncertainty.

For the foregoing reasons, the NRC staff finds that the applicant's ITS safety interlock subsystems designs that are in accordance with IEEE Standards 308-2001, 384-1992, 379-2000, and 603-1998, without exceptions, as described in SAR Section 1.2.2.4, are acceptable. Given that the applicant stated that it may take exceptions to these IEEE Standards, the NRC staff proposes the following condition on the construction authorization:

**Proposed Condition of Construction Authorization [10 CFR 63.32(a)]**

> DOE shall not, without prior NRC review and approval, take or implement any exception to the IEEE Standards 308–2001, 384–1992, 379–2000, and 603–1998 in the design of the ITS safety interlock subsystems.
>
> Any amendment request must include the design basis for the use of the exception(s), including the ability of structures, systems, and components to perform their intended safety functions assuming the occurrence of event sequences in accordance with 10 CFR 63.112(e)(8).

ITS SSCs safety functions in the area of personnel radiation protection are described further in SER Section 2.1.1.7.3.7.

**NRC Staff's Conclusion**

Based on the NRC staff's evaluations in SER Section 2.1.1.6.3.2.8.2.1 and the proposed condition of construction authorization above, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(8) for the mechanical handling equipment ITS are satisfied because (i) the applicant's design information for the mechanical handling systems is adequate to demonstrate that these systems will perform their intended safety functions; (ii) the applicant provided a description of its reliability-centered maintenance programs for ITS SSCs, which the NRC staff reviews and finds acceptable in SER Section 2.1.1.6.3.2.10; and (iii) the applicant's PCSA adequately considered the ability of the ITS mechanical handling systems to perform their intended safety functions.

2.1.1.6.3.2.8.2.2 Heating, Ventilation, and Air-Conditioning Systems Important to Safety

The applicant provided information to show how the Heating, Ventilation, and Air-Conditioning Systems (HVAC) Important to Safety (ITS) will be able to perform their intended safety functions, assuming the occurrence of event sequences. The HVAC intended safety functions include (i) filtering air in the confinement areas to mitigate the consequences of a radionuclide release and (ii) cooling and ventilation of ITS electrical equipment and battery rooms in nonconfinement areas to support their safety functions (SAR Sections 1.2.2.3, 1.2.4.4, 1.2.5.5, and 1.2.8.3). This information included identifying procedural safety controls (PSCs), and measures to ensure the availability of the ITS HVAC systems (SAR Table 1.9-10). The applicant described HVAC ITS SSCs in SAR Section 1.9. SAR Table 1.9-1 identified the portions (i.e., subsystems) of the surface nonconfinement HVAC system in the Emergency Diesel Generator Facility (EDGF) that it found to be ITS and the portions of the surface nuclear confinement HVAC systems in the CRCF and WHF that it found to be ITS. The applicant identified the nuclear safety design bases for the ITS HVAC SSCs in SAR Tables 1.9-3 (for the CRCF) and 1.9-4 (for the WHF).

**Procedural Safety Controls**

The applicant described procedural safety controls (PSCs) in SAR Section 1.9.3. The applicant identified PSC–7 for the ITS HVAC subsystems in the CRCF and WHF in SAR Table 1.9-10. In PSC–7, the applicant stated that one ITS HVAC train will be required to be operating with the other one in standby before waste-handling operations begin. The applicant also discussed PSC–7 in SAR Sections 1.2.4.4.4 (for the CRCF) and 1.2.5.5.4 (for the WHF) and discussed PSC–8 in SAR Section 1.4.1.2.4. The applicant stated in SAR Section 1.2.8.3.1.4 that it did not identify any PSCs for the EDGF HVAC system. In response to NRC staff's request for additional information (RAI) (DOE, 2009fq), the applicant stated that PSC–7 extends to specific ITS HVAC components as well as electrical distribution equipment required to operate ITS HVAC components. Also, as part of this response, the applicant stated that the EDGF ITS HVAC system is covered by PSC–8 (two diesel generators are aligned to start upon detection of undervoltage).

**NRC Staff's Evaluation**

The NRC staff reviewed the identification of procedural safety controls (PSCs) using the guidance in YMRP Section 2.1.1.6. The NRC staff finds that the applicant adequately identified Procedural Safety Controls (PSC) for the ITS HVAC systems in the surface facilities because the applicant described that (i) PSC–7 will ensure the HVAC system's availability to mitigate the consequences of an event sequence by requiring one HVAC train to be in the operational mode and the second train to be in the standby mode before commencing waste-handling operations; and (ii) PSC–8 extends to ITS diesel generator support systems, including EDGF ITS HVAC, and ensures that the ITS diesel generators would be available to provide power to the surface nuclear confinement HVAC system.

**Means To Limit the Concentration of Radioactive Material in Air**

The applicant stated that the heating, ventilation and air-conditioning (HVAC) systems in the surface facilities will control the air flow from areas of low potential for radioactive contamination to areas of higher potential for radioactive contamination (SAR Section 1.9.1.1). Additionally, the applicant classified the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF) HVAC subsystems as ITS because they will be relied on to mitigate the consequences of radionuclide releases resulting from the event sequences related to canister breaches.

In SAR Table 1.2.2-12, the applicant provided the principal design codes and standards applicable to the HVAC system regarding air flow between areas with low potential for radioactive contamination and areas with higher potential for radioactive contamination ANSI/ANS–57.9–1992, ANSI/ANS–57.7–1988 (ANS, 1992aa; ANS, 1988aa).

**NRC Staff's Evaluation**

The NRC staff reviewed the process the applicant proposed to limit the concentration of radioactive material in air using the guidance in YMRP Section 2.1.1.6. The NRC staff finds that control of the air flow from areas with low to higher potential for radioactive contamination is acceptable because this approach is consistent with the guidance related to control of airborne contaminants and gaseous radiation sources in Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff reviewed the applicant's HVAC design in SER Section 2.1.1.7.3.3 where the NRC staff finds that the ITS HVAC exhaust subsystem design with two stages of high efficiency particulate air (HEPA) filtration is adequate to achieve the applicant-specified overall filtration efficiency. The NRC staff finds that, based on industry operating experience, two stages of HEPA filters (rated at 99.97% minimum efficiency for 0.3 μm particles) are capable of removing airborne radioactive material for event sequences below the applicable 10 CFR Part 20 limits. The NRC staff also finds that the applicant adequately specified the overall filtration efficiency for this subsystem because this level of efficiency is consistent with the safety needs determined through the PCSA. Although ANSI/ANS–57.7–1998 (ANS, 1988aa) was withdrawn in October 2007 for lack of funding by the ANS to continue to maintain the standard, the NRC staff determines that it is applicable to the design of the ITS HVAC systems because the design specifications in this version of the standard provide nuclear industry accepted guidance for design of HVAC systems. Therefore, the NRC staff finds that the applicant appropriately described the means to limit the concentration of radioactive material in air. The NRC staff's review of the applicant's means to inspect, test, and maintain ITS HVAC SSCs is discussed later in this SER Section (2.1.1.6.3.2.8.2.2).

**Means To Control the Dispersal of Radioactive Contamination**

The applicant stated in SAR Section 1.9.1.4 that the heating, ventilation and air-conditioning (HVAC) systems will be relied on to control the dispersal of radioactive contamination. The applicant stated that it will minimize the spread of contamination by having filtration zones and by controlling the air flow from areas with low potential for contamination to higher potential for contamination. The applicant described the confinement zoning in SAR Table 1.2.2-13 and defined nonconfinement zones as noncontaminated (i.e., clean) areas, tertiary confinement zones as areas where airborne contamination is not expected during normal operations, and secondary confinement zones as areas with a potential for airborne contamination during normal operations. The applicant stated that this designation is in accordance with DOE–HDBK–1169–2003, as described in Section 2.2.9 (DOE, 2003ae).

In its response to an NRC staff RAI (DOE, 2009fo), the applicant stated that confinement areas are designated as ITS if the area has an identified Category 2 event sequence in which a loaded canister may be breached and an ITS HVAC system will be used to mitigate the potential release. In addition, the applicant stated that it will not rely on seals through walls and slabs to maintain confinement and will update SAR Section 1.9.1.10 to reflect this position (DOE, 2009fo). Furthermore, the applicant also described in this response that air will flow from non-ITS confinement areas into ITS confinement areas.

**NRC Staff's Evaluation**

The NRC staff reviewed the means to control dispersal of radioactive contamination using the guidance in YMRP Section 2.1.1.6; ANSI/ANS–57.9–1992; ANSI/ANS–57.7–1988 (ANS, 1992aa; ANS, 1988aa). Specifically, the NRC staff reviewed Section 2.2.9 (DOE, 2003ae) and finds that the applicant's means to control dispersal of radioactive contamination is acceptable because (i) the DOE handbook (DOE, 2003ae) is consistent with nuclear industry standard guidance [i.e., ASME AG-1-2012 Code on Nuclear Air and Gas Treatment (ASME, 2012aa)], (ii) the applicant's confinement zoning as described in the SAR is also in accordance with this handbook (DOE, 2009fo) and standard practices in the nuclear industry (ASME, 2012aa), and (iii) the applicant's confinement zoning approach is consistent with the air-cleaning practices in commercial nuclear facilities (ASME, 2012aa) and DOE complexes as stated in Section 2.2.9 (DOE, 2003ae). On the basis of the NRC staff's evaluation, the NRC staff finds that the applicant appropriately described the means for the HVAC system to control the dispersal of radioactive contamination.

**Redundancy Within the ITS HVAC Systems**

The applicant stated that the ITS heating, ventilation, and air-conditioning (HVAC) system will be designed to have more than one HVAC train and will have redundant components within HVAC trains. As part of its design criteria for the ITS HVAC subsystems in the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF), the applicant identified two full-capacity independent trains that will be used to exhaust from areas where there is a potential for a canister to be breached (SAR Tables 1.2.4-4 and 1.2.5-3). The applicant stated in SAR Section 1.9.1.12 that one train will be in operation with the other one in standby and that the trains will alternate between these modes. Independent trains will also be a part of the applicant's design criteria for the ITS HVAC subsystems in the CRCF, WHF, and Emergency Diesel Generator Facility (EDGF), which will provide cooling for ITS electrical equipment and battery rooms (SAR Tables 1.2.4-4, 1.2.5-3, and 1.4.1-1). In addition, the applicant included redundancy within a train by specifying operating and standby components (e.g., operating and standby HEPA filter plenums and operating and standby exhaust fans). For example, the applicant showed this redundancy in SAR Figure 1.2.4-104 for the subsystem that will provide cooling to ITS electrical equipment and battery rooms in the CRCF where standby units start automatically if the operating units fail.

**NRC Staff's Evaluation**

The NRC staff reviewed redundancy within the ITS HVAC systems using the guidance provided in YMRP Section 2.1.1.6. The NRC staff determines that the applicant adequately addressed redundancy in the ITS HVAC systems because the applicant specified the use of independent trains as part of its design criteria. Additionally, the applicant further addressed the redundancy of the ITS HVAC systems by stating in an RAI response (DOE, 2009fs) that the HVAC trains will be independent because components in one train cannot cause failure of both trains, as further discussed in SER Section 2.1.1.7.3.3.

**Means To Inspect, Test, and Maintain ITS HVAC SSCs**

The applicant stated that it will monitor and maintain ITS SSCs and, if required, take corrective actions to ensure the required reliabilities are achieved. The applicant's description of inspection, test, and maintenance programs are in SAR Section 1.9.1.13.

For ITS HVAC systems, the applicant identified independent trains and standby (or backup) components within individual trains. For example, in the CRCF, for the ITS HVAC subsystem serving ITS electrical equipment and battery rooms, SAR Figure 1.2.4-104 showed the system will be designed with backup units in case the operating units are not available due to servicing or maintenance. However, for the ITS HVAC subsystem serving the ITS switchgear and battery rooms in the EDGF, the applicant did not show standby air handling units in SAR Figure 1.2.8-26. The NRC staff requested additional information (RAI) regarding (i) how the maintenance on this subsystem would be performed without backup air-handling units and (ii) whether the unavailability of this subsystem during maintenance periods would adversely affect the reliability of ITS systems or subsystems in other facilities. In response to the NRC staff's RAI (DOE, 2009fo), the applicant stated that backup air-handling units are not necessary because (i) the EDGF ITS HVAC is a nonconfinement HVAC system and (ii) the PCSA accounts for the effect of regularly scheduled maintenance (e.g., procedures to limit specific activities such that those components and systems that are out of service are not relied on for safety during the maintenance periods).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's means to inspect, test, and maintain ITS HVAC SSCs using the guidance provided in YMRP Section 2.1.1.6. The applicant explained in its response to the NRC staff's RAI (DOE, 2009fo) that the maintenance unavailability of ITS components or systems (including ITS HVAC) is considered in the PCSA. The NRC staff finds that this is consistent with the information in the PCSA, as discussed above. Therefore, the staff finds that the applicant's PCSA accounted for the unavailability of ITS SSCs due to maintenance. Additionally, the applicant described in SAR Section 1.9.1.13 its inspection, testing, and maintenance programs for ITS SSCs using a reliability-centered maintenance approach. The NRC staff reviewed the applicant's description of its means to inspect, test, and maintain the ITS SSCs in SER Section 2.1.1.6.3.2.10, where the NRC staff finds, with reasonable assurance, that the reliability-centered maintenance program will provide an adequate means to ensure availability of safety functions of ITS SSCs. Additional discussions about the applicant's description of inspection, testing, and maintenance of SSCs is provided in SER Section 2.5.6. On the basis of the above evaluation, the NRC staff finds that the applicant adequately addressed the means to inspect, test, and maintain ITS HVAC SSCs.

**Ability of the ITS HVAC SSCs to Perform Their Intended Safety Functions**

The applicant specified nuclear safety design bases for the ITS heating, ventilation, and air conditioning (HVAC) systems in SAR Tables 1.9-3 (for the Canister Receipt and Closure Facility or CRCF), 1.9-4 (for the Wet Handling Facility or WHF), and 1.9-3 and 1.9-4 (for the Emergency Diesel Generator Facility or EDGF). The applicant stated that the safety functions of the ITS HVAC systems will be able to (i) filter air in the confinement areas to mitigate the consequences of a radionuclide release and (ii) provide cooling and ventilation of ITS electrical equipment and battery rooms in nonconfinement areas to support their safety functions (SAR Sections 1.2.2.3, 1.2.4.4, 1.2.5.5, and 1.2.8.3). SAR Section 1.2.2.3.8 lists the principal codes and standards used for design of the ITS HVAC equipment. As a part of the nuclear safety design bases, the applicant provided the controlling parameters for reliability of the HVAC system to mitigate the consequences of a radiological release.

**NRC Staff's Evaluation**

The NRC staff reviewed the HVAC system information using the guidance in YMRP Section 2.1.1.6. The NRC staff reviews the nuclear safety design bases (including the safety functions) of the HVAC system design in SER Section 2.1.1.7.3.3 to determine how the ITS HVAC systems will provide filtration to mitigate the consequences of a radionuclide release or provide cooling and ventilation to ITS electrical equipment and battery rooms. In addition, the NRC staff evaluates the controlling parameters for the nuclear safety design bases in SER Section 2.1.1.4.3.3.2.1. The NRC staff finds that the reliability the applicant quantified for the surface nuclear confinement ITS HVAC system is adequate for the system to perform its safety function of mitigating the consequences of a radiological release because (i) this reliability value is consistent with that of similar HVAC systems used in other nuclear facilities and (ii) the principal codes and standards for the design of the ITS HVAC systems are consistent with nuclear industry practices for HVAC systems with the same or similar safety functions.

**NRC Staff's Conclusion**

Based on the evaluations in SER Section 2.1.1.6.3.2.8.2.2, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(1), 63.112(e)(4), 63.112(e)(8), 63.112(e)(12), and 63.112(e)(13) for the HVAC system are satisfied because the applicant adequately considered the design of the ITS HVAC systems in its PCSA in the following areas: (i) means to limit concentration of radioactive material in air; (ii) means to control dispersal of radioactive contamination; (iii) ability of structures, systems, and components to perform their intended safety functions if an event sequence occurs; (iv) means to provide redundant systems necessary to maintain the capability of the HVAC system; and (v) means to inspect, test, and maintain SSCs ITS to ensure their continued functioning and readiness.

2.1.1.6.3.2.8.3 Transportation Systems Important to Safety

The applicant discussed the ability of the ITS transport and emplacement vehicle (TEV) and ITS Intrasite operations transportation equipment (site transporter, cask tractor, cask transfer trailer, and site prime mover) to perform their intended safety functions in SAR Section 1.9.1.

**ITS TEV**

The applicant identified the safety functions in SAR Tables 1.9-2 through 1.9-7 and the procedural safety control (PSC) (PSC–10) for the ITS TEV in SAR Table 1.9-10 that are designed to prevent an event sequence from occurring. The applicant will use PSC–10 to assess deviation of the (i) observed time that a waste form spends in each process area or in a given process operation; (ii) component failures per demand; and (iii) component failures per time period, from those used in the PCSA (SAR Table 1.9-10). As a part of this PSC, the applicant will determine the risk significance of these deviations. The applicant stated that these safety functions will be used to define specifications for the TEV design and validate TEV reliability. The purpose of the reliability validation is to ensure that the TEV will be functional and available through the preclosure period.

The applicant considered a spectrum of seismic events to address the TEV's ability to perform under seismic conditions. The applicant conducted a study to assess risk and quantify the mean frequency of seismic-related event sequences (BSC, 2008bg). This study identified three potential TEV failure scenarios: (i) derailment (frequency of failure of $1 \times 10^{-4}$ per year, Table 6.2.2); (ii) tipover (frequency of failure of $5.3 \times 10^{-7}$ per year, Table 6.2.2); and (iii) ejection of waste package from the shielded enclosure (frequency of failure of $2.3 \times 10^{-4}$ per year, Table 6.2.2). The TEV fragility estimates were expressed as probabilities of unexpected performance as a function of a ground motion parameter. The applicant provided an analysis (BSC, 2008co) to support the fragility calculations.

The applicant stated that the TEV maintenance plan will be developed and implemented using the approach described in SAR Section 5.6. The applicant further stated that the maintenance process will be centered on reliability and developed before the receipt and possession of HLW. Furthermore, the applicant stated that periodic tests will be performed at scheduled intervals to detect and replace parts subject to degradation before equipment deterioration reaches an unacceptable condition. The applicant stated that it will design the TEV in accordance with the ASME NOG–1–2004 (ASME, 2005aa).

The applicant stated that the TEV will include redundant design features. For example, the electrical enclosures aboard the TEV will be protected by redundant automatic fire detection and suppression systems (SAR Section 1.3.3.5.1.4). Additionally, redundant programmable logic control (PLC) components will be used to ensure high reliability and availability (SAR Section 1.3.3.5.2.3). Redundant or diverse design features for load-bearing components and braking processes will also be included in the design of TEV (SAR Section 13.2.1).

**NRC Staff's Evaluation**

The NRC staff reviewed the TEV information using the guidance in YMRP Section 2.1.1.6. Specifically, the NRC staff reviewed the reliability calculations the applicant performed Attachment H (BSC, 2008co) to assess the TEV's ability to withstand the seismic events. The NRC staff finds that the applicant's calculations are adequate because the calculations used methodologies that are consistent with the standard engineering practices for structural fragility assessment and site-specific seismic data (see SER Section 2.1.1.4.3.3.1.2 "Passive Structures, Systems, and Components Reliability for Seismic Events"). The NRC staff also finds that the calculations showing that the TEV cannot tip over at any credible ground motion level, even after the TEV's seismic event restraints fail, are reasonable because of the TEV's low center of gravity and wide base. According to the calculations, the only credible condition in which the waste package may be damaged is during waste package transfer from the waste package transfer trolley (WPTT) to the TEV at the docking station. The NRC staff finds that the waste package will not be breached during this credible condition because any impact to the waste package due to the TEV sliding would be much less than an impact velocity of 6 m/s [20 ft/s], a value at which the waste package would not be breached, as described in further detail in the NRC staff's event sequence evaluation in SER Section 2.1.1.4.3.4.1.1.

The NRC staff finds that the applicant's reliability analyses for the TEV ITS components are adequate because these analyses included all identified ITS components, such as the drive motors, drive shafts, wheels, gearboxes, door components (e.g., actuators, locks, and hinges), hardwired interlock circuitry, and seismic restraints to ensure the TEV's availability and reliability to perform its intended safety functions. The NRC staff also reviewed how the applicant applied a component reliability assessment to show the TEV's overall ability to perform its intended safety functions (BSC, 2008bk) and finds that the applicant's approach is adequate because the applicant (i) represented the TEV design in the fault trees and (ii) included component reliability values on the basis of available component reliability databases. Therefore, the NRC staff determines that the applicant's reliability estimates adequately show the TEV's ability to perform the intended safety functions.

The NRC finds that the applicant adequately described the means to inspect, test, and maintain TEV ITS, as necessary, to ensure their continued function and readiness because the applicant described the consideration of maintenance in the TEV design, as detailed in Section H6.2.2 (BSC, 2008co), including (i) the restraint system on the TEV chassis to facilitate maintenance and inspection at regular intervals and (ii) construction of the TEV's wheels with a lower surface hardness than the drift rails to induce wear or damage to the wheels rather than to the drifts rails, which are more difficult to repair.

**Surface Transportation Equipment ITS**

The applicant identified 14 distinct safety functions (SAR Tables 1.9-2 through 1.9-7) and 2 procedural safety controls (PSCs) (i.e., PSC–2, and PSC–10, SAR Table 1.9-10) that are necessary for the ITS surface transportation equipment to prevent event sequences. The surface transportation equipment includes the site transporter, cask tractor, cask transfer trailer, and site prime mover. The applicant stated that these safety functions will be used to define design specifications for the surface transportation equipment. The applicant stated that it will require qualified vendors, using applicable sections of codes and standards, to identify and define operational requirements and limits, as described in Section 1.1 (DOE, 2009ez,fg).

The applicant considered a spectrum of seismic events to address the ability of the site transporter to perform under seismic conditions. The applicant conducted a study to quantify the mean frequency of seismic-related event sequences (BSC, 2008bg). The study identified two potential site-transporter scenarios that could damage the waste package: (i) tipover (including those at locations of five percent grade in the direction of travel and two percent grade transversely) and (ii) impacts to the waste package due to the site transporter sliding into a wall. The assessment required site transporter fragility estimates (BSC, 2008co) that are probabilities of unexpected performance of the site transporter as a function of a ground motion parameter. For other site transportation equipment, the applicant estimated the fragility related to tipover failure on the basis of conservative engineering judgments supported by general earthquake experience with railcars and truck trailers. The applicant stated that the transportation cask is designed to withstand impacts associated with tipover events. Accordingly, the applicant did not identify any safety function related to tipover for the cask tractor, cask transfer trailer, or site prime mover.

The applicant identified procedural safety controls (PSCs) to ensure the transportation equipment's ability to remain in a safe state under certain conditions. The applicant determined that PSC–2 will be necessary to limit spurious movement, potentially causing a collision or tipover during the operation of the site transporter, cask tractor, cask transfer trailer, or site prime mover. The applicant also identified PSC–10, which will compare both residence times in a given process operation and the actual SSCs failure rates with the assumed values used in the PCSA. The applicant stated that it will analyze any significant deviation to determine risk significance.

The applicant described its plans for inspection, testing, and maintenance of the equipment to assess the availability of the surface transportation equipment to perform its intended safety functions. The applicant stated that the maintenance process will be centered on reliability (DOE, 2009dk). In addition, the applicant stated that periodic tests will be performed at scheduled intervals to detect and replace parts subject to degradation before equipment deterioration reaches an unacceptable condition. The applicant stated that, in the event of a malfunction or warning-light condition, the site transporter and site prime mover will be immediately recovered, removed from service, and properly repaired. In addition, in SAR Section 1.13, the applicant provided a description of plans for environment, equipment, and seismic qualification programs that can validate the availability of the surface transportation equipment during the preclosure period. The applicant stated that it will develop and conduct the programs following guidelines from accepted industry standards, such as IEEE 323–2003, and IEEE 344–2004 (IEEE, 2004aa; IEEE, 2005aa). The applicant's plans include conditioning monitoring to determine whether the qualified equipment will remain in a qualified condition.

**NRC Staff's Evaluation**

The NRC staff reviewed the important to safety (ITS) surface transportation equipment information using the guidance in YMRP Section 2.1.1.6. The NRC staff finds that the applicant's assessment of the surface transportation equipment's overall availability to perform the intended safety functions (BSC, 2008au) is acceptable because the applicant described in sufficient detail the functions of the site transporter, cask tractor, cask transfer trailer, and site prime mover in the fault trees and included component reliability values on the basis of available component reliability databases with conservative factors of safety. On the basis of this evaluation, the NRC staff finds that the applicant's reliability estimates provide a reasonable technical basis to demonstrate the intrasite transportation equipment's ability to perform the intended safety functions, as described below.

The NRC staff reviewed the applicant's assessment Attachment G (BSC, 2008co) of the site transporter's ability to withstand the seismic events during operations in the Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Receipt Facility (RF), and during transport of an aging overpack (AO) to and from the Aging Facilities. The NRC staff finds that the applicant's assessment is acceptable because the applicant (i) provided a sufficient basis to support the conclusion that the governing scenario for the site transporter is the seismically-induced sliding of the site transporter into a concrete wall, potentially leading to a breach of the waste package due to induced stresses on the waste package and (ii) estimated the sliding displacement of the site transporter to be smaller than the minimum clearance from the concrete wall to the site transporter by a factor-of-safety of more than 2.5. Additionally, the applicant has specified the controlling parameters in the nuclear safety design bases to limit the probability of occurrence of such an event to beyond the Category 2 event sequence (SAR Tables 1.9.3, 1.9-4, and 1.9-5). The NRC staff also finds that it is acceptable for the applicant not to assign any safety function related to tipover for the cask tractor, cask transfer trailer, or site prime mover because the applicant stated that it would use the transportation cask design to mitigate potential effects caused by seismically-induced tipover events.

The NRC staff reviewed the applicant's information related to PSC–2 (i.e., securing transportation equipment prior to waste-handling operation) and finds that PSC–2 is adequate because PSC–2 requires deactivating the surface transportation equipment with the brakes applied and detaching the site prime mover when performing waste handling, so as to eliminate the potential for spurious movement during canister loading and unloading activities. The applicant would further augment the effectiveness of PSC–2 by requiring redundant and independent verification of the deactivation and detachment steps before the waste loading and unloading operations.

The NRC staff finds that the applicant's description of the plans for inspection, testing, and maintenance of the equipment to ensure the availability of the surface transportation equipment to perform its intended safety functions is acceptable because the applicant described how it will incorporate redundancy, notifications, safety measures, and qualification considerations to ensure proper detection, repair, and maintenance-related activities.

**NRC Staff's Conclusion**

Based on the NRC staff evaluations in SER Section 2.1.1.6.3.2.8.3, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112 (e)(8) and 10 CFR 63.112(e)(13) for the TEV and the surface transportation equipment are satisfied because the applicant's PCSA adequately considered (i) the ability of the TEV and surface transportation equipment to perform their intended safety functions during an event sequence and (ii) the means to inspect, test, and maintain the TEV and the surface transportation equipment.

2.1.1.6.3.2.8.4 Electrical Components and Emergency Power Systems Important to Safety

The applicant provided information on electrical components and ITS electrical power systems (EPS) in SAR Sections 1.2.4, 1.2.5, 1.2.8, 1.4.1, 1.4.2, 1.9.1.8, 1.9.1.11, and 1.13. The NRC staff notes that the applicant describes electrical power systems for radiation protection as "ITS electrical power systems" rather than as ITS emergency power systems–this section uses this same terminology as the applicant; this is further discussed in BSC (2008cq). The applicant also provided information in SAR Sections 1.9.1.12 and 1.9.1.13 on the redundant systems for the ITS EPS SSCs and the means to maintain, inspect, and test the ITS EPS SSCs and, in particular, ITS diesel generator SSCs. As described in this section, the NRC staff focused its review on the performance of ITS EPS, which includes ITS electrical power distribution systems, ITS diesel generators, ITS diesel generator mechanical support systems, ITS direct-current (battery) power, and ITS uninterruptable power supplies (UPS). The ITS distribution system distributes power to ITS loads within the GROA. The objective of the review was to determine whether the ITS EPS SSCs can perform their intended safety functions (e.g., power ITS HVAC and other systems) and that the ITS diesel generators can provide reliable and timely ITS electrical power when required.

**Means to Inspect, Test, and Maintain ITS Electrical Power Systems**

The applicant stated that it will use Regulatory Guide 1.9 (NRC, 2007ag) and IEEE 387–1995 (IEEE, 1996aa) to design the ITS diesel generators (DOE, 2009fc). These codes and standards include provisions for regular maintenance, inspections, and tests of the ITS diesel generators. Additionally, the applicant stated in SAR Section 5.5 (Table 5.5-1) that it will perform periodic functional tests [e.g., tests described in Regulatory Guide 1.118 (NRC, 1995aa) for electrical distribution systems]. Furthermore, the applicant stated that it will use reliability-centered maintenance methodology to develop maintenance programs, including periodic inspecting and testing, to ensure the availability of the ITS EPS.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of inspection, testing, and maintenance of the ITS EPS using the guidance in YMRP Section 2.1.1.6. The NRC staff reviewed the applicant's description of its means to inspect, test, and maintain the ITS SSCs in SER Section 2.1.1.6.3.2.10, where the NRC staff finds that the reliability-centered maintenance program will provide a reasonable means to ensure availability of safety functions of ITS SSCs. The NRC staff's evaluation of the applicant's description of inspection, testing, and maintenance of the SSCs is provided in SER Section 2.5.6, where the NRC staff finds that the applicant adequately described plans to conduct maintenance, surveillance, and periodic testing that would be implemented before the applicant receives, processes, stores, or disposes of high-level radioactive waste. The NRC staff finds that the applicant's description of plans to inspect, test, and maintain ITS EPS are acceptable because the applicant described how it will (i) perform periodic functional tests consistent with tests described in Regulatory Guide 1.118 (NRC, 1995aa); (ii) use a reliability-centered maintenance program; and (iii) use codes, standards, and regulatory guides for regular maintenance, inspections, and tests of the ITS diesel generators and the electrical distribution systems that are consistent with the standard engineering practices for similar electrical power distribution systems and diesel generators used in the nuclear industry.

**Reliable and Timely Electrical Power**

The applicant provided a calculation to show that the ITS heating, ventilation, and air-conditioning (HVAC) high efficiency particulate air (HEPA) filtration function could be lost for up to 8 hours during or after a bounding Category 2 event sequence without resulting in offsite public doses exceeding preclosure performance objectives (DOE, 2009fp). Because of this calculation, the applicant stated (DOE, 2009fp) that the ITS diesel generators will be designed to start and accept load within the 8-hour time period after a loss of offsite power. The applicant stated that it will follow IEEE–387–1995 Section 4.1 to determine the time interval between receipt of a "start signal" by the ITS diesel generator SSCs and the availability of power from the ITS diesel generators (DOE, 2009fp). The applicant further stated that this time interval is expected to be less than 3 minutes.

The applicant discussed the use of ITS uninterruptable power supplies (UPS) in SAR Section 1.4.1.3.1 and in its response to the NRC staff's RAI (DOE, 2009gj). The applicant stated that the preclosure safety analysis does not require any ITS loads to be fed from the ITS UPS upon the loss of offsite power; however, the UPS does provide additional flexibility and capability for the EPS (e.g., improve the voltage regulation of the EPS and provide a contingency power supply beyond the ITS diesel generator) (DOE, 2009gj). The applicant further stated that it will develop the ITS UPS maximum power requirements allowing for future expansion of the ITS UPS.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on the timeliness and reliability of electrical power using the guidance in YMRP Section 2.1.1.6. The applicant estimated the time interval between receipt of a "start signal" by the ITS diesel generator and the availability of power from the ITS diesel generators to be less than 3 minutes (DOE, 2009fp) using the criteria provided in IEEE–387–1985 (IEEE, 1996aa) standard. The NRC staff finds that the ITS electrical power systems (EPS) design will be able to provide timely and reliable electrical power because (i) the ITS diesel generator SSCs will be capable of starting and accepting intended loads within less than 3 minutes (DOE, 2009fp) using IEEE–387–1995 (IEEE, 1996aa) to estimate the diesel generator start time, which is consistent with nuclear industry practice; and (ii) the applicant estimated diesel generators are needed to start and accept the intended loads within an 8-hour time period after a loss of offsite power during an event sequence. The applicant's estimate is based on a conservative calculation that assumes the HVAC is unavailable for the first 8 hours after the release; and thus, radionuclide releases during the first 8 hours are assumed to be unfiltered.

The applicant's information on system redundancy is evaluated next and provides further support for the reliability of the ITS EPS.

**System Redundancy**

The applicant stated in SAR Section 1.9.1.12 that redundant, independent, and physically separated systems will be available for ITS diesel generators. In addition, the applicant stated that each ITS diesel generator will have a rated load-carrying capacity of 5 MVA and the estimated approximate demand is 3.9 MVA; hence, there will be a nearly 25 percent design margin (DOE, 2009dk). Redundant trains for ITS diesel generators; multiple ITS diesel generator mechanical support systems; major ITS distribution SSCs [up to and including ITS motor control centers (MCCs); and ITS load centers within the Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Emergency Diesel Generator Facility (EDGF), and the non-ITS Receipt Facility (RF)] were described in the SAR and the applicant's RAI response (DOE, 2009dk).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on system redundancy using the guidance in YMRP Section 2.1.1.6. The NRC staff finds that the applicant's consideration of redundant systems for ITS EPS is acceptable because the applicant's design for the ITS diesel generators includes (i) a 25 percent load-carrying design margin, (ii) redundant trains for ITS diesel generators, and (iii) multiple ITS diesel generator mechanical support systems as necessary to provide adequate capacity and capability of utility systems ITS.

The Emergency Diesel Generator Facility (EDGF) will include identical ITS SSCs for distributing and controlling ITS power to the CRCF, WHF, and non-ITS RF; however, the RF power-distribution channel can be isolated from the ITS switchgear in the EDGF, and this power connection will not be automatically restored when normal or emergency power becomes available after a power outage. The NRC staff determines that the applicant's design configuration for the ITS EPS includes a redundant central ITS diesel generator and main distribution system from which power is distributed through multiple electrical connections and physical power flow paths to multiple redundant combined ITS EPS/ITS HVAC trains in specified facilities (for a detailed discussion, see SER Section 2.1.1.7.3.6). On the basis of this review, the NRC staff finds that the applicant adequately described redundant features for the major ITS EPS structures, systems, and components, including the ITS diesel generators, ITS EPS main switchgear, ITS Motor Control Centers (MCCs), and ITS load centers located in each facility the ITS EPS will serve.

The NRC staff also finds that the load-carrying capacity for ITS diesel generators, which includes a 25 percent load-carrying design margin, is acceptable because the design provides spare capacity for future growth that is consistent with established engineering practice in the nuclear power industry.

**Ability to Perform Intended Safety Functions**

The applicant identified the performance requirements for the ITS electrical power distribution systems and diesel generators for its CRCF and WHF operations in SAR Tables 1.9-3 and 1.9-4. SAR Table 1.4.1-1 listed the related design criteria for the ITS electrical power distribution systems and diesel generators. The controlling parameters and values for the ITS power generation and distribution systems were also specified in SAR Table 1.4.1-1. The applicant stated that the performance requirement for the SSCs distributing electrical power to ITS surface nuclear confinement HVAC systems in the CRCF is an allowance of 0.007 failures during a 720-hour period following a radionuclide-release event. Similarly, for the CRCF, there is an allowance of 0.3 failures during a 720-hour period following a radionuclide-release event for the ITS diesel generator SSCs to supply ITS electrical power. The applicant stated it would use Regulatory Guide 1.89 for ITS SSC environmental qualification.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information on the ability of the ITS electrical power distribution systems and diesel generators to perform intended safety functions using the guidance in YMRP Section 2.1.1.6. The NRC staff finds that the applicant's information is acceptable because the applicant's inclusion of redundant systems for the ITS diesel generator improves the ability of the ITS diesel generators and electrical power distribution systems to provide electrical power to the needed ITS SSCs so they can perform the intended safety functions if the offsite power is lost during or after an event sequence. Additionally, the NRC staff finds that the ITS electrical power distribution systems and ITS diesel generators will perform their intended safety functions because the applicant plans to follow the guidelines in Regulatory Guide 1.89 (NRC, 1984aa) and IEEE 323–2003 (IEEE, 2004aa). Regulatory Guide 1.89, which concerns environmental qualification of electrical equipment important to safety for nuclear power plants, is equally applicable for these systems, which provide the same or similar functions. These references are used in the nuclear power industry to seismically and environmentally qualify ITS active electrical equipment (SAR Section 1.13).

**NRC Staff's Conclusion**

Based on the NRC staff evaluations in SER Section 2.1.1.6.3.2.8.4, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(8), 63.112(e)(11), 63.112(e)(12), and 63.112(e)(13) for the electrical components and the ITS EPS are satisfied because the applicant's PCSA adequately considered electrical components and ITS EPS in the following areas: (i) ability to perform their intended safety functions during an event sequence; (ii) reliable and timely emergency power to instruments, utility service systems, and operating systems important to safety; (iii) redundant systems necessary to maintain, with adequate capacity, the capability of utility services that are important to safety; and (iv) means to inspect, test, and maintain electrical components and ITS EPS to ensure their continued function and readiness.

2.1.1.6.3.2.8.5 Fire Protection Systems Important to Safety

The applicant provided design descriptions and safety classifications for the double-interlock preaction (DIPA) sprinkler systems. The descriptions and safety classifications were provided in SAR Sections 1.6, 1.7, and 1.9, and in the applicant's response to an NRC staff RAI (DOE, 2009fr).

The applicant stated that it will rely on the DIPA sprinkler systems to protect moderator-controlled areas within the geological repository operations area (GROA) [e.g., Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF)] and subsequently identified these systems as ITS (SAR Table 1.9-1 and SAR Section 1.4.3.2.1.2). The ITS DIPA systems will include spot detectors, sprinkler piping, sprinkler heads, solenoids, sprinkler valves, and a main actuation panel (SAR Figure 1.4.3-21).

The NRC staff evaluated the applicant's basis for the safety classification of the detection and suppression systems. The NRC staff also evaluated the applicant's event-sequence analyses to identify sequences initiated by fire events and assess the role of the DIPA to prevent criticality events.

**Selection of System**

The applicant classified the double-interlock preaction (DIPA) sprinkler systems as ITS because they provide fire protection in specific facilities and minimize the accidental discharge of water. The accidental discharge of water in the Canister Receipt and Closure Facility (CRCF) and Wet Handling Facility (WHF) raises criticality concerns for event sequences that include the potential for a canister breach followed by a spurious activation of the sprinkler system with the potential for water (a moderator) to enter the breached canister. The DIPA sprinkler system is designed to suppress a fire while minimizing accidental discharge of water. The applicant stated that the DIPA systems were selected to provide an additional layer of facility protection because they were not credited in the PCSA due to other fire protection features (e.g., passive fire barriers) at the facilities.

The PCSA showed that a moderator could be introduced into a container following a canister breach and a subsequent spurious activation of the sprinkler system. According to the applicant, the DIPA systems will require a positive fire detection interlock to be made (e.g., confirmation of a fire from a series of fire and smoke detectors) in conjunction with sufficient heat buildup to trigger an actual sprinkler head. As a result, failure of the piping or of the detection system alone will not be sufficient to release water into an area.

**NRC Staff's Evaluation**

The NRC staff reviewed the DIPA sprinkler system design using the guidance in YMRP Section 2.1.1.6 to evaluate the selection of the DIPA system as a suitable means to provide the designated ITS function of fire suppression. On the basis of this evaluation, the NRC staff finds that the applicant's use of the DIPA systems is acceptable because they are commonly used in areas (e.g., telecommunication centers) where spurious water delivery is undesirable. Additionally, these systems are based on standard designs, which use components that have been tested and listed by the Underwriter Laboratories (UL) and Factory Mutual for the intended function of delivering water upon positive identification of fires while minimizing spurious activation of sprinkler systems. The UL is a nationally and internationally recognized independent not-for-profit product safety testing and certification organization. The NRC concludes the fire suppression components with a UL listing carry a level of reliability suitable for GROA applications.

**Ability to Perform Intended Safety Function**

The applicant stated that the design of the DIPA system would be consistent with National Fire Protection Association (NFPA) NFPA 13 (NFPA, 2007ab) and NFPA 72 (NFPA, 2007af) standards. The applicant concluded that the low probability of inadvertent water introduction from a DIPA results in a low overall probability of moderator intrusion in criticality-related event sequences. As stated in SAR Table 1.4.3-2, the applicant cited design failure probabilities of $10^{-6}$ over a 720-hour (30-day) period following radionuclide release in the CRCF, and $6 \times 10^{-7}$ over a 720-hour (30-day) period following radionuclide release in the WHF as its nuclear safety design bases.

On the basis of the fault tree analysis provided in the response to an NRC staff RAI (DOE, 2009fr), the applicant stated that the mean probability of failure of spurious activation of the double-interlock sprinkler systems is $2 \times 10^{-7}$ over a 720-hour (30-day) period following radionuclide release. The applicant further stated that this probability is less than the design basis failure probability provided in SAR Table 1.4.3-2.

**NRC Staff's Evaluation**

The NRC staff reviewed the design information using the guidance in YMRP Section 2.1.1.6 to assess the ability of the DIPA system to perform its intended safety function of fire suppression. The NRC staff finds that the applicant's description of the DIPA system to perform its safety function is acceptable because (i) the DIPA design uses applicable NFPA standards and (ii) the applicant's fault tree analysis of the PCSA demonstrates that the proposed system will achieve the design bases probability of inadvertent introduction of fire suppression water into a canister (SAR Table 1.4.3-2).

**Assessment of Continued Functionality and Means to Inspect, Test, and Maintain**

The applicant stated that the double-interlock preaction (DIPA) sprinkler systems will be designed, installed, and maintained in accordance with NFPA standards. The applicant stated that it will use the suppression system design guidelines outlined in NFPA 13 for installation of sprinkler systems (NFPA, 2007ab), in conjunction with scheduled maintenance in accordance with NFPA 25 for the inspection, testing, and maintenance of water-based fire protection systems (NFPA, 2008ac) to ensure reliable suppression systems are provided. In addition, the applicant stated that the fire detection components used in the DIPA systems will be designed and inspected in accordance with NFPA 72 National Fire Alarm Code (NFPA, 2007af).

**NRC Staff's Evaluation**

The NRC staff reviewed the design information using the guidance in YMRP Section 2.1.1.6 to assess the functionality of the designed system and the applicant's ability to properly inspect, test, and maintain the designed system. The NRC staff finds that the applicant's description of the standard DIPA systems is acceptable because it explained that these systems will be designed in accordance with the nationally accepted industry codes and standards in NFPA 25 13, NFPA 72 (NFPA, 2007ab; NFPA, 2008ac). These code and standards have long been used at nuclear power plants and DOE nuclear facilities as the fire sprinkler and alarm standards. The NRC staff also determines that the applicant's plan to follow the installation, inspection, and maintenance procedures in the referenced standards is acceptable because this approach will help ensure continued functionality of these systems.

**NRC Staff's Conclusion**

Based on the NRC staff evaluations in SER Section 2.1.1.6.3.2.8.5, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(8), 63.112(e)(9), and

63.112(e)(13) for the ITS fire protection system are satisfied because the applicant's PCSA adequately considered the fire detection and suppression systems important to safety in the following areas: (i) ability of the fire protection systems to perform their intended safety function, assuming the occurrence of sequences; (ii) fire detection systems and appropriate suppression systems; and (iii) means to inspect, test, and maintain the DIPA sprinkler systems important to safety to ensure their continued function and readiness.

2.1.1.6.3.2.8.6 Transportation, Aging, and Disposal Canisters

The applicant provided a description of its approach (including the credited safety functions and procedural safety controls for ITS SSCs) to illustrate how ITS SSCs will perform their intended safety functions during the occurrence of event sequences in SAR Section 1.9.1.8. The applicant classified the proposed transportation, aging, and disposal (TAD) canister as ITS because the TAD canisters will contain the spent nuclear fuel (SNF) during the occurrence of an event sequence (SAR Tables 1.9-2 through 1.9-6). SAR Tables 1.9-2 through 1.9-6 provide the design bases for a "representative canister," which includes the TAD canister. SAR Table 1.5.1-7 provided the preclosure nuclear safety design bases for the TAD canister and specified the probability of breach (loss of containment) during both structural and thermal challenges, which are consistent with the design bases in SAR Tables 1.9-2 through 1.9-6.

SAR Section 1.7.2.3.1 covered the methodology for determining passive component reliability and discussed loss of containment of a waste form from the canister (e.g., TAD canister) due to structural challenges. The structural challenges consisted of vertical and off-axis drop, tipover, slap-down, and horizontal drops. The applicant performed explicit finite element analyses, using a model of a representative canister, to simulate different structural challenges. As discussed in BSC (2008ac,cp), the representative canister used the average dimensions of several existing dual-purpose canisters (DPCs), naval canisters, and TAD canisters. The material used for the representative canister was a stainless steel alloy, consistent with that specified in the TAD canister performance specifications (DOE, 2008aa). The characteristics of the representative canister, which included the characteristics of the TAD, were used in making the numerical (finite element) models to evaluate the TAD canister's reliability (BSC, 2008cp). The applicant determined the canister failure probabilities by utilizing a fragility curve for the stainless steel material along with the maximum effective plastic strains obtained from the finite element analyses.

The applicant also classified the aging overpack and casks (e.g., transportation casks and shielded transfer casks) as ITS due to the shielding they provide from the spent fuel (e.g., spent fuel in TAD canisters) during both structural and thermal challenges (SAR Tables 1.9-2 through 1.9-6). SAR Section 1.7.2.3 covered the methodology for determining passive component reliability and discussed loss of shielding of the aging overpack and casks due to structural and thermal challenges.

**NRC Staff's Evaluation**

The NRC staff reviewed the TAD canister information using the guidance in YMRP Section 2.1.1.6. In addition, the NRC staff used the review results from SER Section 2.1.1.4, which focuses on the development and quantification of event sequences, to assist in its evaluation.

The NRC staff reviewed the capability of the representative canister to provide containment for event sequences that challenge the structural integrity of the canister. The characteristics of the representative canister also include characteristics of the TAD. The NRC staff's evaluation of the finite element analyses discussed in BSC (2008ac,cp) is presented in SER Section 2.1.1.4.3.3.1.1. The NRC staff finds in SER Section 2.1.1.4.3.3.1.1 that the finite element analyses results are acceptable because the applicant used the computer software LS-DYNA, which is consistent with standard engineering practices for modeling dynamic responses of mechanical structures. Based on the finite element analysis results, the NRC staff finds that the representative canister will have the ability to withstand the structural challenges and perform the intended safety functions during an event sequence. The NRC staff also finds that the analysis results for the representative canister are applicable to the proposed TAD canister because, according to the applicant, the proposed TAD canister will have dimensions, material, and weight requirements similar to those of the representative canister. Accordingly, the NRC staff finds that a TAD canister, similar to the representative canister, would have similar performance and, therefore, the ability to perform its intended safety functions during an event sequence.

The NRC staff also reviewed the capability of the aging overpacks and casks to provide shielding for event sequences that challenge the structural integrity of the canister. The NRC staff's evaluation of the finite element analyses discussed in BSC (2008ac,cp) is presented in SER Section 2.1.1.4.3.3.1.1. The NRC staff finds in SER Section 2.1.1.4.3.3.1.1 that standard engineering practices and modeling techniques were used appropriately for estimating the reliability of the aging overpacks and casks to withstand the structural and thermal challenges and perform the intended safety functions during an event sequence.

**NRC Staff's Conclusion**

Based on the NRC staff evaluation above, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(8) for the TAD canister, the aging overpacks, and casks are satisfied because the applicant (i) adequately considered the TAD canister's ability to contain the SNF during the occurrence of an event sequence in the TSPA (ii) adequately considered the ability of the aging overpacks and casks to provide shielding during the occurrence of an event sequence in the TSPA and (iii) evaluated the abilities of the representative canister, aging overpack and casks to perform their intended safety functions during an event sequence.

2.1.1.6.3.2.8.7 Waste Packages

The applicant provided information relative to waste package design and performance to illustrate that the waste package will have the capability to perform its intended safety functions if event sequences occur. This information was presented in SAR Sections 1.2.1.4.1, 1.2.4.2.3, 1.3.1.2.5, 1.5.2, and 2.3.6.

The applicant proposed to use the waste package as an engineered barrier for disposal of commercial spent nuclear fuel (SNF), high-level waste (HLW), and DOE and naval SNF and classified the waste package as ITS because it will be relied upon to prevent radioactive gas or particulate releases during normal operations and Category 1 and Category 2 event sequences. The applicant defined a list of safety functions (SAR Table 1.5.2-6) that the waste packages will be required to perform. The waste package design will have six configurations (SAR Table 1.5.2-1): (1) 21–PWR/44–BWR TAD (capacity: one TAD canister containing either 21 pressurized water reactor fuel assemblies or 44 boiling water reactor fuel assemblies); (2) 5–DHLW/DOE Short Codisposal (capacity: five short HLW canisters and one short DOE SNF canister); (3) 5–DHLW/DOE Long Codisposal (capacity: five long HLW canisters and one long DOE SNF canister); (4) 2-MCO/2-DHLW (capacity:  two DOE MCOs and two long HLW canisters); (5) Naval Short (capacity:  one short naval SNF canister); and (6) Naval Long (capacity:  one long naval SNF canister).  The applicant presented structural analyses for three waste package configurations:  21–PWR/44–BWR TAD, 5–DHLW/DOE Short Codisposal, and Naval Long.  The applicant stated that the other three waste packages will be qualified using the methodology in SAR Section 1.5.2.1.1, and an Administrative Control is proposed as a subject for a license specification in SAR Section 5.10 to ensure that each waste package configuration is fully analyzed and qualified for repository operations prior to use Enclosure 1 (DOE, 2009er).

The applicant evaluated the waste package performance using analytical methods, including elastic-plastic finite element analyses and conduction and radiation analyses.  In its response to an NRC staff RAI, the applicant provided representative finite element analysis for three waste package configurations:  (i) 21–PWR/44–BWR TAD (capacity:  one TAD canister containing either 21 pressurized water reactor fuel assemblies or 44 boiling water reactor fuel assemblies); (ii) 5–DHLW/DOE Short Codisposal (capacity:  five short HLW canisters and one short DOE SNF canister); and (iii) Naval Long (capacity:  one long naval SNF canister) (DOE, 2009er).  For the structural analysis, the applicant calculated the stress intensities in the waste package outer corrosion barrier and invoked the tiered screening criteria method (SAR Table 1.5.2-10) that was based on elastic-plastic analysis methods identified in ASME 2001, Section III, Appendix F (ASME, 2001aa).  For the thermal analysis, the applicant calculated time histories of the radial temperature distributions in the waste package and compared them to the temperature limits for accident conditions.  In addition, the applicant developed reliability estimates for the outer corrosion barriers using the energy absorption methodology.  Based on these estimates, the applicant calculated the probabilities of radionuclide release from waste packages during event sequences.  Also, the applicant stated that procedural safety controls (PSCs) are not needed for the waste package to prevent or mitigate event sequences.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's waste-package-related information using the guidance in YMRP Section 2.1.1.6.  The NRC staff finds the waste package information provided by the applicant in SAR Sections 1.2.1.4.1, 1.2.4.2.3, 1.3.1.2.5, 1.5.2, and 2.3.6 and the associated RAI response (DOE, 2009er) is acceptable because it describes an adequate basis to demonstrate that waste packages will have the ability to perform their intended safety functions during event sequences.

The NRC staff finds that the waste package configurations for (i) 21–PWR/44–BWR TAD; (ii) 5-DHLW/DOE Short Codisposal; and (iii) Naval Long SNF will have the ability to perform the intended safety functions because the applicant's analyses demonstrated, using standard engineering methods, that the calculated stresses in the waste package outer corrosion barrier satisfy the tiered screening criteria in SAR Table1.5.2-10, which are based on elastic-plastic analysis methods identified in ASME 2001, Section III, Appendix F (ASME, 2001aa).  The NRC staff determines that the applicant's structural stress calculations for the three WP configurations are consistent with standard industry practices and applicable codes and standards.  For example, the applicant's tiered screening criteria method (SAR Table 1.5.2-10) for the allowed stresses beyond a material's elastic range is a deterministic approach based on elastic-plastic analysis methods provided in "Rules for Evaluation of Service Loadings with Level D Service Limits" ASME 2001, Section III, Appendix F (ASME, 2001aa).  For this method, the wall-average total stress intensity value (twice the maximum shear stress) is derived from the analytical or finite element analyses and is compared against failure criteria that are based on the material ultimate tensile strength.

The NRC staff also finds that the applicant's thermal analysis results showed that the calculated temperature inside the waste package will stay below the temperature limit for accident conditions. The NRC staff finds that the waste package thermal controls will be such that the fuel cladding temperature will be sufficiently low to prevent cladding failure (i.e., commercial SNF cladding prevents fission product and actinide release).

**NRC Staff's Conclusion**

Based on the NRC staff evaluation above, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(8) for the waste packages are satisfied because the applicant's PCSA adequately considered the waste packages' ability to perform its safety functions during the occurrence of an event sequence.

2.1.1.6.3.2.9 Radioactive Waste and Effluents Control

The applicant provided information in SAR Sections 1.9.1.10 and 1.4.5.1.1 regarding radioactive waste and effluents control. The applicant provided information to explain how the PCSA addressed (i) liquid and solid waste management systems to handle the expected volume of potentially radioactive liquid waste generated during normal operations; (ii) Category 1 and 2 event sequences; and (iii) off-gas treatment, filtration, and ventilation systems for control of airborne radioactive effluents. The applicant provided information in SAR Sections 1.9.1.10 and 1.4.2 regarding the termination of operations and evacuation of personnel.

**Liquid Low-Level Waste Management**

The applicant stated that it will include a subsystem to collect low-level radioactive waste (LLW) liquids and potentially radioactive waste liquids in the waste-handling facilities. Liquid low-level waste will include effluent from decontamination activities, actuation of a fire suppression system that could generate water contaminated with radioactive material, and liquids deposited into drain or sump collection systems that gather water from any other activities that could generate low-level waste (SAR Section 1.4.5.1.1.2). The applicant stated that while liquid waste is expected to be free of radioactive contamination, all waste water will be collected in the equipment drainage system in each facility and monitored for radioactive contamination before being managed as nonradioactive industrial wastewater. The applicant stated that no radioactive liquid effluents will be discharged from the repository to the environment. Should the liquid waste from any of these sources be contaminated, the liquid waste will be transferred to a liquid waste collection tank, then processed to remove solid radioactive waste. The resulting solids will be managed as solid low-level waste. In addition, the applicant stated that all liquid waste facilities in which liquid low-level waste is detected will be decontaminated before normal activities are restored (SAR Section 1.4.5.1.1.2). The applicant identified these liquid waste facilities as non-ITS.

The applicant described the capacity of the effluent systems to contain the largest credible volume of fire-water discharge from fire suppression systems (DOE, 2009fo). Based on the criteria in NFPA 801; Section 5.10 (NFPA, 2003aa), which provides standards for fire protection for facilities handling radioactive materials, the applicant determined a maximum of 34,069 L [9,000 gal] of effluent for use in sizing the liquid effluent collection system for fire-water discharge. Additionally, the applicant stated that the holding tanks for all five facilities will be sufficiently large to concurrently accommodate the following factors: design margins, freeboard capacity, a week's capacity for custodial maintenance and decontamination, sampling tank, and rounding errors. The collection tanks located outside the Initial Handling Facility (IHF), Canister

Receipt and Closure Facility (CRCF), and Receipt Facility (RF) will provide a cumulative working volume of 58,901 L [15,560 gal]. Similarly, the WHF will have two collection tanks, each with a working volume of 57,917 L [15,300 gal]. The Low-level Radioactive Waste Facility (LLWF) will have two collection tanks and one process tank, each with a capacity similar to those of the IHF, RF, and CRCF {i.e., 57,917 L [15,300 gal]}. The applicant stated that the working volume of each of these three tanks will be 86,875 L [22,950 gal] after accounting for freeboard capacity, design margin, and rounding error.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's descriptions of its liquid low-level waste management system using the guidance in YMRP Section 2.1.1.6 to determine whether the applicant's PCSA included adequate evaluation of radioactive waste and radioactive effluents and whether the application adequately described prompt termination of operations and evacuation of personnel during an emergency. The NRC staff finds the applicant adequately described its plans for prompt termination of operations and evacuation of personnel because the applicant provided descriptions of: (i) the radiation and radiological monitoring systems for radioactive effluents (reviewed and found acceptable in SER Section 2.1.1.6.3.2.4); (ii) the use of monitoring information to alert personnel of any need to evacuate specific areas and facilitate a controlled termination of operations (SAR Section 1.9.1.10); and (iii) operational control plans to control and monitor systems and devices for the surface and subsurface facilities, including the Digital Control and Management Information Systems (DCMIS) that would provide a redundant high-speed communication network between operators and monitors, alarms, and the ability to control and terminate operations, and the transmission of data offsite (reviewed and found acceptable in SER Section 2.1.1.2.3.6.2)

The NRC staff also evaluated the applicant's technical basis for managing liquid low-level waste, discussed in SAR Section 1.4.5.1.1, to determine whether the applicant described an adequate plan for managing the liquid waste. Design features and procedures for these systems were also evaluated to determine whether they will minimize liquid waste generation and the possibility of spills. More specifically, the NRC staff reviewed SAR Table 1.4.5-1, which provided the anticipated annual volume of low-level waste generated at the repository during expected normal operations to determine whether (i) the stated volumes reasonably represent expected normal operations and (ii) the table includes all potential sources of low-level waste (SAR Section 1.4.5.1). The NRC staff also reviewed the fire-suppression systems of the five facilities at the proposed repository [Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Receipt Facility (RF), and Wet Handling Facility (WHF), and low-level radioactive waste facility (LLWF)] to determine whether sufficient capacity will exist to contain and process the liquid waste generated fire-water discharge.

On the basis of these evaluations, the NRC staff finds that the applicant's capacity of 34,069 L [9,000 gal] for the holding tanks to contain the volume of fire-water discharge from fire-suppression systems is acceptable because this capacity is based on industry practice for (i) sprinkler operation (NFPA, 2007ab) for an Ordinary Hazard Group II facility like the GROA, and (ii) the applicant's description is consistent with criteria in NFPA 801; Section 5.10 (NFPA, 2003) for sizing of the collection systems. The NRC staff further finds that each facility will have sufficient capacity to also contain and process the liquid waste generated by decontamination and custodial maintenance activities because the applicant included additional capacity beyond the fire-water discharge volume to account for (i) design margins, (ii) freeboard capacity, (iii) one week's capacity for custodial maintenance and decontamination, (iv) a sampling tank, and (v) rounding errors. Finally, the NRC staff finds that the applicant's designation of the collection, holding, and process tanks for liquid low-level waste management as non-ITS is acceptable because these tanks will handle liquid low-level waste only, and damage to these tanks will not cause radiological releases exceeding the exposure limits specified in 10 CFR 63.111(a) and 10 CFR 63.111(b).

**Solid Low-Level Waste Management**

The applicant described the subsystem that it will use to manage low-level radioactive solids and potentially radioactive solids in the waste handling facilities. The applicant stated that potential sources of solid (dry and wet) low-level waste will include (i) water processing or decontamination activities that require some processing activity to meet waste disposal criteria at a disposal facility and (ii) the empty dual-purpose canisters (DPCs) (SAR Section 1.4.5.1.1.1). The applicant stated that dry and wet solid low-level waste, except wet spent resins associated with the pool water treatment, will be collected; transferred to the Low-level Radioactive Waste Facility (LLWF), which is classified as non-ITS; and stored until processed to ensure that the final waste form meets the acceptance criteria of the offsite disposal facility. Spent resin will be dewatered at the Wet Handling Facility (WHF) first and then handled as dry solid low-level waste. The applicant stated that processing units used to process the spent resins from the pool water treatment system ion exchangers would be designed using the methods and practices of ANSI/ANS-40.37-1993.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's descriptions of solid low-level waste management using the guidance in YMRP Section 2.1.1.6 and guidance on solid low-level waste management in Regulatory Guide 1.143 (NRC, 2001ab). In addition, the NRC staff evaluated the applicant's technical basis for managing solid low-level waste and examined the capacity of the applicant's facilities. The NRC staff finds that DOE's technical basis is acceptable because the low-level radioactive waste facility (LLWF) would have sufficient capacity to manage, package, and ship these waste streams under normal operating conditions.

The NRC staff reviewed the WHF operation for fluidizing the resin bed, transferring the fluidized resin to the mobile processing container, and using berms or diked areas/rooms to contain spillage or system leakage from WHF storage tanks and processing equipment (SAR Section 1.4.5.1.1.1). The NRC staff finds that (i) spent resins associated with pool water treatment system ion exchangers would be processed using the methods and practices of ANSI/ANS-40.37-1993, an industry standard for low-level waste processing for nuclear facilities like the GROA, whose application here the NRC staff finds acceptable; and (ii) the applicant adequately addressed any spillage or system leakage that may occur during this process, consistent with standard nuclear industry practices.

In addition, the NRC staff finds that the applicant's solid low-level waste management process is acceptable because the applicant's description of its management plan includes waste-management systems to handle the expected volume of potential solid low-level waste (e.g., HEPA filters) generated during normal operations, consistent with guidance on solid low-level waste management in Regulatory Guide 1.143 (NRC, 2001ab).

Therefore, the NRC staff finds that the applicant provided an adequate description of the management plans to control the solid low-level waste generated during handling operations of spent nuclear fuels and high-level wastes because (i) the solid low-level waste management description is consistent with NRC guidance in Regulatory Guide 1.143 and (ii) the solid low-level waste management description includes a subsystem with sufficient capacity to handle the expected volume of solid low-level waste.

**Gaseous Low-Level Waste Management**

The applicant stated that surface facilities will be designed to mitigate the potential release of radioactivity if an event sequence includes a radionuclide release from casks or canisters containing high-level waste (HLW) or spent nuclear fuel (SNF). In particular, the applicant stated the design of the gaseous effluent treatment and ventilation systems are in accordance with the DOE's Air Cleaning Handbook, which refers to industry codes and standards (DOE, 2003ae). HVAC systems will pass exhaust from the confinement zones through high-efficiency-particulate air (HEPA) filters before it is discharged to the atmosphere (SAR Section 1.4.5.3). According to the applicant, these confinement measures will control airborne radioactive waste and effluents in the handling facilities. The applicant stated that it will use the radiation/radiological monitoring system (SAR Section 1.4.2.2), the digital control and management information system (SAR Section 1.4.2.3), and the communications system (SAR Section 1.4.2.4) to facilitate a prompt and controlled termination of operations and evacuation of personnel, if required. The applicant stated that it will design normal repository operations to control gaseous low-level radioactive effluents to an acceptable level and to keep any exposures ALARA. The applicant further stated that HEPA filters will be used to remove radioactive particulates in gaseous effluent. Nonradioactive service gases, such as argon and helium, will be discharged to the nuclear heating, ventilation, and air-conditioning (HVAC) systems and then discharged to the atmosphere through the HVAC exhaust; however, any radioactive particulates in the service gases will be removed by the HEPA filters in the HVAC system.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the gaseous low-level waste management systems using the guidance in YMRP Section 2.1.1.6. The NRC staff evaluated the information the applicant provided in SAR Section 1.4.5.1.1.3, which described the potential sources of gaseous low-level waste and the proposed mechanisms and processes to capture these wastes. The waste streams expected to contribute to gaseous low-level waste at the proposed repository will be from operations involving casks; transportation, aging, and disposal (TAD) canisters; and DPCs. The NRC staff finds that the applicant's use of high-efficiency-particulate air filters (HEPA), in conjunction with the HVAC system to treat gaseous effluents, is appropriate to remove radioactive particulates from these waste streams because the use of these systems is consistent with standard nuclear industry practices. The applicant stated its gaseous effluent treatment and ventilation systems will be designed in accordance with the DOE Air Cleaning Handbook (DOE, 2003ae), which refers to industry codes and standards for GROA ITS SSCs. In SER Section 2.1.1.6.3.2.8.2.2, the NRC staff finds that the ITS HVAC system can perform the intended safety functions.

**NRC Staff's Conclusion**

The NRC staff finds the applicant's information for a prompt termination of operations and the evacuation of personnel is adequate because the applicant described (i) the radiation-monitoring system that will be used to provide the data for the status and alarm information for use in operations and emergency management; (ii) radiation monitors will be designed to operate on a continuous basis and provide both visual and audible alarms; (iii) adequate environmental and meteorological monitoring systems, which include the monitoring of seismic parameters; and (iv) the communications systems for both normal and emergency conditions.

Based on the evaluations in SER Section 2.1.1.6.3.2.8.9, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(10) for radioactive waste and effluent controls and to permit prompt termination of operations and evacuation of personnel during an emergency are satisfied because the applicant provided adequate descriptions for (i) the subsystems that it will use to handle solid, liquid, and gaseous low-level waste; and (ii) systems for monitoring, alarms, and communications to support emergency termination of operations and evacuation.

2.1.1.6.3.2.10 Structures, Systems, and Components Important to Safety Inspection, Testing, and Maintenance

The applicant provided information in SAR Section 1.9.1.13 on considerations of its means to inspect, test, and maintain ITS SSCs to ensure availability of the SSCs' safety functions.

SAR Section 1.9.1.13 stated the applicant will provide license specifications that include the limiting conditions for operation of selected SSCs. According to the applicant, limiting conditions will include specific surveillance requirements, appropriate functional testing, and other inspections. The applicant stated in SAR Section 1.9.1.13 that SAR Sections 1.2, 1.3, 1.4, 1.5, and 5.6 provided information regarding inspection, testing, and maintenance of SSCs. The applicant also stated in SAR Section 5.6.1 that the waste-handling manager will write, test, and approve plans and procedures for operations, maintenance, surveillance, and periodic testing of SSCs before receipt of waste.

In response to the NRC staff request for additional information to identify the inspection, testing, and maintenance needs for ITS SSCs, the applicant (DOE, 2009dk) stated that the reliability-centered maintenance process will be used to develop plans and procedures for inspection, testing, and maintenance of ITS SSCs. According to the applicant, the inspection, testing, and maintenance needs for each component will be based on manufacturer's recommendations, industry codes and standards, equipment qualification, and reliability values used in the PCSA (SAR Section 1.2.1.3).

**NRC Staff's Evaluation**

The NRC staff reviewed the description of the applicant's plans to inspect, test, and maintain ITS SSCs using the guidance in YMRP Section 2.1.1.6. On the basis of its review, the NRC staff determines that the descriptions of the maintenance programs the applicant discussed in SAR Sections 1.2, 1.3, 1.4, 1.5, and 5.6 are adequate to inspect, test, and maintain ITS SSCs to detect degradation and adverse trends so that corrective actions can be taken prior to component failure. This approach is consistent with standard engineering practices for equipment or component inspection and maintenance. Additionally, the applicant identified probable subjects of license specifications for limiting conditions for operation of selected SSCs that include specific surveillance requirements, appropriate functional testing, and other inspections. Therefore, the NRC staff finds that the applicant's description of the reliability-centered maintenance programs provided adequate information to show that the maintenance programs will provide a reasonable means to ensure availability of ITS SSCs. Additional discussions of the applicant's description of inspection, testing, and maintenance of the SSCs is provided in SER Section 2.5.6, where the NRC staff finds that the applicant adequately described plans for the conduct of maintenance, surveillance, and periodic testing that would be implemented before the applicant receives, processes, stores, or disposes of high-level radioactive waste. The NRC staff finds the applicant's approach of basing the inspection, testing, and maintenance of SSCs on manufacturer's recommendations, industry codes and standards, and equipment qualification acceptable because this approach is consistent with the inspection, testing, and maintenance practice in the nuclear industry.

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(13) for testing and maintenance of SSCs ITS are satisfied because the applicant adequately described its means to inspect, test, and maintain the ITS SSCs at the proposed facility.

#### 2.1.1.6.3.3 Administrative or Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

The applicant described procedures that will be developed to prevent event sequences or mitigate their effects in SAR Section 1.9.3. The applicant's description referred to the management controls and procedures that will be implemented to ensure that administrative controls and procedural safety controls (PSCs) will function properly.

The applicant stated that the preclosure PSCs will be used to regulate human activities to ensure preclosure operations will be maintained within the baseline conditions (limits). The applicant also stated that the preclosure PSCs were identified from the initiating event screening analyses, event sequence quantification analyses, consequence analyses, and criticality control analyses, and these PSCs were listed in SAR Table 1.9-10. According to the applicant, preclosure PSCs will be implemented through individual procedures, normal operating procedures, administrative controls, or a radiation-protection program.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information regarding its administrative controls and procedural safety controls (PSCs) using the guidance in YMRP Section 2.1.1.6 and Interim Staff Guidance–04 (NRC, 2007ad). The NRC staff finds that the applicant adequately identified the preclosure PSCs that it will rely on because the applicant (i) provided PSCs to reduce the likelihood of an initiating event or of an event sequence; (ii) provided PSCs to mitigate the consequences of an event sequence; and (iii) derived the PSCs from initiating-events-screening analyses, event-sequence-quantification analyses, radiological-consequence analyses, and criticality-control measures.

The NRC staff reviewed the PSCs that will be applied to ITS SSCs. Specifically, the NRC staff reviewed the relevant facility reliability and event-sequence-categorization analysis reports and consequence analysis reports and finds that the PSCs for ITS SSCs are acceptable because these PSCs (i) will be implemented through individual procedures, normal operating procedures, administrative controls, or a radiation protection program, as appropriate, and (ii) are consistent with the safety functions of the ITS SSCs identified in the PCSA. The NRC staff finds that the applicant's approach for management controls and procedures is acceptable because the approach is consistent with guidance in Interim Staff Guidance–04 (NRC, 2007ad) on human reliability analyses and nuclear industry-wide practices.

Further information is provided in SER Section 2.5.5 (Plans for Startup Activities and Testing) and SER Section 2.5.6 (Plans for Conduct of Normal Activities including Maintenance, Surveillance, and Periodic Testing).

**NRC Staff's Conclusion**

Based on the evaluation above, the NRC staff finds, with reasonable assurance, that the requirements of 10 CFR 63.112(e)(13) for administrative controls and procedural safety controls for ensuring the continued functioning and readiness of SSCs ITS are satisfied because the applicant adequately identified appropriate preclosure procedural safety controls (PSCs) consistent with the PCSA, to ensure safe repository operations.

### 2.1.1.6.4 Evaluation Findings

The NRC staff reviewed the applicant's Safety Analysis Report and other information submitted in support of the license application and has found, with reasonable assurance, that the requirements of 10 CFR 63.112(e) are satisfied subject to the below proposed condition of construction authorization. An adequate preclosure safety analysis of the performance of the structures, systems, and components important to safety has been provided. In particular, this analysis finds that

(1) Structures, systems, and components important to safety are identified

(2) Criteria for categorization of structures, systems, and components important to safety are adequately developed and categorization of items is acceptable

(3) Controls that will be relied on to limit or prevent potential event sequences, or mitigate their consequences, are acceptable

(4) Measures are adequate to ensure the availability and reliability of structures, systems, and components important to safety

**Proposed Condition of Construction Authorization [10 CFR 63.32(a)]**

> DOE shall not, without prior NRC review and approval, take or implement any exception to the IEEE Standards 308–2001, 384–1992, 379–2000, and 603–1998 in the design of the ITS safety interlock subsystems.
>
> Any amendment request must include the design basis for the use of the exception(s), including the ability of structures, systems, and components to perform their intended safety functions assuming the occurrence of event sequences in accordance with 10 CFR 63.112(e)(8).

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# CHAPTER 7

## 2.1.1.7 Design of Structures, Systems, and Components Important to Safety and Safety Controls

### 2.1.1.7.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.7 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's ("DOE" or the "applicant") proposed design of important to safety (ITS) structures, systems, and components (SSCs) and safety controls (SCs) in the geologic repository operations area (GROA). The review in this section is based upon the results of the NRC staff's review of the applicant's Preclosure Safety Analysis (PCSA), as documented in SER Sections 2.1.1.1 through 2.1.1.6. This evaluation considers DOE's description of its implementation of the repository design to ensure that the preclosure performance objectives specified in 10 CFR 63.111(a) and (b) are met, as required by 10 CFR 63.112(f). The objective of the review is to determine whether DOE has provided adequate design information of ITS SSCs and SCs for both the surface and the subsurface facilities of the GROA, including (i) the design bases and design criteria and (ii) how the design criteria ensure the capability of ITS SSCs and SCs to perform their intended safety functions. This chapter also provides the NRC staff's review of the explosion and fire detection systems and suppression systems described by the applicant to ensure the availability of ITS SSCs. The NRC staff evaluates the information in the applicant's Safety Analysis Report (SAR) Sections 1.2 through 1.5, and 1.9 (DOE, 2008ab), supplemental documents referenced in the SAR, and information the applicant provided in response to the NRC staff's requests for additional information (RAIs) (DOE, 2009dk,dl,do,dq,dv,dw,dy,eh,er–ew,ez,fa–fe,fg,fh,fs; DOE, 2010ak–an).

This chapter builds upon the NRC staff's findings in previous SER sections. The NRC staff's Safety Evaluation Report (SER) Sections 2.1.1.3 and 2.1.1.4 address the sufficiency of the applicant's PCSA information, with respect to identifying hazards and initiating events and identifying event sequences, respectively, including the reliability of the SSCs to perform their safety functions. SER Section 2.1.1.6 evaluates the applicant's identification of ITS SSCs. In SER Section 2.1.1.6.3.1, the NRC staff reviewed the basis for the applicant's identification of SSCs as ITS or non-ITS.

The ITS SSCs reviewed in this chapter include: (i) surface facilities where high-level radioactive waste (HLW) is handled; (ii) mechanical handling transfer systems; (iii) heating, ventilation and air conditioning (HVAC) systems; (iv) other mechanical systems; (v) transportation systems used to move high-level waste (HLW); (vi) electrical power systems; (vii) instrumentation and controls (I&C) systems; (viii) fire protection systems; (ix) canister and overpack systems; and (x) criticality prevention and shielding systems. The following sections address the regulatory requirements and the NRC staff's technical review of the applicant's design, including the design bases, design criteria, design methods, design analyses, and the relationship between design criteria and the preclosure performance objectives for the ITS SSCs.

### 2.1.1.7.2 Regulatory Requirements

The regulatory requirements for the design of important to safety (ITS) structures, systems, and components (SSCs) and safety controls (SCs) are in 10 CFR 63.21(c)(2), 10 CFR 63.21(c)(3), 10 CFR 63.112(f), and 10 CFR 63.112(e)(9).

- Section 10 CFR 63.21(c)(2) requires that the SAR must include information relative to materials of construction of the geologic repository operations area (including geologic media, general arrangement, and approximate dimensions), and codes and standards that DOE proposes to apply to the design and construction of the geologic repository operations area.

- Section 10 CFR 63.21(c)(3) requires that the SAR must include a description and discussion of the design of the various components of the geologic repository operations area and the engineered barrier system, including (i) dimensions, material properties, specifications, analytical and design methods used, along with any applicable codes and standards; (ii) the design criteria used and their relationships to the preclosure and postclosure performance objectives, specified at 10 CFR 63.111(b), 10 CFR 63.113(b); and 10 CFR 63.113(c); and (iii) the design bases and their relation to the design criteria.

- Section 10 CFR 63.112(e)(9) requires that the preclosure safety analysis of the geologic repository operations area must include consideration of explosion and fire detection systems and appropriate suppression systems.

- Section 10 CFR 63.112(f) requires that the preclosure safety analysis of the geologic repository operations area must include a description and discussion of the design, both surface and subsurface, of the geologic repository operations area, including (i) the relationship between design criteria and the requirements specified at 10 CFR 63.111(a) and (b) and (ii) the design bases and their relation to the design criteria.

The NRC staff reviewed the applicant's design information using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.7 (NRC, 2003aa). The relevant acceptance criteria in YMRP Section 2.1.1.7.3 are as follows:

- Section 2.1.1.7.3.1 Design Criteria and Design Bases

    1. The relationship between the design criteria and the requirements specified in 10 CFR 63.111(a) and (b); the relationship between the design bases and the design criteria; and the design criteria and design bases for structures, systems, and components important to safety are adequately defined.

- Section 2.1.1.7.3.2 Design Methods

    1. Geologic repository operations area design methods are adequate.

- Section 2.1.1.7.3.3(i) Designs and Design Analyses for Structures, Systems, and Components, Equipment, and Safety Controls that are Safety Related for Surface Facilities

1. Design codes and standards used for the design of surface facility structures, systems, and components important to safety are identified and are appropriate for the design methodologies selected.

2. The materials to be used for structures, systems, and components important to safety related to surface facility design are consistent with the design methodologies.

3. Design analyses use appropriate load combinations for normal and categories 1 and 2 event sequence conditions.

4. Design analyses are properly performed and documented.

- Section 2.1.1.7.3(iii) Designs for Structures, Systems, and Components and Safety Controls that are Safety Related for Waste Package/Engineered Barrier System

1. Waste package and engineered barrier system structures, systems, and components and their controls are adequately designed.

- Section 2.1.1.6.3 Acceptance criterion 1(2)(i) for compliance with 10 CFR 63.112(e)(9) is as follows:

1. The analyses used to identify structures, systems, and components important to safety, safety controls, and measures to ensure the availability and reliability of the safety systems include adequate consideration of explosion and fire detection systems and appropriate suppression systems.

Description and design of non-ITS SSCs of the subsurface facilities are evaluated in SER Section 2.1.1.2.

In addition to YMRP Section 2.1.1.7, the NRC staff used other NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance to perform its review. Often, this NRC guidance was written for the regulatory oversight of nuclear power plants or other nuclear safety applications, such as transportation of spent fuel casks. As discussed below, the NRC staff used methods, information, or guidance in these documents after determining that they are applicable to and appropriate for the activities and systems proposed at the GROA. The applicability of such NRC guidance is discussed further in Section 2.1.1.7.3, below.

### 2.1.1.7.3 Technical Review

The NRC staff reviewed the applicant’s design of the ITS SSCs and SCs to determine whether the information the applicant provided in its SAR adequately demonstrates that the ITS SSCs and SCs are designed to perform consistent with the safety functions identified in the PCSA, as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6. Specifically, the NRC staff’s evaluation assesses whether the design of ITS SSCs included necessary information, as required by 10 CFR 63.21(c)(3) and 10 CFR 63.112(f), on (i) dimensions, material properties, the analytical and design methods, and applicable codes and standards; (ii) the relationship between the proposed design criteria and the performance objectives; and (iii) the design bases and their relation to the design criteria. As defined in 10 CFR 63.2, the design bases are that information that identifies the specific safety functions to be performed by ITS SSCs (e.g., an ITS surface facility building maintains structural integrity during an earthquake), and specific values or ranges of values chosen for controlling parameters that bound the design (e.g., the ITS SSC failure probability). The design criteria are specific numerical values of design parameters (e.g., tornado wind speed and seismic design basis ground motions) and specific design codes that are used for ITS SSC design to ensure that they have the necessary capacities to perform their intended safety functions (design bases) and thus meet the GROA preclosure performance objectives in 10 CFR 63.111(a) and (b).

The NRC staff's evaluation of the design of ITS SSCs and SCs is provided in the following SER Sections: (1) Section 2.1.1.7.3.1 Surface Facilities; (2) Section 2.1.1.7.3.2 Mechanical Handling Transfer Systems; (3) Section 2.1.1.7.3.3 Heating, Ventilation, and Air Conditioning System; (4) Section 2.1.1.7.3.4 Other Mechanical Systems; (5) Section 2.1.1.7.3.5 Transportation Systems; (6) Section 2.1.1.7.3.6 Electrical Power System; (7) Section 2.1.1.7.3.7 Instruments and Controls Systems; (8) Section 2.1.1.7.3.8 Fire Protection Systems; (9) Section 2.1.1.7.3.9 Canisters and Overpacks; and (10) Section 2.1.1.7.3.10 Criticality Prevention and Shielding Systems.

The NRC staff's evaluation of ITS SSC design in each SER section is structured as follows: (1) the applicant's description of its design, including design criteria, design bases, design methods, and design analyses (2) the NRC staff's evaluation of the applicant's Design Criteria and Design Bases; (3) the NRC staff's evaluation of the applicant's Design Methods; and (4) the NRC staff's evaluation of the applicant's Design and Design Analyses.

Additionally, the NRC staff's review of the applicant's explosion and fire detection systems and suppression systems [10 CFR 63.112(e)(9)] associated with hydrogen accumulation in battery rooms is provided in Section 2.1.1.7.3.3; review of the explosion and fire detection systems and suppression systems associated with gas fuel tanks in the Site Transporter, Cask Tractor, and Cask Transfer Trailers, is provided in Section 2.1.1.7.3.5.

DOE has referenced numerous industry codes and standards, Regulatory Guides, and other documents, which provide design criteria, methods, and analysis that DOE proposes to use in the design of the GROA facilities. In addition, the NRC staff has referenced Standard Review Plans, Regulatory Guides, Interim Staff Guidance, and other references to support its evaluation of the LA. As part of its review, the NRC staff has evaluated whether the criteria, methods, and analysis in the referenced documents are applicable and appropriate to use for the GROA design information. This evaluation is necessary to determine whether the information DOE provided is consistent with Section 2.1.1.7.3.3(i) of the YMRP Acceptance Criteria.

Table 7-1 summarizes the industry codes and standards, Regulatory Guides, Interim Staff Guides (ISGs), Standard Review Plans, and other documents used by the NRC staff to support its review of the design criteria, methods, and analysis provided by DOE to evaluate the applicant's information for compliance with 10 CFR Part 63 requirements. This table provides the scope of the codes, standards, and other documents and discusses the bases for their applicability to the GROA design review in this chapter. There are three ways that the basis for the applicability to the GROA review is addressed in Table 7-1 (and in associated text where ITS SSCs are evaluated). Codes, standards, and reference documents have been used to support the NRC staff's review, and in these cases, Table 7-1 provides a basis for the sufficiency of the entire reference for its use by the NRC staff. In some cases, the code, standard, or other reference documents has been used for only one or two specific ITS components or parts or for a specific safety function. In these cases, Table 7-1 provides the bases for the sufficiency of the reference for use for those items and does not address the sufficiency of the entire code, standard, or reference. In a few limited cases, the code, standard, or other reference has general applicability to the GROA review but has been used in a detailed fashion in the ITS SSC design review to address many components or parts of an ITS SSC and is referenced in many places in this SER section. In these cases, Table 7-1 provides the general applicability of the code, standard, or document for use in the GROA design review, and specific discussions providing additional bases for the sufficiency of its use by the NRC staff for its review is included in individual sections of this SER section where the component or part of the ITS SSC is evaluated.

#### 2.1.1.7.3.1 Surface Facilities

The applicant provided design information on surface facilities that it concluded were ITS, which included surface facilities buildings, the aging facility (AF), and flood-control features, in SAR Section 1.2. The surface facilities buildings included the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Receipt Facility (RF). These are described by the applicant in SAR Sections 1.2.2 through 1.2.6. The AF is described in SAR Section 1.2.7, and the flood-control features are described in SAR Section 1.2.2.1.6.2. The IHF is designed to (i) receive transportation casks containing naval spent nuclear fuel (SNF) or DOE high-level waste (HLW) canisters, (ii) transfer the canisters into waste packages, and (iii) transfer the completed waste packages onto a transport and emplacement vehicle (TEV) for transporting them to the subsurface for emplacement in a drift. The other facilities used to load waste packages are the three CRCF facilities. The CRCF is designed to (i) receive transportation casks containing SNF canisters [transportation, aging and disposal (TAD), HLW, DOE SNF, and dual-purpose canisters (DPCs)]; (ii) transfer the canisters into waste packages (WP) or aging overpacks (AO); and (iii) transfer the completed WPS or AOs onto TEV for transporting them to subsurface for emplacement or to the AF. The WHF is designed to (i) receive transportation casks containing uncanistered commercial SNF assemblies, (ii) transfer the assemblies into TAD canisters underwater in the pool, and (iii) load the TAD canisters into aging overpacks for transfer to a CRCF or the Aging Facility. The WHF is also designed to (i) receive DPCs in transportation casks, aging overpacks, or shielded transfer casks; (ii) transfer the DPCs into a shielded transfer cask; and (iii) open the DPCs and transfer the commercial SNF assemblies into TAD canisters underwater in the pool. The RF is designed to (i) receive transportation casks containing TAD canisters or DPCs and (ii) transfer the canisters into aging overpacks on a site transporter for movement to a CRCF or the Aging Facility.

The applicant stated that the surface buildings identified as ITS protect the SSCs located inside these buildings; the aging pads at the AF provide a stable foundation for aging casks; and the flood control structures protect the surface facilities from flood hazards (SAR Section 1.9). The NRC staff reviewed the surface facilities design bases and design criteria to determine whether they are consistent with the safety functions identified in the PCSA, which is reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6, and whether the applicant's proposed design methods are adequate.

##### 2.1.1.7.3.1.1 Surface Buildings

The applicant presented, in SAR Section 1.2.2, the structural design information that is common to all the surface buildings, which included design criteria and design bases, design methods, and design analyses. The NRC staff's evaluation of the applicant's information in SAR Section 1.2.2 is provided in SER Sections 2.1.1.7.3.1.1.1, 2.1.1.7.3.1.1.2, and 2.1.1.7.3.1.1.3.1.

The applicant's structural design information unique to each of the ITS surface buildings, such as analysis and design procedures, are in SAR Sections 1.2.3 through 1.2.6, and are evaluated by the NRC staff in SER Section 2.1.1.7.3.1.1.3.2.

2.1.1.7.3.1.1.1 General Design Criteria and Design Bases

The applicant provided the design bases and their relationships to the design criteria for the IHF, CRCF, WHF, and RF in SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3, respectively. These SAR tables provided design bases for structural integrity of the buildings to protect ITS SSCs inside the buildings from adverse consequences due to external loads (e.g., seismic, wind loads), as well as against building collapse. To meet these design bases, the applicant specified in SAR Tables 1.2.2-1 through 1.2.2-3 that the ITS surface buildings are designed to withstand the external loads, as discussed in this SER subsection.

***Wind and Tornado***

The applicant stated that the design basis wind load is a 3-second-gust wind speed of 145 km/hour [90 mph] at 10 m [32.8 ft] above ground with an annual probability of occurrence of 0.02 (ASCE, 2000ab) (SAR Section 1.2.2.1.6.1). Since the Yucca Mountain site is a "Special Wind Region" (ASCE, 2000ab), the applicant examined site-specific wind data to confirm the design basis wind speed of 145 km/hour [90 mph] (BSC, 2007dc) for the surface facilities.

The applicant stated that the design basis tornado wind parameters are a maximum wind speed of 304 km/hour [189 mph], a pressure drop of 5,585 Pa [0.81 psi], and a rate of pressure drop of 2,068 Pa/s [0.30 psi/s] (SAR Section 1.2.2.1.6.1.2 and SAR Table 1.2.2-1). The applicant developed the design basis tornado parameters using the guidance in NRC Regulatory Guide 1.76 (NRC, 2007ai) and NUREG/CR–4461 (Ramsdell and Rishel, 2007aa). The applicant stated that the tornado-generated missiles are consistent with Spectrum II from Section 3.5.1.4 of NUREG–0800 (NRC, 1981ad). The applicant treated the effects of tornado missile impacts as live loads with impact.

***Explosion***

For the hazard from potential explosions at nearby facilities and transportation routes, the applicant used a maximum overpressure of 6,895 Pa [1 psi] (SAR Table 1.2.2-1) for the ITS SSCs facilities, below which no significant damage to SSCs is expected, consistent with NRC Regulatory Guide 1.91 (NRC, 1978ac).

***Volcanic Ash***

The applicant stated that the roof live load caused by volcanic ash fall is 1,005 Pa [21 pounds of pressure per square foot (psf)] (SAR Section 1.2.2.1.6.5). The live load, in contrast to the dead load from self-weight, is a temporary or short-term load.

***Snow and Ice***

The applicant stated that the maximum design daily snowfall and ice at the GROA is 152 mm [6 in], and the maximum design monthly snowfall is 168 mm [6.6 in]. The applicant based these values on records during the period from January 1, 1983 through February 28, 2005, from Desert Rock Weather Service Meteorological Observatory, Nevada, which is located approximately 45 km [28 mi] southeast of the repository. The applicant stated that these records are representative of the site (SAR Section 1.2.2.1.6.4).

***Seismic***

The applicant described in SAR Section 1.2.2.1.6.3.1 two levels of site-specific seismic ground motions for use in the design of ITS SSCs: (i) Design Basis Ground Motion 1 (DBGM–1), associated with a mean annual probability of exceedance (MAPE) of $10^{-3}$ and (ii) Design Basis Ground Motion 2 (DBGM–2), associated with a MAPE of $5 \times 10^{-4}$ (SAR Table 1.2.2-3). According to the applicant, the applicable DBGM depends on the functions and risk significance of the ITS SSCs, as determined in the PCSA, which the NRC staff evaluates in SER Sections 2.1.1.4.3.4 and 2.1.1.6.3.1. If the postulated loss of function of the ITS SSCs due to a seismically initiated event would result in a dose to the public or workers that would exceed the performance objectives of 10 CFR 63.111(a) and 10 CFR 63.111 63.111(b)(1) but would not exceed those of 10 CFR 63.111(b)(2), the ITS SSC are designed for DBGM–1 ground motions, whereas, if the postulated dose to the public would exceed the performance objectives of 10 CFR 63.111(b)(2), the ITS SSCs are designed for the DBGM–2 ground motions.

The applicant stated that the ITS surface buildings are designed for a site-specific seismic ground motion level DBGM–2. The applicant further stated that the horizontal and vertical peak ground accelerations (PGAs) for DBGM–2 seismic events are 0.45g and 0.32g, respectively, where “g” is the acceleration due to gravity (SAR Table 1.2.2-3). The applicant showed seismic ground motion response spectra for different seismic levels in SAR Figures 1.2.2-8 to 1.2.2-13. A response spectrum is a plot of the peak or steady-state response (displacement, velocity or acceleration) of a series of oscillators of varying natural frequency, which are forced into motion by the same base vibration or shock. The NRC staff’s review of the seismic design spectra is provided in SER Section 2.1.1.1.3.5.2.

**NRC Staff’s Evaluation of the General Design Criteria and Design Bases for ITS Surface Facilities Buildings**

The NRC staff evaluated the applicant’s information on the general design bases and design criteria for the ITS surface buildings in SAR Section 1.2 and SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3, and finds that the applicant’s use of a 3-second-gust wind speed of 145 km/hour [90 mph] as the design basis wind speed is acceptable because the 3-second wind speed is consistent with applicable industry standards (ASCE, 2000ab). The staff notes that the maximum wind speed for a probability of occurrence of $2 \times 10^{-2}$ per year is less than 90 mph (BSC 2007dc). The applicant’s calculation is based on the local meteorological data reviewed and found to be acceptable by the NRC staff in SER 2.1.1.1.

The NRC staff finds that the tornado wind parameters {speed of 304 km/hour [189 mph], pressure drop of 5,585 Pa [0.81 psi], and rate of pressure drop of 2,068 Pa/s [0.3 psi/sec]} is acceptable because these design basis wind parameters conservatively exceed the recommendations in NRC Regulatory Guide 1.76 (NRC, 2007ai). The NRC staff also finds that the use of the Spectrum II tornado-generated missiles considered as live load with impact is acceptable because it is consistent with NRC staff guidance for nuclear power plants in NUREG-0800 (NRC, 2007al, Section 3.5.1.4). Applicability of this guidance to the GROA is discussed further in Table 7-1.

The NRC staff finds the applicant’s use of a maximum overpressure of 6,895 Pa [1 psi] (SAR Table 1.2.2-1), below which no significant damage to ITS SSCs facilities is expected, is acceptable because it is consistent with applicable NRC guidance in Regulatory Guide 1.91 (NRC, 2013af). Applicability of this guidance to the GROA is discussed further in Table 7-1.

The NRC staff finds that the design volcanic ash load on the roof of surface buildings of 1,005 Pa [21 psf] (SAR Section 1.2.2.1.6.5) is acceptable because an areal ash fall density greater than 10 $g/cm^2$ [20 psf] from volcanic activities at the GROA before permanent closure is well below $10^{-6}$/year, as evaluated by the NRC staff in SER Section 2.1.1.3.3.1.3.1.

The NRC staff finds that the maximum design daily snowfall and ice loadings of 152 mm [6 in], and the maximum design monthly snowfall and ice loading of 168 mm [6.6 in] at the GROA is acceptable because the NRC staff evaluated this hazard in SER Section 2.1.1.1.3.3, and found that the methodology the applicant used to estimate the proposed maximum daily and monthly snowfall and ice loading is consistent with NRC guidance specified in NUREG–0800, Section 2.4.3 (NRC, 2007ak). The applicability of NRC guidance in NUREG-0800, Section 2.4.3 to the GROA is further discussed in Table 7-1.

The NRC staff finds that the design of surface buildings for the seismic loads at the site-specific DBGM–2 level with a MAPE of $5 \times 10^{-4}$ is acceptable because this probability level has been determined by the NRC staff as adequate for seismic design of nuclear facilities with similar risk profiles, including fuel cycle facilities and independent spent fuel storage installations (ISFSIs). Regulatory Guide 3.73 (NRC, 2003ae) provides the rationale for use of this MAPE as applied to ISFSI design. The applicability of NRC guidance in Regulatory Guide 3.73 to the GROA is further discussed in Table 7-1. The applicant’s information on the development of the seismic hazard at various levels of MAPE values, including the DBGM–2 horizontal and vertical PGAs of 0.45g and 0.32g, respectively, is evaluated by the NRC staff in SER Section 2.1.1.1.3.5.2, and is found to be acceptable.

On the basis of the evaluation described above of the design information presented by the applicant in SAR Section 1.2 and SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3 for surface buildings and the NRC staff’s evaluation in SER Sections 2.1.1.1 through 2.1.1.3, the NRC staff finds that the design bases and design criteria for wind, tornado, explosion, volcanic ash, snow and ice, and seismic hazards developed by the applicant for the design of ITS surface buildings are acceptable because (1) the design bases and design criteria are consistent with the site-specific information reviewed by the NRC staff in SER Section 2.1.1.1, and found acceptable and (2) the loads were developed using industry standards (ASCE, 2000ab) or based on acceptable NRC guidance for the design of NRC-licensed nuclear facilities (e.g., NRC, 2003ae; NRC, 2007ai).

2.1.1.7.3.1.1.2 Design Methods

The applicant described the proposed facilities and operations in SAR Section 1.2, and stated that the Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Receipt Facility (RF) buildings are reinforced concrete structures, with interior and exterior shear walls, reinforced concrete floors, and roof slab diaphragms acting to transmit lateral forces to vertical-resisting elements (shear walls), and reinforced concrete mat foundations. The Initial Handling Facility (IHF) building has two structures: (1) a main braced-frame steel structure with an internal reinforced concrete structure on a common reinforced concrete mat foundation and (2) a reinforced concrete structure on a separate reinforced concrete mat foundation. The structures are supported on alluvium or engineered fill above the alluvium layer, depending on the location of the particular facility. The engineered fill has material properties equivalent to or better than that of the underlying alluvium. The applicant stated that it designed the ITS surface facilities using the following design method: (1) analyzing structures to determine forces and moments for various loads using computer codes (e.g., SAP2000); (2) combining the seismic loads and nonseismic loads obtained from Step 1, as provided by applicable design codes; and (3) designing reinforcing steel for concrete walls or sizing steel members for these combined loads to ensure that the design capacity exceeds the demand (predicted forces and moments) (SAR Section 1.2.2.1).

The applicant's design method used for the seismic design of ITS buildings was based on elastic analyses using the SAP2000 software (Computers and Structures, 2005aa). The applicant used the Tier #1 analyses, as outlined in BSC (2007ba, Section 7.1.3), for structural design and to evaluate seismic performance of the CRCF, WHF, and RF buildings (SAR Sections 1.2.4, 1.2.5, and 1.2.6). The Tier #1 analyses were based on lumped-mass, multiple-stick models, in which the building walls were modeled as beam-column elements using cross section properties. The ends of the beams were constrained to a master node at each floor diaphragm level and, thus, the floors were considered to be rigid in all three directions, as described in BSC (2007ba, Section 7.2.1.1). In the Tier #1 analyses, the foundation was modeled as soil springs representing soil stiffness to account for soil-structure interaction (SSI) effects, the best-estimate, upper-bound, and lower-bound soil properties.

Since the IHF building structure is a braced-frame steel structure, the applicant stated that the Tier #1 type of analysis for concrete shear-wall type structures is not applicable. Therefore, for the IHF design (SAR Section 1.2.2.1.6.3.2.4), the applicant created finite element analysis models in SAP2000, in which the concrete components were modeled by shell elements and the components of the steel frame structure were modeled as beam-column elements, with the base of the structure assumed as fixed.

**NRC Staff's Evaluation of the Design Methods for ITS Surface Facilities Buildings**

The NRC staff evaluated the applicant's information on the design method of the ITS surface buildings discussed in the SAR Section 1.2. The NRC staff finds that the approach of using numerical models for a linear elastic analysis with the SAP2000 computer code to determine design forces and moments, and then using industry codes to design the surface facilities buildings, is acceptable because (i) the models were developed consistent with established industry practices (ASCE, 2000aa; ASCE, 2005aa) and guidance acceptable to the NRC staff for nuclear power plants (Section 3.7.2 of NUREG–0800 (NRC, 2013ac); (ii) the modeling method used computer codes that are commonly used for structural modeling; and (iii) site-specific seismic ground motion data was used as hazard input data. Applicability of Section 3.7.2 of NUREG–0800 to the GROA is discussed further in Table 7-1.

The NRC staff also finds that the applicant's design method of analyzing the CRCF, WHF, and RF concrete buildings using a Tier #1 lumped-mass, multi-stick model of a building in the SAP2000 code, with the foundation modeled as soil springs, is appropriate for the design of the ITS surface buildings subjected to a DBGM–2 seismic event because it is consistent with standard engineering practices for structural seismic analyses of concrete shear-wall type structures (ASCE, 2000aa; ASCE, 2005aa) and NRC guidance for nuclear power plants (Section 3.7.2 of NUREG–0800 (NRC, 2013ac). The applicant also considered the uncertainties in soil properties in the SSI analyses (SAR Section 1.2.2.1.6.3.2.1), consistent with the guidance in NUREG–0800, Section 3.7.2 (NRC, 2013ac).

The NRC staff also finds that the finite element analysis used to model the steel-braced frame structure of the IHF is acceptable because the finite element analysis method allows behavior of various structural elements to be appropriately modeled, and is consistent with the established industry practice for structural numerical modeling of such structures (ASCE, 2005aa). As described above, the Tier #1 lumped-mass, multi-stick model of a building in the SAP2000 code is appropriate for concrete buildings, but not for a steel-braced frame structure.

Based on the NRC staff evaluation described above, the NRC staff finds that the applicant's proposed design methods for the surface ITS buildings are adequate, consistent with established industry practices, and appropriately address uncertainties. The adequacy of the analytical models of the specific ITS surface buildings is evaluated later in SER Section 2.1.1.7.3.1.1.3.2.

2.1.1.7.3.1.1.3 Design and Design Analyses

2.1.1.7.3.1.1.3.1 General Analysis and Design Procedures

This section presents the NRC staff's evaluation of the general analysis and design procedures that are common for all ITS surface facilities buildings, described by the applicant in SAR Section 1.2.2. The NRC staff's evaluation of the facility-specific aspects of analysis and design procedures of the ITS surface facilities buildings, provided by the applicant in SAR Sections 1.2.3 through 1.2.8, are evaluated in SER Section 2.1.1.7.3.1.1.3.2.

**Design Codes and Standards**

The applicant listed the codes and standards for the structural design of surface buildings in SAR Sections 1.2.2.1.8 and 1.2.2.1.6.3. The applicant stated that it assigned live loads to the floors according to the American Society of Civil Engineers ASCE 7–98 (ASCE, 2000ab). The applicant stated that it developed seismic analysis models of the building structures in accordance with ASCE 4–98 methods (ASCE, 2000aa) (SAR Section 1.2.2.1.6.3). The applicant stated that it designed reinforced concrete surface buildings using the strength design method, as detailed in BSC (2007ba, Section 8.2) and specified in ACI 349–01 (ACI, 2001aa). For steel design, the applicant stated that the allowable stress design method was used, as described in BSC (2007ba, Section 8.2) and specified in ANSI/AISC N690–1994 (AISC, 1994aa).

**Consistency of Materials with Design Methods**

The applicant's design methods identified the principal material properties used in the construction of the ITS surface buildings in SAR Section 1.2.2.1.7. These material properties were (i) concrete compressive strength ($f'_c$) of $3.45 \times 10^7$ Pa [5,000 psi] for shear walls, diaphragms, and foundations; (ii) reinforcing steel yield strength ($f_y$) of $4.14 \times 10^8$ Pa [60,000 psi]; and (iii) Mohr-Coulomb friction angle ($\varphi$) of $39^o$ for soil and $42^o$ for engineered backfill.

**Load Combinations**

The applicant listed the load combinations to be used in ITS surface building design in SAR Section 1.2.2.1.9.2 and BSC (2007av, Sections 4.2.11.4.5 and 4.2.11.4.6). The applicant stated that these load combinations are in accordance with ACI (2001aa) and AISC (1994aa) codes for the design of reinforced concrete and steel structures of nuclear facilities, respectively. In response to the NRC staff's RAI (DOE, 2009es), the applicant stated that some of these load combinations are not applicable to the design of specific structures (e.g., there are no fluid loads in the CRCF) and some individual loads are bounded by other load sources (e.g., wind, ash loads). The applicant stated that the surface building design was governed by load combinations that included gravity loads (dead load and 25 percent of live load) and seismic loadings.

In response to the NRC staff's RAI (DOE, 2009es), the applicant provided its rationale for using 25 percent of the live load for seismic load combinations, instead of the full live load otherwise proposed in the load combinations presented in SAR Section 1.2.2.1.9.2, and stated that, on the basis of its analysis, the consideration of the full live load factor would not impact the overall design because the magnitude of the live load is relatively small compared to the other load sources.

For the seismic load combinations, the applicant considered several sub-combinations using the 100-40-40 component factor method, which assumes that when the maximum earthquake acceleration in one direction occurs, the earthquake accelerations from the other two orthogonal directions are 40 percent of the maximum acceleration, as detailed in BSC (2007ba, Appendix A); BSC (2007af, Section 6.1); and BSC (2007ae; Section 6.3).

**NRC Staff's Evaluation of the Codes and Standards, Consistency of Materials with Design Methods, and Load Combinations for ITS Surface Facilities**

The NRC staff evaluated the applicant's information on the proposed design codes and standards for the ITS surface buildings. The NRC staff finds that the codes and standards the applicant has proposed are appropriate because they are consistent with the standard engineering practices for design and construction of nuclear power plants [ACI (2001aa) and AISC (1994aa)]. Applicability of these standards to the GROA is discussed further in Table 7-1.

The NRC staff also evaluated the material properties of concrete, steel, soil properties, and engineered backfill properties the applicant proposed to use for the ITS surface buildings. The NRC staff finds that the proposed properties of the steel and concrete materials are acceptable because they conform to the applicable design codes, and the soil properties are acceptable because they were appropriately developed based upon properties specific to the GROA as reviewed and found acceptable by the NRC staff in SER Section 2.1.1.1.3.5.1.2. The NRC staff also verified that the applicant used these material properties appropriately in designing the ITS surface buildings to the codes and standards and the design methods described in SER Section 2.1.1.7.3.1.1.2.

The NRC staff evaluated the information on the load combinations the applicant proposed for the design of ITS surface buildings. The NRC staff finds that the proposed load combinations are appropriate because the load combinations are consistent with the codes and standards the applicant used to design concrete and steel structures (SAR Sections 1.2.2.1.8 and 1.2.2.1.6.3), whose use the NRC staff finds acceptable, as discussed further in Table 7-1. The NRC staff finds that the applicant's use of a load combination that includes seismic loading, dead load, and 25 percent of the live load as bounding the design of surface buildings is appropriate because the seismic loads are significantly greater than other loads, such as wind loads; thus resulting in the applicant's design basis conservatively using the greatest demand on the structural components for its design.

The NRC staff also finds that the applicant's use of the 25 percent of the design live load of {4.8 kPa [100 psf]} (BSC, 2007ba) in load combinations that include seismic loads, is appropriate as described in the following section, "Seismic Analysis Method."

The NRC staff finds that the applicant's use of the 100-40-40 methodology to combine responses from the three orthogonal (two horizontal and one vertical) earthquake components acceptable, because this approach is consistent with standard nuclear industry practice (ASCE, 2000aa) and is appropriate for use in designing GROA facilities.

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's proposed use of: (1) design codes and standards, (2) materials, and (3) load combinations is appropriate for the analyses and design of the ITS surface buildings.

**Seismic Analysis Method**

For the seismic analyses of ITS surface buildings subjected to DBGM–2 events, as described in BSC (2007ba, Section 5.1), the applicant used a response spectrum method based on site response spectra (SAR Figures 1.2.2-10, 1.2.2-11). The seismic analysis model (Tier #1) for elastic analysis was a lumped-mass multi-stick model, as discussed earlier in this SER in the Design Methods Section, in which the building structure is modeled as beam-column elements, and the soil is modeled using spring elements. The applicant's design method considered 25 percent of the design live load as mass equivalent in the seismic analysis model, consistent with the recommendations in International Code Council (2003aa) (DOE, 2009es, Enclosure 3). The applicant's method modeled the concrete structures as uncracked. The applicant determined that its primary design parameter was the in-plane response and that concrete cracking was found not to significantly affect the in-plane response, as described in BSC (2007ba, Section 7.1.1). In response to the NRC staff's RAI (DOE, 2009eu), the applicant indicated that the shear wall stiffness corresponding to the uncracked concrete properties was generally satisfactory for determining wall design forces, as stated in Section C3.1.3.1of ASCE 4–98 (ASCE, 2000aa).

The applicant based the percentage of structural damping of 7-percent used in response spectrum analysis (SAR Table 1.2.2-4) on ASCE/SEI 43–05 (ASCE, 2005aa) recommendations. For the soil foundation, the applicant's design method computed the soil damping ratio as greater than 100 percent for surface facilities (BSC, 2008af; Sections 6.1.3.3 and 7.1.5; Hadjian and Ellison, 1985aa). However, the applicant's design method used the soil damping value of 20 percent, as detailed in BSC (2008af, Section 7.1.5), on the basis of its interpretation of ASCE 4–98 (ASCE, 2000aa; Section 3.1.5.4). In response to the NRC staff's RAI (DOE, 2009ev), the applicant provided a comparative study of soil damping values calculated using different design methods for the CRCF to show that the soil damping ratio of 20 percent was conservative. The design methods the applicant used were: (1) SASSI 2000 software (University of California, 2000aa), which modeled the soil foundation as layered media and a massless, rigid foundation for the building and (2) the simplified empirical method based on strong motion data obtained from sites with instrumented structures and free-field accelerographs (Stewart, et al., 1998).

To account for differences in soil (20-percent) and structural (7-percent) damping values in the structural analyses for seismic loads, the applicant's design method generated "hybrid" spectra that combined the 20-percent and 7-percent damped spectra, as outlined in BSC (2007af, Section 6.1). To show the applicability of hybrid response spectra (DOE, 2009et) and the conservatism of the response spectrum analysis design method, the applicant compared the CRCF structural member forces obtained from the response spectrum analysis based on a hybrid spectrum, where 4-percent structural damping and 20-percent soil damping were used, and the time history analysis, where 4-percent Rayleigh damping was used (ASCE, 2000ab). The applicant determined that the forces obtained from the response spectrum analyses were on average 11 to 13 percent higher than those calculated using time history design method analyses.

To calculate total seismic loads, the applicant stated that its design method used the square-root-of-the-sum-of-the-squares (SRSS) method to combine: (i) maximum modal responses of interest in design (stress, strain, moment, shear, or displacement) for modes that are not closely spaced; (ii) maximum values of the structural responses to each of the three components of earthquake motion, as described in NRC (1989ac). The modal responses at closely-spaced frequencies (within 10 percent) were combined using the Ten Percent Method, as described in Regulatory Guide 1.92 (NRC, 1976ad).

**NRC Staff's Evaluation of the ITS Surface Facilities Buildings Seismic Analysis Method**

The NRC staff evaluated the applicant's information on the seismic analysis method for the ITS surface buildings, and finds that the applicant's seismic analysis method of using the response spectrum and the lumped-mass multi-stick model for Tier#1 elastic analysis is appropriate because the method is consistent with the standard engineering practice for seismic analysis [ASCE 4-98 (ASCE, 2000aa)] and guidance acceptable to the NRC staff for nuclear power plants [Section 3.7.2 of NUREG–0800 (NRC, 2013ac)] that is applicable to the proposed activities at the GROA, as discussed further in Table 7-1.

The applicant's use of 25 percent of the design live load as mass equivalent in the seismic analysis models of surface buildings is acceptable because it is consistent with NRC guidance for nuclear power plants (NRC, 2013ac) that the NRC staff finds to be appropriate for use at the GROA because the surface buildings at the GROA are similar in design to those at a nuclear power plant. The NRC staff finds that the applicant's approach of considering concrete as uncracked in the seismic analysis model is conservative, and thus is acceptable because the consideration of concrete cracking in the model would lead to a more flexible structure, resulting in lower spectral accelerations and reduced design forces than those based on the uncracked concrete properties.

The NRC staff finds that the percentage of structural damping the applicant proposed for the analysis and design of ITS surface buildings is acceptable because the damping value used is consistent with ASCE/SEI 43–05 (ASCE, 2005aa). The NRC staff determines that the applicant's use of the soil-damping value of 20 percent for the seismic analysis is acceptable because it is consistent with ASCE 4–98 (ASCE, 2000aa). Additionally, the NRC staff finds that the use of the hybrid spectra, where the applicant used soil damping of 20 percent for the first three soil vibration modes, and the structural damping of 7 percent for the remaining structural modes, is acceptable because it is consistent with the standard industry guidance in ASCE 4–98 (ASCE, 2000aa).

The NRC staff finds that the methods used by the applicant to combine modal responses for one earthquake component (SRSS and the Ten Percent Method) and for the three earthquake components (SRSS method) follows standard industry practice (ASCE, 2000aa) and NRC guidance for nuclear power plants in Regulatory Guide 1.92 (NRC, 2012ac) whose use the NRC staff finds appropriate for use at the GROA. Applicability of this standard and guidance to the proposed activities at the GROA is discussed further in Table 7-1.

Based on the review above, the NRC staff finds that the applicant's seismic analysis method for the surface buildings is acceptable.

***Structural Design Method***

The following subsections present the applicant's approach to the design of the structural components of the surface buildings. The applicant based the structural design of the surface buildings presented in SAR Section 1.2 and the supporting documents on a demand-to-capacity (D/C) ratio equal to or less than unity. The applicant computed demand on a structural component based on the applied loads and computed the capacity based on the applicable design codes and standards. In response to the NRC staff's RAI (DOE, 2009eu), the applicant stated that the design of surface buildings was based on seismic analysis results (BSC, 2007af,da,cx,aq) using bounding soil properties for an alluvium depth of 10.7 m [35 ft].

***Shear Wall Design Method***

The applicant based its design method for shear wall design, as outlined in BSC (2007ba, Appendix D), on ACI 349–01 (ACI, 2001aa) using site-specific input data. The applicant determined that the predominant load path of seismic load was through the diaphragms and shear walls to the base of the concrete slab. The applicant's design method for the shear walls considered the combined effects of in-plane shear loads, axial loads, in-plane bending moments, out-of-plane bending moments, and transverse shear loads. The applicant also considered shear friction (i.e., the capacity of the wall to transfer horizontal loads into the base slab) in the shear wall design (e.g., BSC, 2007ba,cv).

Based on the applicant's design method, the applicant stated that reinforced concrete shear walls (BSC, 2007dh,cy,cv,aq) would have sufficient capacity to withstand the design loads, including seismic events. The applicant included a torsional factor in the design forces [BSC (2006ak, Section 6.3); BSC (2007cv, Section 6.2)], as recommended by ASCE 4–98 (ASCE, 2000aa). The applicant stated that the torsional factor accounted for load eccentricity and resulted in an increase in the design forces used for shear wall design.

***Slab Design Method***

The applicant stated that the design method for the reinforced concrete slabs in surface buildings is in accordance with the provisions of the ACI 349–01 (ACI, 2001aa) code [BSC (2007ba, Appendix D)]. Concrete slabs of the ITS buildings have a thickness varying from 0.46 m [1.5 ft] to 1.22 m [4.0 ft] and include a structural steel support, which consists of a steel deck, as well as steel beams, girders, and trusses. The 76-mm [3-in]-thick steel corrugated deck only supports construction loads. The concrete slabs were considered by the applicant as noncomposite with the structural steel, as described in BSC (2007cz, Section 6.6). Shielded rooms were designed with thicker concrete slabs in which the structural steel support is not credited in the structural analyses, such as in the case of the 1.2-m [4-ft]-thick concrete slab of the CRCF (BSC, 2007ct) and WHF (BSC, 2007cw).

The applicant computed the reinforcement of the concrete slabs for (i) out-of-plane bending loads, (ii) in-plane diaphragm shear, and (iii) in-plane diaphragm moments. To obtain the out-of-plane seismic forces in the diaphragm design, the vertical accelerations obtained from the seismic analyses were amplified by a factor of two, as outlined in BSC (2007ct, Assumption 3.1.6). This amplification was used to account for the effects of vertical floor flexibility, given that the slabs were considered to be rigid diaphragms in design analyses. The applicant used this factor based on a study performed for the Canister Handling Facility (BSC, 2005ao).

The applicant considered the reinforced concrete slabs as one-way slabs (e.g., BSC, 2008cj; BSC, 2007ct). The reinforcing steel computed in the slab span direction was also provided in the orthogonal direction. For the in-plane loads, the applicant analyzed multiple-span diaphragms as simple spans, based on the structural analysis of the largest span [e.g., BSC (2007ct, Assumption 3.2.2)]. The applicant also stated that composite action between the concrete slabs and the supporting structural steel beams was not considered (BSC, 2007cz; Section 3.2).

The steel beams supporting the concrete slabs consisted of W-shaped members. The applicant considered that top flanges of the beams were laterally supported by the steel deck during construction and by the concrete slabs during service. The applicant stated that the deflection limits of the structural steel members were developed consistent with ANSI/AISC N690 (AISC, 1994aa) and International Code Council (2003aa) (see BSC, 2007cz) codes. For the design, the applicant assumed that the structural steel components provided support for the concrete slabs and superimposed loads for all applicable service and extreme load combinations. The applicant's design provided demand/capacity (D/C) ratios for flexure, which is the controlling failure mechanism for this type of structural component.

***Foundation Design Method***

The applicant's design method for the foundation design of the ITS surface buildings was based on finite element method analyses of the mat foundation, as described in BSC (2007ae, Section 6.1), using the SAP2000 software (Computers and Structures, 2005aa), and ACI 349–01 (ACI, 2001aa). The applicant's numerical model of the ITS building foundations was coupled with the superstructure. The design method applied finite element analyses and used an approximate mesh size of 1.5 m [5 ft] × 1.5 m [5 ft] for the mat foundation. In response to the NRC staff's RAI (DOE, 2009ev), the applicant stated that the finite element mesh size was sufficiently refined and that a more closely spaced mesh would not change the calculated design forces significantly. Additionally, the applicant stated that design of the mat foundation has more than 30 percent margin, which would accommodate increases in the calculated design forces from an analysis using a more closely spaced finite element mesh.

For the foundation design method for seismic loads, the applicant modeled the mat foundations using shell elements. The applicant modeled the soil foundation as nonlinear translational springs at each node of the basemat model to represent the soil stiffness. The applicant obtained global soil spring stiffness values from the impedance functions, consistent with Section 3.3.4.2 of ASCE 4–98 (ASCE, 2000aa), based on an alluvium depth of 30.5 m [100 ft] to maximize the design forces in the basemat. The applicant then used these stiffness values to determine individual springs at each basemat node in the model. The applicant calculated horizontal soil spring stiffness per unit area by dividing the horizontal global soil stiffness by the area of the basemat. The applicant calculated vertical soil spring stiffness per unit area three ways, one using the vertical soil global stiffness, and the other two by using the global rotational springs about two horizontal axes. The applicant then used the minimum of the three vertical soil spring stiffness values to maximize the basemat design forces. In response to the NRC staff's RAI (DOE, 2009eu), the applicant stated that the effect of not including rocking global soil springs was not significant, because of the relatively small contribution from rocking modes result from vibration.

The applicant's design method estimated the maximum soil-bearing pressure for the CRCF foundation mat for extreme loading (seismic) conditions to be 546 kPa [11.4 ksf] (BSC, 2007ae), which is less than the applicant's allowable soil-bearing pressure of 2,394 kPa [50 ksf], as described in BSC (2007ba, Section 6.2.3) and reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.3.2.

**NRC Staff's Evaluation of the Structural Design Methods for the ITS Surface Facilities Buildings**

The NRC staff evaluated the applicant's information on the shear wall design method of surface buildings and finds that the shear wall design method is acceptable because the design method is consistent with the standard engineering practice for the design of shear walls for surface buildings at nuclear power plants (ASCE, 2005aa). The surface buildings at the GROA are similar in design to those at a nuclear power plant; and therefore, the NRC staff finds use of this practice acceptable for use at the GROA, as further described in Table 7-1. The NRC staff also evaluated the applicant's determination of the demand/capacity (D/C) ratios of the concrete shear walls for buildings to confirm that these structures would be able to withstand DBGM–2 seismic events (D/C ratios ≤ 1.0). The NRC staff evaluated the applicant's RAI response (DOE, 2009eu) on the use of the seismic analysis results used for the surface building shear walls design (BSC, 2006ak; BSC, 2007da,cx,aq), which were based on bounding soil properties. The NRC staff determines that the seismic analyses of surface facilities buildings using bounding soil properties for the alluvium depth of 10.7 m [35 ft] is conservative because the smaller alluvium depth results in relatively greater amounts of earthquake energy transmitted to the structures, thus, resulting in higher seismic forces than for the seismic analyses with alluvium depths greater than 10.7 m [35 ft].

The NRC staff evaluated the applicant's information on the slab design method for the ITS surface facilities buildings and finds that the applicant's slab design method is adequate for ITS buildings subjected to the DBGM–2 seismic events because the design method is based on the codes and standards that are consistent with the standard engineering practice [ACI 349–01 (ACI, 2001aa)] that the NRC staff finds is acceptable for use at the GROA. Applicability of this standard to the proposed activities at the GROA is discussed further in Table 7-1.

The applicant considered various loads and load combinations, including dead, live, and seismic loads, to determine the load demand. Then the applicant compared that value with the slab design capacity, in accordance with the ACI 349–01 code. The seismic loads at the DBGM–2 level were considered at a MAPE of $5 \times 10^{-4}$ year, which the NRC staff evaluates in SER Section 2.1.1.7.3.1.1 and finds to be acceptable. The NRC staff also reviewed the D/C ratios of the concrete shear walls and finds that these structures are adequate to withstand DBGM–2 seismic events because the applicant performed its analyses and design consistent with existing codes, thus providing sufficient design margin.

The NRC staff finds that the applicant's use of an amplification factor of two as an equivalent static factor for out-of-plane seismic force calculations to design the slab is acceptable because it is consistent with the NUREG–0800 Section 3.7.2 II.1.B.iii Acceptance Criteria (NRC, 2013ac).

The NRC staff finds that the applicant's approach of considering the two-way slab as a one-way slab and providing the same reinforcement in both directions is conservative because the design forces and moments for two-way slab would be less than those for a one-way slab. The NRC staff further finds that the applicant's approach of considering the multiple-span diaphragms as simple spans for in-plane loads and using the largest span for the analysis is conservative because it will result in larger design moments and shear forces. The NRC staff also finds that the applicant's approach of not considering the W-shape steel beams as composite with the slab is conservative, because the potential contribution of the slab to the capacity of the beams is not considered.

The NRC staff evaluated the applicant's information on the foundation design method of the ITS surface facilities buildings, and finds that the applicant's foundation design is acceptable because it used codes and standards that are consistent with NRC guidance [Section 3.8.5, NUREG–0800 (NRC, 2013ag)] for foundation design methods (see Table 7-1 for additional information on the applicability of this guidance to the GROA). Specifically, the modeling of the foundation mat as shell elements, the structural walls as beam-columns, and the soil as nonlinear compression-only springs in the SAP2000 software for the finite element analysis of the mat foundation is consistent with the standard industry practice of using ASCE 4–98 (ASCE, 2000aa) for nuclear power plants. The applicability and acceptability of ASCE 4–98 for use at the GROA is further described in Table 7-1. The NRC staff also finds that the applicant's mesh size of 1.5 m [5 ft] × 1.5 m [5 ft] for the mat elements is appropriate for estimating the design moments and shears in the mat (BSC, 2007ae), based on the size of the mesh relative to the dimensions of the mat.

The NRC staff determines that the applicant's method for computing nodal spring stiffness values based on global spring stiffness of the foundation mat is acceptable because the applicant distributed the vertical and horizontal global stiffness values to the foundation nodes, based on the area of the mat of each node. Also, the NRC staff finds that the applicant's approach of not considering the effects of rotational global springs in the vertical nodal springs is acceptable because the rocking behavior of the buildings during a seismic event would not be significant, based on the height of the building compared the width and length of the buildings (i.e., low height and large area buildings are not subject to significant rocking during seismic events, consistent with the applicant's evaluation of vertical and rocking global spring stiffness), and thus would not affect foundation mat design forces. Further, the NRC staff's review of the applicant's use of the allowable bearing pressures of 479 kPa [10 ksf] for normal loads and 2,395 kPa [50 ksf] under extreme loading (seismic) conditions is provided in SER Section 2.1.1.1.3.5.3.2. Therefore, the NRC staff finds that the applicant's evaluation of foundation design is adequate because the applicant used bounding allowable bearing pressures.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's proposed structural design methods for the analyses and design of (i) shear walls, (ii) slabs, and (iii) foundation of the ITS surface buildings is acceptable.

2.1.1.7.3.1.1.3.2 Facility-Specific Analysis and Design Procedures

This section provides the NRC staff's evaluation of specific aspects of analysis and design that are unique to each of the surface ITS buildings, as discussed by the applicant in SAR Sections 1.2.3 through 1.2.8.

**CRCF and RF**

The CRCF and RF buildings are reviewed together in this section because these buildings are both concrete shear wall structures, analyzed and designed using the same design methods as proposed by the applicant. These buildings do not have unique aspects like the pool in the WHF and structural steel framed structures for the IHF, which are evaluated separately, below. According to the design information provided by the applicant, the CRCF and RF are multistory structures, consisting of reinforced concrete shear walls, floor slabs, roof diaphragms, and a mat foundation. The CRCF dimensions are approximately 119 m [392 ft] wide by 128 m [420 ft] long by 30.5 m [100 ft] high. The reinforced concrete shear walls are 1.2 m [4 ft] thick, and most of the reinforced concrete foundation mat is 1.8 m [6 ft] thick, except in the waste package loadout room, where it is thicker to accommodate the operation of the waste package transfer trolley (SAR Section 1.2.4.1.1). The diaphragm slabs are 0.5 m [1.5 ft] and 0.8 m [2.75 ft] thick for roof, and 0.5 m [1.5 ft] and 1.2 m [4 ft] thick for floors (BSC, 2007ct).

Similarly, the RF building footprint dimensions are approximately 96 m [315 ft] wide by 97 m [318 ft] long. The part of the RF building above grade, where the radioactive material is received and handled, is ITS and has dimensions of 61 m [200 ft] wide by 73 m [240 ft] long by 30.5 m [100 ft] high (SAR Figure 1.2.6-2), as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6.3.1. The superstructure has 1.2-m [4-ft]-thick exterior and interior concrete walls. The internal shielded rooms have 1.2-m [4-ft]-thick concrete walls and roof slabs. The RF foundation mat is 2.1 m [7 ft] thick, except for the East Wing and West Wing areas {approximately 22.6 m [74 ft] × 13.1 m [43 ft]}, where the mat is 1.2 m [4 ft] thick (BSC, 2007ax). The elevated floor diaphragm slabs are 0.46 m [1.5 ft] thick.

***CRCF and RF Structural Analyses***

The applicant performed seismic structural analyses for the CRCF and RF, based on lumped-mass stick models subjected to the DBGM–2 seismic events, in which SSI was represented with global soil springs with six degrees of freedom placed at the center of mass of the basemat foundation, as described in BSC (2007bx, Section 6.1) and BSC (2007az, Section 6.1).

For the structural analysis of the CRCF building, the applicant performed response spectrum modal analysis for six soil conditions corresponding to soil springs representing 30.5 m [100 ft] and 61 m [200 ft] depths of alluvium for lower bound, median, and upper bound soil properties (BSC, 2008af). The applicant presented the story shears of the CRCF for the DBGM–2 seismic events in BSC (2007af, Table 18), showing that shear forces were controlled by the 30.5 m [100 ft] upper bound soil case. The applicant also presented the interstory drift ratios, $\gamma$ (story displacement divided by the story height, radians) for the upper bound soil condition in BSC (2007bx, Table 16), where the maximum drift ratio was calculated to be 0.0122 percent. The applicant compared this value to the drift ratio limit of 0.4 percent, as detailed in BSC (2007av, Section 4.2.11.4.10), and recommended in ASCE 43–05 (ASCE, 2005aa) for systems designed to experience limited permanent distortion (e.g., Limit State C in ASCE 43–05).

For the structural analysis of the RF building, the applicant performed a response spectrum modal analysis for soil conditions representing a 40 m [131.2 ft] alluvium depth for lower bound, median, and upper bound stiffness values (BSC, 2008bf). The applicant indicated in BSC (2007az, Section 7.1) that the upper bound soil case for alluvium depth of 10.7 m [35 ft] provided the highest reactions and accelerations. The largest interstory drift ratio was calculated to be 0.0127 percent, as shown in BSC (2007az, Table C6). This value was lower than the recommended interstory drift ratio limit of 0.4 percent (ASCE, 2005aa).

**NRC Staff’s Evaluation of CRCF and RF Structural Analyses**

The NRC staff evaluated the applicant’s structural analyses information on the CRCF and RF buildings seismic analyses for DBGM–2 seismic event and finds the buildings seismic analyses acceptable because (i) the method is based on the approach reviewed previously and found to be acceptable by the NRC staff in SER Section, “Seismic Analysis Method” (Section 2.1.1.7.3.1.1.3.1), which is applicable to the CRCF and RF buildings because they are concrete shear wall type buildings; and (ii) the structural analyses were performed using a method that is consistent with the standard engineering practice for seismic analysis of nuclear surface facility buildings for a nuclear power plant [Section 3.7.2, NUREG–0800 (NRC, 2013ac)].

The NRC staff determines that the seismic analysis of the RF building using the upper bound soil case for alluvium depth of 10.7 m [35 ft] is acceptable because it results in the highest forces for use in the design, and thus is conservative, as described above in SER Section 2.1.1.7.3.1.1.3.1. Additionally, the NRC staff determines that the interstory drift ratios calculated by the applicant are acceptable because they are less than the recommended limit of 0.4 percent (ASCE, 2005aa) and are, therefore, conservative.

***<u>CRCF and RF Shear Wall Design</u>***

For the shear wall design of the CRCF, the applicant computed the maximum D/C ratios of the shear walls as 0.74 and 0.83 for in-plane shear and bending, respectively (BSC, 2007dh). For the RF shear wall design, the maximum D/C ratio for in-plane shear was 0.57, whereas the maximum D/C ratio for bending and axial loads was 0.76, as described in BSC (2007cy, Section 7). The design calculations were based on the ACI 349–01 (ACI, 2001aa) code.

***<u>CRCF and RF Slab Design</u>***

For the CRCF concrete slab design, the applicant computed reinforcement requirements for out-of-plane bending loads, in-plane diaphragm shear, and in-plane diaphragm moments, as detailed in BSC (2007ct, Section 4.3). The applicant stated that it performed design calculations based on the ACI 349–01 (ACI, 2001aa) recommendations. The applicant calculated that the maximum ratio of the reinforcement required (demand) and the reinforcement provided (capacity), or the Demand/Capacity (D/C) ratio, was 0.71.

For the design of the structural steel framing supporting the concrete slabs (BSC, 2007cu), the applicant stated that it used the ANSI/AISC N690–1994 (AISC, 1994aa) code. The structural steel framing for the CRCF included a Type 2 construction with steel floor and roof decking, simply supported beams on girders and columns, and trusses supporting a portion of the roof. Design loads and load combinations were described in the “Project Design Criteria” (BSC, 2007av) and the “Seismic Analysis and Design Approach Document” (BSC, 2007ba). The analysis of the framing was based on the assumption that the beams and girders are noncomposite with the slab. The applicant stated that the calculated maximum demand/capacity (D/C) ratios were less than 0.85 for beams/girders in flexure, 0.81 for truss members under axial loads, and 0.82 for axial loads on columns.

The applicant provided the RF concrete slab design results in BSC (2008cj, Section 7). The applicant stated that it performed the design calculations based on the ACI 349–01 (ACI, 2001aa) code. For the RF concrete slab design, reinforcement requirements were computed for out-of-plane bending loads, in-plane diaphragm shear, and in-plane diaphragm moments, as detailed in BSC (2008cj, Section 4.3). The applicant calculated that the maximum ratio of the reinforcement required and the reinforcement provided, or the D/C ratio, was 0.66.

The applicant provided the design of the RF structural steel framing that would support the reinforced concrete slabs (BSC, 2007cz), and stated that the structural steel framing for the RF included floor and roof decking supported on steel beams. The applicant also stated that the structural steel framing for the RF was designed using the same approach as for the CRCF. The applicant calculated that the maximum D/C ratio was 0.82 for the steel beams in flexure. The applicant also stated that the structural steel framing did not include steel columns or the trusses.

***CRCF and RF Foundation Design***

For the seismic analyses of the CRCF and RF foundations design, the applicant developed finite element models of the basemat and coupled them with the superstructure, as outlined in BSC (2007ae, Section 6.1) for CRCF and BSC (2007ax, Section 4.3) for RF. As described previously in SER Section 2.1.1.7.3.1.1.3.1, the applicant (i) used the SAP2000 software (Computers and Structures, 2005aa) to analyze the CRCF and RF foundation mats and (ii) modeled the soil supporting the mats using nonlinear spring elements.

The applicant's foundation designs for CRCF and RF were based on the seismic analysis results, as described previously in SER Section 2.1.1.7.3.1.1.3.1. For the RF foundation flexural reinforcement, a standard rebar pattern from ACI (2001aa) was selected by the applicant, and contour plots of bending moments were used to design reinforcement. The applicant computed the shear capacity of the concrete (without any shear reinforcing) and used the shear contour plots to determine the areas of the basemat foundation requiring transverse shear reinforcing. For the CRCF 1.8-m [6-ft]-thick mat foundation, the maximum moment and shear D/C ratios were calculated to be 0.69 and 0.67, respectively, as outlined in BSC (2007ae, Section 6.5). For the RF foundation, the maximum bending moment and shear D/C ratios for the 2.1 m [7 ft] mat were calculated to be 0.85 and 0.56, respectively (BSC, 2007ax). For the 1.2 m [4 ft] mat of the RF, the maximum bending moment and shear D/C ratios were calculated to be 0.92 and 0.87, respectively.

For the design of CRCF and RF foundations, the applicant calculated the maximum bearing pressure on the mat foundation by dividing the maximum reaction force of the individual springs by the area of the corresponding shell element, as described in BSC (2007ax, Section 6.4.2). For the CRCF foundation, the applicant calculated the maximum bearing pressure under the mat foundation as 545 kPa [11.4 ksf], as detailed in BSC (2007ae, Section 6.4.2). For the RF foundation, the applicant calculated the maximum bearing pressure as 488 kPa [10.2 ksf]. For both foundations, the applicant calculated the maximum bearing pressure to be less than the allowable bearing pressure of 2,394 kPa [50 ksf] that the applicant proposed for the extreme seismic loading condition, as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.3.2. The applicant modeled the foundation as soil springs effective for compressive loads only.

For the CRCF foundation design, the applicant computed a safety factor of 3.1 against the structure overturning, showing that the restoring moments from the weight of the building structure about the building edge was 3.1 times the overturning moments from the horizontal seismic forces. For the foundation's resistance against sliding, however, the applicant computed that the safety factor was 0.68, which indicated that the CRCF building may slide during a seismic event. The applicant computed the expected sliding displacement using the approximate reserve energy approach (ASCE, 2005aa). This analysis yielded a sliding displacement of 5.1 mm [0.2 in] during DBGM–2 seismic events. For the RF, the safety factor for overturning stability was computed as 2.99, but the safety factor against sliding was 0.727

for DBGM–2 seismic events (BSC, 2007ax). The applicant calculated an RF sliding displacement of 4.6 mm [0.18 in], using the approximate reserve energy approach (ASCE, 2005aa). To ensure that the potential building displacement during a seismic event does not adversely impact the utility systems entering and exiting the building, the applicant stated in BSC (2007ba, Section 11.1.1) that the connections would be designed to accommodate a sliding displacement of at least 10.2 mm [0.4 in], as outlined in BSC (2007ae, Section 6.6.1.2) and BSC (2007ax, Section 6.6.1).

**NRC Staff's Evaluation of the CRCF and RF Shear Walls, Slabs, and Foundation Design**

The NRC staff evaluated the applicant's information on the design of the CRCF and RF buildings' shear walls and finds it acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously and found to be acceptable in SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the CRCF and RF buildings because they are also concrete shear wall type buildings; (ii) the design used the codes and standards that are consistent with the standard engineering practices and that the NRC staff finds acceptable for use at the GROA, as further discussed in Table 7-1; and (iii) D/C ratios were less than 1.0, thus the provisions of the design code are met.

The NRC staff evaluated the applicant's information on the CRCF and RF buildings' slabs design (BSC, 2007ct; BSC, 2008cj) and finds that the slab design the applicant provided is acceptable for DBGM–2 seismic events because (i) the applicant used the design method that is evaluated and found to be acceptable by the NRC staff in SER Section 2.1.1.7.3.1.1.3.1; (ii) the design used the codes and standards (ACI, 2001aa, AISC, 1994aa) that are consistent with the standard engineering practices and that the NRC staff finds acceptable for use at the GROA, as further discussed in Table 7-1; and (iii) the design has adequate margins to the code-specified limits because the maximum D/C ratio is 0.71. The NRC staff evaluated the applicant's structural steel framing designs for the CRCF and RF (BSC, 2007cu,cz) and finds the design of the steel framing for the CRCF and RF acceptable because (i) the design inputs (e.g., DBGM–2 vertical accelerations), (ii) structural steel materials, (iii) analysis method, and (iv) the load combinations are consistent with the project design documents (BSC, 2007av,ba; BSC, 2006ak) and the referenced design code (AISC, 1994aa). Therefore, the NRC staff concludes that the information provided by the applicant on the design of the CRCF and RF concrete slab and supporting structural steel framing is acceptable.

The NRC staff evaluated the applicant's information on the CRCF and RF buildings foundation analyses and designs and finds that the applicant's seismic analyses and design of the CRCF and RF foundations for the Tier #1 analyses of DBGM–2 seismic events are acceptable because the analysis method, design method, design codes and standards, load combinations, and materials used are the same as the general analyses and design procedures reviewed previously by the NRC staff in SER Section 2.1.1.7.3.1.1.3.1 and found acceptable. The NRC staff evaluated the applicant's information to verify that the maximum D/C ratios for bending moments and shears were less than 1.0, and is, therefore, consistent with the design code (ACI, 2001aa), which the NRC finds acceptable for use at the GROA, as further discussed in Table 7-1.

The NRC staff finds that the applicant's design analyses with respect to the overturning of the CRCF and RF buildings is acceptable because the estimated safety factors against the overturning of the structure are greater than 1.5, a margin which is used for similar facilities at a nuclear power plant [NUREG–0800 Section 3.8.5 (NRC, 2013ag)]. The NRC staff also finds that the applicant's design analyses, with respect to sliding displacement, is acceptable because connections entering the structure would have sufficient flexibility to accommodate a sliding displacement of 10.2 mm [0.4 in.], which provides for a margin of 5.1 mm [0.2 in] for sliding displacement when compared with the maximum calculated sliding displacement of 5.1 mm [0.2 in]. Additionally, the NRC staff determines that (i) the CRCF and RF, as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.2.3.1, would be separated by a sufficient distance from other surface facilities (SAR Figure 1.1-2) and that there would not be any interaction between them due to sliding of the building during a seismic event and (ii) use of a nonlinear time-history analysis to estimate the sliding displacement is consistent with ASCE 43–05 (ASCE, 2005aa), and appropriate for the GROA, as discussed in Table 7-1. Therefore, the NRC staff finds that the CRCF and RF foundation analyses and designs are acceptable.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's proposed structural design and supporting analyses of the CRCF and RF buildings' shear walls, slabs, and foundation are acceptable.

**Wet Handing Facility**

The Wet Handling Facility (WHF) is a reinforced concrete structure that consists of concrete shear walls, roof slab diaphragms, mat foundations, and a pool (SAR Section 1.2.5). The overall footprint of the WHF building is approximately 117 m [385 ft] by 120 m [395 ft], and the ITS portion of the building is about 117 m [385 ft] by 91 m [300 ft]. The maximum height of the building is 30.5 m [100 ft] above grade. The below grade portion of the structure consists of 2.4-m [8-ft]-thick exterior earth retaining walls and 1.2-m [4-ft]-thick walls separating interior rooms from the pool. The below-grade pool substructure is approximately 35 m [116 ft] by 35 m [116 ft]. The internal dimensions of the pool are 22.5 m [74 ft] wide, 19 m [61 ft] long, and 16 m [52 ft] below grade. The normal pool water level is 1.2 m [4 ft] below grade, thus providing normal pool water depth of 14.6 m [48 ft]. The at-grade foundation mat is 1.8 m [6 ft] thick, whereas the pool foundation mat is 2.4 m [8 ft] thick. The main WHF superstructure has 1.2 m [4 ft] thick concrete walls. The floor diaphragm slabs are 0.46 m [1.5 ft] to 0.61 m [2.0 ft] thick, except the internal shielded rooms, whose slabs are 1.2 m [4 ft] thick.

***WHF Structural Analyses***

The applicant analyzed the WHF building based on a Tier #1 lumped-mass multi-stick model for the response spectrum modal analysis (BSC, 2007cx). To incorporate the pool in the structural model, an additional set of global soil springs was attached to the pool foundation. The applicant's analysis model did not consider the lateral restraining effect of the soil on the underground pool. The water mass of the pool was divided equally and included in the mass at the building floor level and at the bottom of the pool.

The applicant performed a response spectrum modal analysis of the WHF (BSC, 2007cx) using the analysis method described in the SAR Section 1.2.2.1.6.3, and the hybrid spectra, as shown in BSC (2007bm, Tables 1–12). The applicant performed the analyses for six soil conditions corresponding to soil springs that represent alluvium depths of 9.1 m [30 ft] and 30.5 m [100 ft] for upper, median, and lower bound soil properties. The model with the upper bound 9.1 m [30 ft] alluvium soil case was the stiffest model, as shown in BSC (2007bm, Tables 1–6), resulting in a fundamental period of vibration of $T_1$ = 0.157 s. The maximum interstory drift ratio occurred for the 30.5 m [100 ft] lower bound soil condition, where $T_1$ = 0.26 s. The applicant stated that the maximum interstory drift ratio was 0.018 percent as described in BSC (2007bm, Table 15), which is one order of magnitude lower than the interstory drift ratio of 0.4 percent.

The applicant indicated in BSC (2007bm, Table 23) that the upper bound 9.1 m [30 ft] alluvium soil case controlled the shear forces in the east-west direction, whereas the upper bound 30.5 m [100 ft] alluvium condition controlled the structural response in the north-south direction.

The applicant's analyses included the effects of water sloshing in the WHF pool due to a seismic event (BSC, 2007di). This analysis by the applicant determined water pressures imposed on the pool walls and the amount of freeboard required for preventing pool water spilling caused by sloshing during a seismic event. The applicant performed the analysis using the ACI 350.3 (ACI, 2001ab) code, assuming the pool walls embedded in the ground were rigid. The sloshing effects were evaluated for DBGM–2 and Beyond Design Basis Ground Motion (BDBGM) at mean Annual Probability of Exceedance (MAPE) of $10^{-4}$ earthquake events (BSC, 2007di; SAR Section 1.2.2.1.6.3.2.3). The applicant concluded that the sloshing during a DBGM–2 event may raise the water level by 0.76 m [2.5 ft], and for the BDBGM event, 2.2 m [7.2 ft]. The free board provided in the pool is 1.22 m [4 ft], which is sufficient to prevent pool water from sloshing onto the WHF floor during a DBGM–2 seismic event. During a BDBGM seismic event, however, the pool water may overflow onto the WHF floor. The applicant stated (BSC, 2007di) that it would include sloped floors and curbs to contain any fluid that may slosh out of the pool within the WHF. Additionally, the applicant stated that ample water would remain in the pool to provide the necessary shielding because, under the conditions when fuel elements are not being handled, a minimum of 6.4 m [21 ft] of water covers the casks and related fuel elements. The applicant also stated that the occurrence of a BDBGM event at the same time fuel elements are being handled has extremely low probability.

**NRC Staff's Evaluation of the Structural Analyses for the WHF**

The NRC staff evaluated the applicant's structural analysis information on the WHF building seismic analyses for the DBGM–2 seismic events (BSC, 2007cx) and finds it acceptable because (i) the analysis method is based on the approach reviewed previously and found to be acceptable by the NRC staff in the SER section, "Seismic Analysis Method" (Section 2.1.1.7.3.1.1.3.1), and this method is applicable to the WHF building because it is a concrete shear wall type building and (ii) the applicant considered a range of soil properties (upper, median, and lower bound) and alluvium depths of 9.1 m [30 ft] and 30.5 m [100 ft] and used the bounding results from that evaluation for design from north-south and east-west direction earthquake analyses. The NRC staff finds that the range of alluvium depths of 9.1 m [30 ft] and 30.5 m [100 ft] and the range of soil properties (upper, median, and lower bound) used by the applicant are acceptable, as described in the NRC staff's evaluation in SER Section 2.1.1.1.3.5.1.1. The NRC staff determines that the modeling of the pool water as added mass with the floor and foundation levels in the seismic analysis of the WHF is acceptable because it is consistent with the analysis method of using the Tier #1 lumped-mass multi-stick model, described above and found to be acceptable by the NRC staff.

Additionally, the NRC staff determines that the interstory drift ratios calculated by the applicant are acceptable because they are less than the recommended limit of 0.4 percent (ASCE, 2005aa), and is, therefore, conservative.

The NRC staff also reviewed the applicant's structural analysis information to evaluate the applicant's estimate of the WHF pool sloshing effects (BSC, 2007di) for DBGM–2 and BDBGM seismic events, and finds it acceptable because the analysis is consistent with the standard industry practices in ACI 350.3 (ACI, 2001ab), which is acceptable for use at the GROA, as discussed in Table 7-1. The NRC staff finds that the freeboard of 1.2 m [4 ft] is sufficient to contain water during a DBGM–2 event, and thus is acceptable. The NRC staff also finds that potential overflow of pool water onto the WHF floor caused by sloshing effects during a BDBGM seismic event, is acceptable, because (i) the applicant's design (BSC, 2007di) would include sloped floors and curbs to contain any fluid that sloshes out of the pool during a BDBGM seismic event within the WHF; and (ii) the depths of pool water, 7.0 m [23 ft] (SAR Table 1.9-10 PSC-22) when spent fuel is not being handled, and 3.2 m [10.5 ft] (SAR Section 1.10.3.5.5) when the spent fuel is being handled, are sufficient to provide shielding, as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.2.3.1).

***WHF Shear Wall Design***

The applicant described the WHF shear walls design for seismic loads, as detailed in BSC (2007cv, Section 7), and calculated that the maximum D/C ratio for in-plane shear was 0.71, whereas the maximum D/C ratio for bending and axial loads was calculated to be 0.77. The applicant stated that the design calculations were based on the ACI 349–01 (ACI, 2001aa) code.

***WHF Slab Design***

The applicant provided the WHF design calculation of five representative reinforced concrete slabs at various floors, bounding a range of lateral dimensions and slab thickness values, using ACI 349–01 (ACI, 2001aa) code in BSC (2007cw, Section 7). The applicant provided the design of the structural steel framing that would support the reinforced concrete slabs in BSC (2007cu). The structural steel framing for the WHF included steel floor and roof decking, steel beams, girders, columns, and trusses (BSC, 2007dj). The applicant stated that the design of the structural steel framing for the WHF used the ANSI/AISC N690–1994 (AISC, 1994aa) code. The applicant stated that the structural steel framing for the CRCF included a Type 2 construction with steel floor and roof decking, simply supported beams on girders and columns, and trusses supporting a portion of the roof. Design loads and load combinations were described in the "Project Design Criteria" (BSC, 2007av) and the "Seismic Analysis and Design Approach Document" (BSC, 2007ba). The applicant stated that the analysis of the framing was based on the assumption that the beams and girders are noncomposite with the slab. The applicant stated that the calculated maximum D/C ratio was 0.69 for beams/girders in flexure, 0.70 for axial loads in truss members, and 0.68 for axial loads on columns.

***WHF Foundation Design***

According to the applicant, the WHF foundation consisted of a 1.83-m [6-ft] -thick reinforced concrete slab at the grade level, a 2.44-m [8-ft]-thick pool base slab, and 1.8-m [6-ft]-thick retaining walls. Design of the WHF foundation was based on the finite element model for the basemat foundation and the pool structure, coupled with the Tier #1 model, as outlined in BSC (2007bl, Section 4.3). Shear walls on top of the grade basemat were included in the applicant's model to consider the stiffening effects of the walls. The applied loading combinations included dead, live, hydrostatic, lateral earth pressure, surcharge pressure, hydrodynamic, and seismic loads, as described in BSC (2007bl, Sections 4.3 and 6.3).
To account for SSI, the applicant calculated soil spring constants based on the impedance functions, as described previously in the SER Section, "Foundation Design Method" (Section 2.1.1.7.3.1.1.3.1). The applicant stated that the design calculations for the WHF foundation were based on the ACI 349–01 (ACI, 2001aa) code.

The applicant analyzed the WHF foundation basemat, as described in BSC (2007bl, Sections 3.1 and 6), using the WHF building seismic analysis results for the upper bound soil properties of 10.7-m [35-ft]-thick alluvium (BSC, 2007cx), which were bounding, as stated by the applicant in response to the NRC staff's RAI (DOE, 2009eu). The applicant then generated moment and shear contour plots that were used to design the WHF foundation and to compute the shear and flexural reinforcement in the foundation basemat. The applicant proposed a typical reinforcing bars pattern from the ACI 349–01 code (ACI, 2001aa) in the basemat, with additional reinforcement in critical regions computed based on the moments and shear forces. For the grade basemat, the applicant computed maximum moment and shear D/C ratios of 0.85 and 0.82, respectively, as shown in BSC (2007bl, Tables 10 and 11). The maximum moment and shear D/C ratios were 0.79 for the exterior and pool walls and 0.83 for the exterior wall, as shown in BSC (2007bl, Tables 10 and 11).

The applicant calculated maximum bearing pressure on the basemat by multiplying the maximum vertical deflections of joints connecting the link elements and the equivalent subgrade moduli, as described in BSC (2007bl, Section 6.5.3). The maximum bearing pressure, based on linear elastic characteristics, was estimated to be 713 kPa [14.9 ksf] for the grade basemat and 2,006 kPa [41.9 ksf] for the pool basemat. The applicant compared and found that these bearing pressures were smaller than the allowable bearing capacity of 2,394 kPa [50 ksf], reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.4.

The applicant evaluated the overall stability of the WHF, as outlined in BSC (2007bl, Section 6.7), obtaining an overturning safety factor of 2.66. However, the critical foundation resistance against sliding during a seismic event resulted in a safety factor of 0.363. The applicant used the reserve energy approach (ASCE, 2005aa), obtaining a sliding displacement of 3.5 mm [0.14 in]. The applicant concluded that sliding of the WHF building would not impact the intended safety function of the structure, because any commodities or utilities entering the structure, or clearance of any adjacent structures, would accommodate a displacement of 10.2 mm [0.4 in], thus providing a safety margin of [10.2–3.5] = 6.7 mm [0.26 in] in the design to accommodate building sliding displacement.

**NRC Staff's Evaluation of the WHF Shear Walls, Slabs, and Foundation Design**

The NRC staff evaluated the applicant's information on the WHF building shear walls design and finds it acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously and found to be acceptable in SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the WHF building because it is a concrete shear wall type building; (ii) the design used the codes and standards that are consistent with the standard engineering practices, and the NRC staff finds their use to be acceptable at the GROA, as further discussed in Table 7-1; and (iii) the NRC's staff's evaluation of the applicant's design (BSC, 2007cv) finds that the D/C ratios for the shear wall are less than 0.77, which is less than 1.0, consistent with the design code.

The NRC staff evaluated the applicant's information on the WHF building concrete slab design and finds that the design of slabs for the DBGM–2 seismic events is acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously and found to be acceptable in SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the WHF because the WHF building is a concrete shear wall type building; (ii) the design used the codes and standards that are consistent with the standard engineering practice [Section 3.8.4 of NUREG–0800 (NRC, 2013ad), ASCE 4–98 (ASCE, 2000aa)], the applicability of which to the GROA is discussed in Table 7-1; and (iii) the NRC staff's evaluation of the applicant's information (BSC, 2007dj) confirmed that the D/C ratios for the shear wall were less than 0.70, and thus less than 1.0, consistent with the code.

The NRC staff evaluated the applicant's structural steel framing design supporting the concrete slab for WHF (BSC, 2007dj) and finds the design of the steel framing for WHF acceptable because (i) the design inputs (e.g., DBGM–2 vertical accelerations), (ii) structural steel materials, (iii) analysis method, and (iv) the load combinations are consistent with the project design documents (BSC, 2007av,ba; BSC, 2006ak) and the selected and appropriate design code (AISC, 1994aa). Therefore, the NRC staff concludes that the information provided by the applicant on the design of the WHF concrete slab and the supporting structural steel framing is acceptable.

The NRC staff evaluated the applicant's information on the WHF foundation design and finds that the analysis and design of the WHF foundation for the DBGM–2 seismic events is acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously and found to be acceptable in SER Section 2.1.1.7.3.1.1.3.1, which is applicable to the WHF building because it is a concrete shear wall type building; (ii) the design and analyses are consistent with the standard engineering practices [Section 3.8.5 of NUREG–0800 (NRC, 2013ag)], reviewed for use at the GROA, as discussed in Table 7-1; and (iii) the seismic analyses results of the WHF using bounding soil properties for the alluvium depth of 10.7 m [35 ft] is conservative because the smaller alluvium depth results in relatively greater amounts of earthquake energy transmitted to the structures, thus, resulting in higher seismic forces than for the seismic analyses with alluvium depths greater than 10.7 m [35 ft].

The NRC staff finds that the applicant's design of the connections entering the structure and clearance between the WHF and other structures, to accommodate a sliding displacement of 10.2 mm [0.4 in] is adequate because this design provides for a margin of 6.7 mm [0.26 in] for sliding displacement when compared with the applicant-calculated WHF sliding displacement of 3.5 mm [0.14 in]. Additionally, the NRC staff determines that the WHF is sufficiently far from other surface facilities (SAR Figure 1.1-2), as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.2.3.1, and thus would not interact with other buildings due to sliding during a seismic event. The NRC staff finds that, based on the NRC staff's experience and judgment, that sliding of the building of a conservative magnitude of less than 5.1 mm [0.2 in] as a rigid body would not affect the building function, and thus is acceptable.

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's proposed structural analyses and design of the WHF building shear walls, slabs, and foundation is acceptable.

**Initial Handling Facility**

According to the applicant, the IHF would be composed of two seismically independent ITS surface structures, the main structure and the waste package loadout room. The IHF cask handling process area main structure is a braced-frame steel structure approximately 52 m [170 ft] wide, 57 m [187 ft] long, and 32 m [105 ft] high. The main structure includes an internal reinforced concrete structure consisting of 1.2-m [4-ft]-thick walls and roof that provides the shielding and structural support for canister transfer and waste package loading/closure operations within the main structure. The main steel structure and the internal concrete structure have 1.8-m [6-ft]-thick common mat foundation. The IHF waste package loadout room is a reinforced concrete structure approximately 12.5 m [41 ft] wide, 43 m [140 ft] long (excluding external north–south concrete buttresses), and 18.3 m [60 ft] high. The concrete mat foundation for the IHF waste package loadout room is structurally and seismically separate from the IHF main structure concrete mat foundation, and is 1.8 m [6 ft] thick.

***IHF Structural Analyses***

The applicant analyzed the proposed IHF structures using the response spectrum method with DBGM–2 ground motions, as outlined in BSC (2008am, Section 1) for the steel frame structure, and BSC (2007aq, Section 1) for the concrete structures. For the steel frame structural analysis, the applicant did not include SSI, and assumed a fixed-base support (restrained in translation, but free to rotate) at the basemat, because the structure supports were designed as pinned connections, as described in BSC (2008am, Section 3.1.2). In response to the NRC staff's RAI (DOE, 2009ev), the applicant stated that SSI can be excluded for the braced-frame steel structure of IHF, based on ASCE 4–98 (ASCE, 2000aa), which is applicable to the GROA, as discussed in Table 7-1. The applicant stated that the design of column base connections would conform to ANSI/AISC 341–02 (AISC, 2002aa). For the response spectrum method, the analytical model included the mass of gravitational loads, snow load, and the crane payload, as described in BSC (2008am, Section 4.3 and 6.6), and structural damping of 7 percent. The maximum accelerations of the steel frame (2.26 g) occurred in components at elevations of 8.15 m [26.75 ft] and 11.3 m [37 ft], described in (BSC, 2008am; Section 7.1.1). The maximum displacement for the building was 50 mm [1.97 in] for a component at an elevation of 32 m [104.5 ft], as detailed in BSC (2008am, Table 7.1.11), which corresponded to the drift ratio (displacement/height) of 0.16 percent, and thus is less than the 0.4 percent maximum recommended in ASCE 43–05 (ASCE, 2005aa).

For the IHF reinforced concrete structures, the applicant created finite element models and performed response spectrum method dynamic analyses using DBGM–2 seismic design spectra data, as outlined in BSC (2007aq, Section 1.0). SSI effects were not included in the analysis and design of the concrete structures (BSC, 2007aq). In response to an NRC staff RAI (DOE, 2009ev), the applicant stated that the design in-plane shear forces for the walls would not be significantly affected by the inclusion of SSI effects, because (i) the dominant fixed-base mode response in the north-south direction was at or near peak spectral acceleration levels; and (ii) the design of the walls in the east-west direction was controlled by the minimum reinforcement standards of the ACI 349–01 code (ACI, 2001aa), and not by the calculated design forces for various load combinations, including those from the DBGM–2 seismic events, and thus, had significant margins in the design. The applicant stated that the demand/capacity (D/C) ratios for various structural elements varied from 0.10 to 0.69. The applicant also stated (DOE, 2009ev) that the finite element models used for the detailed design for construction will include soil-structure interaction effects.

**NRC Staff's Evaluation of the Structural Analyses for the IHF**

The NRC staff evaluated the applicant's information on the analyses of the IHF steel frame and reinforced concrete building structures and finds it acceptable because (i) the analysis method is based on the approach reviewed previously and found to be acceptable by the NRC staff in SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the IHF steel frame and reinforced concrete building structures because they are of the same type and (ii) the analysis is consistent with the standard engineering practice for the design of buildings for a nuclear power plant [Section 3.8.4 of NUREG–0800, (NRC, 2013ad)], which is applicable to the GROA, as discussed in Table 7-1.

The NRC staff evaluated the applicant's analyses of the steel structure (BSC, 2008am) and concrete structures (BSC, 2007aq), and finds that the applicant's assumption of a fixed-base support for the dynamic analyses of the IHF structures is acceptable because (i) the soil-structure interaction of the flexible steel structure with the concrete foundation would be minimal, which is consistent with the industry guidance in ASCE 4–98 (ASCE, 2000aa), the use of which is acceptable for the GROA, as discussed in Table 7-1 and (ii) the NRC staff evaluated the IHF concrete structures analysis and design and finds that the concrete structures design have sufficient margins to accommodate potential increase of seismic analysis design forces as a result of SSI.

***IHF Steel Frame Design***

The applicant stated that the design of the IHF steel braced-frame in BSC (2008am, Section 7.1.4), is consistent with the ANSI/AISC N690–1994 (AISC, 1994aa) code. The applicant analyzed the steel frame for various loads, including the seismic events, using the SAP 2000 software. The applicant calculated the D/C ratios for columns to be below 0.6 and the maximum D/C for the roof bracing group was 0.77.

***IHF Shear Walls Design***

The applicant stated that the use of in-plane and out-of-plane forces and moments in the IHF shear walls for the reinforced concrete component design is consistent with ACI 349–01 (ACI, 2001aa) code. The applicant further stated that the design forces and moments were obtained from the analysis of concrete structures using the SAP2000 software, as described in BSC (2007aq, Section 4.3.2.1). The applicant stated that the maximum D/C ratios for shear walls subjected to out-of-plane shear varied from 0.16 to 0.69, with the majority being less than 0.37. The applicant further stated that the maximum D/C ratios for the in-plane shear with tension varied from 0.10 to 0.53, with the majority being less than 0.41.

***IHF Slabs Design***

The applicant stated that the forces and moments used for the IHF slab design are consistent with the ACI 349–01 (ACI, 2001aa) code. The applicant further stated that the IHF slabs design forces and moments were based on the analysis of concrete structures using the SAP2000 software, as described in BSC (2007aq, Section 4.3.2.2). The results in BSC (2007aq, Section 6.6) indicated that the maximum D/C ratio for out-of-plane shear in the slabs was 0.68.

***IHF Foundation Design***

According to the design information provided by the applicant, the IHF foundation consisted of two individual basemats, which were modeled using finite element analysis. The applicant included the SSI effects in the foundation analysis, including the uncertainties in the soil properties, consistent with Section 3.7.2 of NUREG–0800 (NRC, 2013ac). To account for SSI, the applicant calculated soil spring constants based on the impedance functions, as described previously in this SER section, consistent with Section 3.3.4.2 of ASCE 4–98 (ASCE, 2000aa).

For the IHF foundation analysis, the applicant developed soil springs for an alluvium depth of 9.1 m [30 ft] and 30.5 m [100 ft], as stated in BSC (2008ar, Section 4.3.1). The applicant assumed in BSC (2008aq, Assumption 3.2.1) that the use of soil springs for the 30.5 m [100 ft] alluvium depth, which had the lower stiffness, would result in the maximum bending moments and shear forces in the mat foundation due to larger deformations than those based on using the soil springs for the 9.1 m [30 ft] alluvium depth. Also, according to SAR Figure 1.1-130, the alluvium thickness for the IHF facility varies from about 9.1 m [30 ft] to 27.4 m [90 ft], a soil condition that is not necessarily represented by uniform soil conditions at 9.1 m [30 ft] or 30.5 m

[100 ft]. The applicant stated further that the use of soil springs based on the 9.1 m [30 ft] alluvium depth would be conservative because the lower soil spring stiffness was used for the mat everywhere.

The applicant computed global soil springs stiffness values (BSC, 2008ar) to obtain the nodal spring stiffness values per unit area, as detailed in BSC (2008aq, Section 6.2). The applicant obtained soil spring stiffness per unit area solely from the global translational springs, according to the tributary areas for each joint. The applicant's calculations in BSC (2008aq, Section 6.5.3) stated that the maximum D/C ratios in the IHF foundation mat for moment and shear forces from the DBGM–2 seismic events varied from 0.25 to 0.97.

The applicant stated in BSC (2008aq, Section 6.7) that the soil-bearing pressure for the small mat could reach 2,107 kPa [44 ksf] when subjected to the DBGM–2 seismic events. This result led to a D/C ratio of 0.88, considering an allowable soil-bearing capacity of 2,394 kPa [50 ksf] for seismic loads, reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.3.2.

**NRC Staff's Evaluation of the IHF Steel Frame, Shear Walls, Slabs, and Foundation Design**

The NRC staff evaluated the applicant's information on the design of the IHF steel frame building and finds it acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously and found to be acceptable in the SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the IHF structural steel framing and concrete structure and component design; and (ii) the design used the codes and standards that are consistent with the standard engineering practices for design of buildings with similar waste handling operations, such as nuclear power plant facilities [Section 3.8.4 of NUREG–0800 (NRC, 2013ad)], as further discussed in Table 7-1.

The NRC staff evaluated the applicant's information on the IHF building shear walls design and finds it acceptable because: (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated and found to be acceptable previously in SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the IHF shear walls design because the IHF concrete structures are shear wall type buildings; (ii) the design used the codes and standards that are consistent with standard engineering practice (ACI, 2001aa; ASCE, 2000aa), the use of which is acceptable for the GROA, as discussed in Table 7-1; and (iii) the maximum D/C ratio is less than 0.69, which is less than 1.0, consistent with the design codes (ACI, 2001aa), the use of which for the GROA is acceptable, as discussed in Table 7-1.

The NRC staff evaluated the applicant's information on the IHF building slabs and finds it acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated and found to be acceptable in SER Section 2.1.1.7.3.1.1.3.1, and is applicable to the IHF building slabs design because the evaluation in SER Section 2.1.1.7.3.1.1.3.1 includes the slab design of surface buildings similar to the IHG building slabs; (ii) the design used codes and standards that are consistent with standard engineering practice [Section 3.8.4 of NUREG–0800 (NRC, 2013ad)], the use of which for the GROA is acceptable, as further discussed in Table 7-1; and (iii) the maximum D/C ratio is less than 0.68, which is less than 1.0, consistent with the design codes (ACI, 2001aa).

The NRC staff evaluated the applicant's information on the IHF building foundation design and finds it acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously in this SER Section 2.1.1.7.3.1.1.3.1, and this method is applicable to the IHF building foundation design because the evaluation in SER Section 2.1.1.7.3.1.1.3.1 includes the slab design of surface buildings; (ii) the analysis and design are consistent with Section 3.8.5 of NUREG–0800 (NRC, 2013ag), the use of which for the GROA is acceptable, as further discussed in Table 7-1; (iii) D/C ratios were less than 1.0, and therefore, the building would withstand a DBGM–2 event.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's proposed structural analyses and design of the IHF building steel frame building, shear walls, slabs, and foundation is acceptable.

2.1.1.7.3.1.2 Aging Facility

The applicant provided information related to the design of the Aging Facility (AF) in SAR Section 1.2.7, including the design bases and design criteria, design methods, and design analyses. The AF consists of two areas of 0.91-m [3-ft]-thick reinforced concrete mat foundation or pads at grade level, to be designed to support vertical aging overpacks (AO) and horizontal aging modules (HAM). The aging pad areas were designated as 17P {L-shaped 397 m [1,302 ft] × 360 m [1,180 ft], with a cutout of 158 m [519 ft] × 95 m [312 ft]} and 17R {rectangular shaped 506 m [1,661 ft] × 274 m [900 ft]}, as depicted in SAR Figure 1.2.7-2. The AF is designed to accommodate (i) 2,400 vertical AOs containing TAD canisters or dual-purpose canisters (DPCs, which are used for storage and transportation of commercial SNF; and (ii) 100 concrete HAMs containing only DPCs. The applicant stated that the reinforced concrete mat foundation or the aging pads are designed in accordance with ACI 349–2001 (ACI, 2001aa) code, the use of which at the GROA is acceptable, as discussed in Table 7-1.

**Design Criteria and Design Bases**

In SAR Section 1.2.7.5 and Table 1.2.7-1, the applicant provided the design bases and their relationship to the design criteria of the AF. SAR Table 1.2.7-1 provided nuclear safety design bases as (i) structural integrity of the aging pad to protect the ITS SSCs (AO, HAM) from external events such as earthquakes, extreme winds, and tornado winds and (ii) protection against aging overpack tipover and sliding, reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6.3.1. The applicant determined that the AO is required to be designed to prevent it from sliding into another AO on an aging pad for a beyond DBGM seismic event (SAR Table 1.2.7-1).

**Design Methods**

In SAR Section 1.2.7.6, the applicant described the design methods used for the structural design of the aging pads. Each aging pad slab is a reinforced concrete mat, with the top of the mat being at the grade level. The applicant stated that the pads are designed to withstand loads and load combinations imposed by natural phenomena, such as earthquakes, extreme winds, and tornado winds.

The applicant's design method for the concrete mats of the AF includes a finite element static analysis using the SAP2000 software (Computers and Structures, 2005aa). In response to an NRC staff RAI (DOE, 2009ew), the applicant provided bases for the assumptions and approaches used in the finite element analysis. In particular, the applicant stated that their design method incorporates the following: (i) use of a small representative area 26.5 m [87 ft] × 35 m [114 ft] of the aging pad supporting 16 vertical AOs to represent the behavior of the actual pad, which is conservative because the design for the aging pad has repeating arrays of 16 vertical casks on a continuous concrete slab; (ii) design of the pads for the HAMs were bounded by the design of the pads for AOs because the distributed loadings on the pads for the vertical AOs and HAMs were similar, while the pads supporting the HAMs are smaller than the pads supporting the AOs; (iii) modeling the concrete pad using shell elements with a 0.91 m [3 ft] × 0.91 m [3 ft] mesh size was used because resulting shear and moment contour diagrams reflected gradual distribution of forces between supports, showing the areas of maximum positive and negative forces and points of inflection; and (iv) soil stiffness properties were computed based on the lower bound soil properties to compute conservative estimate of design forces. The applicant stated that the lower bound soil stiffness properties (i.e., vertical and horizontal subgrade moduli) would result in larger deflections in the mat, and the higher conservative design mat forces and moments were selected to simulate softer (or less stiff) springs for the pad models.

**Design and Design Analyses**

The applicant's design and design analyses for the AF considered the effects of flooding loads due to high-intensity rainfall that could potentially impact the AF. In SAR Section 1.2.2.1.6.2.2, the applicant stated that the AF is protected against the probable maximum flood (PMF) by locating the structures above the PMF or by engineered barriers, such as dikes or drainage channels. The general layout of the AF and flood protection barriers were depicted in SAR Figures 1.2.7-2 and 1.2.2-7. However, SAR Figure 1.2.2-7 did not show the elevation of the AF or the planned slopes in the area. In response to the NRC staff's RAI (DOE, 2009ew), the applicant provided the results of its flood inundation analysis and the planned drainage channels. This information showed that the AF concrete pads are at higher elevation {more than 3.0 m [10 ft]} above the PMF level of approximately 1,138.5 m [3,735.4 ft] mean sea level near the AF. Design of the applicant's flood protection barriers is reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.7.3.1.3.

The applicant also analyzed the pad of the AF for dead loads (self-weight, AOs, and site transporter) and live loads of 7.2 kPa [150 psf] to account for other loads expected during the placement of AOs on the pad. The seismic loads were based on the assumption of PGA values of 0.45 g in the horizontal direction and 0.32 g in the vertical direction for the mass of the concrete pad (dead load plus 25 percent of the live load). In response to the NRC staff's RAI regarding SSI effects that may amplify the seismic accelerations and increase the mat design forces (DOE, 2009ew), the applicant recognized the potential effects of amplification of vertical seismic accelerations resulting from the SSI effects but qualitatively considered this to be bounded by the design margins (i.e., the maximum D/C (demand-to-capacity) ratios of 0.56 in flexure and 0.69 in shear) (BSC, 2007aa).

For the AF overpacks on the concrete pad, the applicant used seismic accelerations of 1.03 g and 0.716 g in the horizontal and vertical directions, respectively (BSC, 2007aa). However, the applicant stated that the overpacks would slide at horizontal seismic accelerations beyond 0.35 g because of the coefficient of friction (COF) of 0.35 between the concrete pad and the AO, and thus would not experience greater than 0.35g horizontal seismic accelerations. In response to the NRC staff's RAI (DOE, 2009ew), the applicant stated that a COF of 0.35 was achievable, as the value was well within the identified ranges reported in published literature [e.g., mean of 0.43 in NUREG–1864 (NRC, 2006ab), a range between 0.35 and 0.40 in NUREG/CR–6865 (Luk, et al, 2005aa)]. The applicant also calculated cask displacement and rotation during a

DBGM–2 seismic event to conclude that the casks would remain stable and would not tip over, and the sliding displacement would be small. The applicant determined that the wind and tornado-initiated damage to the AOs could not initiate event sequences at the AF and excluded these initiating events from the PCSA. This conclusion was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.4.

The applicant calculated the AF design forces for seismic loading by equivalent static analysis and the 100-40-40-component factor method outlined in ASCE 4–98 Section 3.2.7.1.2 (ASCE, 2000aa). Considering various load combinations, the applicant stated that it used the maximum forces (bending moments and shear forces) to design the flexural and shear reinforcing steel, in accordance with the ACI 349–01 (ACI, 2001aa) code. The applicant concluded that shear strength of concrete is greater than the demand and that shear reinforcement is not required. Nevertheless, the applicant proposed to use #5 reinforcing bars at 610-mm [24-in] spacing, which is greater than the minimum spacing of approximately 381 mm [15 in], as specified in ACI 349–01 Section 11.5.4.1 (ACI, 2001aa). In response to the NRC staff's RAI (DOE, 2009ew) on the amount and spacing of shear reinforcement of the concrete pad, the applicant reiterated its position that shear reinforcement was not required in the aging pad design, because $V_u < \varphi V_c$, where, $V_u$ is the factored shear force at a section, $V_c$ is nominal shear strength provided by concrete, and $\varphi$ is the strength-reduction factor for shear. Therefore, the applicant stated that the shear reinforcement spacing standard in Section 11.5.4.1 of ACI 349–01 (ACI, 2001aa) code was not applicable.

The maximum soil-bearing pressure for the AF was computed to be approximately 192 kPa [4 ksf], which is less than the allowable bearing capacity of 2,394 kPa [50 ksf] reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.3.2. The predicted maximum displacement was approximately 8.1 mm [0.32 in].

**NRC Staff's Evaluation of the Aging Facility Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the applicant's information on the AF design criteria and design bases, and the consistency of the AF design criteria and design bases (SAR Section 1.2.7.5 and Table 1.2.7-1) with the applicant's site characterization information and the PCSA was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1, and SER Sections 2.1.1.4 and 2.1.1.6, respectively. The NRC staff finds that the design bases and design criteria for the ITS AF are adequately defined and are acceptable because the (i) design bases and design criteria are consistent with site characterization information and are adequate to address the site-specific hazards; (ii) the design bases and design criteria are consistent with the PCSA used to identify SSC ITS, which the NRC staff evaluates in SER Sections 2.1.1.3.3.1.3.2, 2.1.1.4.3.4.2, and 2.1.1.6.3.2.8.3; and (iii) the relevant safety functions (e.g., protection against AO tipover or direct exposure to personnel) are linked to the nuclear safety design bases and design criteria. The NRC staff reviewed the applicant's selection and use of design codes [e.g., ACI 349–2001 (ACI, 2001aa)] and found them acceptable, as further evaluated in Table 7-1.

The NRC staff evaluated the applicant's information on the design method for the aging pad of the AF and finds that the applicant's design method for the aging pad of the AF is acceptable because (i) the design method considered external events that could affect the structural integrity of the aging pads; (ii) the numerical models developed for the finite element analyses of the pad to support the design method were based on procedures that are consistent with standard engineering practices (ASCE, 2000ab); and (iii) computer software SAP2000

(Computers and Structures, 2005aa) used for the finite element analyses is standard software used in the nuclear industry for structural analysis.

Regarding the applicant's assumptions in the computer analysis for the AF, the NRC staff finds that (i) the applicant's rationale for the model size is acceptable because the continuity of the slab, which was not considered in the analysis, would reduce the design bending moments and shear forces, and thus is conservative; (ii) the distributed load for the HAMs would be smaller than that for the vertical AOs based on the weight comparison between a HAM and an AO, and the area over which the weight is distributed, and thus the concrete slab design for vertical AOs conservatively bounds the design of the mats supporting the HAMs; (iii) the finite element analysis of the concrete slab using shell elements with a 0.91 m [3 ft] × 0.91 m [3 ft] mesh size is appropriate for estimating the design moments and shears in the concrete slab (BSC, 2007aa), based on the size of the mesh relative to the dimensions of the concrete slab; and (iv) the applicant's modeling of the soil as springs with assigned stiffness values is appropriate because this approach is consistent with standard engineering practice, as described previously in this SER for the surface buildings (SER Section 2.1.1.7.3.1.1.3.1), and this approach is applicable to the aging pads design because the pads are concrete components, the design of which is addressed in SER Section 2.1.1.7.3.1.1.3.1 and found to be acceptable. Also, the NRC staff finds the applicant's use of the lowest soil stiffness properties in analyzing the pad acceptable because it would result in larger soil deformations and conservative pad design force values.

The NRC staff evaluated the applicant's information on the load analysis and design of the AF for the DBGM–2 seismic events and finds that the applicant's information on the analysis and design of the AF for the DBGM–2 seismic events is acceptable because (i) the analysis method, design method, load combinations, and materials used are consistent with those evaluated previously in SER Section 2.1.1.7.3.1.1 for the surface buildings, and this method is applicable to the AF design for the DBGM–2 seismic events because the pads are concrete components, the design of which is addressed in SER Section 2.1.1.7.3.1.1.3.1 and found to be acceptable; and (ii) the analyses and design are consistent with the standard engineering practice [ASCE, 2000ab, Section 3.8.5 of NUREG–0800 (NRC, 2013ag)], the use of which for the GROA is found to be acceptable, as discussed in Table 7-1.

For the AF seismic loads analysis, the NRC staff finds that the applicant performed the structural design of the pad properly because the design analysis is consistent with standard engineering practice (ASCE, 2000aa), as described previously in SER Section 2.1.1.7.3.1.1 for the surface buildings, and this approach is applicable to the design of the AF pads because the pads are concrete components, the design of which is addressed in SER Section 2.1.1.7.3.1.1.3.1 and found to be acceptable. The applicant's use of the DBGM–2 PGA values for the design of the concrete slab without potential amplification caused by the SSI is acceptable because the NRC staff determines that the applicant's design load capacity is larger than the calculated demands from seismic loads, and potential increased loads caused by the SSI.

The NRC staff finds that use of the applicant's proposed AF COF value of 0.35 between a concrete pad and aging cask (i.e., between steel and concrete) and 0.7 between a concrete pad and horizontal aging module (i.e., between concrete and concrete) as described in BSC (2009aa, Table 6-37) for seismic analysis is acceptable because the COF value of 0.35 used by the applicant is within the values of 0.3 to 0.5 recommended in published reports in the literature [NUREG–1864 (NRC, 2006ab), NUREG/CR–6865 (Luk, et al., 2005aa)]. Additionally, the NRC staff determined that the values of COF in the range of 0.3 to 0.5, and resulting potential increase of horizontal forces to the concrete pad from AOs and HAMs, would not affect the concrete slab design, because these forces are small compared to the horizontal seismic forces because of the mass of the concrete pads. The NRC staff finds that the applicant's assessment that the casks would remain stable and would not tip over and that sliding displacement is small is acceptable for the reasons noted above, and because the assessment is consistent with the results of parametric studies of the seismic behavior of dry cask storage systems conducted by the NRC staff (e.g., Luk, et al., 2005aa).

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the aging facility are acceptable.

2.1.1.7.3.1.3 Flood-Control Features

The applicant discussed the flood-control features for the GROA in SAR Section 1.2.2, and Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, 1.2.6-3, and 1.2.7-1, and stated that the GROA surface facilities would be located in two distinct areas: the North Portal area and the Aging Facility area. The applicant also stated that because of the steeply sloping terrain west of the North Portal area at Exile Hill and west to north of the Aging Facility, the GROA area is prone to flooding by storm runoff. The applicant provided design information for the proposed flood-control features for preventing inundation of the surface facilities from the PMF at the site in SAR Figure 1.2.2-7 and in its responses to the NRC staff's RAIs (DOE, 2010ak,am,an; DOE, 2009ew). The applicant also provided PMF and flood inundation analyses for the proposed flood-control features (BSC, 2007db), as reviewed and found to be acceptable in SER Sections 2.1.1.1.3.4 and 2.1.1.3.3.1.

The applicant's design of the flood-control features include the following structures to control the PMF runoff: (i) a dike, or levee, and channel system west, north, and east of the Aging Facility; (ii) a dike and channel system located between the North Portal and Aging Facility areas; (iii) a dike and channel system east and south of the North Portal area; (iv) two diversion ditches in Exile Hill west of the North Portal area; and (v) three storm water detention ponds southeast of the North Portal.

**Design Criteria and Design Bases**

In SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, 1.2.6-3, and 1.2.7-1, the applicant provided the nuclear safety design bases and design criteria for the GROA site flood-control features, which provide that the flood protection features be located and sized to prevent the ITS structures from being inundated by a flood associated with the Probable Maximum Precipitation (PMP) event. On this basis, the applicant stated that all the dikes and channel systems, diversion ditches, and detention ponds are to be located and sized to convey or attenuate the design basis PMF flow with sufficient freeboard, which is the distance from the PMF water surface elevation and the dike crest, to prevent inundation of the surface facilities. Additionally, the applicant stated that the design of levees and open channels did not consider seismic ground motions concurrent with the PMF, as this combined event has a more than 1-million-year return period for a Category 2 event (DOE, 2010an).

**Design Methods**

The design methods the applicant used for the flood-control features included estimation of the design basis PMP at the GROA site, which would produce the PMF runoff (BSC, 2007db). The dike crest elevations were based on the PMF water surface elevations and the desired freeboard. The applicant assumed a layout of channels, dikes, and diversion ditches and estimated the PMF water surface elevations for channel segments in the dike and channel system proposed in BSC (2007db). The applicant estimated the PMF water surface elevations based on the peak PMF flows in the channel segments. The applicant stated in BSC (2007db, Section 7.2.1) that the peak flow in the channel would increase along the downstream direction because of the contribution from new drainage areas along the downstream direction of the channel. Also, the applicant stated that the PMF peak flows calculated by the HEC-1 model for subareas and concentration points were applied to the appropriate channel cross-sections in the HEC-RAS model (BSC, 2007db).

The applicant addressed design methods for the diversion ditches and storm water detention ponds in its responses to two NRC staff RAIs (DOE, 2010an,ak). In DOE (2010an), the applicant stated that, consistent with the nuclear safety design basis (SAR Section 1.9, Table 1.9-6), the diversion ditches would be sized to transport the PMF provided in the site drainage report BSC (2007db); this was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6.3.1. In DOE (2010ak), the applicant stated that the detention ponds are downhill from ITS surface facilities of the GROA, and significant land is available for locating the detention ponds. Further, the applicant stated that the final design parameters of the storm water detention ponds (e.g., storage capacity, maximum flood detention time) are part of the applicant's detailed design.

In response to the NRC staff's RAI (DOE, 2010an), the applicant provided a description of the geotechnical design aspects of the flood-control features and stated that the detailed design will address geotechnical engineering aspects of the flood-control features. The applicant further stated it will use guidance and engineering practices in U.S. Army Corps of Engineers (2000aa,1994aa), Federal Highway Administration (2005aa), and Regulatory Guide 1.102 (NRC, 1976ac) regarding the detailed design of dikes (levees) and channels of flood-control features at the GROA site (DOE, 2010an).

**Design and Design Analyses**

The applicant stated that the final grading of the Aging Facility site, bounded by the dikes of the proposed flood-control system, is expected to influence PMF water surface levels in that area. In response to the NRC staff's RAI on the Aging Facility design described in SAR Sections 1.2.2 and 1.2.7 (DOE, 2009ew), the applicant stated that final grading of the existing topography and associated cross sections through the Aging Facility site will be provided in the detailed design. Further, the applicant presented results of a flood-inundation analysis using the HEC-RAS model (BSC, 2007db), to show that the aging pad area would remain above the PMF water surface elevation.

On the basis of the applicant's estimated peak PMF flow of $1.42 \times 10^6$ L/s [50,219 $ft^3/s$], the applicant stated that it would design the flood protection features to accommodate a flow of $1.56 \times 10^6$ L/s [55,240 $ft^3/s$] to provide 10 percent allowance for the bulking factor (BSC, 2007db). In response to the NRC staff's RAI to clarify this discrepancy, the applicant stated that its design basis PMF flow of $1.56 \times 10^6$ L/s [55,240 $ft^3/s$] exceeds the peak flood flow of $1.13 \times 10^6$ L/s [40,000 $ft^3/s$], corresponding to the 1-million-year return period screening criterion for Category 2 event sequences by a margin of approximately 38 percent (DOE, 2010ak).

In response to the NRC staff's RAI (DOE, 2010am), the applicant explained the method it used in applying the peak PMF flows to cross-sections of the three channel segments shown in BSC (2007db, Figures 6-1 and 7-1). The applicant also explained how it conservatively determined the PMF water surface elevations used in mapping the combined floodplain at the confluences between these channel segments. Instead of dividing the combined flow among the channel segments that join at a confluence, the applicant adopted a more conservative approach of applying the flow to each channel segment and mapping the floodplain based on the channel segment with the highest water surface elevation.

In SAR Section 1.2.2.1.6.2.2, the applicant stated that the protection against flooding is consistent with Regulatory Guide 1.102 (NRC, 1976ac). The applicant stated that ITS structures are located at or near the highest elevations of the North Portal and Aging Facility areas that are protected by a dike and channel system and that adequate slopes would be provided in these areas to preclude inundation of any structures (SAR p. 1.2.2-6).

**NRC Staff's Evaluation of the Flood-Control Features Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the applicant's information on the design criteria and design bases, design methods, and design and design analyses of the flood-control features and finds that the design bases and design criteria used in the design of the flood-control features for the PMF are adequate because the design bases and design criteria are based on the site conditions for determining the PMF, and the flood-control features are consistent with the NRC Regulatory Guide 1.102 (NRC, 1976ac), which the NRC staff finds acceptable for use at the GROA, as further evaluated in Table 7-1.

The NRC staff evaluated the applicant's information on the design methods for the flood-control features (ditches, dikes or levees, storm water detention ponds) and finds that the design methods for flood control are adequate because (i) design of the flood-control features are consistent with the standard industry guidance (U.S. Army Corps of Engineers, 2000aa; U.S. Army Corps of Engineers,1994aa; Federal Highway Administration, 2005aa) that is applicable for the design and construction of earth levees, flood-control channels, and culverts, and are applicable to the GROA, as discussed in Table 7-1; (ii) the PMF flow and water surface elevation calculations were based on site properties; and (iii) the PMF flow and water surface elevations were calculated using the HEC-1 and HEC-RAS models, consistent with standard engineering practice and Regulatory Guide 1.102 (NRC, 1976ac). The NRC staff further finds that detention ponds downhill from the ITS SSCs surface facilities provide the feature to control flooding of the GROA.

The NRC staff evaluated the design and design analyses of the flood-control features the applicant proposed and finds that the design and design analyses of the flood-control features the applicant proposed are acceptable because the design is based on Regulatory Guide 1.102 and the flood-protection features are designed to accommodate a peak PMF flow of $1.56 \times 10^6$ L/s [55,240 $ft^3$/s], which provides a margin of 38 percent over a Category 2 flood event (i.e., the PMF has a annual return period of less than a 1-million-year return period). The NRC staff also finds that the floodplain mappings are acceptable because the floodplain mapping was determined by the highest water surface elevations calculated in various segments of the channel system, and is therefore conservative.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the flood-control features is acceptable.

**NRC Staff's Conclusion of the ITS Surface Facilities Buildings Design**

On the basis of the NRC staff's evaluation of the applicant's information of the GROA surface facilities ITS SSCs (CRCF, RF, WHF, and IHF surface buildings, AF, and flood-control features) design, described above, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the information provided by the applicant adequately (i) provides information on materials of construction, dimensions, proposed codes and standards, analytical and design methods and (ii) identifies the relationship between the design bases and the design criteria.

### 2.1.1.7.3.2 Mechanical Handling Transfer Systems

The applicant provided design information for the ITS mechanical handling transfer systems equipment to be used at the GROA in SAR Sections 1.2.2.2, 1.2.3.2, 1.2.4.2, 1.2.5.2, and 1.2.6.2. The ITS mechanical handling systems are (i) the Canister Transfer Machine (CTM), (ii) the Waste Package Transfer Trolley (WPTT), (iii) the Spent Fuel Transfer Machine (SFTM), and (iv) the Canister Transfer Trolley (CTT). The applicant stated that these mechanical handling transfer systems would be located in multiple surface facilities to lift and transport canisters, casks, and overpacks containing canistered or uncanistered waste. In the following section, the NRC staff evaluates the design bases and design criteria, design method, and design and design analyses for the mechanical handling transfer systems.

#### 2.1.1.7.3.2.1 Canister Transfer Machine

The applicant stated that the Canister Transfer Machine (CTM) is used in the surface facilities to transfer a waste canister from a cask or overpack to a waste package, overpack, or shielded transfer cask (SAR Section 1.2.2.2.1). The applicant further stated that the CTM is used in the canister transfer areas of the surface facilities and located on the second floor of the IHF, CRCF, WHF, and RF. The CTMs at these facilities share the same design criteria and design bases, design methods, design and design analyses. In SAR Table 1.2.2-11, the applicant specified the rated capacity of the CTM to be 63,502 kg [70 tons] for all facilities. The applicant described design features of the CTM in SAR Section 1.2.4.2 and provided the mechanical envelope diagram in SAR Figure 1.2.4-50. The applicant provided instrumentation and logic diagrams in SAR Figures 1.2.4-51 through 1.2.4-56.

**Design Criteria and Design Bases**

The applicant presented the nuclear design bases for the CTM and their relationship with the design criteria in SAR Tables 1.2.3-3 (IHF), 1.2.4-4 (CRCF), 1.2.5-3 (WHF), and 1.2.6-3 (RF). The applicant provided specific design criteria to meet each of the necessary safety functions, along with controlling parameters and bounding values.

The applicant provided several design criteria for the safety functions to (i) protect against a drop of the load and (ii) protect against the drop of a load onto a canister so that the drop energy does not breach the load or canister. These criteria, based on the American Society of

Mechanical Engineers (ASME) NOG–1–2004 Type I (ASME, 2005aa) cranes, provide for (i) two hoist upper limit switches, (ii) a hoist adjustable speed drive (ASD) at set points that are independent of the hoist upper limit switches, (iii) a load cell to prevent the CTM from lifting a load that is over its rated load-carrying capability, and (iv) a sensor to stop the load when it clears the CTM slide gate.

The applicant design criterion for the CTM includes an upper limit to which the canister is lifted in order to limit the potential canister drop height. For example, the CTM is designed to not be able to lift the bottom of a canister more than 13.7 m [45 ft] above the cavity floor with the CTM hoisting system in a two-block condition, in which the load block comes into physical contact with the head block (upper block) or its supporting structure, preventing further upward movement of the load block and preventing overload of the rope-reeving system and hoisting machinery (ASME, 2005aa) that may result in a canister drop.

The applicant used the following design criteria to protect against spurious movement and to limit travel speed of the CTM: (i) interlocks between the CTM shield skirt and the bridge and trolley drives; and (ii) circuit breakers, which power the speed drives of the bridge and trolley motors for instantaneous overcurrent protection. The design criterion to limit the speed of the CTM trolley and bridge constrains the maximum speeds to 6.1 m/min [20 ft/min].

To ensure that a canister (DPC or TAD) will fall in a vertical orientation with a flat bottom or near flat bottom drop into the receiving container during the canister handling operations, the applicant's design criteria for the CTM include the use of a guide sleeve located inside the shield bell of the CTM (SAR Section 1.7.2.3.1; DOE, 2009fy; PSC-14; SAR Table 1.9-10).

The applicant's design criteria address protection of workers from radiation exposure in the room in which the CTM operates, by including interlocks (ITS controls) between the shield skirt and gates (shield and port) and limit switches to ensure that the shield skirt is in the down position to permit hoist operation. In addition, the applicant described a procedural safety control (PSC) to mitigate radiation exposure to personnel. This PSC (PSC-13) in SAR Table 1.9-10 provides for the applicant developing a procedure for closing the port slide gates when a canister transfer operation is complete. The NRC staff's evaluations of these ITS controls are provided in SER Section 2.1.1.7.3.7.

**Design Methods**

The applicant stated that it used the ASME NOG–1–2004 (ASME, 2005aa) code for the design method of the CTM (SAR Section 1.2.2). The applicant stated that it designed the CTM for DBGM–2 seismic events using the ASME NOG–1–2004 (ASME, 2005aa) codes, and assessed the fragility of the CTM design to show that the CTM has the capacity to perform the safety functions during a seismic event (SAR Section 1.7.1.4).

**Design and Design Analyses**

The applicant stated that the CTM design features and supporting analyses include load path redundancy and design factors, such as limited travel speed and high safety design margins for the grapple, overload protection, redundant braking systems, and over-travel-limit switches to ensure safe operation of the CTM. The applicant stated that these design features and methods are in accordance with the provisions of the ASME NOG–1–2004 (ASME, 2005aa) code. The applicant also used the design provisions of ASME NOG–1–2004 (ASME, 2005aa) code for the choice of CTM materials.

The applicant stated that it designed the CTM for static, dynamic, and environmental loads associated with normal operations, as well as Categories 1 and 2 event sequences. The specific design loads that the applicant used in the design and design analyses of the CTM included those from (i) normal operations, (ii) seismic event, (iii) extreme wind conditions, and (iv) collision of a canister with the CTM structure caused by CTM malfunction. In the case of loads due to a seismic event, the applicant considered dead loads, live loads, and seismic loads of DBGM–2 levels. In the case of extreme wind loads, the applicant considered the dead loads and wind loads. In the case of a collision of a canister with a CTM, the applicant considered dead loads, live loads, and loads associated with a collision event sequence. The applicant stated that the CTM design method is in accordance with the provisions of ASME NOG–1–2004 (ASME, 2005aa) code, which includes guidance for combination of loads (SAR Table 1.2.4-4).

**NRC Staff's Evaluation of the Design Criteria and Design Bases, Design Methods, and Design and Design Analyses for the CTM**

The NRC staff reviewed the applicant's information in SAR Sections 1.2.2.2, 1.2.3.2, 1.2.4.2, 1.2.5.2, and 1.2.6.2, along with the applicable codes and standards, as related to the design criteria and design bases for the CTM and finds it acceptable because it is consistent with the provisions of the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes, which the NRC staff finds acceptable for use at the GROA. Further information about the scope and applicability of the codes and standards used in the staff's evaluations can be found in Table 7-1. The single-failure proof cranes are designed and constructed so that they remain in place and support the critical load during and after a seismic event. The single failure-proof features (e.g., dual brakes) eliminate load drops as a result of failure of a single component. The staff further finds that the ASME NOG–1–2004 (ASME, 2005aa) code provides standards on interlocks/limit switches (Section 6440 "Limit Switches") to prevent a range of accidents [including spurious movement and two-blocking (potential of hoisting beyond the intended upper limit)], load combinations, including seismic (Section 5310 "Load Combinations", speed limits (Section 5330 "Motion Speeds") and material selections (Sections 4200 "Materials and Connections" and 5200 "Materials"). Additionally, the NRC staff finds PSC-13, which was proposed by the applicant to protect personnel from radiation exposure, is acceptable because it provides additional safety beyond the provisions of the ASME NOG–1–2004 (ASME, 2005aa). The NRC staff's review of the use of a guide sleeve during canister handling operations to ensure a flat bottom or near flat bottom drop of a canister inside the shield bell of the CTM is provided in SER Section 2.1.1.4.3.3.1.1, where the NRC staff finds its use acceptable.

The NRC staff finds that the applicant's design method for the CTM is acceptable because it is consistent with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes, which is a standard industry practice for the design of cranes in a nuclear power plant, and has design features to minimize the likelihood of a drop of a canister and potential collision of a canister with CTM that may result in radioactivity release. Use of this code at the GROA for the design of cranes is appropriate because the handling of SSCs containing radioactive materials at the GROA are similar to those at a nuclear power plant, as discussed further in Table 7-1. Additionally, the applicant's seismic fragility assessment of the mechanical handling equipment, such as CTM, is reviewed and found to be acceptable in SER Section 2.1.1.4.3.2.2.

The NRC staff evaluated the applicant's information on the design and design analyses of the CTM and finds that the CTM design and design analyses are acceptable because (i) the design method is consistent with the ASME NOG–1–2004 code (ASME, 2005aa), which is appropriate for use at the GROA, as described above and in Table 7-1; (ii) loads to be used for the design and design analyses of the CTM are consistent with normal loads and loads resulting from structural challenges during CTM operations; (iii) load combinations are consistent with the design provisions of ASME NOG–1–2004 (ASME, 2005aa), which includes dead, live, impact, and wind loads during normal plant operations, as well as loads associated with seismic and abnormal events; and (iv) the applicant's use of single-failure features is consistent with the code, which contains standards for the single-failure features (e.g., Section 5416.1, "hoist drive system" for Type I cranes). The use of ASME NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa).

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the CTM used in surface facilities are acceptable.

2.1.1.7.3.2.2 Waste Package Transfer Trolley

The applicant stated that the Waste Package Transfer Trolley (WPTT) consists of two main components: the shielded enclosure and the trolley that operates on rails (SAR Fig. 1.2.4-88). The trolley performs three main functions: (i) transferring a waste package between areas for the purpose of loading the waste package; (ii) accepting a waste package from the CTM; and (iii) positioning the waste package to permit its transfer to the Transport and Emplacement Vehicle (TEV). The WPTT is to be used in the IHF (SAR Section 1.2.3.2.4.1.3) and the CRCF (SAR Section 1.2.4.2.4.1.3). The WPTT, with a payload rating of 90,718 kg [100 tons] and a top speed of 4 km/hour [2.5 mph], is part of the waste package loadout subsystem of these surface facilities. It operates between the waste package positioning room, waste package closure room, and waste package loadout room. The WPTT transfers an empty waste package to the CTM when positioned under the waste package port in the vertical position. Before the loaded waste package is transferred to the TEV, the waste package would be rotated to the horizontal position.

**Design Criteria and Design Bases**

The applicant presented the nuclear safety design bases for the WPTT, describing the safety functions and associated controlling parameters and values in SAR Tables 1.2.3-3 (IHF) and 1.2.4-4 (CRCF). The applicant also provided the specific design criteria for each of the safety functions, along with their relationships to design bases.

The applicant's nuclear safety design bases for the WPTT included (i) preventing rapid tilt-down (rotational movement of the enclosure at uncontrolled speed), (ii) limiting speed, (iii) protecting against spurious movement, and (iv) protecting against the tip over of the WPTT while holding a loaded waste package. The applicant stated in SAR Tables 1.2.3-3 (IHF) and 1.2.4-4 (CRCF) that the WPTT will be designed in accordance with the standards of the ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes. The WPTT will also be designed with two redundant drive trains to rotate the shielded enclosure (SAR Tables 1.2.3-3, 1.2.4-4). The applicant stated that either one of the WPTT drive trains alone can handle the shielded enclosure. Further, the applicant stated that electrical power is needed to rotate the enclosure in either direction (SAR Table 1.2.4-4). The applicant stated (SAR Section 1.2.4.2.4.1.3) that the shielded enclosure is rotated by electric motors and gear assemblies (one on each side of the enclosure), and that the mechanical gears provide positive engagement at all times, preventing slippage or other free movement of the shielded enclosure.

To limit travel speed and protect against spurious movement, the applicant designed the WPTT with a top speed of 4 km/hour [2.5 mph] and employs interlocks between its drive mechanism and the waste package port slide gate. The interlock interrupts power to the trolley drive when the waste package port slide gate is opened, thereby halting the WPTT.

To protect the WPTT from tipping over or rocking during a seismic event while holding a loaded waste package, the applicant added a design criterion providing that the WPTT be designed to the ASME NOG–1–2004 (ASME, 2005aa) for loads and accelerations associated with a DBGM–2 seismic event. Further, the applicant stated that it designed the rails on which the WPTT travels with seismic restraints to prevent the WPTT from tipping over during a seismic event.

**Design Methods**

The applicant stated that the WPTT design method is in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes (SAR Section 1.2.2.2.1). In addition, the applicant stated that the WPTT would be designed for loads and accelerations associated with the site-specific ground motions of DBGM–2 seismic events (SAR Table 1.2.3-3). The applicant also assessed the fragility of the WPTT design to show that the WPTT has the capacity to perform its safety functions during a seismic event (SAR Section 1.7.1.4).

**Design and Design Analyses**

The applicant's main design features of the WPTT include redundant drives, gear engagement features, and power interrupt interlocks. The applicant stated that these design features are in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code (SAR Table 1.2.4-4). The applicant stated that the design load combinations for the WPTT rails are also in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code.

The applicant's load combinations to be used for the design of the WPTT include normal operating conditions, structural challenges during operations, and effects of natural phenomena (SAR Section 1.2.4.2.4.9). In SAR Section 1.2.2.2.9.2.4, the applicant specified the design loads related to normal operations, seismic events, extreme wind conditions, and potential loads from event sequences, including collision of the WPTT with a loaded waste package. To evaluate the design of the WPTT, the applicant's design analyses included load combinations of loads from normal conditions, event sequences, and the seismic events (SAR Section 1.2.4.2.4.1.3).

**NRC Staff's Evaluation of the Design Criteria and Design Bases, Design Methods, and Design and Design Analyses for the WPTT**

The NRC staff evaluated the applicant's information in SAR Sections 1.2.3.2.4.1.3 and 1.2.4.2.4.1.3, along with the applicable codes and standards, related to the design bases and the relationship between the design bases and design criteria, design methods, and design and design analyses for the WPTT using the standards in the ASME NOG–1–2004 (ASME, 2005aa) code, which the NRC staff has found to be acceptable for use at the GROA. Further information about the scope and applicability of the codes and standards used in the staff's evaluations can be found in Table 7-1. The NRC staff also reviewed the DBGM–2 seismic ground motions for WPTT design in SER Section 2.1.1.7.3.1.1.1, where the NRC staff finds the applicant's seismic analysis acceptable.

The NRC staff also finds that the design criteria and design bases for the WPTT are adequate because the applicant's design bases and design criteria are consistent with the safety functions for the WPTT, as identified in the PCSA. In SER Section 2.1.1.6.3.2.8.2.1, the NRC staff evaluated the applicant's design information and found that the information is adequate to demonstrate that the WPTT will perform its intended safety functions. For instance, in order to achieve the safety function of protecting against tipover of the WPTT while holding a loaded waste package, the applicant stated that the WPTT would be designed in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes for loads and accelerations associated with DBGM–2 seismic events (SAR Table 1.2.4-4). Further, the applicant stated that the WPTT is also equipped with seismic restraints to prevent a tipover during a seismic event. The applicant's load combinations for the normal operation follow the load combinations sections [i.e., Sections 4140, 5310, and Table 5453.1(a)-1 of ASME NOG–1–2004 (ASME, 2005aa)]. The NRC staff finds the applicant's seismic restraint design acceptable because the applicant designed it using applicable ASME NOG–1 (ASME, 2005aa) code provisions (e.g., Section 7416). Additionally, the applicant's seismic fragility assessment of the mechanical handling equipment, such as the WPTT, is reviewed by the NRC staff and found to be acceptable in SER Section 2.1.1.4.3.2.2.

The NRC staff finds that the applicant's design method, design and design analyses for the WPTT are acceptable because the method, as well as the design and design analyses, are consistent with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I single-failure proof cranes. The NRC staff finds that the use of ASME NOG–1–2004 (ASME, 2005aa) is acceptable for use at the GROA because this code provides guidance on construction of overhead gantry cranes to be used at nuclear facilities, as further discussed in Table 7-1. Additionally, the use of ASME NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa).

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the WPTT are acceptable.

2.1.1.7.3.2.3 Spent Fuel Transfer Machine

In SAR Section 1.2.2.2.1, the applicant stated that the Spent Fuel Transfer Machine (SFTM) would be used to transfer spent nuclear fuel assemblies from a DPC or transportation cask to a TAD (either canister) or the SNF staging rack in the WHF pool. With a rated capacity of 1,361 kg [1.5 tons] (SAR Table 1.2.2-11), the applicant stated that the main function of the SFTM is to transfer commercial SNF (CSNF) assemblies to an empty TAD canister, previously staged in the pool or, alternatively, to a staging rack inside the pool. The applicant stated that the SFTM is designed in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes, and includes safety features such as overload protection and redundant braking systems to minimize the likelihood of a load drop.

**Design Criteria and Design Bases**

The applicant's nuclear safety design bases of the SFTM, which included the safety functions (e.g., protect against drop of a spent fuel assembly) and controlling parameters (e.g., probability of failure of dropping an assembly due to equipment failure), were presented in SAR Table 1.2.5-3. For each of these design bases, the applicant presented multiple design criteria to show the relationship between the design bases and the design criteria. For example, the applicant provided two design criteria for the nuclear safety design basis to protect against drop of an SNF assembly. The first criterion included designing the SFTM to ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes. The second criterion included one interlock to prevent the SFTM lifting device from operating if it is not properly connected to the hoisting system, and another interlock to prevent the hoisting motion if the lifting device is not fully engaged or disengaged. Additional design bases and design criteria included

(i) Protection against lifting an SNF assembly above the limits for workers' safety, the applicant included a design criterion of installing a mechanical stop on the SFTM to limit the maximum lift height.

(ii) Protection against a SFTM collapse during a seismic event by designing the SFTM to withstand loads and accelerations associated with a DBGM–2 seismic event. Further, the design of the SFTM ensures that a seismic event does not cause derailment or loss of any main structural components.

**Design Methods**

The applicant stated that the SFTM is designed in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes (SAR Section 1.2.2.2.1). Specifically, the applicant referenced Sections 4140 ["Load Combinations" (structural)], 5310 ["Load Combinations (mechanical)], Tables 5415.1-1 ("Load Combinations–Hoist Drive Shafting") and 5453.1(a)-1 ("Load Combinations–Bridge and Trolleys Axles") when evaluating load combinations for normal operation (SAR Section 1.2.2.2.9.2.1). The applicant stated that it designed the SFTM to DBGM–2 seismic events. The applicant also evaluated the fragility of the SFTM design to show that the SFTM has the capacity to perform the safety functions during a seismic event (SAR Section 1.7.1.4).

**Design and Design Analyses**

The applicant stated that the SFTM is designed to the ASME NOG–1–2004 (ASME, 2005aa) Type I crane, which includes redundant safety design factors (e.g., dual brakes). The applicant's design analysis of the load combinations for the SFTM design includes normal operating conditions, event sequences, and effects of natural phenomena. In SAR Section 1.2.2.2.9.2.1, the applicant specified the design loads from normal operations, seismic events, extreme wind conditions, and potential loads from event sequences. To evaluate the design of the SFTM, the applicant stated that it combined the design loads, as appropriate, consistent with the guidance in ASME NOG–1–2004 Sections 4140 and 5310 and Table 5453.1(a)-1 "Load Combinations–Bridge and Trolleys Axles" (ASME, 2005aa).

**NRC Staff's Evaluation of the Design Criteria and Design Bases, Design Methods, and Design and Design Analyses for the SFTM**

The NRC staff reviewed the applicant's information in SAR Section 1.2.2.2.1, along with the applicable codes and standards, as related to the design criteria and design bases, design methods, and design and design analyses for the SFTM using the design standards of the ASME NOG–1–2004 (ASME, 2005aa) code. The NRC staff has found use of this code to be acceptable for use at the GROA. Further information about the scope and applicability of the codes and standards used in the NRC staff's evaluations can be found in Table 7-1. The NRC staff finds that the applicant's design criteria and design bases are adequate because they correspond to the safety functions identified in the PCSA for the SFTM, as evaluated by the NRC staff in SER Section 2.1.1.6.3.2.8.2.1. In this evaluation, the NRC staff finds that the applicant's design information is adequate to demonstrate that the SFTM will perform its intended safety functions. For instance, in order to protect against an SNF assembly drop, the SFTM was designed in accordance with ASME NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) cranes code, which provides standards on single-failure proof features (e.g., Sections 5416 "Single Failure-Proof Features" [mechanical] and 6110 "Single Failure-Proof Features" (Type I Cranes) [electrical]). Additionally, the applicant's seismic fragility assessment of the mechanical handling equipment is reviewed and found to be acceptable in SER Section 2.1.1.4.3.2.2.

The NRC staff finds that the design of the SFTM is capable of protecting against an SNF assembly drop because the design is consistent with the ASME NOG–1–2004 (ASME, 2005aa) Type I cranes with single-failure proof features, which are designed and constructed to preclude a load drop because of the failure of a single component. The NRC staff finds that the applicant's design method, design and design analyses for the SFTM are acceptable because the method, as well as the design and design analyses, are consistent with ASME NOG–1–2004 (ASME, 2005aa) for Type I single-failure proof cranes. The NRC staff finds use of this code acceptable for use at the GROA because this code provides guidance on construction of overhead gantry cranes to be used at nuclear facilities, as further discussed in Table 7-1.

The ASME NOG–1–2004 (ASME, 2005aa) includes numerous single-failure proof features, such as (i) a minimum of two holding brakes for a dual hoist drive with each holding brake rated at a minimum of 125 percent full-load torque (Section 5416 "Single-Proof Features); and (ii) dual load paths for the reeving system with each path capable of supporting the load upon a rope failure (Section 5420 "Reeving System"). It also provides standards for (i) seismic loads (Section 4136 "Seismic and Abnormal Events Loads"), and (ii) load combinations [Sections 4140 "Load Combinations" (structural) and 5310 "Load Combinations" (mechanical)]. The use of ASME NOG–1–2004 (ASME, 2005aa) Type I cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa).

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the SFTM are acceptable.

#### 2.1.1.7.3.2.4 Cask Transfer Trolley

The applicant stated in SAR Section 1.2.2.2.1 that the cask transfer trolley (CTT) is used to move a transportation cask from the cask preparation area to the cask unloading room and back in the IHF, CRCF, WHF, and RF. The applicant stated that in the cask preparation room, the casks are transferred to the CTT using the cask handling crane (CHC) and then transferred to the cask unloading room on the CTT for canister transfer using the CTM. The applicant states that the CTT drive units and air bearings would be controlled and monitored locally and powered by an onboard battery, and that the operator would use pendant controls to operate the trolley.

The applicant stated that the CTT in the IHF has a capacity of $2.4 \times 10^5$ kg [265 ton], whereas the CTT for all other surface facilities (CRCF, RF, and WHF) would have a capacity of $1.8 \times 10^5$ kg [200 ton] (SAR Table 1.2.2-11). The applicant described the CTT design features in SAR Sections 1.2.3.2.1, 1.2.4.2.1, 1.2.5.2.1, and 1.2.6.2.1, and a mechanical envelope diagram is in SAR Figure 1.2.3-20. The applicant also provided a process and instrumentation diagram in SAR Figure 1.2.4-27.

**Design Criteria and Design Bases**

The applicant presented the nuclear design bases and their relationship with the design criteria in SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3. The applicant stated that, except for the pneumatic components, the CTT would be designed in accordance with the provisions of the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes.

Since the ASME NOG–1–2004 (ASME, 2005aa) code does not address pneumatic components, the applicant stated that the pneumatic components are designed in accordance with specific design codes and standards that address pneumatic valves, pressure relief valves, and piping (DOE, 2009dq). These codes and standards are

- ASME B16.34–2004 (ASME, 2005ab) (for ball, gate, and throttle valves)
- ASME Boiler and Pressure Vessel Code, Section VIII, Paragraph UG–131 (ASME, 2007aa) (for safety relief valves)
- ASME B31.3–2004 (ASME, 2004ab) (process piping)
- American Petroleum Institute 526 *Flanged Steel Pressure Relief Valves* (API, 2002aa)
- American Petroleum Institute 527 *Seat Tightness of Pressure Relief Valves* (API, 1991aa)

The applicant further stated that the design of other commercial pneumatic components, such as air cylinders, air motors, and air bearings/casters, would follow the manufacturer's standards.

The applicant also identified the following design criteria for the CTT:

- To limit the CTT speed, the applicant's design criterion for the pneumatic-powered traction drives precludes travel speeds from being greater than 4 km/hour [2.5 mph] (SAR Table 1.2.3-3) using the shutoff valves in the air supply of the drive units.
- To protect against spurious movement, the applicant imposed a design criterion providing for disconnecting the pneumatic power supply during cask unloading so that the CTT is firmly on the floor (SAR Table 1.2.3-3).
- To protect against waste container impact as a result of the CTT sliding into a wall or structural column and to minimize seismically induced sliding or rocking, the applicant's CTT design includes operating clearance and energy-absorbing features to minimize the effect of seismically induced sliding or rocking (SAR Table 1.2.3-3).

Further, the applicant proposed several other procedural safety controls (PSCs) to ensure the safe operation of the CTT (SAR Section 1.2.3.2.1.4). For example, the CTT is equipped with a warning system to notify the operator in the case of deflation of the CTT air supply. The operator independently verifies that the CTT is resting on its landing pads during cask loading and unloading operations. This is a safety step identified in the PCSA, which ensures that the CTT is at rest. In addition, the applicant stated that the CTT is designed to a DBGM–2 seismic event (SAR Table 1.2.2-11). The applicant stated that the trolley would be free to slide without encountering an obstruction (SAR Section 1.2.2.2.1). Finally, the applicant stated that redundant systems, speed limitations, and protective features (e.g., operating clearance and energy absorbing features) ensure that tipover, collision, or uncontrolled movements are minimized.

### Design Methods

The applicant stated that its design method for the CTT follows the ASME NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) crane code, except for unique features of the CTT associated with the pneumatic components (SAR Section 1.2.2.2.1), such as valves, air bearings, and piping. In response to the NRC staff's RAI (DOE, 2009dq, Enclosure 2), the applicant identified design codes and standards for these pneumatic components. The applicant stated the design method for the CTT incorporates the site-specific ground motions described in SAR Section 1.2.2.1.6.3, and that it is designed not to tip over, but may slide during these seismic events (SAR Section 1.2.2.2.1). The applicant also assessed the fragility of the CTT design to show that the CTT will perform its safety functions during a seismic event (SAR Section 1.7.1.4).

### Design and Design Analyses

The applicant stated that the CTT design includes several safety features, including restraint arms to hold a cask during a DBGM–2 seismic event, limiting trolley travel speed, fail-safe features, air pressure monitoring, onboard battery controls, and continuous monitoring of the CTT drive units. The applicant stated that the CTT design is in accordance with the ASME NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) cranes (SAR Table 1.2.2-11).

The applicant's design analyses of the load combinations for the CTT include normal operating conditions, event sequences, and effects of natural phenomena. In SAR Section 1.2.2.2.9.2.4, the applicant stated that the design loads for the CTT encompassed normal operations, seismic events, extreme wind conditions, and potential loads from event sequences, including collision of the CTT with (i) a waste package, (ii) casks containing waste, or (iii) a structure, such as the shield door. For each of the load combinations, the applicant compared the demand with the values specified in the ASME NOG–1–2004 (ASME 2005aa) code.

### NRC Staff's Evaluation of the Design Criteria and Design Bases, Design Methods, and Design and Design Analyses for the CTT

The NRC staff evaluated the applicant's information in SAR Section 1.2.2.2.1, along with the applicable codes and standards, related to the CTT design criteria and design bases, design methods, and design and design analyses using the ASME NOG–1–2004 (ASME, 2005aa) code, which the NRC staff has found to be acceptable for use at the GROA. Further information about the scope and applicability of the codes and standards used in the NRC staff's evaluations can be found in Table 7-1. The NRC staff finds that the design criteria and design bases for the CTT the applicant provided are adequate because they correspond to the safety functions identified in the applicant's PCSA, as evaluated by the NRC staff in SER Section 2.1.1.6.3.2.8.2.1, where the NRC staff finds that the applicant's design information is adequate to demonstrate that the CTT will perform its intended safety functions. The use of ASME NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa), and includes standards on single-failure features (e.g., Section 5416.1, "hoist drive system" for Type I cranes).

The NRC staff finds that the applicant's identification of additional industry codes and standards for the pneumatic components of the CTT outside the scope of the ASME NOG–1–2004 (DOE, 2009dq) is acceptable. The applicant specified the capacity of the safety valves in accordance with Paragraph UG–131 of ASME B&PV Section VIII Code (ASME, 2007aa), which the NRC staff finds acceptable (see Table 7-1).

The NRC staff finds that the provision of safety features (e.g., shutoff valves and disconnecting air supply) proposed by the applicant is an acceptable way to limit the CTT speed and protect against spurious movement because no pneumatic equipment is capable of moving without adequate air supply. The NRC staff also finds that the proposed operating clearance and energy-absorbing features are acceptable because these features are capable of minimizing the effect of seismically induced sliding impact or rocking. The NRC staff further finds that these industry codes and standards, when used in conjunction with the ASME NOG–1–2004, would provide acceptable CTT design bases and design criteria. The NRC staff also finds that the design criteria and safety controls for the CTT are in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes code, which includes multiple single-failure proof features (e.g., Sections 5416.1 "hoist drive system" for Type I Cranes and 5420 (a) "rope reeving system" for Type I Cranes). The NRC staff also finds that the procedural safety controls proposed by the applicant are an acceptable means to ensure safe operation of the CTT.

The NRC staff evaluated the applicant's information on the design method for the CTT and finds that the applicant's design method is adequate because it is consistent with the ASME NOG–1–2004 (ASME, 2005aa) code. Further, the NRC staff finds that the applicant's design of the CTT to DBGM–2 is acceptable, as evaluated by the NRC staff in SER Section 2.1.1.7.3.1.1.1. Additionally, the applicant's seismic fragility assessment of the mechanical handling equipment is reviewed and found to be acceptable in SER Section 2.1.1.4.3.2.2.

The NRC staff evaluated the applicant's information on the CTT design and design analyses and compared the design and design analyses with applicable codes and standards. The NRC staff finds that the CTT design and design analyses are acceptable because they are in accordance with the ASME NOG–1–2004 (ASME, 2005aa); B16.34-2004 (ASME, 2005ab); ASME Boiler and Pressure Vessel Code, Section VIII, Paragraph UG–131 (ASME, 2007aa); ASME B31.3–2004 (ASME, 2004ab); API 526 (API, 2002aa); and API 527 (API, 1991aa). The NRC staff finds the use of these codes and standards acceptable for use at the GROA because these codes and standards provide guidance for designing SSCs for their intended use. The applicability of these codes to the GROA is further discussed in Table 7-1.

In addition, the NRC staff finds that the applicant's use of the onboard battery systems and controls to eliminate the potential electrical hazards to workers from a power line tethering to the CTT is an appropriate way to minimize electrical hazards (Rosaler, 1995aa). The applicant's use of speed limit, control and monitoring of the air pressure to limit the potential for the CTT to move in an unpredictable manner, designing the pneumatic power to be fail/safe, and continuous monitoring of the CTT drive units are also consistent with acceptable pneumatic equipment safety practices (Mobley, 2001aa).

Based on the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the CTT are acceptable.

**NRC Staff's Conclusion of the Mechanical Handling Transfer Systems**

On the basis of the NRC staff's evaluation of the applicant's design of the following mechanical handling transfer systems (i) Canister Transfer Machine (CTM); (ii) Waste Package Transfer Trolley (WPTT); (iii) Spent Fuel Transfer Machine (SFTM); and (iv) Canister Transfer Trolley (CTT), described above, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the description provided by the applicant of the mechanical handling transfer systems designs adequately (i) provides information on materials of construction, dimensions, proposed codes and standards, analytical and design methods; (ii) defines the relationship between design criteria and the performance objectives; and (iii) identifies the relationship between the design bases and the design criteria.

### 2.1.1.7.3.3 Heating, Ventilation, and Air Conditioning System

The applicant provided information related to the materials of construction, codes and standards, and design of ITS HVAC systems in SAR Sections 1.2.2.3 (surface facilities), 1.2.4.4 (CRCF), 1.2.5.5 (WHF), and 1.2.8.3 [Emergency Diesel Generator Facility (EDGF)]. The ITS HVAC systems in the CRCF and WHF provide waste form cooling, temperature control, flow control, filtration, and support confinement. In addition, the ITS HVAC system provides cooling to support the equipment in the EDGF. The applicant stated that the ITS HVAC systems include dampers, ductwork (including supports), an exhaust fan with an adjustable speed drive, high-efficiency particulate air (HEPA) filters, demisters, prefilters, air handling units, electrical power supplies, and I&Cs.

In this section, the NRC staff's review focused on the design bases and design criteria, design method, and design analysis for these HVAC systems. The electrical power supplies and instrumentation and controls (I&C) for these systems are reviewed in SER Sections 2.1.1.7.3.6 and 2.1.1.7.3.7, respectively. The NRC staff's review of the HVAC system description is in SER Section 2.1.1.2.3.2.4 and ITS HVAC system performance is evaluated in SER Section 2.1.1.4.3.3.2.1, where the NRC staff finds the failure probability the applicant identified for the surface nuclear confinement ITS HVAC system acceptable.

**Design Criteria and Design Bases**

The applicant listed the nuclear safety design bases of the ITS HVAC system, including portions of the surface nonconfinement HVAC that supports the cooling of ITS electrical equipment and battery rooms in the EDGF in SAR Table 1.9-3 for the CRCF, SAR Table 1.9-4 for the WHF, and SAR Table 1.4.1-1 for the Electrical Systems. The applicant described the relationship between design bases and design criteria in SAR Tables 1.2.4-4 (CRCF), 1.2.5-3 (WHF), and 1.4.1-1 (EDGF). The design criteria were based on industry standard or NRC guidance [e.g., ASME AG–1–2003 (ASME, 2004ac) and Regulatory Guide 1.52 (NRC, 2001ae)]. The applicant stated that the nuclear safety design bases were developed based on the PCSA and includes safety functions and controlling parameters for ITS HVAC systems performance during potential Category 2 event sequences.

The applicant identified the following safety functions for the surface nuclear confinement HVAC systems: (i) mitigate the consequences of radionuclide release (CRCF and WHF) by providing confinement and (ii) provide cooling to support the ITS electrical function (CRCF, WHF, and EDGF). To mitigate the consequences of radionuclide release, the design criteria would provide for two full-capacity, independent trains (one operating and the other in standby). Each train is equipped with an automatic start capability to bring the standby train on-line upon failure of the operating train. To provide cooling to the ITS electrical function, the design criteria would provide for an independent train for the rooms associated with each of the two ITS electrical trains. The applicant also listed the design indoor temperatures for various rooms at the GROA. For instance, in SAR Table 1.2.4-8 (CRCF), the applicant stated that for the electrical rooms, the maximum and minimum temperatures are 32 °C (90 °F) and 18 °C (65 °F) dry bulb, respectively. The NRC staff evaluated the applicant's ITS HVAC in SER Section 2.1.1.6.3.2.8.2.2.

In terms of controlling parameters, the applicant specified a probability of failure ($\leq 4 \times 10^{-2}$) for the ITS HVAC systems over a mission time of 30 days following a potential radionuclide release (SAR Table 1.2.4-4). In response to an NRC staff RAI (DOE, 2009fd) on design bases and design criteria for the ITS HVAC systems, the applicant provided additional information to address the performance requirements for overall filtration efficiency and the cooling requirements for ITS electrical equipment. The applicant stated that the ITS HVAC systems maintain indoor environmental conditions in accordance with the provisions of the American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) 2007 code (ASHRAE, 2007aa). The NRC staff evaluation of the applicant's failure probability evaluation is in SER Section 2.1.1.4.3.3.2.1.

The applicant identified in SAR Tables 1.2.4-4 and 1.2.5-3 that the ITS HVAC systems support the ITS electrical function by cooling ITS electrical equipment and battery rooms. In describing the ITS HVAC subsystems serving the battery rooms, the applicant stated that air is continuously exhausted from each battery room to maintain hydrogen concentrations below the explosive limits (SAR Section 1.2.4.4.1). This hydrogen is generated from electrochemical reactions when the lead-acid batteries charge and discharge. The lower explosive limit for hydrogen in air is 4.1 volume percent (NUREG−1805; NRC, 2004ac; Section 16.2). Additionally, the battery rooms are equipped with hydrogen gas detectors (SAR Section 1.4.1.3.1), and each group of electrical and battery rooms are served by redundant sets of HVAC supply and exhaust equipment (SAR Section 1.2.5.5.1). Thus, the applicant stated that hydrogen accumulation during battery charging is precluded during normal operations (SAR Section 1.2.4.4.2). In response to the NRC staff's RAI (DOE, 2009fd), the applicant indicated that 21 volume air changes per hour for the battery rooms would be performed, exceeding the ASHRAE 2007 (ASHRAE, 2007aa) guidance of five volume air changes per hour.

**Design Methods**

The applicant stated that its design methods for ITS HVAC systems are in accordance with the codes and standards identified in SAR Section 1.2.2.3. In response to the NRC staff's RAI (DOE, 2009fd), the applicant stated that the HEPA filters measure 610 mm × 610 mm × 292 mm [24 in × 24 in × 11.5 in], which is consistent with the ASME AG–1–2003 (ASME, 2004ac) guidance. In addition, the applicant stated that the design method for the adjustable speed drives (ASDs) is consistent with the National Electrical Manufacturers Association (NEMA) ICS 7–2006 (NEMA, 2006ab). The applicant stated that prefilters and high-efficiency filters for air handling units were designed according to ASHRAE 2004 (ASHRAE, 2004aa), with their efficiency calculated using ANSI/ASHRAE 52.1–1992 (ASHRAE, 1992aa), and that sizing criteria for filters and coils are consistent with ASHRAE 2005 (ASHRAE, 2005aa). In addition, the applicant stated that it sized ducts to maintain a fluid velocity of 12.7 m/s [41.7 ft/s], thereby minimizing particulate settlement consistent with DOE–HDBK–1169–2003 (DOE, 2003ae).

## Design and Design Analyses

In addition to the codes and standards described in the previous section, the applicant indicated that it would use NRC guidance documents for analyses and design of the ITS HVAC systems (SAR Table 1.2.2-9). These documents include Regulatory Guides 1.140 and 1.52 (NRC, 2001ad,ae), which provide guidance on the design, inspection, and testing criteria for air filtration and adsorption systems for nuclear power plants. These Regulatory Guides reference outdated standards; therefore, the applicant elected to use updated standards in applying the Regulatory Guides. The applicant indicated that it used ASME AG–1–2003, including ASME AG–1a–2004 (ASME, 2004ac), in lieu of ASME AG–1–1997 (ASME, 1997aa), and ASME N509–2002 (ASME, 2003ab) in lieu of ASME N509–1989 (ASME, 1996aa). In addition, the applicant stated that it used DOE–HDBK–1169–2003 (DOE, 2003ae) in lieu of ERDA 76-21 (Burchsted, et al., 1976aa). The applicant also used Regulatory Guide 3.18 (NRC, 1974ab), which provides guidance on design of confinement barriers and systems for fuel reprocessing plants.

The applicant stated that the construction materials for the ITS HVAC systems are consistent with the codes and standards identified in SAR Section 1.2.2.3. The applicant further identified in SAR Section 1.2.2.3.7 the use of Stainless Steel Type 304L for the ductwork, HEPA filter casings, and HEPA filter housings, referencing ASTM A240/A240M–06c (ASTM International, 2006aa). Additionally, the applicant stated it would use ASME AG–1–2003, including ASME AG–1a–2004 (ASME, 2004ac) for the fan and HEPA filter housing materials.

The applicant identified the use of independent trains in its design criteria for the ITS HVAC systems. For example, the applicant identified two full-capacity, independent trains (one operating and the other in standby mode). Each train is equipped with automatic-start capability to bring the standby train on line upon the failure of the operating train for the subsystem that exhausts from areas with canister breach potential. For the subsystems that provide cooling for the ITS electrical equipment and battery rooms, the applicant stated that there would be an independent train for the rooms associated with each ITS electrical train. Additionally, the applicant described the physical separation of the trains. For example, for the CRCF, the applicant identified Train A HVAC equipment located on the opposite end of the building from the Train B HVAC equipment (BSC, 2008ac). In response to the NRC staff's RAI (DOE, 2009dw) on the independence of trains and the potential for a single point of failure, the applicant stated that individual components such as interlocks and ASDs would not be a potential single point of failure because failure of these components may cause a spurious transfer of the operating train but would not lead to system failure nor prevent the HVAC from fulfilling its ITS function.

The applicant's load combinations include normal operating conditions, event sequences, and the effects of natural phenomena. The ITS HVAC system ducts and supports are designed for concurrent dead weight, seismic load, and pressure load. Additionally, the applicant stated it would use the International Building Code 2000 (International Code Council, 2003aa) for the design of HVAC ducts and duct supports for seismic loads.

To provide waste form cooling, the applicant evaluated the thermal performance of waste forms and waste containers in the facility using standard simulation tools (ANSYS v. 8.0 and FLUENT v. 6.0.12). The applicant simulated the thermal behavior of the waste package and the transfer trolley under normal and off-normal conditions. Simulated off-normal conditions included two different scenarios: (i) ventilation provided by ITS exhaust fans only and (ii) 30-day no airflow conditions. The applicant stated that the calculated peak cladding temperature of the waste form remained below 400 °C [752 °F] for normal and 570 °C [1,058 °F] for off-normal conditions. The applicant stated that the calculations showed that the waste form can remain within these values without the proper functioning of the HVAC system under off-normal conditions (SAR Section 1.2.2.3.6).

**NRC Staff's Evaluation of the ITS HVAC Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the applicant's information in SAR Sections 1.2.2.3, 1.2.4.4, 1.2.5.5, and 1.2.8.3, along with the applicable codes and standards, as related to the design criteria and design bases, design methods, and design and design analyses for ITS HVAC systems. The NRC staff finds that the applicant's design criteria and design bases for the ITS HVAC systems are adequate because first, the design bases and design criteria are consistent with the sources of the reliability data that were evaluated in SER Section 2.1.1.4.3.3.2.1, where the staff finds acceptable the applicant's surface nuclear confinement ITS HVAC failure assessment. In addition, the NRC staff finds that the applicant's surface nuclear confinement ITS HVAC system reliability is adequate for the system to perform its safety function of mitigating the consequences of a radiological release, as evaluated in SER Section 2.1.1.6.3.2.8.2.2. Second, the proposed cooling and filtration specifications are consistent with the industry-accepted guidance, codes and standards, namely, Regulatory Guide 1.52 (NRC, 2012ad); ASHRAE Handbook, (ASHRAE, 2007aa), which the NRC staff finds acceptable for use at the GROA. Further information about the scope and applicability of the codes and standards used in the NRC staff's evaluations can be found in Table 7-1. Third, the electrical room temperatures were developed consistent with accepted industrial design goals (Cummins Power Generation, 2004aa). Finally, the design criteria of limiting hydrogen concentrations by performing 21 air changes per hour in the battery room (DOE, 2009fd) exceeds the five volume changes per hour guidance in ASHRAE 2007 (ASHRAE, 2007aa). The five volume changes per hour guidance limits the hydrogen concentration to less than 4.1 percent, which is the lower flammable limit for hydrogen in air. The NRC staff finds that ASHRAE 2007 is an applicable industry standard to determine volume change requirements to preclude hydrogen accumulation. Furthermore, the staff finds that the rate of 21 air changes per hour exceeds the standard nuclear power plant practice of using 10 air changes per hour to limit the hydrogen concentration below 2 percent of the total volume of the room, as recommended in NUREG–1805 (NRC, 2004ac; Section 16.4.2), and it is therefore acceptable.

The NRC staff finds the applicant's choice of 304L stainless steel [ASTM A240/A240M–06c (ASTM International, 2006aa)] as the material for ductwork, HEPA filter casings, and HEPA filter housings is acceptable due to the material's superior corrosion resistance and strength (Fontana, 1986aa). The NRC staff finds that the applicant's use of the International Building Code (IBC) 2000 (International Code Council, 2003aa) for the design of HVAC ducts and duct supports is acceptable because the HVAC system functions are similar to a building facility HVAC system within the scope of the IBC Code, as discussed further in Table 7-1. The NRC staff further finds, based on the DOE handbook (DOE, 2003ae) and the ASME standards [AG–1–1a–2004 and AG–1–2012, (ASME, 2004ac; ASME, 2012aa)] that the applicant's use of DOE–HDBK–1169–2003 (DOE, 2003ae), in lieu of ERDA 76-21 as recommended by Regulatory Guide 1.52 (NRC, 2012ad), is acceptable because the DOE–HDBK–1169–2003 handbook (i) is based on ERDA 76-21 with updated information and operating experience provided by industry and subject matter experts, and (ii) is consistent with the air cleaning practices in commercial nuclear facilities, as outlined in ASME AG–1–2003, including ASME AG–1a–2004 (ASME, 2004ac) and ASME AG–1–2012 (ASME 2012aa).

The NRC staff evaluated the applicant's information on the design method for ITS HVAC systems and finds that the applicant proposed design method is adequate because the method, including the HEPA filters size, is based on codes and standards that are consistent with the standard engineering practices for design of the nuclear HVAC systems, such as the spent fuel handling area in a nuclear power plant. The NRC staff finds that the applicant's choice of the duct fluid velocity of 12.7 m/s [41.7 ft/s] is adequate because this air velocity is approximately seven times the typical recommended maximum air velocity of 1.8 m/s (Rosaler, 1995aa) for industrial applications. The applicant's choice of air velocity ensures sufficient air flow through the HEPA filter media and maintains differential pressures across different zones in the surface facilities, and thus is acceptable.

The NRC staff evaluated the applicant's information on the HVAC system design using the guidance in Regulatory Guides 1.140, 1.52, and 3.18 (NRC, 2001ad; NRC,2012ad; NRC, 1974ab). The NRC staff determines that the ASME standards the applicant proposed to use are appropriate because the use of these ASME standards is consistent with the NRC Regulatory Guides 1.140 and 1.52. The applicant's design and design analyses of the ITS HVAC confinement barriers and systems are acceptable because they are consistent with NRC Regulatory Guide 3.18 (NRC, 1974ab).

The NRC staff also finds that the applicant's thermal evaluation, showing that the waste form is capable of maintaining the established temperature limit without the functioning of the HVAC system, is acceptable because the applicant used standard computer codes (e.g., ANSYS and FLUENT) that are consistent with NRC guidance, as further discussed in NUREG–2152 (NRC, 2013ae) and NUREG–1922 (NRC, 2010af). The NRC staff also finds that the applicant's normal and off-normal temperature limits (400 °C [752 °F] and 570 °C [1,058 °F]) are consistent with the guidance in NUREG–1536 (NRC, 2010ah) for zirconium-alloy clad spent fuel in storage. In addition, the applicant's evaluation provides information regarding the significance of the HVAC system design relative to maintaining the temperature limit for the waste form. The NRC staff finds that the applicant's design using physically separated HVAC trains is acceptable because this design approach is consistent with Regulatory Guide 1.52 (NRC, 2012ad). In addition, the applicant specified as part of its design criteria the use of independent trains and further stated in its response to the NRC staff's RAI (DOE, 2009fs) that the HVAC trains are independent, so failure of components in one train cannot cause failure of both trains. The NRC staff finds the applicant's response (DOE, 2009fs) acceptable because with two independent trains, the failure of individual components in one train (e.g., interlocks and ASDs) would not lead to a single point of failure involving both trains (DOE, 2009dw).

In summary, the NRC staff finds that the applicant's design and design analyses for the ITS HVAC systems are acceptable because the applicant used acceptable techniques to conduct design analyses, and the design is consistent with acceptable industry standard codes and guidance.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the ITS HVAC system is acceptable.

**NRC Staff's Conclusion of the HVAC System Design**

On the basis of the NRC staff's evaluation of the applicant's design of the HVAC system, including its design to control potential hydrogen accumulation in the Battery Rooms and prevent explosion, described above, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), 63.112(e)(9), and 63.112(f) are satisfied. The NRC staff finds that the description provided by the applicant of the HVAC system acceptably (i) provides information on materials of construction, dimensions, proposed codes and standards, analytical and design methods; (ii) defines the relationship between design criteria and the performance objectives; and (iii) identifies the relationship between the design bases and the design criteria.

**2.1.1.7.3.4 Other Mechanical Systems**

The applicant provided design information for ITS mechanical systems other than the mechanical handling transfer systems (evaluated in SER Section 2.1.1.7.3.2) in SAR Sections 1.2.2 through 1.2.6 and SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, 1.2.6-3, and 1.9-2 through 1.9-5. The applicant classified the other mechanical systems as follows: (i) crane systems; (ii) special lifting devices; and (iii) other mechanical structures (shield and confinement doors, rails, platforms, and racks). The applicant stated that these mechanical systems are located in multiple surface facilities with the same design and functions. Because the mechanical systems share the same design and functions of lifting containers of radioactive materials or are passive SSCs, the NRC staff's evaluation applies to these mechanical systems at different facilities (e.g., CRCF and WHF).

2.1.1.7.3.4.1 Crane Systems

The applicant stated that crane systems are used in the CRCF, WHF, RF, and IHF. The applicant stated that the main function of these specialized crane systems is to upend a cask to a vertical position or move casks from one location to another. The applicant also stated that the crane systems include overhead bridge cranes (e.g., cask handling crane, cask preparation crane, auxiliary pool crane, waste package handling crane, waste package closure remote handling system) and jib cranes. In SAR Table 1.2.2-10, the applicant specified the load ratings of cranes, based on the function of the crane and the load being lifted. For example, the waste package remote handling system has a load rating of 2,721 kg [3 ton], whereas the IHF cask handling crane has a load rating of $2.7 \times 10^5$ kg [300 ton]. The applicant stated that ITS cranes lifting waste forms, for example, cask handling cranes, each is designed in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes. The applicant also stated that for ITS cranes not lifting waste forms, they are designed either to the ASME NOG–1 Type II standards, as in the case of waste package handling cranes, or they are designed to the ASME NUM–1 (ASME, 2005ac) Type IA (single failure-proof) standards, as in the case of DPC cutting jib cranes. Furthermore, the applicant stated that non-ITS cranes that do not lift waste forms, for example, the CTM maintenance cranes, are designed to the ASME NOG–1–2004 (ASME, 2005aa) Type III standards. A Type I crane includes single failure-proof features such that any credible failure of a single component will not result in the loss of capability to stop and hold the critical load. A Type II crane is designed and constructed so that it will remain in place with or without a load during a seismic event. Type III cranes are not used to handle critical loads; therefore, single-failure proof features are not needed for such cranes.

The NRC staff's evaluation encompasses cranes of all classifications because they all perform lifting functions of different types of loads. The focus of the staff's evaluation is on the ASME NOG–1 Type I and ASME NUM–1 Type 1A cranes due to the single-failure proof designation.

**Design Criteria and Design Bases**

The applicant presented the design bases and design criteria for the specialized crane systems, such as the cask handling crane, cask preparation crane, and waste package handling crane, in SAR Tables 1.2.3-3 (IHF) and 1.2.4-4 (CRCF). The applicant presented information for the auxiliary pool and jib cranes in SAR Table 1.2.5-3 (WHF). The applicant presented the design bases and design criteria for the canister transfer machine (CTM) maintenance crane of the RF in SAR Table 1.2.6-3.

To limit the possibility of a load drop, the applicant stated that the cask handling crane would be designed in accordance with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I Cranes (single-failure proof) code.

To limit the drop height, the applicant used a design criterion that limits the hoist lift height. For example, the applicant stated that the 1.8 × $10^5$ kg [200 ton] CRCF cask handling crane's design precludes lifting the cask more than 9.1 m [30 ft] above the floor when the crane hoisting system is in a two-block condition (SAR Table 1.2.4-4).

To limit travel speed of the trolley and bridge, the applicant imposed a speed limitation of 6.1 m/min [20 ft/min] for the 200-ton CRCF cask handling crane (SAR Table 1.2.4-4).

To protect against crane collapse onto a waste container, and/or a cask or heavy object drop, the applicant used a design criterion for ITS cranes to be designed for loads and accelerations associated with a DBGM–2 seismic event (SAR Table 1.2.4-4), as described in SAR Section 1.2.2.1.6.3.

**Design Methods**

The applicant stated in SAR Section 1.2.2.2.7 that the design method for overhead bridge cranes is based on ASME NOG–1–2004 (ASME, 2005aa) for Type I, II, and III cranes. The applicant also stated that the design method for jib cranes would be in accordance with the ASME NUM–1–2004 (ASME, 2005ac) code for Type IA cranes (SAR Section 1.2.2.2.1).

**Design and Design Analyses**

The applicant stated that the design features of cranes in ASME NOG–1–2004 and NUM–1–2004 (ASME, 2005aa,ac) include load path redundancy; design margins, such as allowable stresses being below the material yield strength and hoist speeds being inversely proportional to rated load; overload protection; redundant braking systems; and over-travel limit switches to minimize the likelihood of a load drop. The applicant stated that it would use ASME NOG–1–2004 Sections 4200, 5200, and 6200 (ASME, 2005aa) for the overhead crane material selection and ASME NUM–1–2004 Sections NUM–III–8200, NUM–III–8300, and NUM–III–8400 for the jib crane material selection (SAR Section 1.2.2.2.7).

The applicant defined the crane design loads in SAR Section 1.2.2.2.9.2.1. The applicant stated that it considered the following design loads for the cranes: dead loads, live loads, dynamic loads, seismic loads, environmental loads, and event sequence loads. The applicant defined the design loads for jib cranes in SAR Section 1.2.2.2.9.2.2 and the design loads for the overhead bridge cranes in SAR Section 1.2.2.2.9.2.1.

**NRC Staff's Evaluation of Crane Systems Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the applicant's information on the specialized crane systems design criteria and design bases, design methods, and design and design analyses using guidance in YMRP Section 2.1.1.7 and the industry codes and standards proposed for use by the applicant [ASME NOG–1–2004 and ASME NUM–1–2004 (ASME, 2005aa,ac)]. The NRC staff finds that the applicant's classification of the specialized crane systems into Type I, II, and III classes, on the basis of lifting waste forms, is acceptable because this classification is consistent with the definition of critical load in ASME NOG–1–2004 (ASME, 2005aa), which the NRC staff finds acceptable for use at the GROA, as further discussed in Table 7-1. The NRC staff finds that the design criteria and design bases the applicant provided for the specialized crane systems are adequate because (i) the design criteria include appropriate design bounding limits (e.g., trolley speed and lift hoist height) for cranes; the applicant's speed limit of 6.1 m/min [20 ft/min] is less than the maximum trolley and bridge speeds of 7.6 m/min [25 ft/min] for trolley and 12.2 m/min [40 ft/min] for bridge recommended by ASME NOG–1–2004 (ASME, 2005aa) for a 200-ton crane (Tables 5332.1-1 and 5333.1-1), thereby adding margin to the crane operations; (ii) the design criteria for the crane category of mechanical systems are based on Type I, II, and III features of ASME NOG–1–2004 (ASME, 2005aa) as well as ASME NUM–1–2004 Type IA (ASME, 2005ac) for the design of the jib cranes; (iii) the design criteria include single-failure proof features, such as dual load path (redundant reeving), conservative design factors, overload protection, redundant braking systems, over-travel limit switches, and inspection and testing provisions (ASME, 2005aa, Table 7200-1; ASME, 2005ac, Table NUM–1–8210–2), which prevent load drop due to the failure of a single component; and (iv) the design criterion for seismic stability for the design of the cranes accounts for loads and accelerations associated with a DBGM–2 seismic event, consistent with the site-specific information, which was evaluated in SER Section 2.1.1.7.3.1.1.1 and found to be acceptable by the NRC staff.

The NRC staff evaluated the applicant's information on the design method for overhead bridge cranes and jib cranes and finds that the applicant's design methods are adequate because they are consistent with ASME NOG–1–2004 (ASME, 2005aa) for the overhead bridge cranes and ASME NUM–1–2004 (ASME, 2005ac) for the jib cranes. These ASME codes are used in the nuclear industry for design and construction of cranes performing lifting functions. They include overload protection, redundant braking systems, over-travel switches, and protective devices to make the likelihood of a load drop by a crane extremely small. The use of ASME NOG–1–2004 (ASME, 2005aa) Type I cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa). In addition, because the ITS overhead bridge cranes and the jib cranes are SSC ITS (as evaluated by the NRC staff in SER Section 2.1.1.6.3.1) they will be designed to maintain their safety functions under seismic loads, based on DBGM–2 (SAR Section 1.2.2.2.1), as evaluated in SER Section 2.1.1.7.3.1.1.1. The NRC staff's evaluation of site-specific ground motions is documented in SER Section 2.1.1.1.3.5.2.5, where the applicant's analysis was found to be acceptable.

The NRC staff evaluated the applicant's information on the design and design analyses information of the overhead bridge cranes and jib cranes and finds that the applicant's design and design analyses for the overhead bride cranes and jib cranes are appropriate because (i) the designs are consistent with ASME NOG–1–2004 for overhead bridge cranes and ASME NUM–1–2004 for jib cranes. Both ASME NOG–1–2004 and ASME NUM–1–2004 (ASME, 2005aa,ac) are accepted codes for the design of crane systems at nuclear facilities. Both standards include recommendations on allowable stresses, hoist speed, over load protection, redundant brakes, and limit switches; and (ii) the applicant's material selections for the crane systems would be consistent with ASME NOG–1–2004 Sections 4200 "Materials and Connections" (structural) and 5200 "Materials" (mechanical) (ASME, 2005aa) and ASME NUM–1–2004 (ASME, 2005ac) Sections NUM–III–8200 (structural), NUM–III–8300 (mechanical), and NUM–III–8400 (electrical). Applicability of these standards to the proposed activities at the GROA is discussed further in Table 7-1.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the specialized crane systems is acceptable.

2.1.1.7.3.4.2 Special Lifting Devices

The applicant provided design information for special lifting devices in SAR Section 1.2.2.2.1. The special lifting devices include cask yokes, canister grapples, and lifting beams located at the end of mechanical handling equipment. Their functions are to lift and transport casks, overpacks, or canisters containing waste. These special lifting components either remove the cask lids or lift HLW and DOE SNF canisters during canister transfer operations. The grapples are equipped with mechanical jaw actuation mechanisms with safety release features.

**Design Criteria and Design Bases**

The applicant presented the design bases and design criteria for special lifting devices in SAR Tables 1.2.3-3 (IHF), 1.2.4-4 (CRCF), 1.2.5-3 (WHF), and 1.2.6-3 (RF). In these tables, the applicant provided specific design criteria to meet each of the design bases. For example, for canister grapples in the CRCF (Table 1.2.4-4), one of the design criteria the applicant specified to meet the design bases of drop protection is that the grapples are designed for loads and ground motions associated with a DBGM–2 seismic event.

To protect against a cask drop or load drop onto a cask/canister, the applicant developed a design criterion based on ANSI N14.6–1993 (ANSI, 1993aa); and NUREG–0612, Section 5.1.1(4) (NRC, 1980aa). Section 5.1.1(4) "Special Lifting Devices" of NUREG–0612 (NRC, 1980aa) clarifies that the stress factor in Section 3.2.1.1 of ANSI N14.6–1978 (ANSI, 1978aa) should be based on the combined maximum static and dynamic loads imparted on the special lifting device, in lieu of the original guideline in Section 3.2.1.1, which bases the stress design factor only on the weight (static load) of the load and of the intervening components of the special lifting devices. In addition, the applicant included special provisions in the lifting device design to prevent a load drop. For example, the inner lid lifting grapple for naval waste packages is designed with three lifting jaws, equally spaced, to engage the lid of the waste package. The three lifting jaws feature is shared by grapples used to lift different waste forms (e.g., DOE SNF canisters, Hanford multicanister overpack, as described in SAR Section 1.2.4.2.2.1.3). Further, raising or lowering the hoist is only possible if the grapple is fully engaged with the load. The grapple with a suspended load is mechanically prevented from unintentional disengagement.

To protect against a load drop during a seismic event, the applicant stated that the special lifting devices would be designed for loads and accelerations associated with a DBGM–2 seismic event.

**Design Methods**

The applicant stated that the design method for the design of special lifting devices is in conformance with the provisions of ANSI N14.6–1993 (ANSI, 1993aa), as modified by NUREG–0612 Section 5.1.1(4) (NRC, 1980aa) (SAR Section 1.2.2.2.1). Section 5.1.1(4) "Special Lifting Devices" of NUREG–0612 (NRC, 1980aa) clarifies that the stress factor in Section 3.2.1.1 of ANSI N14.6–1978 (ANSI, 1978aa) should be based on the combined maximum static and dynamic loads imparted on the special lifting device, in lieu of the original guideline in Section 3.2.1.1, which bases the stress design factor only on the weight (static load) of the load and of the intervening components of the special lifting devices. In addition, the applicant stated that the design method addresses seismic safety by a provision to account for loads and accelerations associated with a DBGM–2 seismic event.

**Design and Design Analyses**

In SAR Section 1.2.2.2.7, the applicant stated that the materials used for the special lifting devices would be consistent with ANSI N14.6–1993 Section 4 "Design" (ANSI, 1993aa). In addition to the design guidelines of ANSI N14.6–1993 (ANSI, 1993aa), the applicant stated in SAR Section 1.2.4.2.1.1.3.1 that the design includes safety features, such as sensors to provide status of load engagement, remote and local control capabilities to engage or disengage a load, and mechanical safety features to prevent grapple disengagement when a load is suspended from the grapple. The applicant stated that these design features would protect against a load drop from the special lifting device.

In SAR Section 1.2.2.2.9.2.3, the applicant defined the following specific design loads for the design and analyses of the special lifting devices, depending on the device and its location: (i) loads related to normal operations, (ii) loads due to a seismic event, and (iii) loads due to event sequences from a collision with an SSC or a canister containing waste. In the case of loads related to normal operations, the applicant used the design specifications of ASME NOG–1–2004 Section 4140 "Load Combinations" (structural), 5310 "Load Combinations" (mechanical), and Table 5415.1-1 "Load Combinations–Hoist Drive Shafting" (ASME, 2005aa). In the case of loads due to a seismic event, the applicant considered dead loads, live loads, and seismic loads of DBGM–2 levels. For a collision, the applicant considered dead loads, live loads, and loads associated with a collision event sequence. The applicant also stated that the hoisting equipment is designed in accordance with ASME NOG–1–2004 (ASME, 2005aa) (SAR Section 1.2.2.2).

**NRC Staff's Evaluation of the Special Lifting Devices Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the applicant's information in SAR Section 1.2.2.2.1, along with the applicable codes and standards, as related to the design criteria and design bases, design methods, and design and design analyses of the special lifting devices using the guidance in YMRP Section 2.1.1.7 and the design guidelines of ANSI N14.6–1993 (ANSI, 1993aa).
The NRC staff finds that the relationship between design bases and design criteria is adequately defined because the design criteria for the special lifting devices are based on the ANSI N14.6–1993 (ANSI, 1993aa) standard, the 1978 version of which was endorsed with a clarification in NUREG–0612 Section 5.1.1(4) (NRC, 1980aa) about the stress-intensity factors to be applied to the combined static and dynamic loads, instead of only the static loads. The NRC staff finds the use of the ANSI N14.6–1993 (ANSI, 1993aa) standard acceptable because (i) the provisions in this version of the standard for the design of the lifting devices are the same

as in the 1978 version, endorsed by the NRC in NUREG–0612 (NRC, 1980aa) and (ii) even though the ANSI N14.6–1993 (ANSI, 1993aa) standard has been withdrawn by the ANSI, the standard is still consistently used in the nuclear industry for designing special lifting devices and is acceptable for use at the GROA, as further discussed in Table 7-1.

The NRC staff evaluated the applicant's information on the design and design analyses for the special lifting devices and finds that the applicant's design and design analyses are appropriate because (i) the materials to be used for the special lifting devices are consistent with ANSI N14.6–1993 Section 4 (ANSI, 1993aa); (ii) the safety features (e.g., sensors to indicate the status of load engagement) provide additional assurance to prevent a load drop from the special lifting device; (iii) the applicant proposed design loads that are consistent with ASME NOG–1–2004 Sections 4140 "Load Combinations" (structural), 5310 "Load Combinations" (mechanical), and Table 5415.1-1 "Load Combinations – Hoist Drive Shafting" (ASME, 2005aa); and (iv) the ASME NOG–1–2004 (ASME, 2005aa) provides standards on the hoisting system (e.g., Sections 5120 "Hoisting Units", 5331 "Hoist Speeds") to which the special lifting devices are attached. The use of ASME NOG–1–2004 (ASME, 2005aa) Type I cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa).

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the special lifting devices are acceptable.

2.1.1.7.3.4.3 Other Mechanical Structures

The applicant described other mechanical structures in the GROA surface facilities as follows: shield and confinement doors, slide gates, rails, platforms, and racks. The shield and confinement doors and slide gates protect facility personnel from direct radiation (SAR Section 1.2.4.2.1.1.3.1). The rails support the WPTT, TEV, and large gantry cranes, such as the bridge crane of the CTM. In addition to the supporting function, the rails also provide electricity to power the TEV traction motors.

The applicant stated that the platforms include multilevel steel structures to provide personnel and tool access to the top of the aging overpacks or transportation cask and provide a single operating platform to access the top of the shielded enclosure of the WPTT for maintenance purposes (SAR Sections 1.2.3.2.1.1.3.1, 1.2.4.2.1.1.3.1, and 1.2.5.2.1.1.3).

The applicant stated that the racks stage SNF assemblies to allow (i) blending of fuel assemblies for thermal management and (ii) flexibility of loading and unloading of SNF assemblies (SAR Section 1.2.5.2.2.1.3). The SNF staging racks contain fixed neutron-absorber plates for criticality control that the applicant stated would be consistent with ANSI/ANS 8.21–1995 (ANS, 1995aa) and ANSI/ANS 8.14–2004 (ANS, 2004aa). The TAD canister staging racks and DOE canister staging racks are steel structures that hold the canisters for staging purposes. The staging racks provide seismic support for the canisters and support canisters at an elevation that the applicant stated minimizes potential drop height. The canisters contain neutron absorber plates to control criticality.

**Design Criteria and Design Bases**

The applicant presented the design bases and design criteria for shield doors, slide gates, and platforms in SAR Tables 1.2.3-3 (IHF), 1.2.4-4 (CRCF), 1.2.5-3 (WHF), and 1.2.6-3 (RF).

The design bases and design criteria for the rails and racks were presented in SAR Table 1.2.4-4 (CRCF) as follows.

- To protect against direct personnel exposure and mitigate radionuclide consequences, the applicant stated that it would use shield doors and slide gates consisting of steel plates with neutron-absorbing material. Further, a staggered door panel edge provides shielding between the mating door panel seams. To prevent impact with other conveyance equipment, the doors are equipped with obstruction sensors that would prevent the door from operating if any object is on its travel path. Additionally, interlocks would be in place to prevent the shield doors from opening if other doors are open or if dedicated radiation monitors are triggered. The motors operating the doors will be designed with insufficient torque to breach a canister if the slide gate were to close on the canister inadvertently.

- To protect against TEV derailment during waste package loading, the applicant stated that the rails are designed for a DBGM–2 seismic event so that a derailment during a seismic event is prevented.

- To prevent a seismic-induced collapse of the platforms or a waste container breach due to seismically induced impact of the cask transfer trolley or site transporter onto the platforms (SAR Table 1.2.4-4), the applicant stated that it would use the design methods and practices provided in the ANSI/AISC N690–1994 (AISC, 1994aa). The applicant further stated that the platform would be equipped with energy-absorbing features to limit impact forces from the platform on the waste container during a seismic event.

- To protect the canisters against tip over or canister impact, the applicant stated that the staging racks are designed to the DBGM–2 seismic loads. In addition, the applicant stated that a protective wall would be adjacent to the SNF staging rack to ensure that large objects, such as canisters, cannot collide with the rack, preventing damage to either the rack or SNF assemblies, or both (SAR Section 1.2.5.2.2.1.3). To protect against a postulated fire-induced canister breach, the applicant included a design criterion for the staging racks to include (i) a thermal barrier that encloses the bottom and sides of the canisters to control canister temperatures during fire scenarios (SAR Section 1.2.4.2.2.1.3).

**Design Methods**

In SAR Section 1.2.4.2, the applicant stated that the design method for shield and confinement doors, slide gates, steel platforms, and racks are in accordance with ANSI/AISC N690–1994 Section Q1.2 (AISC, 1994aa) code. In SAR Section 1.2.4.1.6, the applicant stated that the design method for TEV and WPTT rails are in accordance with ASME NOG–1–2004 (ASME, 2005aa). The applicant further stated that the SNF staging racks contain fixed-neutron absorbers for criticality control, in accordance with ANSI/ANS 8.21–1995 (ANS, 1995aa) and ANSI/ANS 8.14–2004 (ANS, 2004aa) (SAR Section 1.2.5.2.2.1.3).

**Design and Design Analyses**

The applicant listed the design features of the shield and confinement doors as follows: (i) the equipment confinement and shield doors with a confinement function are designed with interlocks so that they would not open when there is a potential for radiation, (ii) the facilities operation room is notified of the open or closed state of the shield doors, (iii) confinement doors are operated from the facilities operation room, and (iv) the shield doors are equipped with obstruction sensors that halt door travel when an object is detected in its path (SAR Section 1.2.4.2.1.1.3.1). The applicant stated that the rails are designed to support the WPTT and TEV. For the TEV, the rails also provide electrical power to the traction motors. The main design features of platforms and racks are to provide personnel safety and seismic protection for canisters staged on various racks.

In SAR Section 1.2.4.2, the applicant defined the load combinations for the shield and confinement doors, slide gates, and platforms as per ANSI/AISC N690–1994 Table Q1.5.7.1 (AISC, 1994aa). In addition, the applicant stated that the material to be used for this category of mechanical systems is in accordance with ANSI/AISC N690–1994 Section Q1.4 (AISC, 1994aa) (SAR Sections 1.2.4.2.1.7 and 1.4.2.2.7). In SAR Section 1.2.4.1.7, the applicant stated that the materials of construction and design loads for the TEV and WPTT rails are in accordance with ASME NOG–1–2004 (ASME, 2005aa). The applicant also used the load combinations in ASME NOG–1–2004 (ASME, 2005aa) for the design of the TAD canister staging racks and the truck cask handling frames. For other types of ITS racks or platforms and frames, the applicant stated that it would use ANSI/AISC N690–1994 Table Q1.5.7.1 (AISC, 1994aa) for load combinations.

**NRC Staff's Evaluation of Other Mechanical Structures' Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the applicant's information in SAR Sections 1.2.3.2.1.1.3.1, 1.2.4.2.1.1.3.1, and 1.2.5.2.1.1.3, along with the applicable codes and standards, as related to the design criteria and design bases, design methods, and design and design analyses of the shield and confinement doors, rails, platforms, slide gates, and racks using the guidance in YMRP Section 2.1.1.7. The staff also considered the industry codes and standards proposed for use by the applicant, ANSI/ANS 8.21–1995 (ANS, 1995aa) and ANSI/ANS 8.14–2004 (ANS, 2004aa), which the NRC staff has found to be acceptable for use at the GROA. Further information about the scope and applicability of the codes and standards used in the NRC staff's evaluations can be found in Table 7-1. The NRC staff finds that the design bases and design criteria for the shield and confinement doors, rails, platforms, slide gates, and racks are adequate because the design bases and design criteria addressed the safety functions of protecting against direct exposure of personnel, precluding collapse onto waste containers, and mitigating the consequences of radionuclide release (SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3). The safety features (e.g., obstruction sensors, interlocks and doors incapable of producing sufficient torque to breach a canister) afford additional margins for radiological and operational safety. In addition, the staff finds the applicant's TEV design for derailment prevention acceptable, in part, because the TEV rails are designed to a DBGM–2 seismic event. The NRC's staff's evaluation of the TEV is further described in SER Section 2.1.1.7.3.5.1, where the NRC staff finds the design features to protect the TEV from derailment are acceptable.

The NRC staff also finds the applicant's use of the design methods and practices provided in the ANSI/AISC N690–1994 (AISC, 1994aa) acceptable because it is consistent with standard engineering practice, will be used in concert with other codes and standards, and are sufficient for its intended use at the GROA, as further discussed in Table 7-1. Equipping the platforms with additional structural capacity and energy absorbing features to limit the impact force to the waste container provide additional protection to the waste containers.

The NRC staff also finds that the applicant's use of (i) a protective wall to prevent collision of SNF staging racks with large objects and (ii) a thermal barrier to control canister temperatures during fires are acceptable because the use of these features add additional protection to the SNF during a collision or fire beyond that which is typically used in similar facilities in the nuclear industry.

The NRC staff evaluated the applicant's information on the design methods for the shield and confinement doors, rails, platforms, slide gates, and racks and finds that the design methods for the shield doors, confinement doors, slide gates, platforms, and racks are acceptable because the methods are consistent with ANSI/AISC N690–1994 (AISC, 1994aa). The NRC staff also finds that the applicant's use of ASME NOG–1–2004 (ASME, 2005aa) is appropriate for the design of rails because this code addresses the use of rails in conjunction with overhead bridge cranes and trolleys. The NRC staff further finds that the applicant's design methods for design of platforms and SNF staging racks with fixed-neutron absorbers are consistent with ANSI/ANS 8.21–1995 (ANS, 1995aa) and ANSI/ANS 8.14–2004 (ANS, 2004aa) for criticality control and are, therefore, appropriate, as evaluated in SER Section 2.1.1.6.3.2.6. The applicability of each of these standards to the GROA is further discussed in Table 7-1.

The NRC staff evaluated the applicant's information on the design and design analyses for the shield and confinement doors, rails, platforms, slide gates, and racks and finds that the applicant's design and design analyses are appropriate because the applicant would use the design recommendations of ANSI/AISC N690–1994 (AISC, 1994aa) and ASME NOG–1–2004 (ASME, 2005aa) codes to design the shield doors, slide gates, rails, platforms, and racks. These codes are consistent with the standard engineering practices for designing safety-related steel structures for nuclear facilities. The NRC staff finds that the applicant's use of ASME NOG–1–2004 (ASME, 2005aa) to design the TAD canisters staging racks and the truck cask handling frames are appropriate because ASME NOG–1–2004 (ASME, 2005aa) contains acceptable guidance for load combinations (e.g., Section 4140 "Load Combinations"). The use of ASME NOG–1–2004 (ASME, 2005aa) Type I cranes is also consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa). Similarly, the NRC staff finds that the applicant's use of ANSI/AISC N690–1994 (AISC, 1994aa) Table Q1.5.7.1 for other types of ITS racks, platforms, and frames appropriate because this standard provides recommendations for load combinations suitable for safety-related steel structures, such as racks, platforms, and frames for nuclear facilities. The applicability of ANSI/AISC N690–1994 (AISC, 1994aa) is further discussed in Table 7-1.

On the basis of the NRC staff's evaluation described above, the NRC staff finds that the applicant's design criteria and design bases, design methods, and design and design analyses of the other mechanical structures (shield and confinement doors, rails, platforms, slide gates, and racks) are acceptable.

**NRC Staff's Conclusion of Other Mechanical Systems Design**

On the basis of the NRC staff's evaluation of the applicant's information described above on the design of other mechanical systems: (i) crane systems (e.g., cask handling cranes and jib cranes); (ii) special lifting devices (e.g., yokes, grapples, and lifting beams); and (iii) other mechanical structures (shield and confinement doors, slide gates, rails, platforms, and racks), the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the description provided by the applicant of the designs of other mechanical systems adequately (i) provides information on materials of construction, dimensions, proposed codes and standards, analytical and design methods; (ii) defines the relationship between design criteria and the performance objectives; and (iii) identifies the relationship between the design bases and the design criteria.

**2.1.1.7.3.5 Transportation Systems**

The applicant provided design information for the ITS transportation systems used at the GROA in SAR Sections 1.2.8.4, 1.3.2, 1.3.3, and 1.3.4. The four ITS transportation systems were identified by the applicant in SAR Section 1.9.1, based on the PCSA in SAR Sections 1.6 through 1.9, and reviewed and found to be acceptable by the NRC staff in SER Sections 2.1.1.4.3.2 and 2.1.1.6.3.2.8.3. The transportation systems are the (i) Transport and Emplacement Vehicle (TEV), (ii) Site Transporter, (iii) Cask Tractor and Cask Transfer Trailer (CTCTT), and (iv) Site Prime Mover. The applicant stated that the transportation systems are designed for loads and accelerations associated with DBGM–2 seismic events and as provided in SAR Table 1.2.8-2. The NRC staff evaluated the design bases and design criteria, design method, and design and design analyses of these transportation systems, as follows.

2.1.1.7.3.5.1 Transport and Emplacement Vehicle

The applicant stated that it plans to use the TEV to transport the loaded waste packages from the surface facilities (CRCF and IHF) to their designated locations in the emplacement drifts (SAR Section 1.3.1, Figure 1.3.1-3). The applicant stated that the TEV operation consists of (i) handling the waste packages by accepting, lifting, and securing the waste packages inside the protective structure of the TEV for transport; (ii) shielding personnel from exposure to radiation from waste packages; (iii) transporting the waste packages on the pallets from the surface facilities to the subsurface facility in a controlled manner; (iv) emplacing the waste packages in the emplacement drift; and (v) returning the TEV to the surface facility. The applicant also stated that it plans to use the TEV for waste package retrieval operations (SAR Section 1.3.3.3.2). Furthermore, the applicant stated that, under normal conditions, the retrieval operations consist of performing the reverse sequence of steps that defines the emplacement operations and under off-normal conditions and the TEV performs the retrieval operations after the off-normal condition is restored to normal conditions, as described by the applicant in SAR Section 1.11.1, and reviewed and found to be acceptable in SER Section 2.1.2. If an event or incident occurs during TEV operations, the TEV is stopped remotely, and recovery steps are taken to restore the operations to normal conditions, as discussed by the applicant in SAR Section 1.11.1.

The applicant provided the following descriptions of the TEV: (i) the TEV is a crane, rail-based transporter {4.9-m [16-ft]-diameter operating envelope} (SAR Figure 1.3.4-20) propelled by eight electric motors powered by an electrified third rail; (ii) TEV is manually operated or computer-controlled, fully instrumented handling equipment with sensors and communication networks; (iii) the TEV contains a battery backup system with sufficient capacity to power the sensors and maintain communication with the Central Control Command in the event of a normal power failure; and (iv) the TEV is also equipped with a restraint system to limit the vertical lift of the TEV off the rails during a seismic event (SAR Figure 1.3.1-3), redundant braking systems, and a shielded enclosure that surrounds the waste package.

**Design Criteria and Design Bases**

The applicant provided the nuclear safety design bases and their relationship to the design criteria for the TEV in SAR Section 1.3.2.3.1 and SAR Table 1.3.3-5. In this table, the applicant provided specific design criteria for each design basis, including controlling parameters and bounding values.

To protect against tipover during a DBGM–2 seismic event, the applicant included a design criterion of low TEV center of gravity with the rails widely spaced to prevent the TEV from tipover during a seismic event. The TEV design includes the use of a 3.4-m [11-ft]-wide track to fit within the 4.9-m [16-ft]-wide operating envelope of the emplacement drift (SAR Section 1.3.2.3.1). In addition, the applicant stated the TEV was designed consistent with ASME NOG–1–2004 (ASME, 2005aa) Type I single-failure proof criteria.

To protect against runaway during operations, the applicant specified a design criterion for the TEV to include special drive mechanisms and braking systems. More specifically, the applicant stated that the TEV's design would include drive motors, high-ratio gearbox, and disk brakes to prevent runaway.

To protect against derailment, the applicant used the features in the ASME NOG–1–2004 (ASME, 2005aa) code for Type I (single-failure proof) cranes as the design criteria for protection against derailment. In addition, the TEV and its interface with the rails at the loadout station are equipped with seismic restraints to limit the vertical movement of the TEV during a DBGM–2 seismic event (SAR Section 1.3.2.3.1).

To protect personnel from direct radiation exposure, the applicant specified a design criterion for the use of interlocks and shielded enclosures on the TEV. For example, the applicant specified interlocks to prevent opening the TEV front and rear shield doors in unrestricted areas between the surface handling facility and the emplacement drift turnouts (SAR Table 1.3.3-5).

To protect the waste packages from ejection during a spectrum of seismic events, the applicant included a design criterion providing locks for the TEV front and rear shield doors. The applicant indicated that it would use mechanical switches and hardwired circuitries that mechanically prevent unintentional motion of the shield doors (SAR Section 1.3.3.5.3 and Table 1.3.3-5).

**NRC Staff's Evaluation of the TEV Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the design criteria and design bases of the TEV and finds that the design bases and design criteria for the TEV are adequate because they (i) include a design criterion to keep the TEV's center of gravity low and providing a wide base to protect against a tipover during a DBGM–2 seismic event; (ii) include the single failure criterion in ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes that would prevent a load drop as a result of single component failure; (iii) encompass a design criterion to equip the TEV with special drive mechanisms and braking systems to protect against TEV runaway, and the proposed run-away prevention features are consistent with the industry practice of protecting rail-based vehicles from unintended motions; (iv) adopt the use of ASME NOG–1–2004 (ASME, 2005aa) Type I single-failure proof as a design criterion for protection against the derailment of the TEV during a DBGM–2 seismic event, consistent with the standard engineering practice in nuclear industries for heavy load dropping protection during an accident (e.g., derailment); (v) address protection of personnel from radiation exposures through the use of interlocks and shielded enclosures, which is consistent with the nuclear industry radiation protections measures; and (vi) include a design criterion to protect the waste packages from ejection through the use of locks to the TEV front and rear shield doors.

**Design Methods**

The applicant stated that due to the unique nature of the GROA site and the specialized nature of the TEV operations, consensus codes and standards may not be fully applicable for the design of the TEV (SAR Section 1.3.3.5.8). Therefore, the applicant utilized a design method for the TEV based on several studies. In one study (BSC, 2008ck), the applicant identified codes and standards [ASME NOG–1–2004 (ASME, 2005aa) for Type I cranes, along with six additional supporting standards [e.g., (IEEE, 2006aa)] it found applicable. In a second study (BSC, 2008cl), the applicant (i) evaluated the applicability of the identified standards to the TEV application, (ii) identified technical gaps in the standards, and (iii) defined additional provisions to supplement the existing codes and compensate for the gaps. In the third and final study (BSC, 2008cr), the applicant outlined a design development plan (DDP) for the TEV. This DDP included (i) selection of TEV structures, systems, and components; (ii) engineering calculations; (iii) modeling; (iv) failure mode and effects analysis; (v) fault tree analysis; and (vi) various levels of testing (e.g., bench testing of components, factory acceptance testing at full scale, and off-site integrated testing). On the basis of these studies, the applicant stated that its design method includes (i) performing further reliability analyses and (ii) generating detailed design assemblies, wiring diagrams, process and instrumentation diagrams, and logic diagrams for all SSCs (i.e., drive motors, gearboxes, shield door actuators, door locks, interlock switches, and door hinges) involved in performing the TEV's safety functions (SAR Table 1.3.3-7).

**NRC Staff's Evaluation of the TEV Design Methods**

The NRC staff evaluated the applicant's information on the design methods of the TEV that included three design studies and finds that the applicant's selection of the ASME NOG–1–2004 standard (ASME, 2005aa) with additional supporting IEEE standards (IEEE, 2006aa) in the design is appropriate because this standard includes consideration for dynamic seismic qualifications, materials controls, harsh/radiation environmental-condition requirements, single-failure proof requirements, and testing requirements. The NRC staff finds the applicant's design method for the TEV is adequate because (i) the design method is consistent with the applicant-cited codes and standards (ASME, 2005aa, IEEE, 2006aa), which the NRC staff finds applicable for use at the GROA, as proposed by the applicant; (ii) the three-step process of evaluating the applicability of identified codes to the TEV, identifying gaps and identifying additional necessary provisions to supplement the existing codes and standards is consistent with the industry practice of developing prototypical equipment when appropriate; (iii) the applicant's DDP addresses reliability analyses of the detailed design and provides the basis for the performance specifications, test specifications, and test procedures; and (iv) extended factory acceptance testing at full scale would be performed to demonstrate the TEV performance and reliability of meeting the design criteria (SAR Section 1.3.3.5.6). Therefore, the NRC staff finds that the TEV design methods are acceptable.

**Design and Design Analyses**

In designing the TEV, the applicant first considered specific characteristics of the GROA site (SAR Section 1.3.3.5.1.1), such as the (i) layout and operations of surface facilities and loadout rooms (e.g., TEV rails extending into the surface facilities); (ii) surface-to-subsurface elevation changes (up to 2.15 percent grade) and environmental hazards {such as tail winds of 145 km/hour [90 mph]}; (iii) layout and operations of the subsurface facility {e.g., minimum curve radius of 61 m [200 ft]; 808 m [2,651 ft] maximum travel one-way distance}; (iv) thermal characteristics of the subsurface {such as air temperature of 50 °C [122 °F]}; and (v) waste package sizes {e.g., maximum weight of $2.7 \times 10^5$ kg [300 tons], maximum length of 6,299 mm

[248 in], and maximum height of 2,349 mm [92.49 in]}. The applicant's load combinations include loads from normal operating conditions, event sequences, and the effects of natural phenomena, including seismic events.

The applicant used the ASME NOG–1–2004 (ASME, 2005aa) Type I overhead crane code to define the material specifications, load combinations, and design analyses for the design of the TEV lifting and propulsion functions. The applicant also specified the industry standard in Doman (1988aa) for the design and construction of the TEV shielded enclosure, front shield doors, door drives, hinges, and locks. The applicant stated that it relied on this document (Doman, 1988aa) as guidance for load combinations, design considerations for the hinges, door drive systems, and safety devices associated with the shield doors.

The applicant's TEV design features address each of the following design criteria and design bases:

- To protect the TEV against tipover. The applicant addressed preventing TEV tipover during a DBGM–2 seismic event (SAR Section 1.3.3.5.1.1) by designing the TEV with a wide vehicle base {3.4 m [11 ft]} and a low center of gravity. The applicant stated that its TEV design ensured that the waste package positioning was low in the vertical direction {i.e., less than 356 mm [14 in] from the top of the rail to bottom of the emplacement pallet}. In addition, the applicant specified ASME NOG–1–2004 Section 4457 (ASME, 2005aa) to define a specification for gantry stability during extreme environmental or off-normal conditions, as described in BSC (2008ck, Section 6.12). The standard specifies that the TEV shall have a safety factor of no less than 1.1 against overturning under abnormal event loading. In addition, the TEV design included a seismic restraint between the TEV chassis and the rails (SAR Figure 1.3.3-41), and the TEV operating speed would be limited to 2.7 km/hour [1.7 mph] (SAR Sections 1.3.2.3.1 and 1.3.3.5.3).

- To protect the TEV against runaway during operations. The applicant incorporated five design and control elements: (i) drive components to mechanically limit the speed of the TEV to 2.7 km/h [1.7 mph], (ii) high-torque drive motors, (iii) integral disk brakes in the drive motors, (iv) rail brakes, and (v) high-ratio gearboxes (SAR Section 1.3.3.5.3).

    1. Limit the TEV speed to 2.7 km/h [1.7 mph] with a design tolerance of ±10 percent, as recommended by ASME NOG–1–2004 Table 5333.1-1 (ASME, 2005aa) as the "fast" speed for the largest TEV payload of 80,829 kg [89.1 ton]. This speed is consistent with the waste package envelope information (SAR Sections 1.3.2.3.1, and 1.3.3.5.3). To achieve the desired speed limit, the applicant selected a 1,750-rpm motor coupled to a 914-mm [36-in] wheel. The applicant further stated (SAR Section 1.3.2.3.1) that this speed is within the design bases limit, even during descent of the North Ramp, which has a downgrade of 2.15 percent (BSC, 2008bz; Sections 3.1.1 and 3.2.1.32).

    2. The horsepower requirements for the TEV motor to negotiate a rail grade of 2.5 percent for operations, both up and down on grade and for a curve that is on a grade, were developed using the guidance in Cummins and Given (1973aa). This condition is not addressed in ASME NOG–1–2004 code (SAR Section 1.3.3.5.1). The applicant calculated a traction power of approximately 140 hp (BSC, 2008cb) that it stated would allow the TEV to support waste package transportation for both emplacement and retrieval operations.

3. The applicant included redundancy to mitigate the effect of power loss by incorporating eight integrated double disc brakes, one pair for each drive motor. This was done in consideration of interaction of the TEV with other SSCs that could potentially affect the speed control, such as electrical power failure. The applicant also introduced an additional braking system for parked or off-normal conditions. The applicant selected rail brakes (or “thrusters”) that directly couple the TEV to the rail in a wedge-like braking action. To supplement the disc brakes, the applicant added a second level of runaway-prevention redundancy by adding eight high-gear-ratio (100.75:1) gearboxes, one for each drive motor. These gearboxes have a noncoasting design (DOE, 2009ez).

The applicant described a custom seismic restraint system, as shown in SAR Figure 1.3.3-41 to address TEV derailment prevention at the loadout station during a seismic event. This passive restraint system consists of L-shaped structures located on the underside of the TEV chassis and extending under the railhead. These L-shaped seismic restraints, located at the front and back and left and right sides of the TEV, are designed to prevent derailment by limiting the vertical motion of the TEV during a seismic event (SAR Section 1.3.3.5.1.1 and Figure 1.3.3-41). Outside the loadout station (i.e., during transport), the applicant stated that it reduced the potential for derailment by specifying double-flanged wheels for the TEV. Wheel climb derailment on the tightest curve should not occur. The applicant addressed derailment due to wheel climb on the tightest curve {radius of 61 m [200 ft]}. The applicant performed a geometric assessment to ensure that wheel climb would not occur. If a derailment did occur, the TEV design would reduce the impact on the waste package by limiting the height of wheel drop, thereby minimizing potential breach. The applicant stated (SAR Section 1.3.2.3.1) that the bottom faces of the TEV chassis and the base plate are at the same height from the rails to ensure that a potential drop would only be 76 mm [3 in]. The applicant also performed structural analyses, which showed that, even with a 508-mm [20-in] drop (SAR Table 1.5.2-9), the outer corrosion barrier of the waste package would receive only 25 percent of the energy necessary to break the waste package (SAR Table 1.5.2-9).

- To protect the TEV against waste package ejection and against inadvertent door opening, the applicant included electrically activated door lock systems in the TEV design (SAR Section 1.3.3.5.3). The shield door system would consist of two, outward-swinging doors. One shield door would house the lock solenoids, and the other door would contain structurally featured holes in which the steel shot bolts would penetrate to prevent door motion. These design features protect against waste package ejection resulting from a seismic event, collision, derailment, normal transport, or tipover. The applicant indicated that the cross-sectional area and material strength are to be selected to withstand loads resulting from a DBGM–2 seismic event. The applicant stated that it incorporated a non-ITS collision-avoidance system and physical stops to prevent collisions (SAR Section 1.3.2.1).

1. The applicant incorporated a series of interlocks and a hardwired ITS switch mounted on the TEV that the applicant stated were designed per ASME NOG–1–2004 and IEEE 384–1992, 323–2003, 383–2003, 344–2004, and 336–2005 (IEEE, 2006aa; IEEE, 2005aa; IEEE, 2004aa,ab; IEEE, 1998aa). The interlocks and hardwired ITS switch prevent inadvertent opening of the TEV’s front and rear doors in unrestricted areas.

2. ITS switches are interlocked with the door solenoid circuitry to prevent opening of the doors when the ITS switch is electrically determined to be in the incorrect position. The applicant stated that externally mounted mechanical arms (permanently located only in areas that are safe for door opening) are designed to physically engage and actuate the ITS switches on the TEV before the front and rear doors are opened. The applicant referenced Doman (1988aa) as the industry guidance for the design of these components. In addition, the applicant indicated that operators in the Control Center need to confirm, in accordance with operating procedures [Section 2.5.2 (BSC, 2008bz)], the proper position of the ITS switch before the TEV is allowed to proceed.

**NRC Staff's Evaluation of the TEV Design and Design Analyses**

The NRC staff evaluated the load combinations used by the applicant for the design of the TEV and finds the load combinations acceptable because they are consistent with the ASME NOG–1–2004 (ASME, 2005aa) code, which is applicable to the GROA, as discussed in Table 7-1. The NRC staff evaluated the applicant's information on the TEV design features for meeting each of the nuclear safety design bases and design criteria and finds the applicant's TEV design and design analyses acceptable because the applicant addressed the following in the design of the TEV.

For the tipover scenario, the applicant (i) applied DBGM–2 for seismic loads, consistent with the design criteria for ITS SSCs, (ii) referenced the applicable section of the ASME NOG–1–2004 (ASME, 2005aa) code with respect to gantry stability, and (iii) included a seismic restraint and speed limits to protect the TEV against a tipover. The NRC staff finds the applicant's use of seismic restraints acceptable because they represent a passive design that limits the vertical motion of the TEV during a seismic event (SAR Figure 1.3.3-41); and the ASME NOG–1–2004 code, Section 7000, which the applicant used in its design, includes structural and weld testing for seismic restraint systems. These tests will further ensure that the TEV is able to perform its safety functions during design basis events.

For the runaway scenario, the applicant (i) limited the speed and (ii) provided redundancy using mechanical features (e.g., integral disc brakes and high-ratio gearboxes) into the design. The NRC staff finds the applicant's runaway prevention acceptable because the engagement of the disc brakes is both redundant and independent of a human operator's reaction time or sensor response time.

For limiting the TEV speed, the applicant referenced the applicable table of the ASME NOG–1–2004 (ASME, 2005aa) standard and selected mechanical features (e.g., a motor coupled with a wheel) to match the desired limiting speed. The NRC staff finds that the applicant's approach of restricting the TEV maximum speed to 2.7 km/hr [1.7 mph] is acceptable because this value is consistent with ASME NOG–1–2004 (ASME, 2005aa). This maximum potential crash impact speed is less than an impact speed of 22 km/hour] [13.6 mph], which is the speed that could cause a waste package breach, as detailed in BSC (2008bz, Section 3.2.1.10). In addition, the NRC staff finds the applicant's use of the drive motor appropriate because the drive motor was selected to match the desired maximum speed limit and the traction power was properly sized according to the SME Mining Engineering Handbook (Cummins and Given, 1973aa).

For redundancy in braking, the applicant (i) incorporated integrated double disc brakes and (ii) added additional parking brakes and high-gear-ratio gearboxes. The NRC staff finds the applicant's use of parking rail brakes is adequate because it utilizes the weight of the TEV to produce the frictional normal force rather than relying on an external actuation system. The NRC staff further finds the applicant's use of the high-ratio gearboxes acceptable because they would limit the speed of the TEV, thereby minimizing the likelihood of runaway.

For derailment during a seismic event scenario, the applicant (i) added seismic restraints and (ii) specified double-flanged wheels for the TEV. The NRC staff finds that this design is consistent with rail vehicle design criteria in ASME NOG–1–2004 (ASME, 2005aa). The NRC staff finds that the applicant's use of double-flanged wheels for the TEV would reduce the potential for derailment outside the loadout station (i.e., during transport) because (i) considerable energy would be required to lift the wheels beyond the flanges during a seismic event; and (ii) is consistent with the guidelines in ASME NOG–1–2004 (ASME, 2005aa) code, related to material selection, loading, clearances, and flange width and height for these types of flanged wheels.

For derailment due to wheel climb on the tightest curve scenario, the applicant (i) limited the height of wheel drop; (ii) limited the waste package (WP) height drop to 76 mm [3 in] by design; and (iii) compared the energy absorbed by the WP from a derailment drop to the energy necessary to break the WP. The NRC staff finds that the applicant's mitigation of the TEV derailment effect by limiting the drop height to 76 mm [3 in] is adequate because the loading on the outer barrier of the WP would be significantly less than the loading limit established by the applicant's acceptance criteria of a 508-mm [20-in] drop (SAR Table 1.5.2-9), as evaluated earlier in this section of the SER.

For WP ejection and inadvertent door opening scenarios, the applicant installed shield doors, and incorporated a collision avoidance system as a redundant measure to reduce the frequency of collisions. The applicant also evaluated the applicability of the design codes and standards to ensure that the shield doors will perform their safety functions (BSC, 2008ck). The NRC staff finds that the TEV design for protection against WP ejection and against inadvertent door opening to protect personnel from radiation exposure is adequate because the design would use appropriate design codes and standards for the shield doors to perform their safety functions and incorporates redundancy (such as collision mitigation systems).

For the inadvertent opening of the TEV front/rear doors scenario, the applicant incorporated a hardwired ITS switch and interlocks that are designed to industry-accepted ASME and IEEE standards. The applicant also designed the TEV with externally mounted mechanical arms to physically engage the ITS switch prior to opening the doors (SAR Section 1.3.2.3.1). The NRC staff finds this approach acceptable because it provides hardware-based interlocks and redundancy for this function. The NRC staff's evaluation of interlocks and ITS controls for the GROA is provided in SER Section 2.1.1.7.3.7.

In summary, and based on the evaluation of the applicant's information on the TEV design features described above, the NRC staff determines that the TEV design and design analyses are adequate because the TEV is protected against tipover, runaway, derailment, waste package ejection, and inadvertent shield door opening. The NRC staff's review of these event sequences is provided in SER Sections 2.1.1.4.3.2.1, 2.1.1.4.3.2.2,, 2.1.1.4.3.4.1, and 2.1.1.4.3.4.2. The NRC staff's review of the SSCs ITS for the TEV are provided in SER Section 2.1.1.6.3.2.8.3. Therefore, the NRC staff concludes that the TEV has been designed adequately to meet the design bases and design criteria, and thus is able to perform its intended safety function.

2.1.1.7.3.5.2 Site Transporter

The applicant stated that it plans to use the site transporter in intrasite operations to transport loaded and unloaded aging overpacks and unloaded DPCs inside shielded transfer casks between surface facilities, such as the CRCF, WHF, RF, AF, and the low-level radioactive waste facility (SAR Section 1.2.8.4.1). The applicant provided the design features of the site transporter in SAR Section 1.2.8.4.1.1 and a mechanical envelope diagram in SAR Figure 1.2.8-49. The applicant also indicated that the site transporter would be designed to withstand the natural phenomena described in SAR Table 1.2.2-1, including horizontal and vertical ground motion shown in SAR Figures 1.2.2-8 to 1.2.2-13.

**Design Criteria and Design Bases**

Based on its PCSA in SAR Section 1.6 through 1.9, the applicant presented the nuclear safety design bases for the site transporter and their relationship to the design criteria in SAR Table 1.2.8-2 (SAR Section 1.2.8.4.1.5). In this table, the applicant provided specific design criteria for design basis, including controlling parameters and bounding values. First, the applicant defined procedural safety controls for site transporter motion to prevent spurious movements. Complementing the procedural safety controls, the applicant described a design basis limiting the site transporter speed to 4 km/hour [2.5 mph] to protect against runaway.

Second, the applicant defined a design basis to preclude fuel tank explosion. The applicant stated that the site transporter is equipped with a fuel tank for its conventional engine that carries a maximum of 378L [100 gal] of diesel fuel (SAR Section 1.2.8.4.1.1), which limits the potential consequences of a fire. In addition, the site transporter is equipped with an onboard fire suppression system that mitigates the consequences of a potential fire. Furthermore, the applicant stated that the combustion engine on the site transporter would only be operated outdoors (SAR Section 1.2.8.4.1.1). When indoors, the transporter engine would be shut off, and the site transporter will use electric power for propulsion via an electrical cable. This configuration minimizes the risk for explosion of the fuel tank.

Third, the applicant defined three additional design bases with corresponding design criteria for the site transporter to ensure safety: (i) a design criterion that limits the aging overpack lift height capability of the transporter to 0.91 m [3 ft], thus reducing the severity of a potential drop; (ii) a design criterion specifying a wide base resulting in an inherent vertical stability that protects against tipover; and (iii) a design provision for clearance and energy-absorbing features to minimize sliding impact and inducing stresses on the waste container.

**NRC Staff's Evaluation of the Site Transporter Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on design criteria and design bases for the site transporter and finds that the applicant's design criteria and design bases for the site transporter are acceptable because they (i) addressed the relevant safety functions (SAR Table 1.2.8-2) of protecting against the site transport's spurious movement, runaway, fuel tank explosion, drop of the waste package, tipover, and sliding impact and inducing significant stress on the waste package; and (ii) were derived from the PCSA as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6.3.2.8.3.

**Design Methods**

The applicant based the design method of the site transporter following the specifications of the ASME NOG–1–2004 code for Type I (single-failure proof) overhead cranes (ASME, 2005aa) (SAR Section 1.2.8.4.1.6). In addition, the applicant indicated that the following specifications or codes recommended by ASME NOG–1–2004 would also be followed for the design of the site transporter (DOE, 2009ez): (i) ASTM A 572/A 572M–04 (ASTM International, 2004ac) for the car body, crawler frame, rear lift fork assembly, front support arms, and cask restraint system, all of which will be constructed from steel; (ii) American Welding Society D14.1/D14.1M–2005 (American Welding Society, 2005aa); (iii) ANSI/AGMA 2001–C95 (American Gear Manufacturers Association, 2001aa) for machining tolerance, backlash, and inspection of gearing; (iv) CMAA 70-2004 (Crane Manufacturers Association of America, 2004aa) for track-type limit switches; and (v) NEMA MG-1 (NEMA, 2006aa) for motor size selection.

**NRC Staff's Evaluation of the Site Transporter Design Methods**

The NRC staff evaluated the applicant's information on the design method for the site transporter and finds that the applicant's design method for the site transporter is adequate because the applicant's methods for the design of the site transporter are based on ASME NOG–1–2004 (ASME, 2005aa) and other codes recommended in ASME NOG–1–2004, which include specific guidelines for material specifications; welding specifications; and the design of structural support, lifting, propulsion, braking functions, and other ITS SSC components of the site transporter. The NRC staff finds each of these codes acceptable for use at the GROA, as proposed by the applicant, as further discussed in Table 7-1.

**Design and Design Analyses**

The applicant's design and design analyses for the site transporter included combinations of loads from normal operating conditions, event sequences, and the effects of natural phenomena, including seismic events. The applicant described the site transporter design features to address each of the nuclear safety design bases and design criteria, as follows.

To protect the site transporter against runaway, the applicant specified a safety design provision to limit the site transporter to 4 km/hour [2.5 mph] through sizing of the electric motors and gearboxes to constrain the maximum rotational speed (SAR Section 1.2.8.4.1.1). The applicant acknowledged that the safety design speed limit for the site transporter of 4 km/hour {2.5 mph [220 ft/min]} was higher than the speed recommended in ASME NOG–1–2004 Table 5333.1-1 (ASME, 2005aa) code. ASME NOG–1–2004 recommends a maximum Type I crane speed of 3.2 km/hour [1.98 mph] to transport loads weighing between 0 and 44,452 kg [0 and 49 ton]. To assess the impact of this deviation, the applicant analyzed (BSC, 2008at) the robustness of transportation casks required to withstand a 1,016-mm [40-in] horizontal drop onto an unyielding object. The applicant's structural analysis showed that the impact energy of a collision at a 4 km/hour [2.5 mph] site transporter speed is a factor of 90 less than the impact energy the cask is designed to survive from a 1,016 mm [40 in] drop. Furthermore, the applicant stated that for heavy loads, such as typical 230,000-kg [250-ton] vertical aging overpacks that will be handled by the site transporter, the moving speed of the site transporter would be lower than 4 km/hour [2.5 mph]. As described in its response to the NRC staff's RAI (DOE, 2009ez), the applicant concluded that no breach would occur from a site transporter collision at its design speed limit.

To protect the site transporter against fuel tank explosion, the applicant stated that the site transporter would be equipped with fuel tanks that preclude explosions and are available commercially. The applicant stated that the tanks include features such as flame-resistant coatings, self-sealing polymeric foam, insulating foam, Kevlar®/Dyneema®/Twaron® protective wrap, and internal cell foam that meet the explosion-proof standards [ASME NOG–1–2014 (ASME, 2005aa)] for the site transporter. The applicant stated that the site transporter would be equipped with a fire-suppression system and only carries a maximum of 378 L [100 gal] of diesel fuel (SAR Section 1.2.8.4.1.1). The applicant also stated that the TAD canister would be designed to withstand a fully engulfing fire characterized by an average flame temperature of 1720 °F and lasting for a period to be determined by calculations of a pool spill fire formed by 378 L [100 gal] of diesel fuel (BSC, 2007cs), without failure of its containment function (SAR Table 1.5.1-10). Similarly, the aging overpack would be designed to withstand the same fully engulfing fire without failure of its shielding function. The applicant stated that fully engulfing fire is defined in 10 CFR 71.73(c)(4) and clarified that the burning period is calculated based on a pool fire of all the site transporter hydrocarbon fuel and other combustible lubricating and hydraulic fluids plus other combustible and flammable materials on the site transporter (SAR Section 1.4.3.1.2).

To protect the site transporter from the spurious movement during lifting/lowering operations (SAR Section 1.2.8.4.1.4) involving a loaded canister being placed into or removed from the aging overpack, the applicant follows a Procedural Safety Control (PSC)-2 (SAR Table 1.9-10) to disconnect the electrical power to the site transporter with brakes engaged (SAR Section 1.2.8.4.1.4). As a part of PSC-2, the operator also performs an independent verification of the deactivation of the electrical power.

To reduce the severity of a drop on the aging overpack, the applicant designed the site transporter with a maximum lifting height of 0.3 m [1 ft] (SAR Section 1.2.8.4.1.5). Additionally, the applicant prescribed testing of the lifting system before operation of the site transporter in the GROA. A dynamic load test over the full range of the lift using a test weight at least equal to 110 percent of the lift weight is to be conducted to provide assurance against premature failure of the lifting members. The applicant also referenced an industry standard—ASME NQA–1–2000 Subpart 2.15 (ASME, 2000aa)—to be followed for the hoisting, rigging, and transporting of items. In addition, the applicant indicated in BSC (2007av, Section 4.8.1.2.7) its adoption of NUREG–0612 (NRC, 1980aa) and ANSI N14.6–1993 (ANSI, 1993aa). The former is a standard for the control of heavy loads at nuclear power plants and the latter for the design of special lifting devices for shipping containers weighing 4,536 kg [10,000 lb] or more.

To protect the site transporter against sliding impacts, the applicant described operating clearances and energy-absorbing features (SAR Table 1.2.8-2) to limit the frequency of sliding impact of the site transporter into a wall and the induced stresses on the waste package due to seismic events. Additionally, the applicant stated that it designed the site transporter with a drive system consisting of tracks that provide significant resistance to sliding.

The applicant credited the site transporter with the safety function of protecting against tipover (SAR Section 1.2.8.4.1.1). The applicant designed the site transporter with a wide base to prevent tipover (SAR Table 1.2.8-2). In addition, the two front and two rear lifting forks, which are synchronized to share the load, reduces the probability of both a drop from overloading and a tipover. The applicant stated that the passive cask restraint system provides stabilization during cask movement. The restraint system, which contacts the cask after the cask has been raised to the correct height, was illustrated in SAR Figure 1.2.8-49.

**NRC Staff's Evaluation of the Site Transporter Design and Design Analyses**

The NRC staff evaluated the applicant's information on the site transporter design and design analyses and determines that they are acceptable, for the following reasons:

The NRC staff evaluated the load combinations used by the applicant for the design of the site transporter and finds the load combinations acceptable because they are consistent with the ASME NOG–1–2004 (ASME, 2005aa) code, which is applicable to the GROA, as discussed in Table 7-1.

The NRC staff concludes that the applicant-specified speed limit of 4 km/hour [2.5 mph] is acceptable because it is slow enough to protect the site transporter, as shown by the applicant's analysis. This analysis demonstrated that impacts at 4 km/hour [2.5 mph] would not result in impact energy capable of breaching the cask.

The NRC staff finds that the design of the site transporter for protection against fuel tank explosion is acceptable because the applicant uses commercially available explosion-proof fuel tanks (DOE, 2009fa) that are ISO 9001 certified and designed for more severe environments than those expected at the GROA. Further, the site transporter is equipped with an onboard fire-suppression system. The TAD canister and aging overpack to be transported by the site transporter are to be designed to withstand a fully engulfing fire, as defined in 10 CFR 71.73(c)(4), for a burning period based on the amount of combustible hydrocarbons (e.g., diesel fuel) and the combustible and flammable materials on the transporter (SAR Section 1.4.3.1.2).

The NRC staff finds that the applicant's site transporter design for protection against spurious movement during lifting/lowering operations is acceptable because the applicant specified a PSC (PSC-2, SAR Table 1.9-10) that includes deactivation of power to the site transporter, the setting of the brakes, and operator's independent verification of electrical power deactivation.

The NRC staff finds that the applicant's site transporter design to reduce the severity of a drop on the aging overpack is acceptable because the applicant designed the site transporter's lifting system using the codes and standards that are consistent with the standard engineering practices for similar lifting operations, which the NRC staff finds acceptable for use at the GROA, as further described in Table 7-1. Also, the applicant designed the site transporter to limit the lift height of the Aging Overpack to 0.91 m [3 ft] (SAR Table 1.2.8-2), which is evaluated by the NRC staff in SER Section 2.1.1.4.3.3.1.1 and found to be acceptable. In addition, the applicant's equipment qualification program, as stated in SAR Section 1.13, would ensure that the site transport lifting devices have the ability to perform the intended safety function. The NRC staff evaluates this program in SER Section 2.1.1.2.3.6.2, and finds the applicant's description to be acceptable.

The NRC staff finds that the design for the site transporter the applicant proposed to provide protection against sliding impacts is acceptable because the design considers operating clearance and energy-absorbing features to ensure that the waste container would not be breached from sliding impacts and the track design provides significant resistance to sliding. Therefore, the NRC staff concludes that these measures acceptably minimize the potential for sliding impacts.

The NRC staff finds that the site transporter design the applicant proposed to provide protection against tipover is acceptable because the design incorporates a wide base, as one means of lowering the center of gravity, to prevent overturning. Further, the design includes a restraint system to provide for three-point stability. Therefore, the NRC staff concludes that these measures acceptably minimize the potential for tipover.

On the basis of the evaluation of the applicant's information on the site transporter design features described above, the NRC staff determines that the site transporter design and design analyses are adequate because the site transporter is protected against (i) runaway, (ii) fuel tank explosion, (iii) spurious movement during lifting/lowering operations, (iv) significant impacts in the case of an aging overpack drop, (v) sliding impacts, and (vi) tipover. Therefore, the NRC staff concludes that the site transporter has been designed adequately to meet the design bases and design criteria, and thus would be able to perform its intended safety function.

2.1.1.7.3.5.3 Cask Tractor and Cask Transfer Trailers

The applicant stated that the cask tractor is used in intrasite operations to pull cask transfer trailers carrying (i) a transportation cask containing a horizontal DPC from the RF to the aging pad and (ii) a horizontal shielded transfer cask containing a horizontal DPC from the aging pad to the WHF (SAR Section 1.2.8.4.2). According to the applicant, both types of cask transfer trailers are heavy industrial trailers with a support skid mounted on top. The skid consists of a self-contained hydraulic system, a hydraulic ram, an optical alignment system, and hydraulic jacks. The applicant stated that the skid is designed to raise, level, and stabilize the cask transfer trailer while transferring the DPC at the horizontal aging module. The applicant stated that the functions of the cask tractor and cask transfer trailer (CTCTT) are expected to perform at the GROA are similar to ones performed by similar vehicles utilized in other nuclear facilities, such as spent fuel storage sites.

The applicant described the tractor as a vehicle driven by a human operator. The applicant stated that the seat is equipped with sensors and interlocks that shuts off the engine when the driver leaves the seat. The cask tractor is a diesel-powered, four-wheel drive, four-wheel steering vehicle capable of carrying 378 L [100 gal] of fuel onboard.

The applicant described the design and operational processes for the CTCTT in SAR Section 1.2.8.4.2. The applicant also provided a mechanical equipment envelope in SAR Figure 1.2.8-50 and stated that the CTCTT is designed to withstand the natural phenomena loading parameters provided in SAR Table 1.2.2-1, as applicable.

**Design Criteria and Design Bases**

The applicant presented the nuclear safety design bases for the CTCTT and their relationship to the design criteria in SAR Table 1.2.8-2 (SAR Section 1.2.8.4.2.5). In this table, the applicant provided specific design criteria for each of the nuclear safety design bases.

The applicant defined a design criterion that limits the maximum speed of the CTCTT to protect against runaway. In addition, the applicant specified PSC–2 for the CTCTT (SAR Section 1.2.8.4.2.4 and Table 1.9-10). The PSC–2 stated that the CTCTT is to be deactivated and the brakes set during waste handling operations. The applicant also stated that the materials to be used in the onboard fuel tank design and limited fuel tank capacity would preclude fuel tank explosions. Furthermore, the applicant stated that it specified a criterion for a maximum trailer height to limit the cask drop to less than 1.8 m [6 ft] to reduce the severity of a potential drop. Finally, the applicant indicated that cask puncture is precluded by limiting the force from the hydraulic ram acting against the casks to a value below the minimum required to cause damage.

**NRC Staff's Evaluation of the CTCTT Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the design criteria and design bases for the CTCTT and finds that the applicant's design criteria and design bases for the CTCTT are acceptable because they addressed the relevant safety functions (SAR Table 1.2.8.2) for protection against the CTCTT's runaway, fuel tank explosions, drops, and cask punctures. Further, the design criteria and design bases were derived from the PCSA, as reviewed, and found to be acceptable by the NRC staff in SER Sections 2.1.1.3.3.1.3.5, 2.1.1.4.3.2.2, and 2.1.1.6.3.2.8.3.

**Design Methods**

The applicant stated that it based its cask tractor design method on industry-accepted standards for the design of the CTCTT (SAR Section 1.2.8.4.2.6). These standards provide guidance for the design of safety systems for (i) personnel and burden carriers [ANSI/ITSDF B56.8 (ITSDF, 2006aa)]; (ii) operator-controlled industrial tow tractors [ANSI/ITSDF B56.9 (Industrial Truck Standards Development Foundation, 2006ab)]; and (iii) the design, fabrication, and maintenance of semitrailers employed in the highway transport of weight-concentrated radioactive loads [ANSI N14.30–1992 (ANSI, 1992aa)].

**NRC Staff's Evaluation of the CTCTT Design Methods**

The NRC staff evaluated the applicant's information on the design methods and the applicable standards for the design of the CTCTT and determines that the standards [i.e., ANSI/ITSDF B56.8, ANSI/ITSDF B56.9 (ITSDF 2006aa,ab), and ANSI N14.30–1992 (ANSI, 1992aa) the applicant proposed to use are appropriate for use at the GROA, as further discussed in Table 7-1. In brief, these standards cover the safety aspects of the equipment necessary to meet the nuclear safety design criteria and are applicable to operator-controlled equipment. Therefore, the NRC staff finds that the application of these standards and the design methods for the CTCTT design are acceptable.

**Design and Design Analyses**

The applicant's load combinations for the design and design analyses of the CTCTT include loads from normal operating conditions, event sequences, and the effects of natural phenomena, including seismic events. The applicant described the CTCTT design features that address each of the CTCTT nuclear safety design bases and design criteria, as follows.

To protect the CTCTT against runaway, the applicant specified a safety design specification for the CTCTT to limit travel speed to 4 km/hour [2.5 mph] (SAR Section 1.2.8.4.2.1). The design has the tractor equipped with a dual-brake system and an alarm that notifies the operator of a system failure. The applicant's design also includes a braking system that is independent of the tractor and engages automatically when the trailer is disconnected. This braking system will be designed to hold the trailer on a 5 percent grade with a 2 percent cross-slope (SAR Section 1.2.8.4.2.1). In response to the NRC staff's RAI (DOE, 2009ez), the applicant also stated that ANSI/ITSDF B56.9–2006 Section 7.13 (ITSDF, 2006ab) will be used to design the physical speed controls of the cask tractor.

To protect the CTCTT against a fuel tank explosion, the applicant stated that the CTCTT would be designed for the safety function of precluding fuel tank explosion (SAR Section 1.2.8.4.2.1). The applicant stated that the tractor fuel tank would be designed in accordance with the ANSI/ITSDF B56.9–2006 (ITSDF, 2006ab). In its responses to NRC staff RAIs (DOE, 2009ez,fa), the applicant stated that the ANSI/ITSADF standard precludes fuel tanks explosions, that the designs are commercially available (SER Section 2.1.1.7.3.5.2), and the commercially available fuel tanks are designed for more severe environments than those expected at the GROA. The applicant also stated that the tanks include features, such as flame-resistant coatings, self-sealing polymeric foam, insulating foam, Kevlar®/Dyneema®/Twaron® protective wrap, and internal cell foam that can meet the explosion-proof standard [ANSI/ITSDF B56.9-2006, ITSDF 2006ab] for the CTCTT.

To reduce the severity of a drop, the applicant designed the cask tractor and cask transfer trailers (CTCTT) to preclude dropping a cask from a height greater than 1.8 m [6 ft], measured from the equipment base (SAR Section 1.2.8.4.2.1, Table 1.2.8-2).

To prevent cask puncture, the applicant stated that the CTCTT is designed to preclude transportation cask puncture resulting from a collision or impact of a hydraulic ram (SAR Section 1.2.8.4.2.1). The applicant stated that the transportation cask has a thick steel lid has an inner steel shell, a layer of dense gamma-shielding material, and a thick outer steel shell that together is more than 178 mm [7 in] thick. The applicant further stated that the inherent toughness of the casks provides the necessary puncture resistance for the postulated events at the GROA. In addition, the applicant would include a relief valve in the hydraulic ram design to prevent actuator overpressure (DOE, 2009ez).

**NRC Staff's Evaluation of the CTCTT Design and Design Analyses**

The NRC staff evaluated the applicant's information on the CTCTT design and design analyses for meeting each of the nuclear safety design bases and design criteria and finds the CTCTT design and design analyses acceptable, as follows.

The NRC staff reviewed and evaluated the load combinations used by the applicant for the design of CTCTT and compared these with the load combinations described in ASME NOG–1–2004. The NRC staff finds the applicant's information to be acceptable because the load combinations are consistent with the ASME NOG–1–2004 (ASME, 2005aa) code. This code is applicable and acceptable for use at the GROA, as discussed in SER Table 7-1.

The CTCTT design includes features, such as a dual-brake system and a speed limit to protect against a runaway. Additionally, the CTCTT is (i) equipped with braking systems that operate in tandem when connected and (ii) designed to automatically brake on both the tractor and trailer, should the CTCTT become uncoupled or exceed the speed limit of 4.0 km/hour [2.5 mph], as described in the applicant's response to the NRC staff's RAI (DOE, 2009ez). In addition, the standard (ITSDF, 2006ab) proposed by the applicant to design the speed controls is consistent with standard engineering practice for designing equipment used in nuclear power plants for motion and handling controls of tow tractors with a sit-down rider and powered by an internal combustion engine, and the NRC staff finds it acceptable for use at the GROA, as further discussed in Table 7-1.

The NRC staff determines that the applicant's design of the fuel tank of the CTCTT to preclude fuel tank explosion is acceptable because the design is consistent with the industry-accepted standard ANSI/ITSDF B56.9–2006 (ITSDF, 2006ab). The NRC staff determines that

ANSI/ITSDF B56.9–2006 (ITSDF, 2006ab) is applicable to fuel tank design for the cask tractor because this standard addresses the potential fuel tank explosion condition and provides that the tow tractor comply with UL 558 (Underwriters Laboratories, 1996aa). The NRC staff further finds that UL 558 (Underwriters Laboratories, 1996aa) is applicable because it is an industry standard that addresses fire safety aspects of diesel-fueled industrial tow tractors. The use of the fuel tanks that preclude explosion is also reviewed previously and found to be acceptable by the NRC staff in SER Section 2.1.1.7.3.5.2 for the Site Transporter.

The NRC staff finds that the CTCTT design the applicant proposes for reducing the severity of a drop is acceptable because it has been designed for the proposed maximum cask drop height of 1.8 m [6 ft], following ANSI/ITSDF B56.9-2006 (ITSDF, 2006ab). The applicability of this code is discussed further in Table 7-1. For comparison, the two-block drop height for the cask handling crane is 9.1 m [30 ft] (SER Section 2.1.1.4.3.2.1.2).

The NRC staff finds that the applicant's design consideration for the CTCTT to preclude cask puncture and limit damage to the DPCs inside the cask is acceptable because the applicant designed the transportation cask to provide adequate puncture resistance for design basis events at the GROA. Further, the applicant's design includes a pressure-relief valve in the hydraulic ram designed to prevent actuator overpressure.

Based on the evaluation of the applicant's information on the CTCTT design features described above, the NRC staff determines that the CTCTT design and design analyses are acceptable because the CTCTT is protected against (i) runaway, (ii) fuel tank explosion, (iii) cask drop, and (iv) cask puncture. Therefore, the NRC staff concludes that the CTCTT has been designed adequately to meet the design bases and design criteria, and thus is able to perform its intended safety function.

2.1.1.7.3.5.4 Site Prime Movers

The applicant stated that it would use three types of site prime movers in the GROA to move cask cars and trailers loaded with casks between the buffer area and the handling facilities. According to the applicant, these prime movers are rubber-tired tractors, steel-wheeled locomotives, or a hybrid prime mover that consists of both rubber and steel wheels. The applicant stated that the truck tractor would be used to pull trailers carrying loaded truck casks, whereas the steel-wheeled, rail-based switcher locomotive would move rail cask cars. The applicant indicated that these prime movers would be driven by operators and equipped with a 378-L [100-gal] diesel fuel tank, speed control features, and air-based braking systems with onboard air compressors.

The applicant provided design information in SAR Sections 1.2.8.4.3, including operational processes, PSCs, design criteria and design bases and their interrelationships, design methods, codes and standards, and design load combinations associated with the site prime mover. In response to an NRC RAI, the applicant provided additional design envelope information on the site prime movers (DOE, 2009ez). The applicant indicated that the dimensions of the legal-weight truck or overweight truck cask trailer are 2,591 mm [102 in] wide and 16.2 m [53 ft] long. The applicant specified that the rail carrier would be 3,251 mm [128 in] wide, and the railcar outside length is 27.4 m [90 ft]. According to the applicant, the railcar can accommodate the maximum combined naval transportation cask and rail carrier weight of $3.6 \times 10^5$ kg {395 tons [789,000 lb]} with the loaded naval transportation cask alone having a weight of $2.7 \times 10^5$ kg {295 tons [590,000 lb]}.

## Design Criteria and Design Bases

The applicant presented nuclear safety design bases for the prime mover and their relationships to the design criteria in SAR Table 1.2.8-2 (SAR Section 1.2.8.4.3.5). The applicant also identified two design criteria to address the safety design bases.

First, to protect against runaway, the applicant specified that the site prime mover would include equipment to limit the speed of the site prime mover to the design criteria (SAR Table 1.2.8-2) of 14.5 km/hour [9 mph] within the GROA and 4.4 km/hour [2.75 mph] while approaching the handling facilities (SAR Section 1.2.8.4.3.1). Secondly, to preclude fuel tank explosion, the applicant stated that it would use commercially available fuel tanks that provide explosion protection in environments more severe than those expected at Yucca Mountain, as described in its response to the NRC staff's RAI (DOE, 2009fa). In addition, for the rail-based site prime movers, the applicant stated that PSC-2 would limit spurious movement of the rail-based site prime mover during handling operations with loaded waste containers by specifying that the site prime mover would be detached before a loaded waste container is placed on or taken off the railcar (SAR Section 1.2.8.4.3.4). For the truck-based site prime movers, the applicant stated that the site prime mover operating procedure would specify that the site prime mover must be detached or deactivated with its brakes set prior to waste handling operations (SAR Section 1.2.8.4.3.4).

## NRC Staff's Evaluation of the Site Prime Movers Design Criteria and Design Bases

The NRC staff evaluated the applicant's information on the design criteria and design bases for the site prime mover and finds that it is acceptable because the speed of the prime mover is acceptably limited to 14.5 km/hour [9 mph] within the GROA and 4.4 km/hour [2.75 mph] while approaching the handling facilities, and the applicant's procedural safety control provides that the site prime mover be detached before moving a waste container on or off of a railcar. Further, the design criteria and design bases were derived from the applicant's hazard assessments and are consistent with the event sequence evaluation and identification of ITS SSCs, which are reviewed and found to be acceptable by the NRC in SER Sections 2.1.1.3.3.1.3.5, 2.1.1.4.3.2.2, and 2.1.1.6.3.2.8.3.

## Design Methods

The applicant stated that the design method for the site prime mover followed the American Association of State Highway and Transportation Officials (2004aa), and American Railway Engineering and Maintenance-of-Way Association (2007aa) (SAR Section 1.2.8.4.3.6). These design methods were used to ensure that the site prime movers remained visible during operations, had adequate braking, and proper fuel system integrity.

The applicant also stated that the design method applied the sections of the 49 CFR Part 571, Federal Motor Vehicle Safety Standards, that are related to the lamps, reflective devices, and associated equipment used on the vehicle, to ensure proper motion signaling. The applicant also followed regulatory requirements of 49 CFR 571.106 to design brake hoses and 49 CFR 571.301 to design fuel system integrity (DOE, 2009ez).

## NRC Staff's Evaluation of the Site Prime Movers Design Methods

The NRC staff evaluated the applicant's information on the design method for the site prime movers and finds that it is adequate because the applicant's methods for the design of the site prime movers are based on engineering design codes used to construct roadways and rail lines, including for short-haul transportation of heavy equipment and loads. These standard codes include those from the American Association of State Highway and Transportation Officials (2004aa) and American Railway Engineering and Maintenance-of-Way Association (2007aa). Therefore, the NRC staff finds each of these codes acceptable for use at the GROA because they provide sufficient guidance for carrying these loads safely, as further discussed in Table 7-1.

**Design and Design Analyses**

The applicant's load combinations for the site prime mover include loads from normal operating conditions, event sequences, and the effects of natural phenomena, including seismic events. The applicant described design features for each of the nuclear safety design bases and design criteria for the site prime movers, which are summarized as follows.

To protect the site prime movers against runaway, the applicant specified a design criterion for reducing runaway probability by limiting the maximum vehicle speed to 14.5 km/hour [9 mph] when traveling in the GROA and 4.4 km/hour [2.75 mph] when approaching the handling facilities (SAR Section 1.2.8.4.3.1). According to the applicant, the speed would be controlled by a governor on the engine and a transmission constraint that regulates speed. The applicant indicated that the site prime movers and the cask conveyances are also equipped with braking systems that would operate in tandem when these systems are connected. The applicant stated that these braking systems would be designed such that the brakes would be automatically applied when the 14.5-km/hour [9-mph] design limit is exceeded. For rail-based cars, the applicant stated that it used Association of American Railroads (2004aa, Section M) as guidance to design braking and speed control features of the rail-based site prime movers (DOE, 2009ez).

To protect the site prime movers against fuel tank explosion, the applicant stated that it designed the site prime movers with safety functions that preclude fuel tank explosion (SAR Section 1.2.8.4.3.1). The applicant indicated that it designed the site prime movers with limited fuel capacity {tank size of 378 L [100 gal] of diesel fuel} and protection against fire and explosions (SAR Section 1.2.8.4.3.1; DOE, 2009fa). The applicant indicated that such explosion-proof fuel tanks are commercially available and would be used in the site prime movers. The applicant stated that the tanks include features such as flame-resistant coatings, self-sealing polymeric foam, insulating foam, Kevlar®/Dyneema®/Twaron® protective wrap, and internal cell foam that can meet the explosion-proof standard for the site prime movers.

**NRC Staff's Evaluation of the Site Prime Movers Design and Design Analyses**

The NRC staff evaluated the applicant's information on the design features of the site prime movers that address the nuclear safety design bases and design criteria and finds them acceptable for the following reasons:

The NRC staff evaluated the load combinations used by the applicant for the design of site prime movers and finds the load combinations acceptable because they are consistent with the ASME NOG–1–2004 (ASME, 2005aa) code, which is applicable to the GROA, as discussed in SER Table 7-1.

The speed control approach the applicant described for site prime mover runaway prevention is acceptable because the use of engine governors, transmission constraints, and speed-triggered braking systems are standard engineering practice to maintain speed within safe levels. The NRC staff considers the function of the site prime movers at the GROA similar to analogous safe operations of cask transports at other nuclear facilities, and the design of the site prime movers, as described by the applicant, are sufficient and adequate for use at the GROA. Further, use of the specific standard industry codes to design these features is acceptable to the NRC staff, as described in Table 7-1.

The NRC staff finds that the site prime mover design and design analyses against fuel tank explosions is acceptable because the applicant will use commercially available equipment that precludes fuel tank explosions.

In summary, based on the evaluation of the applicant's information on the site prime mover design features described above, the NRC staff determines that the site prime mover design and design analyses are adequate because the site prime mover is protected against runaway and fuel tank explosion. Therefore, the NRC staff concludes that the site prime mover has been designed adequately to meet the design bases and design criteria, and thus is able to perform its intended safety function.

**NRC Staff's Conclusion of the Transportation Systems Design**

On the basis of the NRC staff's evaluation described above, of the applicant's design of the following transportation systems (i) Transport and Emplacement Vehicle; (ii) Site Transporter; (iii) Cask Tractor and Cask Transfer Trailers; and (iv) Site Prime Movers, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), 63.112(e)(9), and 63.112(f) are satisfied. The NRC staff finds that the description of the transportation systems designs adequately (i) defines the relationship between design criteria and performance objectives; (ii) identifies the relationship between the design bases and the design criteria; and (iii) provides adequate information on dimensions, proposed codes and standards, and analytical and design methods.

#### 2.1.1.7.3.6 Electrical Power Systems

The applicant provided design information in SAR Sections 1.4.1.2 and 1.4.1.3 for the proposed ITS electrical power systems to be used at the GROA. The applicant stated that the ITS electrical power system would receive electric power from the non-ITS normal electrical power system and, in turn, would provide power to ITS SSCs that require electrical power to perform a safety function. According to the applicant, if the non-ITS normal electrical power system becomes unavailable, the ITS electrical power system would receive electric power from the ITS diesel generators.

The applicant stated that the ITS electrical power system consists of three subsystems (i) the ITS Alternating Current (AC) subsystem, (ii) the ITS Direct Current (DC) subsystem, and (iii) the ITS Uninterruptible Power Supply (UPS) subsystem. The applicant further stated that the ITS AC subsystem would provide power to ITS equipment and some non-ITS SSCs; the ITS DC subsystem would provide power for the control actions of the ITS electrical power system switchgear; and the ITS UPS subsystem would provide power to the ITS instruments and controls that must be continuously powered to perform their safety functions.

The applicant described the nuclear safety design bases and design criteria for the ITS electrical power system in SAR Tables 1.9-3 (CRCF) and 1.9-4 (WHF). In addition, the applicant presented the ITS electrical power system design bases and design criteria in SAR

Table 1.4.1-1 and described them in SAR Sections 1.4.1.2.5 and 1.4.1.3.5. The applicant discussed the design criteria of redundancy and reliability attributes of the ITS electrical power system in SAR Sections 1.9.1.11 and 1.9.1.12.

This section contains the NRC staff's evaluation of the proposed ITS electrical power system design. The evaluation determines whether the description of the proposed ITS electrical power system design for surface operations adequately describes the applicant's proposed design criteria for the ITS electrical power system and the relationship between the design criteria and the design bases.

**Design Criteria and Design Bases**

The applicant listed the nuclear safety design bases and their relationships with design criteria for the ITS electrical power system in SAR Tables 1.9-3 and 1.9-4 and presented the information for the ITS electrical power system in SAR Table 1.4.1-1 (SAR Section 1.4.1.2.5). Based on the PCSA, the applicant provided several design criteria for the safety design bases that (i) provide electrical power to ITS nuclear confinement HVAC systems in the CRCF and WHF and (ii) support ITS electrical function in the EDGF. The criteria, which are based on the IEEE standards [IEEE 535–1986, IEEE 741–1997, IEEE 384–1992, IEEE 603–1998, IEEE 308–2001, IEEE 450–2002, IEEE 484–2002, IEEE 336–2005, IEEE 572–2006, and IEEE 650–2006 (IEEE, 2006aa–ac; 2003aa,ab; 2001aa; 1998aa; 1997aa; 1986aa)] for electrical power systems for nuclear facilities, apply when offsite commercial electrical power is available and during loss of offsite power (LOSP) events.

Regarding the availability of ITS electrical power during a loss of offsite power (LOSP) event, the applicant provided the following design criteria (SAR Table 1.4.1-1): (i) two independent ITS diesel generators are included in the ITS electrical power system design, (ii) support systems for each ITS diesel generator are electrically and physically independent of the support systems for the other ITS diesel generator, and (iii) fuel oil storage for each ITS diesel generator is sized for 14 days of continuous operation and would be capable of being refilled while the ITS diesel generators are operating.

To assure continuous availability of the ITS electrical power distribution system, the applicant included design criteria such that (i) the ITS electrical distribution equipment and associated raceways are electrically independent and physically separated; (ii) upon occurrence of an LOSP, the ITS diesel generator switchgear would be isolated from the non-ITS normal electrical power system; and (iii) ITS loads would be automatically sequenced onto the ITS diesel generators; the loading sequence used for ITS buses is EDGF, CRCF1, CRCF2, CRCF3, and WHF (SAR Section 1.4.1.2.1).

The applicant included design criteria for ITS battery-powered DC electrical power SSCs within the EDGF. The applicant stated that the ITS DC electrical power system would maintain continuous function of ITS switchgear and ITS diesel generator startup and operation during the time interval between an LOSP and availability of power provided by the ITS diesel generators.

According to the applicant, certain ITS SSCs would need continuous AC power to support or perform a safety function when AC power becomes unavailable. The applicant stated that the ITS UPS electrical power system design, including ITS UPS SSCs, is subject to design criteria as identified in SAR Table 1.4.1-1. The applicant further stated that battery-operated UPS SSCs would be needed within the CRCF, WHF, and EDGF to provide continuous AC power for these SSCs by using inverters to convert battery power to AC power.

According to the applicant, controlled environmental conditions would be maintained in ITS electrical equipment and battery rooms for continuous operation during normal conditions and during an LOSP with ITS diesel generator power. The applicant stated that battery rooms in the EDGF, CRCF, and WHF would be temperature-controlled and ventilated. The applicant also stated that the HVAC systems in the EDGF, CRCF, and WHF would include an ITS HVAC subsystem that would provide an independent HVAC train capable of maintaining environmental conditions during normal and LOSP conditions with ITS diesel generator power for each of the rooms associated with the two ITS electrical trains. In response to NRC staff RAIs (DOE, 2009fb,fc) on design bases and design criteria and bounding limits for this ITS electrical power system performance, the applicant indicated that the reliability for the ITS electrical power system SSCs supplying power to ITS HVAC systems in the EDGF would be the same as the reliability of the CRCF and WHF electrical power systems. Reliable electrical power for ITS HVAC for ITS electrical equipment in the EDGF, CRCF, and WHF is needed to provide reliable ITS power to the ITS power distribution system.

**NRC Staff's Evaluation of the ITS Electrical Power Systems Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information in SAR Sections 1.4.1.2, 1.4.1.3, and SAR Table 1.4.1-1, along with the applicable codes and standards, on the design criteria and design bases of the ITS electrical power systems and finds that the design bases and design criteria the applicant provided for the ITS electrical power system are adequate because (i) the design criteria are consistent with IEEE standards [IEEE 535–1986, IEEE 741–1997, IEEE 384–1992, IEEE 603–1998, IEEE 308–2001, IEEE 450–2002, IEEE 484–2002, IEEE 336–2005, IEEE 572–2006, and IEEE 650–2006 (IEEE, 2006aa–ac; 2003aa,ab; 2001aa; 1998aa; 1997aa; 1986aa)] for electric power systems for nuclear facilities; (ii) the design criteria address both when offsite commercial electrical power is available and when loss of offsite power (LOSP) events occur; (iii) two independent ITS diesel generators with sufficient fuel supply support the ITS electrical power system; (iv) ITS electrical loads are automatically sequenced onto the ITS diesel generators upon loss of normal power; (v) DC electrical power and battery-operated UPS SSCs within the EDGF were identified as ITS; (vi) battery-operated UPS provide continuous power to provide a contingency power supply, if needed; (vii) the ITS electrical equipment and battery rooms are temperature controlled and ventilated by two independent ITS HVAC trains; and (viii) the design bases for the ITS electrical power system supporting the operation of ITS SSCs were appropriately identified and derived from the PCSA, as evaluated in SER Sections 2.1.1.3.3.1.3.5 and 2.1.1.4.3.3.2.1. Further information concerning the NRC staff's evaluation of the applicant's proposed use of the codes, standards, and regulatory guidance cited in the SAR and RAI responses may be found in Table 7 -1.

**Design Methods**

The applicant stated that the design method of the ITS electrical power system is consistent with industry codes and standards, as described in SAR Section 1.4.1.2.8. These industry codes and standards include IEEE 535–1986, IEEE 741–1997, IEEE 384–1992, IEEE 603–1998, IEEE 308–2001, IEEE 450–2002, IEEE 484–2002, IEEE 336–2005, IEEE 572–2006, and IEEE 650–2006 (IEEE, 2006aa–ac; 2003aa,ab; 2001aa; 1998aa; 1997aa; 1986aa). Additionally, in response to the NRC staff's RAI (DOE, 2009fc), the applicant stated that (i) the Project Design Criteria Document (BSC, 2007av) provides that the ITS electrical power system be designed in accordance with IEEE 379-2000 (IEEE, 2001ab), which provides methods for application of the single-failure proof criterion and (ii) this standard has been applied to the design of the ITS electric power system. The applicant also stated that the design method for the ITS AC and DC electrical power systems are consistent with Regulatory Guide 1.41 (NRC, 1973ad). Furthermore, in response to the NRC staff's RAI (DOE, 2009fc), the applicant stated that Regulatory Guide 1.9 (NRC, 1993ab) is incorporated in the design method for the ITS AC and DC electric power systems.

The applicant further stated that the design method of the ITS diesel generators is consistent with additional industry codes and standards, including the National Fire Protection Association NFPA 70 and 110 (NFPA, 2005ab,ac) and IEEE 387–1995 and 446–1995 (IEEE, 1996aa,ab). In addition to the codes and standards listed previously, the applicant stated that the design method of the ITS 125-V DC supply is consistent with IEEE 485–1997 and 946–2004 (IEEE, 2005ab IEEE, 1997ab). The applicant also stated that the design method of the ITS UPS SSCs would be consistent with ANSI/IEEE 944–1986 and IEEE 1184–1994 (IEEE, 1995aa; IEEE, 1986ab).

The applicant stated that the design method (SAR Section 1.4.1.2.6) establishes ampacity ratings for ITS electrical power system cable and conductor SSCs is in accordance with IEEE 835-1994 (IEEE, 1994aa) and NEMA WC 51-2003 (National Manufacturers Association, 2003aa). The applicant further stated that the design method includes the sizing of the ITS electrical power system distribution cabling based on 125 percent of the full anticipated load current at a 100 percent load factor.

In response to the NRC staff's RAI, the applicant stated that the ITS power distribution subsystem would include ITS current carrying components, such as cables and electrical conductors (DOE, 2009gj). The applicant classified noncurrent carrying components of the electrical distribution system such as conduits, raceways, enclosures, fittings, and cable trays that either support or provide physical protection for ITS and non-ITS electric power and signal distribution cables as non-ITS. The applicant stated that these components would be designed and installed using the methods and practices of NFPA 70 (NFPA, 2005ab). The applicant further stated that, for ITS electric conductors within the electrical power distribution system, the design method incorporates redundant ITS electrical power conductors routed in separate conduits, cable trays, or ducts associated with each ITS electrical power train. The applicant stated that, for ITS conductors, the design method of raceways, conduits, cable trays, or ducts and associated support and physical protection components would provide physical separation and protection from adverse interactions between redundant ITS circuits and between ITS and non-ITS circuits, and would apply the single-failure criterion (SAR section 1.4.1.2.6), consistent with IEEE 384-1992 and IEEE 379-2000 (IEEE, 1998aa; IEEE, 2001ab). The applicant also described its equipment qualification program for all ITS electrical power system components, including wiring and cabling, in SAR Section 1.13. The applicant also described its equipment qualification program, based on NRC Regulatory Guides 1.100 and 1.89 (NRC, 1988aa; NRC 1984aa), which endorse IEEE 344-2004 (IEEE, 2005aa) and IEEE 323-2003 (IEEE, 2004aa), respectively, with clarifications, for all ITS electrical power system components, including wiring and cabling, in SAR Section 1.13.

The applicant stated that its design method incorporated selected design criteria from the cited codes and standards. The applicant applied the selected criteria contained in the cited codes and standards and the results of the PCSA to design the ITS electrical power systems. In response to the NRC staff's RAI (DOE, 2009dl), the applicant described the specific criteria in the identified industry codes and standards to be used, the safety function of the ITS electrical power systems, the applicability of the codes and standards to each ITS subsystem, and the rationale for any exceptions or alternate methods that may be taken. The applicant stated that the design method for the on-site ITS AC and DC electrical power systems would be in accordance with each of the codes and standards described previously, as identified in SAR Sections 1.4.1.2.6 and 1.4.1.2.8, confirming that the design would fully conform to the provisions of these codes and standards, as described in SAR Section 1.2.2.4, "Regulatory Guidance and Other Design Codes and Standards." The applicant also stated (SAR Section 1.4.1.2.8) that it will design the ITS AC and DC electrical power system in accordance with Regulatory Guide 1.41 (NRC, 1973ad) and Regulatory Guide 1.9 (NRC, 1993ab); and in accordance with IEEE 379-2000 (IEEE, 2001ab) (DOE, 2009fc). Additionally, in response to the NRC staff's RAI (DOE 2009fd, Enclosure 3), the applicant stated that any non-ITS components whose malfunction could cause an ITS SSC to fail to perform its safety function, the applicant's design method classifies those SSCs as ITS. Further, any design features that may be implemented to prevent the unacceptable interaction between the ITS and non-ITS components are also classified as ITS.

**NRC Staff's Evaluation of the ITS Electrical Power Systems Design Methods**

The NRC staff evaluated the applicant's information in SAR Sections 1.4.1.2 and 1.4.1.3 on the design method for the ITS electrical power systems to assess (i) whether the design criteria are consistent with the PCSA and (ii) the codes and standards to be used in the design and construction of facility ITS electrical power systems have been identified. The NRC staff finds that the applicant's design methodology for the electrical power systems is adequate because the applicant (i) adequately described proposed design methods and their relationships to design criteria for the ITS electrical power system; (ii) provided detailed information relative to the IEEE codes and standards [IEEE 308-2001, IEEE 379-2000, IEEE 384-1992, IEEE 603-1998 (IEEE, 2001aa,ab; IEEE, 1998ab; IEEE, 1992aa)] that the applicant proposes to apply to the design and construction of the ITS electrical power system at the GROA using the design criteria and design bases developed from PCSA; (iii) uses the IEEE, NEMA, and NFPA codes and standards and NRC regulatory guidance [IEEE 535-1986, IEEE 741-1997, IEEE 384-1992, IEEE 603-1998, IEEE 308-2001, IEEE 323-2003, IEEE 344-2004, IEEE 450-2002, IEEE 484-2002, IEEE 336-2005, IEEE 572-2006, IEEE 650-2006, IEEE 387-1995, IEEE 446-1995, IEEE 485-1997, IEEE 946-2004, IEEE 944-1986, IEEE 1184-1994, IEEE 835-1994, NEMA WC 51-2003, NFPA 70, NFPA 110, NRC Regulatory Guide 1.89, and NRC Regulatory Guide 1.100 (IEEE, 2006aa-ac; IEEE, 2005aa,ab; IEEE, 2004aa; IEEE, 2003aa,ab; IEEE, 2001aa; IEEE, 1998ab; IEEE, 1997aa,ab; IEEE, 1996aa,ab; IEEE, 1995aa; IEEE, 1994aa; IEEE, 1986aa,ab; NEMA, 2003aa; NFPA, 2005ab,ac; NRC, 1988aa; NRC, 1984aa)] that are consistent with the engineering practice for the design of electrical power systems of nuclear facilities, and whose use at the GROA, as proposed by the applicant, the NRC staff finds acceptable; and (iv) proposed to follow NRC Regulatory Guides 1.41 and 1.9 (NRC, 1973ad, 1993ab) for preoperational testing of electrical power systems and testing of emergency diesel generators, respectively. Further information concerning the NRC staff's evaluation of the applicant's proposed use of the codes, standards, and regulatory guidance cited in the SAR and RAI responses may be found in Table 7-1.

**Design and Design Analyses**

The applicant described the power supply specifications and proposed design criteria for power supply feeder sizing and margin provisions to be used in the design of the system distributing ITS power to ITS electrical power system loads in the EDGF, CRCF, and WHF (SAR Section 1.4.1.2.1). According to the applicant, the features of the proposed ITS electrical power system design include independent, redundant, separate trains of ITS electrical power that would provide power to designated independent, redundant, separate trains of ITS loads, such as ITS HVAC systems in the fuel handling facilities. The applicant also provided information for the ITS diesel generators and their associated ITS mechanical support and 13.8-kV distribution systems to describe provisions for physical protection, cooling, separation, and redundancy criteria for both ITS electrical power system Trains A and B within and between the EDGF and the CRCF and WHF.

The applicant stated that the redundant ITS electrical power system design includes an independent Train A and B configuration for the combined ITS electrical power system and ITS HVAC systems within the proposed GROA. According to the applicant, the Train A ITS electrical power system would power only Train A ITS HVAC SSCs in all CRCFs, WHF, EDGF, and for the (non-ITS) RF. An identical relationship would exist between the Train B ITS electrical power system and Train B ITS HVAC SSCs in the same facilities.

The applicant described that the ITS electrical power system design configuration would include redundant power sources (commercial power or diesel generators) and distribution systems that would provide power through multiple dedicated electrical connections and physical power flow paths to respective ITS SSCs (electrical power systems and HVAC) in the CRCFs, WHF, EDGF and non-ITS SSCs (electrical power system and HVAC) in the RF. The applicant described that, during normal operations, Train A and Train B ITS electrical power systems and related HVAC in each facility are designed to run continuously. The applicant also described that redundant confinement HVAC systems in the CRCFs, WHF, and RF would be independently capable of selecting operation on either Train A or Train B, as needed to maintain performance. Accordingly, the ITS electrical power system redundant Trains A and B would both be providing power to designated loads simultaneously. The applicant further described that the amount of ITS electrical power that is provided by each power train at any time would depend on which confinement HVAC Train (A or B) is operating in each CRCF, the WHF, and the RF, and the demand for power in each facility.

The applicant stated that the ITS electric power system would be normally powered by the GROA normal electrical power system. To provide reliable and timely backup ITS power when needed, the applicant described that a series of automatic operations would be initiated upon occurrence of an under/degraded voltage condition at the ITS 13.8 kV switchgear train (A or B). According to the applicant, the automatic operations would be designed to (i) disconnect (isolate) the normal electrical system and the ITS power switchgear; (ii) disconnect output loads; (iii) start the ITS diesel generator; and (iv) sequentially connect ITS power to designated loads. In response to the NRC staff's RAIs (DOE, 2009fb,fc), the applicant stated that the application of IEEE 308-2001 and IEEE 741-1997 (IEEE, 2001aa; IEEE, 1997aa) standards to ITS electrical equipment at the GROA is equivalent to applying these standards to Class 1E equipment for the design of the ITS electric power subsystem in a nuclear power plant and that the design of the ITS electrical power supply, including cable raceway features that support ITS cables (SAR section 1.4.1.2.6), will be in accordance with IEEE 379-2000 (IEEE, 2001ab). In the event of an LOSP, the applicant described that the automatic operations would be applied to both ITS electric power trains.

**NRC Staff's Evaluation of the ITS Electrical Power Systems Design and Design Analyses**

The NRC staff evaluated the applicant's information in SAR Sections 1.4.1.2 and 1.4.1.3 on the design and design analyses of the ITS electrical power systems and finds that the design and design analyses the applicant proposes to perform for the ITS electrical power systems are acceptable for several reasons. First, the proposed systems include redundancy of ITS electrical power that will enhance reliability of the ITS electrical power supply to the ITS SSCs to perform their intended safety functions. Second, the applicant adequately described the major functional and architectural attributes of the ITS electrical power system and major ITS electrical power system SSCs. Third, the applicant provided detailed information regarding the IEEE codes and standards [IEEE 308-2001, IEEE 379-2000, and IEEE 741-1997 (IEEE, 2001aa,ab; IEEE, 1997aa)] that it proposes to apply to the design and construction of the ITS electrical system at the GROA; the NRC staff finds the use of the IEEE 308-2001, 379-2000, and IEEE 741-1997 (IEEE, 2001aa,ab; IEEE, 1997aa) standards acceptable because the application of these standards, which are for Class 1E electrical systems at nuclear power plants, to ITS electrical system design at the GROA, is conservative, as the analogous electrical systems at nuclear power plants are more risk-significant than those for which these design codes will be used at the GROA. Further information about the scope and applicability of codes and standards used in the NRC staff's evaluation in this SER section can be found in Table 7.1.

**NRC Staff's Conclusion on ITS Electrical Power Systems Design**

On the basis of the NRC staff's evaluation of the applicant's information on design of the ITS electrical power systems described above, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the description of the ITS electrical power systems designs adequately (i) provides information on materials of construction, dimensions, proposed codes and standards, and analytical and design methods; (ii) defines the relationship between design criteria and the performance objectives; and (iii) identifies the relationship between the design bases and the design criteria.

#### 2.1.1.7.3.7 Instrumentation and Controls

The applicant provided information on ITS I&C equipment in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, 1.4.2, and 1.9.1 (DOE, 2008ab) to describe the safety functions of ITS I&C equipment for proper operations of the repository processes and also discussed how the ITS I&C equipment enables facility operators to continuously monitor the status of all packaging and emplacement functions. The applicant described that the ITS I&C equipment is designed to sense conditions indicative of the onset of event sequences and to initiate actions to prevent or mitigate those event sequences. The NRC staff performed a review of the applicant's description of the design bases, design criteria, proposed design methods, and design analyses of the ITS I&C equipment and circuitry used to initiate preventive or mitigative safety actions.

**Design Criteria and Design Bases**

In SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, 1.4.2, and 1.9.1 (DOE, 2008ab), the applicant described its general control philosophy for the GROA, stating that repetitive operations would utilize automation to support the facility operators. To facilitate this, the applicant proposed using various non-ITS local digital control systems. The applicant stated that facility operators stationed in the Central Control Center (CCC) Facility and in the operations rooms of the various surface facilities would operate these systems using human–machine interface (HMI) consoles. The applicant further stated that the local digital control systems can be monitored and controlled through a GROA-wide Digital Control and Management Information System (DCMIS) to convey the normal (non-ITS) control and monitoring commands and signals between the local sensors and controllers to the HMI consoles located in each surface facility and the CCC. According to the applicant, if the CCC becomes uninhabitable, the surface facilities can continue operations using the local HMI consoles. The applicant stated that the control system is designed such that active operator control can occur from only one location at a time; controls sequentially closer to the equipment being operated would take priority. Operators in the facility operations room or in the CCC can stop an activity that is locally controlled. After this action, CCC can reset the stop command to allow the facility operations to resume with local commands.

The applicant stated that human actions and digital controllers would be used for operational purposes, but are not relied on to reduce the frequency or mitigate the consequences of Category 1 or Category 2 event sequences. The applicant stated that ITS functions will be implemented using mechanical, electromechanical, or electrical devices with known reliability. In addition, facility operators using the Digital Control and Management Information System cannot override the automatic performance of safety functions by the ITS controls or the actions of a local operator.

The applicant described the intended normal operations, safety functions, and applicable design criteria associated with these ITS controls within the descriptions of the various electro-mechanical SSCs. In SAR Section 1.4.2, the applicant indicated that all ITS controls consist of individual hardwired devices, instead of being driven by software or programmable devices. The applicant stated that the use of programmable components are limited to normal operating functions, and the hardwired ITS controls are integrated into the design of the ITS SSCs in a way that prevents the ability of other normal use or non-ITS controls from overriding any of the ITS control functions. According to the applicant, the ITS safety functions may be performed either by a single component, or a group of components working in tandem. These components include limit switches, load sensors, and interlocks for adjustable speed drive (ASD) controllers, among other devices. The applicant further stated that, to facilitate maintenance and surveillance activities, or to facilitate recovery from a spurious actuation of an ITS control function, key-locked switch bypasses are used under administrative controls to override an ITS control function. According to the applicant, when programmable logic controllers are used, their use is constrained by the operation of the hard-wired ITS controls associated with the system under control.

The applicant stated that the ITS I&C equipment operates as part of the ITS SSCs to accomplish ITS functions. These SSC functions were summarized in tables within SAR Section 1.9.1. For example, the applicant indicated that ITS I&C equipment that is a part of a crane or hoist system may work together with other SSCs to prevent lifting a canister or TAD higher than allowed safety limits. The applicant also described other categories of ITS interlocks that provide interlock functions to ensure that the interactions between SSCs do not result in conditions adverse to safety. For example, at locations where potential radiological hazards could exist to facility personnel from inadvertent exposure to radioactive canisters and waste packages, the applicant designed equipment and personnel access shield doors with safety interlocks to prevent inadvertent opening when radioactive materials are present. The applicant may also choose to establish additional administrative procedural controls to further limit potential personnel radiation exposure.

The applicant described the ITS I&C equipment functions needed to ensure the SCCs achieve the safety functions and described the design bases and design criteria for these ITS I&C systems within SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.3.3, 1.4.2 and 1.2.8. In these SAR sections, the applicant described the design concepts and the intended normal operations and safety functions for the proposed ITS SSCs (i.e., ITS mechanical handling systems; ITS HVAC systems; ITS emergency diesel generators and their support systems; and the GROA communications and monitoring systems). According to the applicant, these systems function within the IHF, CRCF, WHF, RF, and EDGF, and on transport systems that travel among these facilities or between the surface facilities and the subsurface emplacement facilities. For the

ITS HVAC and ITS emergency diesel generators and support systems, the applicant provided criteria for redundancy, independence, and single-failure designs, based on the IEEE standards [IEEE 308-2001, IEEE 379-2000, IEEE 384-1992, and IEEE 603-1998 (IEEE, 2001aa,ab; IEEE, 1992; IEEE, 1998ab)].

The applicant stated that it presented the nuclear safety design bases for the ITS I&C equipment and their relationship with the design criteria in SAR Tables 1.9-2 through 1.9-7. SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, 1.4.2, and 1.9.1 (DOE, 2008ab) describe specific design criteria that would be applied in the design in order to meet each of the nuclear safety design bases, along with controlling parameters and bounding values. SAR Table 1.4.2-1 summarized the safety functions of ITS SSCs that are implemented through the use of 29 key ITS controls. SAR Table 1.9-1 presented the results of the preclosure safety classification of SSCs as ITS and non-ITS within the GROA. SAR Tables 1.9-2 to 1.9-7 identified the nuclear safety design bases, comprising safety functions and controlling parameters, and values of the nuclear safety design bases, and design criteria for the ITS SSCs. The applicant also described various design considerations and design criteria in SAR Sections 1.9.1.1 through 1.9.1.13.

To design the ITS controls, a set of design criteria were selected by the applicant to ensure the ability of SSCs to perform their intended safety functions. According to the applicant, the safety functions include measures to (i) protect personnel from inadvertent direct exposure to radiation, (ii) support initiation or generation of emergency ITS electrical power supply, (iii) support operation of components required during a loss of electric power, and (iv) protect against the drop of a canister containing radioactive materials.

**NRC Staff's Evaluation of the ITS Instrumentation and Controls Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, and 1.4.2 regarding the design criteria and design bases of the ITS I&C equipment. To support the evaluation of the design bases and the relationship between the design bases and design criteria for the ITS I&C equipment, the NRC staff examined the design and the intended operations of CRCF to confirm that the prevention or mitigation of potential event sequences is consistent with the relationship of the design bases and design criteria provided in the SAR. Since the applicant's description of the event sequences associated with operations of the CRCF addressed preclosure activity related events that are typical of the activities in surface facilities, the NRC staff reviewed several CRCF event sequences to evaluate consistency of the design bases and design criteria of the ITS I&C equipment interactions with material handling equipment, shield door, and cask port slide gate operations. The NRC staff also examined event sequences in conjunction with the review of ITS controls needed to support continued ITS electrical power system operation. From the review of the event sequences, the NRC staff determines that the design bases and design criteria are consistent with the safety function of the HVAC confinement and electrical equipment cooling and ventilation capability for the CRCF following a radionuclide release because the design of the HVAC and electrical equipment cooling trains included redundant trains of equipment designed to meet the independence and single-failure proof criteria of IEEE–308-2001, 379-2000, 384-1992, and 603-1998 (IEEE, 2001aa,ab; IEEE, 1992aa; IEEE, 1998ab) standards for safety controls.

The NRC staff finds that the design bases and design criteria for the ITS I&C supporting the operation of ITS SSCs are acceptable because (i) the ITS safety functions are implemented through the use of mechanical, electromechanical, or electrical devices with known reliability; (ii) the use of interlocks to ensure personnel exposure safety is a conservative design practice; (iii) the design bases (including safety functions and controlling parameters and values ) and their relationships with the design criteria are adequately defined, including the proposed ITS controls and circuitry that are depicted in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, and 1.4.2, (iv) the applicant would comply with the proposed condition of construction authorization in SER Section 2.1.1.6.3.2.8.2.1 regarding potential exceptions to the use of these codes; and (v) the design criteria have been selected to ensure the ability of SSCs to perform their intended safety functions based on the analyses used to identify ITS SSCs, as reviewed and found acceptable by the NRC staff in SER Section 2.1.1.6.3.2.8.2.1.

**Design Methods**

The applicant stated that the design method of the ITS I&C components that support operations of ITS SSCs will be based on the PCSA controlling parameters and applicable industry codes and standards to ensure that the ITS controls are available to perform their safety actions, when needed, to prevent occurrence of a Category 1 or a Category 2 event sequence, or mitigate consequences of a Category 2 event sequence. In SAR Sections 1.2.3 through 1.2.6, 1.4.2, and 1.13.2, the applicant identified industry design codes and standards that it proposed to use for ITS control design. The design method incorporates these codes/standards. They include rules for Type I cranes in ASME NOG–1–2004 (overhead crane controls) (ASME, 2005aa); and standard design criteria within IEEE 344-2004 (seismic qualification), IEEE 323-2003 (environmental qualification), and IEEE 603-1998 (criteria for safety systems—applicable to ITS safety interlocks, ITS electrical power system controls, and to ITS HVAC controls) (IEEE, 2005aa; IEEE, 2004aa; IEEE, 1998ab); guidance within Regulatory Guide 1.100 (seismic qualification) (NRC, 1988a) and Regulatory Guide 1.89 (environmental qualification) (NRC,1984aa) (which endorse the use of IEEE Standards IEEE-344 and IEEE 323, respectively); and applicable portions of NFPA 70 (National Electrical Code) (NFPA, 2005ab).

In responses to the NRC staff's RAIs (DOE, 2009dl,do) related to ITS controls, the applicant stated that it may take exceptions or alternatives to portions of the IEEE–308-2001, 379-2000, 384-1992, and 603-1998 (IEEE, 2001aa,ab; IEEE, 1992aa; IEEE, 1998ab) standards. The applicant stated further that these standards will be either adopted in whole or in part during the detailed design phase, based on whether they were determined applicable or appropriately adapted for use in the Yucca Mountain repository design. In the RAI responses, the applicant provided tables identifying specific exceptions or clauses within these principal codes and standards applicable to ITS controls where it may take these exceptions or implement alternate approaches.

**NRC Staff's Evaluation of the Instrumentation and Controls Design Methods**

The NRC staff evaluated the applicant's information in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, and 1.4.2, 1.9.1.12 and 1.13.2 regarding the design method of the ITS I&C equipment. The NRC staff reviewed the applicant's proposed method for the ITS I&C equipment for determining the appropriate design criteria to be applied to the ITS controls design, as described in the SAR, as well as for implementing specific criteria of the nuclear industry's safety design codes and standards.

The NRC staff determines that a design methodology based on adherence to the design criteria provisions of the ASME, IEEE, and NFPA codes and standards [ASME NOG 1-2004 (Type 1), (ASME 2005aa); IEEE 344-2004, IEEE 308-2001, IEEE 379-2000, IEEE 384-1992, and

IEEE 603-1998 (IEEE 2005aa; IEEE, 2001aaab; IEEE, 1992aa; IEEE, 1998ab); and NFPA 70 (National Electrical Code), (NFPA 2005ab)] without exceptions or alternatives for the design of ITS controls is appropriate and acceptable because these codes and standards (i) describe design criteria that are standard nuclear industry practice for design of ITS I&C to protect worker safety, which the NRC staff finds appropriate for use at the GROA, as discussed in SER Table 7-1; (ii) the application of such standards serves to reduce uncertainties in the accomplishment of safety functions; and (iii) the provisions regarding ITS I&C qualification testing are consistent with NRC guidance in Regulatory Guides 1.100 for seismic qualification and 1.89 for environmental qualification (NRC, 1988aa; NRC, 1984aa) (which endorse the use of IEEE Standards IEEE 344 and IEEE 323, respectively).

For the ITS interlock controls, the applicant's statement that exceptions to the IEEE 308-2001, IEEE 379-2000, IEEE 384-1992, and IEEE 603-1998 (IEEE, 2001aa,ab; IEEE, 1992aa; IEEE, 1998ab) standards may be taken is addressed in SER Section 2.1.1.6.3.2.8.2.1, where the NRC staff proposes a condition of construction authorization that DOE shall not, without prior NRC review and approval, take or implement any exception to the IEEE Standards 308–2001, 384–1992, 379–2000, and 603–1998 in the design of the ITS safety interlock subsystems. Any amendment request requesting exceptions to these IEEE standards must include the design basis for the use of the exception(s), including the ability of structures, systems, and components to perform their intended safety functions assuming the occurrence of event sequences, in accordance with 10 CFR 63.112(e)(8). Further information concerning the NRC staff's evaluation of the applicant's proposed use of the codes, standards, and regulatory guidance cited in the SAR and RAI responses can be found in Table 7-1.

**Design and Design Analyses**

The applicant described the design and design analyses of the ITS controls in SAR Table 1.4.2-1, which summarized the safety functions of SSCs that are implemented through 29 key groups of ITS controls. To facilitate the discussion of the NRC staff's evaluation of the 29 key groups of ITS controls provided by the applicant, the NRC staff divided these 29 groups of ITS controls into three broad categories, based on similarity of (i) process equipment and processes, (ii) operations and normal process controls, and (iii) types of ITS controls and interlock applications. The three categories of ITS controls discussed below are (i) doors, materials handling cranes, and WPTT; (ii) HVAC (CRCF, WHF, and EDGF); and (iii) diesel generator. The ITS controls category for doors, materials handling cranes, and WPTT includes the ITS controls for the equipment shield door, port slide gate (single and double), personnel access and shield door, equipment confinement door-double, cask preparation crane, cask handling crane, jib crane, and auxiliary and pool crane (SAR Table 1.4.2-1). The ITS controls category for HVAC (CRCF, WHF, and EDGF) includes controls for confinement and ITS electrical system cooling and ventilation in these facilities. The applicant classified the HVAC systems in IHF and RF as non-ITS (SAR Table 1.9-1, SER Section 2.1.1.6.3.2.4). The ITS controls category for the diesel generator include ITS controls for the ITS diesel fuel oil system, ITS diesel starting air system, and the ITS diesel generator itself (SAR Table 1.4.2-1).

The evaluation of other groups of ITS controls in SAR Table 1.4.2-1 is provided in SER Sections as follows: (i) CTM and SFTM in SER Section 2.1.1.7.3.2; (ii) TEV in SER Section 2.1.1.7.3.5; (iii) cask handling yoke and pool handling yoke in SER Section 2.1.1.7.3.4; and (iv) cask lid lifting grapple, CTM canister grapple, SNF canister grapple, WP inner lid grapple, HLW canister grapple, jib crane lid lifting grapple, pool lid lifting grapple, and PWR and BWR lifting grapple in SER Section 2.1.1.7.3.4.

Doors, Materials Handling Cranes, and WPTT

According to the applicant, the cask port slide gate (door) is located in the floor of the canister transfer room between the canister transfer room (lower level) and canister staging area (upper level). The applicant provided mechanical outline drawings, piping and instrumentation diagrams, and logic diagrams for the port slide gate in SAR Figures 1.2.4-20, 1.2.4-51, 1.2.4-53, and 1.2.4-57 through 1.2.4-62. The applicant stated that the two safety functions of the ITS SSCs are to (i) protect against inadvertent direct exposure of personnel to radiation and (ii) maintain DOE SNF canister separation from other canisters to prevent criticality events. Both rely on prohibiting the opening of the slide gate unless the CTM shield skirt is in place (DOE, 2009dk).

The applicant stated that the CTM (materials handling crane) transfers HLW from different types of canisters into waste packages in the IHF, CRCF, WHF, and RF. The applicant stated that the CTM materials handling crane will be designed in accordance with ASME NOG 1–2004 (ASME, 2005aa) for Type I cranes. The applicant described CTM in SAR Sections 1.2.3.2.2, 1.2.4.2.2, 1.2.5.2.5, and 1.2.6.2.2, with piping and instrumentation diagrams in SAR Figures 1.2.4-44, 1.2.4-48, 1.2.4-51, and 1.2.4-64 and logic diagram figures in SAR Figures 1.2.4-45, 1.2.4-49, 1.2.4-52 through 56, and 1.2.4-65. Identification of initiating events for the canister and cask handling operations in the GROA facilities is evaluated in SER Section 2.1.1.3.3.2. The methodology for quantification and identification of event sequences, as well as event sequence development and identification of safety functions, is evaluated in SER Section 2.1.1.4.3.2.

The applicant described five safety functions for the ITS controls, as follows, which the NRC staff reviewed and found acceptable in SER Section 2.1.1.6.3.1.

- Protect against a load drop by ensuring that power to the CTM hoist motor is shut off if the “no final hoist upper limit” switch or “no rope mis-spool” switch trips.

- Limit drop height by preventing hoist raising/lowering without safety permissive, limiting lift heights, and requiring a “grapple engaged” signal to allow the load to be lifted further.

- Protect against spurious movement using the hoist holding brake that does not release unless the ASD is given a raise or lower command. The CTM hoist trolley cannot move forward or reverse unless the CTM shield skirt raised interlock and the canister hoist trolley and “shield bell not locked” interlocks are satisfied.

- Protect against inadvertent radiation exposure of personnel and maintain DOE SNF canister separation from personnel at the canister staging area (upper level) through the use of automatic interlocks, which ensure that the CTM shield skirt cannot be raised unless the CTM slide gate is closed. The CTM slide gate cannot be opened unless the CTM shield skirt is lowered.

- Protect against dropping a canister due to a spurious closure of the slide gate by requiring the force of the closing slide gates to be power limited such that the maximum slide gate closing force is insufficient to sever the hoisting ropes.

According to the applicant, the WPTT operates in the waste package loadout subsystem of both the IHF (SAR Section 1.2.3.2.4.1.3) and the CRCF (SAR Section 1.2.4.2.4.1.3). The applicant stated that the safety function for ITS controls implemented for the WPTT would protect against spurious movement while the CTM is lowering the canister by employing interlocks between its drive mechanism and the waste package port slide gate. The interlock interrupts power to the trolley drive when the waste package port slide gate is opened, thereby halting the WPTT.

<u>HVAC (CRCF, WHF, and EDGF)</u>

The applicant stated that the CRCF HVAC systems provide temperature control, flow control, and filtration during normal CRCF operation. According to the applicant, ITS portions of the HVAC system for the CRCF ensures confinement and filtration of radiological releases from event sequences involving breach of waste containers or damaged SNF assemblies and provides environmental conditions, as stated in SAR Table 1.2.4-8, for cooling and ventilation of ITS electrical and mechanical equipment to support the filtration function. The safety functions for ITS controls for CRCF HVAC the applicant discussed in the SAR are as follows.

*Mitigate the Consequences of Radionuclide Release*

SAR Section 1.2.4.4.1 stated that the CRCF surface nuclear confinement HVAC is designed to limit the release of radioactive contaminants to protect workers and the public. The specific safety function the applicant identified is the need to be able to detect the failure of one train (A/B) and to initiate the other train ASD fan motor (B/A), ensuring confinement area exhaust fans are running.

SAR Figure 1.2.4-99 depicted the CRCF 1 Composite Ventilation Flow Diagram Tertiary Confinement ITS Exhaust and non-ITS HVAC Supply Subsystems. SAR Figures 1.2.4-101 and 102 depicted the CRCF 1 ITS Confinement Areas HEPA Exhaust System–Train A and B Ventilation and Instrumentation Diagram. SAR Figure 1.2.4-103 depicted the CRCF and WHF ITS Confinement Areas HEPA Exhaust Fan (Trains A and B) Logic Diagram. The applicant stated that the ITS confinement areas would be maintained at a negative pressure relative to the atmosphere. According to the applicant, ITS exhaust fans exhaust the air through two stages of HEPA filters before discharging it to the atmosphere. The applicant further stated that ITS ASDs vary fan speed, as necessary, to maintain proper differential pressure relative to the atmosphere. According to the applicant, a duct-mounted differential pressure sensor and transmitter monitors the differential pressure of the main exhaust duct and signals a differential pressure controller to adjust the exhaust ASD signal.

The applicant proposed two independent trains (A and B) that are interconnected by an ITS interlock, which is provided to shut down the operating exhaust fan (in Train A) and start the standby unit (in Train B) upon detection of low differential pressure across the fan coincident with low flow, a high HEPA filter train differential pressure, or a low HEPA filter train differential pressure. Within the description of these HVAC trains, the SAR indicated that the HVAC inlet and discharge dampers for each train would automatically close when their associated operating supply fan shuts down to isolate the HVAC envelope so the other train can draw air through its HEPA filter train.

*Support the ITS Electrical Function by Providing Cooling*

The applicant provides in SAR Figure 1.2.4-104 the CRCF 1 Composite Ventilation Flow Diagram Tertiary Confinement ITS HVAC Systems, Electrical, and Battery Rooms. In SAR Figures 1.2.4-105 through 1.2.4-108, the applicant provides CRCF 1 Confinement ITS Electrical Room and Battery Room HVAC System—Train A and B Ventilation and Instrumentation

Diagrams. In SAR Figures 1.2.4-109 through 1.2.4-111, the applicant provides logic diagrams related to the ITS fan coil unit and battery room exhaust fan related to CRCF and WHF (Trains A and B). These diagrams depict redundant HVAC trains (Trains A and B) and the ITS I&C equipment for both trains needed to start and maintain the cooling and ventilation functions necessary to support the ITS electrical functions.

The applicant stated in SAR Section 1.2.4.4.1 that redundant sets of HVAC supply and exhaust equipment serve each group of ITS electrical rooms and battery rooms (Train A and Train B).

<u>Diesel Generator</u>

The applicant stated that two independent ITS diesel generators (Train A and Train B) and supporting ITS mechanical systems will be positioned in the EDGF (SAR Section 1.4.1.2.1). The ITS control safety function of the ITS diesel generator and associated mechanical supporting system are summarized next.

*ITS Electrical Power*

In SAR Section 1.4.1.2.1 and Table 1.4.1-1, the applicant described the ITS diesel generators and their associated safety functions. According to the applicant, ITS electrical power is provided to (i) ITS surface nuclear confinement HVAC system and electrical power HVAC system in the CRCF, WHF, and EDGF; and (ii) the non-ITS HVAC system and electrical power HVAC system in the RF.

*Two Independent Diesel Generators*

The applicant proposed two independent, 100-percent load diesel generators. The applicant responded to the NRC staff's RAI (DOE, 2009dv) that the controls to be used in conjunction with the ITS diesel generators conform to IEEE 387–1995 and IEEE 741–1997 (IEEE, 1996aa; IEEE, 1997aa) for circuit breaker interlocks, in addition to the design codes and standards included in the response to the NRC staff's RAI (DOE, 2009do) (IEEE Std 308-2001, IEEE Std 384-1992, and IEEE Std 603-1998) (IEEE, 2001aa; IEEE, 1992aa; IEEE, 1998ab). Circuit breaker electrical interlocks are provided to prevent automatic closing of an ITS diesel generator circuit breaker to an energized or faulted bus. The applicant also described (SAR Section 1.4.1.2.1) a solid state type undervoltage device for sensing low or degraded voltage on an ITS 13.8 kV power bus. The applicant further stated that a logic signal would be provided to trip the breaker carrying normal power to the ITS 13.8 kV bus upon detection of undervoltage or otherwise degraded voltage conditions.

The applicant stated that each ITS diesel generator design has ITS support systems, including related ITS I&C and ITS interlock equipment, which is electrically and physically independent from the support systems for the other ITS diesel generator. The applicant further stated that each ITS diesel generator fuel oil storage tank is sized for 14 days of continuous operation and would be capable of online refueling.

SAR Section 1.2.8.2.1 stated that there are two independent, underground diesel fuel oil storage tanks, two redundant diesel fuel oil transfer pumps, and two diesel fuel oil day tanks (one for each train). SAR Figure 1.2.8-17 depicted the engine-mounted fuel oil pump and the connections to the engine. SAR Figure 1.2.8-18 depicted the EDGF ITS diesel fuel oil system Train A piping and instrumentation diagrams; though this diagram is demonstrative of Train B as well, SAR Figure 1.2.8-19 depicted the ITS diesel generator fuel oil transfer pump logic diagram.

The applicant provided a list of its standards and codes that would be applied for the design of the ITS diesel generators and associated supporting systems.

**NRC Staff's Evaluation of the ITS Instrumentation and Controls Design and Design Analyses**

The NRC staff evaluated the applicant's information in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.4.1, and 1.4.2 regarding the ITS control design and design analyses for meeting each of the nuclear safety design bases and design criteria. In the area of doors, cranes, and the WPTT, the NRC staff finds that the applicant's proposed design and design analyses of ITS I&C to accomplish safety functions, described previously, are acceptable for the following reasons. First, the applicant adequately described the major design and functional attributes of these ITS SSCs (e.g., protection against a drop and radiation exposure). Second, the applicant's design criteria for controls and safety interlocks governing safety actions of the specific mechanical handling equipment functions are based on established safety-related design criteria for Type I cranes in ASME NOG–1–2004 and IEEE–308, IEEE-379, IEEE-384, and IEEE-603 (IEEE, 2001aa,ab; IEEE,1992aa; IEEE, 1998ab). These standards are consistent with nuclear industry engineering practice for the design of controls and safety interlocks, and are acceptable for use as described by the applicant, with the proposed condition of construction authorization described in SER Section 2.1.1.6.3.2.8.2.1. Further, the applicant's proposed use of these IEEE codes would ensure that the reliability values derived from the applicant's PCSA, as evaluated in SER Section 2.1.1.4.3.4, would be met because the NRC staff has found that the use of such codes and standards in the design of safety controls for nuclear facilities reduces modeling uncertainties and provides for a reliable design.

For the HVAC for CRCF, WHF, and EDGF, the NRC staff finds that the applicant stated (DOE, 2009dl) that the reliability design criteria of principles of independence, redundancy, and single-failure proof will be applied to those systems with a provision for multiple safety trains. The NRC staff finds that the applicant's design and design analyses of instrumentation, controls, and interlock systems supporting the operations of CRCF/WHF/EDGF ITS HVAC SSCs are acceptable because the applicant would implement the applicable IEEE reliability design criteria for the design of safety functions to ensure reliability of performance. Further information concerning the NRC staff's evaluation of the applicant's proposed use of the codes, standards, and regulatory guidance cited in the SAR and RAI responses may be found in Table 7-1.

**NRC Staff's Conclusion of the ITS Instrumentation and Controls Design**

On the basis of the NRC staff's evaluation of the applicant's information on the design of the ITS Instrumentation and Controls systems described above, the NRC staff concludes, with reasonable assurance, subject to the proposed condition of construction authorization in SER Section 2.1.1.6.3.2.8.2.1, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 10 CFR 63.112(f) are satisfied. The NRC staff finds that the description of the ITS Instrumentation and Controls systems designs adequately (i) provides information on the codes and standards and analytical and design methods proposed for use by the applicant, (ii) defines relationships between design criteria and the performance objectives, and (iii) identifies the relationships between the design bases and the design criteria.

#### 2.1.1.7.3.8 Fire Protection Systems

The applicant provided design information for the ITS GROA fire protection systems in SAR Section 1.4.3.2.1, and Table 1.4.3-2. The fire protection systems include fire detection and double-interlock preaction (DIPA) sprinkler suppression systems. The applicant stated that these ITS fire detection and DIPA sprinkler systems are used in moderator-controlled areas of the CRCF and WHF and have the safety function of maintaining moderator control by preventing spurious actuation and inadvertent introduction of fire suppression water in areas where the breach of a loaded canister and water intrusion could lead to a criticality event (DOE, 2008ab). The NRC staff reviewed and found acceptable the applicant's description of the fire protection systems in SER Section 2.1.1.2.3.2.7. The fire detection and DIPA sprinkler suppression systems in the CRCF and WHF were identified as ITS through the PCSA, as reviewed and found acceptable by the NRC staff in SER Section 2.1.1.6.3.2.8.5. The NRC staff's review in this section focuses on the applicant's ITS fire protection system design bases, design criteria, design methodology, and design analysis to determine if these systems would achieve their intended safety function.

### Design Criteria and Design Bases

The applicant provided the nuclear safety design bases and design criteria for ITS fire protection systems in SAR Table 1.4.3-2. The nuclear safety design controlling parameters and values to maintain moderator control were calculated by the applicant as the mean probability of inadvertent introduction of fire suppression water into a canister for the ITS fire protection systems in the CRCF and WHF. The applicant stated that the mean probability must be less than or equal to $10^{-6}$ over a 720-hour period following a radionuclide release in the CRCF and less than $6 \times 10^{-7}$ over a 720-hour period following a radionuclide release in the WHF. The NRC staff's evaluation of DIPA reliability is documented in SER Section 2.1.1.6.3.2.8.5, where the NRC staff finds the applicant's PCSA demonstrated that the proposed fire protection systems will appropriately prevent the inadvertent introduction of fire suppression water into a canister. The applicant specified that to meet these design bases, the fire protection systems are designed as a DIPA system, consistent with NFPA 13 (NFPA, 2007ab) and NFPA 72 (NFPA, 2006aa) standards (SAR Table 1.4.3-2 Design Criteria).

The applicant stated that the intended safety function of the ITS DIPA sprinkler systems is to maintain moderator control by preventing spurious activation and inadvertent introduction of fire suppression water into a breached canister. For a DIPA sprinkler system to activate, an independent heat detection (individual sprinkler head response) signal and a supplemental fire detection (e.g., smoke, flame, or other form of fire detection) signal are necessary before the discharge conditions are met and water is delivered. The DIPA sprinkler system is a variation of a traditional wet pipe system and is used in spaces (e.g., telecommunication centers) where the inadvertent introduction of water is undesirable.

### Design Methods

The applicant stated in SAR Section 1.4.3.2.1 that the design methods and design analyses to be used in designing the ITS DIPA fire protection systems, components, and equipment are consistent with Regulatory Guide 1.189 (NRC, 2009ac), to the extent it applies to fire suppression systems, and are consistent with the NFPA 13 (NFPA, 2007ab) standard for fire suppression sprinkler systems and the NFPA 72 (NFPA, 2006aa) standard for fire alarm systems.

### Design and Design Analyses

The applicant provided fire protection system design information in SAR Sections 1.4.3.2.1 (surface facilities) and 1.4.3.2.2 (subsurface emplacement area) and in Table 1.4.3-2. In addition, the applicant provided responses to the NRC staff RAI in DOE (2009fg, Enclosures 1 and 2). The applicant stated in SAR Section 1.4.3.2.1 that, consistent with standard industry practice, the fire protection equipment and materials are approved by recognized testing laboratories (e.g., Factory Mutual or Underwriters Laboratories) in their published lists. These listings indicate that fire protection equipment and materials meet nationally recognized standards and have been tested and found suitable for use at the GROA.

The applicant indicated in its response to the NRC staff's RAI [DOE (2009fg, Enclosure 1)], that preventing boron pool dilution resulting from a spurious sprinkler system actuation in the WHF was not considered a nuclear safety design basis. The applicant stated that the boron content of the pool in the WHF would still be at a sufficient level to control criticality, even if suppression system water were to hypothetically drain into the pool.

**NRC Staff's Evaluation of the Fire Protection Systems Design Criteria and Design Bases, Design Methods, and Design and Design Analyses**

The NRC staff evaluated the information the applicant provided in SAR Section 1.4.3.2.1, Table 1.4.3-2, and Table 1.9-1 on the fire protection systems design criteria and design bases, design methods, and design and design analyses, and finds that the applicant's design criteria for the ITS DIPA systems are appropriate to meet the design bases for the following reasons. First, the ITS DIPA sprinkler systems include independent heat detection (individual sprinkler head response) and supplemental fire detection (e.g., smoke, flame, or other form of fire detection) interlocked signals before water is introduced to the sprinkler piping. Second, the applicant provided controlling parameters and values (e.g., mean probability of inadvertent introduction of fire suppression water into a canister $\leq 10^{-6}$ over a 720-hour period following a radionuclide release), thus maintaining moderator control such that criticality is a beyond category 2 event. The NRC staff also finds it acceptable for the applicant to not include boron dilution in the WHF pool resulting from introduction of fire suppression system water into the pool as one of the design bases. This is because the pool boron concentration is sufficiently high to control criticality even with the introduction of suppression system water, as described in the applicant's RAI response [DOE (2009fg, Enclosures 1 and 2)]. The NRC staff concluded in SER Section 2.1.1.3.3.2.6 that the applicant's exclusion of boron dilution in the WHF pool is acceptable because the amount of water needed to dilute the boron to below subcriticality concentrations exceeds the total pool volume.

The NRC staff finds that the applicant's design methods and analyses of the ITS fire protection systems are acceptable because they are consistent with (i) Regulatory Guide 1.189 (NRC, 2009ac); (ii) the NFPA 13 (NFPA, 2007ab) standard for the design, installation, inspection, testing, and maintenance of various sprinkler systems (including DIPA systems); and (iii) NFPA 72 (NFPA, 2006aa) standard on the design, inspection, testing, and maintenance standards for various fire alarm and sprinkler monitoring systems. The fire protection program elements (fire hazard analysis, compensatory measures) outlined in RG 1.189 are applicable to all nuclear facilities, including the GROA. NFPA 13 and 72 are nationally-accepted standards used for safety-related fire protection systems in the nuclear industry. The use of these standards is consistent with the design codes and standards for fire protection systems in YMRP (NRC, 2003aa), and the NRC staff finds their use at the GROA acceptable, as proposed by the applicant, and further described in Table 7-1.

**NRC Staff's Conclusion of the Fire Protection Systems Design**

Based on the NRC staff's evaluation of the applicant's information on the ITS fire protection system designs described above, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the description of the fire protection system design adequately (i) provides information on materials of construct ion, dimensions, proposed codes and standards, and analytical and design methods; (2) defines the relationship between design criteria and the performance objectives; and (3) identifies the relationship between the design bases and the design criteria.

### 2.1.1.7.3.9 Canisters and Overpacks

The applicant provided design information for ITS overpacks and canisters to be used at the GROA. The applicant identified the overpacks and canisters as ITS through the PCSA, incorporating site-specific data and consequence analyses. These were reviewed by the NRC staff and found acceptable in SER Section 2.1.1.6.3.1. The ITS overpacks and canisters are categorized into the following three groups by the NRC staff to facilitate its review: (i) waste packages; (ii) transportation, aging, and disposal (TAD) canister; and (iii) other canisters, overpacks, and casks, which include DOE SNF standardized and HLW canisters, DPCs, naval canisters, aging overpacks and shielded transfer casks (STCs), as well as the transportation cask.

The applicant stated in SAR Section 1.2.8.4.5 that the transportation cask, which contain the canisters (TAD, DOE SNF standardized canister, naval canisters, DPCs), is designed and certified under 10 CFR Part 71 requirements that bound the GROA site-specific performance criteria for Category 1 and 2 event sequences in the PCSA. The applicant applied GROA-specific conditions to demonstrate compliance with the requirements for the PCSA at 10 CFR 63.112 in SAR Sections 1.6 to 1.9. The applicant's PCSA for these components is reviewed and found to be acceptable by the NRC staff in SER Sections 2.1.1.4.3.4 and 2.1.1.6.3.1. The following evaluation contains the NRC staff's review of the applicant's design bases and design criteria, design methodology, and design and design analyses for these canisters and overpacks.

#### 2.1.1.7.3.9.1 Waste Package

The applicant described the waste package design in SAR Sections 1.5.2, 1.2.1.4.1, 1.2.4.2.3.1.3, 1.3.1.2.5, and 2.3.6.7.4 and Tables 1.5.2-6 and 1.5.2-7. The applicant stated that the ITS waste package is an engineered barrier for disposal of CSNF, HLW, and DOE and naval SNF. The applicant further stated that the waste packages are designed to accommodate six different loading configurations, depending on the waste form. According to the applicant, the waste package can contain a TAD canister with CSNF, a short or long codisposal canister with defense HLW and DOE SNF, or a short or long naval canister with naval SNF (see SER Section 2.1.1.2.3.5.1 for more information on waste package configurations). According to the applicant, all waste package configurations have a single design that consists of two concentric cylinders (i.e., the inner vessel and the outer corrosion barrier) with the upper and lower sleeves at the ends of the outer corrosion barrier for additional structural support (SAR Figures 1.5.2-3 through 1.5.2-8). Although all waste packages have a single design, the applicant stated that different waste package configurations have different internal structures and different external dimensions to accommodate various waste forms.

**Design Criteria and Design Bases**

The applicant presented the nuclear safety design bases for the waste packages and their relationship with the design criteria in SAR Table 1.5.2-6. In this table, the applicant also provided the specific design criteria for each of the design bases, along with controlling parameters and bounding values.

The applicant provided several design criteria for the safety design bases to (i) provide containment for a sealed waste package for an event sequence resulting from an impact, a drop of a load onto the waste package, or a spectrum of fire ranging from a local fire confined to a single fire zone to a large fire that could propagate to an entire facility (BSC, 2008ac,as,bk) and (ii) protect against breach of the waste package from a rock or vibratory ground motion impacts. According to the applicant, these design criteria, based on 2001 ASME Boiler and Pressure Vessel Code, Section II (Materials and Specifications) and Section III (Rules for Construction of Nuclear Power Plant Components) (ASME, 2001aa), use ultimate tensile strength limits based on material energy absorption capabilities. Also, the applicant indicated, the controlling parameters imposed on each design basis are a mean conditional probability or a mean frequency of breach of the waste package.

The applicant stated that the waste package consists of two concentric cylinders: an inner vessel of Stainless Steel Type 316 (UNS S31600, with further compositional restrictions as described in SAR Section 1.5.2.7) designed for structural support, and a corrosion-resistant outer shell made of Alloy 22 (UNS N06022, a nickel-chromium-molybdenum alloy with further compositional restrictions, as described in SAR Section 2.3.6.7). The applicant further stated that the design of the waste packages includes the following characteristics: (i) a minimum thickness of 25 mm [1 in] for the outer corrosion barrier; (ii) a minimum 2 mm [0.08 in] and maximum 10 mm [0.40 in] difference between the waste package inner vessel outer diameter and the outer corrosion barrier inner diameter for the as-fabricated waste package; (iii) a minimum 30 mm [1.2 in] difference between the waste package inner vessel overall length and the outer corrosion barrier cavity length, from the top surface of the interface ring to the bottom surface of the top lid; and (iv) a design pressure of 1 MPa [150 psi] at 343 °C [650 °F] for the inner vessel to accommodate internal pressurization of the waste package, filled with inert helium gas, including effects of high temperature and fuel rod fill-gas release.

**NRC Staff's Evaluation of the Waste Package Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the design criteria and design bases for the waste package and its components and finds that the design criteria and design bases for the waste package and its components are adequate because they (i) were derived from the specific site characteristics and consequence analyses; (ii) address the relevant events identified in the PCSA (i.e., impact from collisions and drop of a load onto the waste package, fire, and breach of the waste package from a rock or vibratory ground motion impacts) that could affect the waste package; and (iii) are based on the 2001 ASME Boiler and Pressure Vessel Code, Sections II and III design criteria, which the NRC staff finds acceptable for use at the GROA, as further described in Table 7-1.

**Design Methods**

The applicant stated that the design methodology for the waste package incorporates structural and thermal design methods presented in the waste package component design methodology report (BSC, 2007bi). The design methodology includes the following: (i) understanding the requirements imposed on the design, (ii) formulating a design concept, (iii) gathering all the design input information, (iv) making defensible assumptions, (v) selecting analytic methods and computational tools, and (vi) demonstrating that the design requirements are met. The applicant presented the structural design methods, including the analyses performed for various load combinations (normal loads and event sequences loads), and applicable acceptance criteria in SAR Tables 1.5.2-8 and 1.5.2-9. The applicant performed parametric studies to analyze waste package response to accidental fires, assuming the worst fire conditions not exceeding those defined in the NRC regulations for transportation casks.

The applicant's design methodology includes elastic finite element analyses for normal loads to estimate (i) the tensile stresses imposed on the waste package outer corrosion barrier while the waste package is statically resting on a waste package pallet, (ii) the contact stresses imposed on the waste package from axial and radial thermal expansion of the inner vessel and outer corrosion barrier, and (iii) the tensile stresses imposed on the waste package outer corrosion barrier due to internal pressurization from increased temperature and decreased volume between the inner vessel and outer corrosion barrier. The applicant stated that, for the normal loads, its acceptance criterion was for the generated stresses to remain in the elastic range and below the threshold for stress corrosion cracking of Alloy 22.

The applicant's design methodology applies elastic-plastic finite element analyses and analytic methods to evaluate waste package performance for structural challenges from seismic and operational event sequences, as reviewed and found to be acceptable by the NRC staff in SER Sections 2.1.1.4.3.2.1.1, 2.1.1.4.3.2.1.3, 2.1.1.4.3.3.1.1, and 2.1.1.4.3.3.1.2.2. The applicant's design methodology calculates the stress intensities in the waste package outer corrosion barrier for the following cases: (i) the waste package subjected to dynamic forces inside the TEV due to seismic ground motion, (ii) collision of the TEV with the emplaced waste package, (iii) oblique drop of the waste package onto the TEV surface, (iv) damage to the waste package while oriented horizontally inside the WPTT that is subjected to the dynamic loads from vibratory ground motion, (v) the waste package subjected to loads produced by drift collapse in the lithophysal portions of the repository caused by vibratory ground motion, (vi) the waste package subjected to loads produced by rockfall in the nonlithophysal portions of the repository, and (vii) horizontal drop on the emplacement pallet and invert. The applicant stated that, for event sequence loads, the acceptance criteria of Level D Service Limits of the 2001 ASME Boiler and Pressure Vessel Code, Division 1, Class 2, Appendix F, are used to limit the consequences of the specified event. The Level D Service Limits of the allowable stresses in Section F-1341.2, Appendix F for plastic analysis are intended to assure that waste package breach from an event sequence will not occur, even though stresses may be generated beyond the elastic range. The applicant invoked a tiered screening criteria method for material failure (SAR Table 1.5.2-10). The tiered method uses a step-by-step approach to evaluate the stresses for compliance with the 2001 ASME Boiler and Pressure Code, starting with the most conservative analysis, and refining the analysis, as appropriate, to ensure that the provisions of the code are met.

**NRC Staff's Evaluation of the Waste Package Design Methods**

The NRC staff evaluated the information provided by the applicant on the structural and thermal design methods for the waste package and its components and finds that the proposed design methods are adequate because they are consistent with applicable codes and standards. In particular, the applicant's tiered screening criteria method (SAR Table 1.5.2-10) for the allowed stresses beyond the material's elastic range is a deterministic approach based on elastic-plastic analysis methods provided in ASME 2001, Section III, Appendix F "Rules for Evaluation of Service Loadings with Level D Service Limits" (ASME, 2001aa). For this method, the wall-average total stress intensity value (twice the maximum shear stress) is derived from the analytical or finite element analyses and is compared against failure criteria that are based on the material ultimate tensile strength. Additionally, the NRC staff finds the applicant's use of the Level D Service Limits of stresses in the 2001 ASME Code, as described in Appendix F, Section F-1341.2 for plastic analysis, for the design of the waste package acceptable because it will assure that the inner and outer canisters breach will not occur, even though material may undergo inelastic deformations. Therefore, the NRC staff finds that the waste package would perform its safety function of providing containment. The NRC staff's evaluation of the applicant's threshold values for stress corrosion cracking (SCC) of the waste package outer corrosion barrier is documented in SER Section 2.2.1.3.1.3.2.3, where the NRC staff finds the applicant reasonably accounted for SCC in its model.

**Design and Design Analyses**

The applicant stated that the waste package provides containment and protects against the release of radioactive gases or particulates during normal operations and Category 1 and Category 2 event sequences during the preclosure period.

The applicant also stated that the waste package inner vessel is a load-bearing component (i.e., a pressure vessel) for internal pressure and deadweight loads and is designed consistent with 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (ASME, 2001aa) for Class 2 components. In addition, the applicant stated that the outer corrosion barrier serves as a corrosion-resistant component and is not a pressure vessel (BSC, 2007bi). According to the applicant, the design uses the applicable technical specifications of 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (ASME, 2001aa) for Class 2 components.

The applicant stated that the materials, design, fabrication, testing, and examination of the waste packages (both the inner vessel and the outer corrosion barrier) meet the specifications in the following codes and standards in ASME (2001aa):

- Section II, "Materials"
- Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components"
- Section V, "Nondestructive Examination"
- Section IX, "Welding and Brazing Qualifications"

The applicant presented the waste package fabrication materials and process in SAR Section 1.5.2.7.1 and described the fabrication procedure and the welds for the waste package in SAR Figure 1.5.2-11. In addition, the applicant described the methods for the closure welds in SAR Section 1.2.4.2.3.

The applicant stated that the waste package inner vessel material is ASME SA–240 (UNS S31600) with additional controls on nitrogen and carbon, referred to as Stainless Steel 316. According to the applicant, the waste package outer corrosion barrier material is ASME SB–575 (UNS N06022) with limited constituents of chromium, molybdenum, tungsten, and iron, which is referred to as Alloy 22 (BSC, 2007bi).

Using these design methods, the applicant provided analyses of three waste package configurations (TAD canister, DOE short codisposal canister, and naval long canister; SAR Table 1.5.2-9) to be used at the GROA (SAR Section 1.5.2.1.1). In response to the NRC staff's RAI (DOE, 2009er), the applicant stated that the remaining three waste package configurations (5-DHLW/DOE long codisposal, 2-MCO/2-DHLW codisposal, and the naval canistered SNF short waste package) will be analyzed prior to their use in repository operations.

The applicant provided structural and thermal finite element analyses of the three waste package configurations (TAD canister, DOE short codisposal canister, and naval long canister), including performance under normal and event sequence load combinations (DOE, 2009er). These included the following analyses by the applicant: (i) naval canistered SNF long waste package oblique impact inside the TEV (BSC, 2007cn); (ii) nonlithophysal rockfall impacts on waste packages (BSC, 2007co,cr); (iii) emplacement pallet lift and degraded static analysis (BSC, 2007cp); (iv) naval canistered SNF long waste package vertical impact on the emplacement pallet and invert (BSC, 2007cq); and (v) thermal responses of TAD and 5-DHLW/DOE SNF waste packages to a hypothetical fire accident (BSC, 2007cs). The applicant stated that these analyses were bounding for the three waste package configurations because (i) the three waste packages have identical outer barrier thickness of 2.54 cm [1 in], thus, the behavior of the waste packages for impact loads and fire would be similar; (ii) the heaviest waste package was selected for the impact analyses (SAR Table 1.5.2-9) to evaluate the waste packages for the worst loading; (iii) thermal analysis for a hypothetical fire event was performed for the waste packages containing the TAD and 5-DHLW/DOE SNF canisters, thus bounding the applicant's evaluation for the DOE short codisposal canister, Naval Long canister, and the TAD. The staff notes that the Naval Long canister and the TAD canister have identical physical dimensions.

In SAR Section 1.5.2.7.3, the applicant stated that all major fabrication welds of the waste package will be nondestructively examined (NDE) after final machining, surfacing, and heat treatment. The applicant stated that the welds for the inner vessel and outer corrosion barrier are examined by either Level II or Level III NDE personnel, in accordance with the 2001 ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NC–5000 "Examination" (ASME, 2001aa). The applicant would use radiographic examination, liquid dye penetrant testing, and ultrasonic examination to examine the outer corrosion barrier longitudinal weld, circumferential weld, bottom lid weld, and upper sleeve to outer corrosion barrier weld. According to the applicant, the liquid dye penetrant testing method is only used to examine the lower sleeve to outer corrosion barrier weld, inner vessel support ring to outer corrosion barrier weld, inner vessel lid lifting feature weld, outer lid lifting feature weld, and divider plate assembly weld to inner vessel. The applicant would use radiographic examination and liquid dye penetrant testing methods to examine the inner vessel longitudinal weld, inner vessel circumferential weld, and inner vessel bottom lid weld (SAR Table 1.5.2-13). According to the applicant, the final closure welds is inspected using visual, eddy current, and ultrasonic inspection techniques (SAR Table 1.5.2-7).

**NRC Staff's Evaluation of the Waste Package Design and Design Analyses**

The NRC staff evaluated the applicant's information on the codes and standards for the materials, design, fabrication, testing, and examination of the waste package and its components, and design analysis, and finds that the cited codes and standards [ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (ASME, 2004aa)] for Class 2 components) are appropriate because they are consistent with standard industry practices, and the NRC staff finds their use acceptable at the GROA, as proposed by the applicant, as further discussed in Table 7-1. Specifically, the NRC staff finds that the design codes and standards are appropriate for the waste packages design and construction because Subsection NC is modeled after Subsection NB (ASME, 2004aa) for Class 1 components and contains provisions for material design, fabrication, examination, testing, overpressure relief, marking, stamping, and reports.

The NRC staff also evaluated the information provided regarding proposed fabrication materials, fabrication processes, and closure methods for the waste package. The NRC staff finds that the processes the applicant proposed to use for the waste package fabrication, assembly, and closure are acceptable because the processes (i) are consistent with applicable sections of 2001 ASME Boiler and Pressure Vessel Code (ASME, 2001aa) for fabrication of steel pressure vessels accepted by the nuclear industry and (ii) the use of this code for the GROA is appropriate, as discussed further in Table 7-1.

The NRC staff also finds that the selection of waste package materials (i.e., Stainless Steel 316 for the load-bearing component of the waste package and of Alloy 22 for the corrosion-resistant component of the waste package) is acceptable because (i) Stainless Steel 316 exhibits resistance over a wide range of environmental conditions (e.g., pitting, acids, high temperatures) (Fontana, 1986aa); and (ii) Alloy 22 is resistant to localized corrosion, stress corrosion cracking and oxidizing, and reducing chemicals (Pope, 1997aa).

The NRC staff reviewed representative samples of the applicant's structural and thermal finite element analyses to evaluate waste package performance under normal operations and event sequence (Category 1 and 2 event sequences) load combinations. The applicant evaluated waste package configurations containing the 21–PWR/44–BWR TAD, 5-DHLW/DOE short codisposal and naval canistered SNF long canisters. For these analyses, the NRC staff finds that the calculated stresses in the waste package outer corrosion barrier meets the applicant's tiered screening criteria method to evaluate material failure for mechanical loading (SAR Table 1.5.2-10) and are consistent with the 2001 ASME Boiler and Pressure Vessel Code (ASME, 2001aa). Additionally, based on the review of the applicant's thermal analyses of waste packages containing TAD and 5-DHLW/DOE SNF canisters (BSC, 2007cs), the calculated surface temperatures of canisters inside the waste package stayed below the temperature limits for these canisters for accident conditions. The NRC staff finds that the applicant's design analyses on the waste package and its components is acceptable because

- The design analyses conform to established practices for mechanical/structural performance assessment using finite element methods (Bathe, 1996aa);

- The waste package components are designed to sustain loads from normal operations and Category 1 and 2 event sequences;

- The waste package thermal controls are such that the canister surface temperature limits are below the design limits; and thus, the fuel cladding temperature is sufficiently low to prevent thermally induced cladding failure; and

- The representative waste package analyses of the selected waste packages containing naval long canister, TAD canister, and 5-DHLW/DOE SNF canisters for potential event sequence loads are bounding for the three waste packages, because

    (1) all three waste packages containing 21-PWR/44-BWR TAD, 5-DHLW/DOE Short codisposal, and Naval Long canisters have the same outer barrier thickness of 25.4 cm; thus, the behavior of the waste packages for impact loads would be similar;

(2) the heaviest waste package was analyzed to evaluate the worst condition for impact loads; and

(3) the thermal analysis for a hypothetical fire event was performed for the waste packages containing the highest thermal load (TAD and 5-DHLW/DOE SNF canisters) thus representing the worst condition for the waste packages containing 21-PWR/44-BWR TAD, 5-DHLW/DOE Short codisposal, and Naval Long canisters.

The NRC staff evaluated the information provided by the applicant regarding the proposed nondestructive examination (NDE) techniques and NDE examiners qualifications to detect and evaluate fabrication-related defects. The NRC staff finds that the applicant's proposed nondestructive examination techniques and the NDE examiner qualifications are acceptable because these techniques are consistent with the applicable sections of the waste package design code 2001 ASME Boiler and Pressure Vessel Code (ASME, 2001aa) (e.g., the NDE provisions of Section NC-5000 "Examination"), which is reviewed and found to be acceptable by the NRC staff, as described earlier in this SER Section and as discussed in SAR Table 7-1.

Based on the evaluation described above, the NRC staff finds that the applicant's design analyses for the waste package configurations containing the 21–PWR/44–BWR TAD canister, containing the 5-DHLW/DOE Short codisposal canister, and containing the Naval canistered SNF Long canister are acceptable. However, the applicant has not provided the necessary analyses for the three waste package configurations for the 5-DHLW/DOE Long codisposal, 2-MCO/2-DHLW codisposal, and Naval canistered SNF Short waste packages. Additionally, the NRC staff determines that the DOE cannot receive the (four) canisters that would be emplaced in these three waste packages (i.e., DHLW Long, DOE Long, MCO, Naval Short) because the waste packages associated with the disposal of these canisters have not been evaluated by the applicant or found acceptable for disposal by the NRC. The proposed condition of construction authorization in SER Section 2.1.1.2 would provide that DOE cannot accept the MCO canister. The NRC staff finds that, in addition, these specific waste packages (i.e., 5-DHLW/DOE long codisposal, 2-MCO/2-DHLW codisposal, and Naval Short) and canisters (i.e., DHLW long, DOE long, and Naval Short) that were not analyzed by the applicant also shall not be accepted at the repository, without prior NRC review and approval, of information from DOE that either (i) confirms that the current PCSA bounds the intended performance of these waste packages and canisters at the GROA or (ii) demonstrates, through the PCSA, that these waste packages and canisters can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

**Proposed Condition of Construction Authorization [10 CFR 63.32(a)]**

DOE shall not, without prior NRC review and approval, accept the following waste packages: (i) 5-DHLW/DOE long codisposal; (ii) 2-MCO/2-DHLW codisposal; and (iii) Naval Short.

DOE shall not, without prior NRC review and approval, accept the following canisters: (i) DHLW long; (ii) DOE long; and (iii) Naval Short.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these waste packages and canisters at the GROA or (ii) demonstrates, through the PCSA, that these waste packages and canisters can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

2.1.1.7.3.9.2 Transportation, Aging, and Disposal Canister

The applicant stated that it plans to use the TAD canister to dispose of commercial spent nuclear fuel (CSNF). The applicant indicated that the TAD canister may be loaded, sealed, and used for storage at the utilities and then used for transportation to the GROA. The TAD canister may also be loaded with CSNF at the repository. According to the applicant, the TAD canisters are used in surface facilities, including the canister receipt and closure facility (CRCF), receipt facility (RF), and wet handling facility (WHF), and in the subsurface facility where it is inside a waste package. The applicant stated that the evaluation of the TAD canister system and components (aging overpack, STC, transportation cask, site transporter) is based on the performance specification (DOE, 2008ag), as described in SAR Section 1.5.1.1.1.2.1.3. The applicant described in SAR Table 1.5.1-10 how the TAD performance specification requirements are met in the applicant's evaluation of the TAD canister system for the GROA. The applicant provided the design features of the TAD canister in SAR Section 1.5.1.1.1.2.1.3 and a conceptual representation in SAR Figure 1.5.1-5. The TAD canister structural and containment characteristics were described by the applicant in SAR Sections 1.5.1.1.1.2.6.1.1 and 1.5.1.1.1.2.6.1.2, respectively. The applicant also indicated that the TAD canister is designed to withstand the natural phenomena listed in SAR Table 1.2.2-1 and horizontal and vertical ground motion shown in SAR Figures 1.2.2-8 to 1.2.2-13.

**Design Criteria and Design Bases**

The applicant identified the TAD canister as ITS because it is relied upon to prevent or mitigate the consequences of an event sequence (SAR Section 1.9), which the NRC staff reviewed and found to be acceptable in SER Section 2.1.1.6.3.1. In SAR Table 1.5.1-7, the applicant provided the nuclear safety design bases for the TAD canister and their relationship to TAD canister structural characteristics. Specifically, the TAD canister provides containment to radioactive materials when subjected to structural challenges, such as drop of the canister or a load onto the canister, a side impact or collision, and seismic events. The TAD canister also provides containment when subjected to thermal challenges over a spectrum of fires [a local fire confined to a single fire zone to a large fire that could propagate to an entire facility (BSC, 2008ac,as,bk)] while contained within a cask, waste package, aging overpack, or the CTM shield bell. The TAD canister is designed to meet the requirements of 10 CFR Part 72 for storage, 10 CFR Part 71 for transportation, and 10 CFR Part 63 for repository disposal.

In SAR Tables 1.9-3 and 1.9-4 and Table 1.5.1-7, the applicant identified TAD canister design criteria and design bases, which include design bases to provide containment and protect against TAD canister breach from drops, impact, collision, and fire events, based on the PCSA in SAR Sections 1.6 to 1.9. The applicant stated in SAR Section 1.5.1.1.1.2.5.2 that, for criticality safety, the TAD canister would be designed to meet the requirements of 10 CFR Part 72 for storage, 10 CFR Part 71 for transportation, and 10 CFR Part 63 for repository disposal. Further, the applicant stated that the TAD canister would provide moderator control to ensure subcriticality during all possible event sequences for handling operations that are important to criticality. In addition, the applicant stated that the TAD canister has thermal characteristics such that the cladding temperature shall not exceed 400 °C [752 °F] for normal operations of storage, transportation, and handling, and 570 °C [1,058 °F] during draining, drying, and helium backfill operations, as described by the applicant in SAR Section 1.5.1.1.1.2.5.3.

**NRC Staff's Evaluation of the TAD Canister Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design bases and design criteria of the TAD and finds that the design criteria and design bases the applicant used for the TAD canister are adequate because they are (i) derived from the PCSA, reviewed and found acceptable by the NRC staff in SER Sections 2.1.1.4.3.4 and 2.1.1.6.3.1; and (ii) consistent with the canister's intended safety function to provide containment and criticality safety from structural or thermal challenges at the GROA.

**Design Methods**

In SAR Section 1.5.1.1.1.2.6, the applicant presented the design methods to be used for designing the TAD canister to meet the performance specifications. The applicant focused on two parameters: (i) a helium leakage rate and (ii) fuel cladding temperature. The applicant's design methodology specifies that the TAD canister shall maintain a normal condition maximum helium leakage rate of $1.5 \times 10^{-12}$ fraction of canister free volume per second for Design Basis Ground Motion 2 (DBGM–2) and Beyond DBGM–2 (BDBGM) seismic events. During these events, the TAD canister is either suspended by a crane inside a cylindrical steel cavity, contained within a transportation cask (with and without impact limiters), or contained within an aging overpack. The TAD canister shall also maintain the maximum off-normal condition helium leakage rate of $9.3 \times 10^{-10}$ fraction of canister free volume per second for a fully engulfing fire {with a flame temperature of 938 °C [1,720 °F] for 30 minutes} while in an open or closed transportation cask (with or without impact limiters). The applicant's design methodology specifies the maximum cladding temperature for a 2,000-year return period seismic event limited to 400 °C [752 °F] (normal) and for a 10,000-year return period seismic event limited to 570 °C [1,058 °F] (off-normal conditions at the GROA). Similarly, the TAD canister, while contained in an aging overpack must maintain a maximum helium leakage rate of $9.3 \times 10^{-10}$ fraction of canister free volume per second (off-normal) and a maximum cladding temperature of 570 °C [1,058 °F] (off-normal)

**NRC Staff's Evaluation of the TAD Canister Design Methods**

The NRC staff evaluated the information provided by the applicant on the design specifications and methodology for the TAD canister and finds that the applicant's design methods to be used for the TAD canister are adequate because (i) the applicant considered a range of normal and off-normal conditions, for the effect on the helium leakage rates and cladding temperatures; (ii) the normal and off-normal cladding temperature limits for structural integrity are consistent with the guidelines of NUREG–1536 (NRC, 2010ah), which specifies cladding temperature limits of 400 °C [752 °F] (normal) and 570 °C [1,058 °F] (off-normal); and (iii) the applicant's proposed maximum helium leakage rates for normal and off-normal conditions are consistent with ANSI N14.5–1997 (ANSI, 1998aa). Additionally, the PCSA for the TAD configurations at the GROA in a transportation cask, and during transfer operations to and in the aging overpack or the waste package, have been reviewed and found acceptable by the NRC staff in SER Section 2.1.1.4.3.4.

**Design and Design Analyses**

The applicant stated in SAR Section 1.5.1.1.1.2.6.1.2 that NUREG–1536 (NRC, 1997ae) is used as the basis for the TAD canister water draining and drying procedures, which include drying and inerting (filling with pressurized gas) of the canister. Further, the applicant stated that helium is used to inert the TAD canister to prevent oxidation of the spent fuel cladding.

The applicant specified in SAR Section 1.5.1.1.1.2.6.1.2 that the fabrication of the TAD canister will follow the 2004 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (ASME, 2004aa) code for Class 1 components. The applicant stated in SAR Section 1.5.1.1.1.2.7 that the material for the TAD canister shell and structural internals (i.e., basket) are constructed of a 300-series stainless steel, UNS S31603 (which may also be designated as Type 316L), as per ASTM A276–06 (ASTM International, 2006ab). In addition, the TAD canister shell and structural internals are designed to be compatible with either borated or unborated water environments as defined in DOE (2008ag, Table 3.1-4). In addition, the applicant identified a list of prohibited or restricted materials that cannot be used to construct the TAD canister (e.g., organic hydrocarbon-based material). The applicant further stated that all metal surfaces of the TAD canister shall meet the cleanliness specifications, as defined in ASME NQA–1–2000, Subpart 2.1, Classification C (ASME, 2000aa). With respect to criticality, SAR Section 1.5.1.1.1.2.2.2 described the characteristics and materials (e.g., minimum thickness of the neutron-absorber plates and associated range of boron contents in the neutron-absorber plates) of the neutron absorbers.

In SAR Section 1.14.2.3.1.3, the applicant listed the TAD canister components that are designed to prevent and control criticality. The applicant indicated that the shell of a sealed canister is most important because it prevents a moderator from being introduced into the SNF. The applicant also stated in SAR Section 1.14.3 that the nuclear criticality safety program at the repository complies with 10 CFR Part 63 and the applicable parts of NRC Regulatory Guide 3.71 (NRC, 2005ac).

The TAD canister containment characteristics were described by the applicant in SAR Section 1.5.1.1.1.2.6.1.2. The applicant established a maximum leakage rate of $1.5 \times 10^{-12}$ fraction of canister free volume per second and a cladding temperature limit of 400 °C [752 °F] after the TAD canister underwent a 0.3-m [12-in] vertical flat bottom drop, consistent with the TAD canister design methods, reviewed and found to be acceptable by the NRC staff, as described above. The applicant specified that the TAD canister closure welds shall conform to the standards set forth in SFPO–ISG–18 (NRC, 2008ae) and that closure weld helium leak testing shall conform to the testing procedures in ANSI N14.5–97 (ANSI, 1998aa). The applicant further stated that the guidance of SFPO–ISG–18 (NRC, 2008ae) shall be followed to ensure the final closure weld integrity.

The applicant stated in SAR Section 1.5.1.1.1.2.8 that the materials, design, fabrication, testing, and examination of the TAD canister shall meet the specifications of the following codes and standards:

- ANSI N14.5–97, American Standard for Radioactive Materials—Leakage Test on Packages for Shipment (ANSI, 1998aa)
- 2004 ASME Boiler and Pressure Vessel Code (ASME, 2004aa)
- ASCE 7–98, Minimum Design Loads for Buildings and Other Structures (ASCE, 2000ab)
- ASTM A276–06, Standard Specification for Stainless Steel Bars and Shapes (ASTM International, 2006ab)
- SEI/ASCE 7–02, Minimum Design for Buildings and Others Structures (ASCE, 2003aa)

**NRC Staff's Evaluation of the TAD Canister Design and Design Analyses**

The NRC staff reviewed the specification [NUREG–1536 (NRC, 1997ae)] the applicant identified for drying and inerting of the TAD canister. The NRC staff finds that this specification is acceptable because NUREG–1536 (NRC, 2010ah) provides guidance for storage of canisters containing SNF at NRC-certified facilities under 10 CFR Part 72, which are similar to the GROA preclosure facilities, as further discussed in Table 7-1.

The NRC staff also evaluated the design codes and standards to be used for the TAD canister design and construction [ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (ASME, 2004aa) for Class 1 components]. The NRC staff finds that the design codes and standards are appropriate for the TAD canister design and construction because Subsection NB provides standards for nuclear pressure vessel material design, fabrication, examination, testing, overpressure relief, marking, stamping, and reports. The NRC staff further reviewed the material specifications and restrictions for TAD canister construction. The NRC staff finds that the specifications and restrictions, including standards to be used, are adequate because they are consistent with the standard engineering practices for similar canister construction applications at NRC-licensed storage facilities [NUREG–1536 (NRC, 2010ah)], and the NRC staff finds their use appropriate, as proposed by the applicant, as further discussed in Table 7-1.

The NRC staff evaluated the characteristics and design specifications of the criticality-significant components of the TAD canister. The NRC staff finds that the TAD canister reasonably prevents and controls criticality because (i) the canister shell is designed to prevent a moderator from entering the canister; (ii) the fixed neutron-absorbers specifications for controlling criticality are consistent with NRC guidance for criticality control at nuclear facilities [NUREG–1536 (NRC, 2010ah)], discussed further in SER Section 2.1.1.7.3.10.1; and (iii) the criticality information the applicant provided is consistent with the criticality standards in Regulatory Guide 3.71 (NRC, 2005ac), which is applicable to the GROA, as discussed in Table 7-1.

The NRC staff evaluated the information provided by the applicant on the design and design analyses for the TAD canister using the applicable guidance in SFPO–ISG–18 (NRC, 2008ae) for the TAD canister containment characteristics. The NRC staff finds that it is acceptable for the applicant to follow the guidance in SFPO–ISG–18 for repository welding applications because this ISG addresses welding flaws of sufficient sizes that could impair the weld structural strength or confinement capability. The NRC staff finds that SFPO–ISG–18 (NRC, 2008ae) was developed to address the qualification of final closure welds on austenitic stainless steel canisters. Additionally, SFPO–ISG–18 states that, when the welding techniques and examination methods conform to guidance given in SFPO–ISG–15 (NRC, 2001ac), there is reasonable assurance that no flaws of significant size will exist such that they could impair the structural strength or confinement capability of the weld.

The NRC staff also evaluated the codes and standards [ASTM A276–06, (ASTM International, 2006ab)] the applicant specified for the materials, design, fabrication testing, and examination of the TAD canister and finds them acceptable because these are industry-accepted design and testing standards, the use of which the NRC staff finds acceptable for use at the GROA as proposed by the applicant, as further discussed in Table 7-1.

2.1.1.7.3.9.3 Other Canisters, Overpacks, and Casks

The NRC staff organized its review and evaluation of other canisters, overpacks, and casks into the following topics: (i) DOE standardized canisters for SNF, (ii) HLW canisters, (iii) DPCs, (iv) naval canisters for U.S. Navy SNF, (v) aging overpacks, and (vi) transportation casks.

2.1.1.7.3.9.3.1 U.S. Department of Energy Standardized Canister

The applicant provided the design information for the DOE standardized canister in SAR Sections 1.5.1.3.1.2.1.1 (Shell), 1.5.1.3.1.2.1.2 (Internals), and a mechanical envelope diagram of a small-diameter standardized canister in SAR Figure 1.5.1-9. The applicant stated that the DOE standardized canister contains DOE SNF generated by DOE production reactors, demonstration commercial power reactors, and domestic and foreign research and training reactors. According to the applicant, the DOE standardized canister design allows two different canister diameters and lengths: the small diameter canister has an outer diameter of 457 mm [18 in], and the large diameter canister has an outer diameter of 610 mm [24 in]. Both the small and large diameter canisters can be either 3.1 m [10 ft] or 4.6 m [15 ft] long. The applicant stated that these standardized canisters are fabricated from Stainless Steel Type 316L. The applicant indicated that the weight of the large diameter canister plus its content (DOE SNF) weighs between 4,077 kg [9,000 lb] for the 3.1-m [10-ft] length and 4,536 kg [10,000 lb] for the 4.6-m [15-ft] length. The applicant further indicated that the weight of the small diameter canister is between 2,265 kg [5,000 lb] for the 3.1-m [10-ft] length and 2,722 kg [6,000 lb] for the 4.6-m [15-ft] length (SAR Section 1.5.1.3.1.2.1.1).

**Design Criteria and Design Bases**

According to the applicant, the safety function of the DOE standardized canister is to provide containment of radioactive materials. In SAR Section 1.5.1.3.1.2.5, the applicant provided the design criteria and design bases for the DOE standardized canisters, with the nuclear safety design bases given in SAR Table 1.5.1-25.

The applicant stated that the canister provides containment when it is subjected to structural challenges, such as the drop of the canister or drop of a load onto the canister, a side impact or collision, drop of a HLW canister onto the DOE standardized canister, and drop of one DOE standardized canister onto another DOE standardized canister. For all of these events, the applicant provided respective design criteria in terms of the maximum effective plastic strain that would result from a structural challenge, then the applicant would determine whether the maximum effective plastic strain would meet the necessary reliability when compared to the DOE standardized canister capacity curve.

The applicant stated that the canisters also provide containment when they are subjected to thermal challenges (i.e., a spectrum of fires). The applicant identified fire as a possible internal initiating event (SAR Section 1.6.3) that might result in an event sequence affecting the canister's structural integrity. SAR Section 1.7.2.3.3 further discussed how the PCSA evaluated the probability of loss of containment (breach) from a fire for the different types of canisters. The applicant stated that the canisters are able to withstand the thermal challenges while being contained within an overpack or a cask. The applicant excluded structural response of bare canisters to fire events outside a waste package or cask based on its evaluation of operational sequences of the facilities, as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.3.1.3. In addition, the DOE standardized canister also is able to withstand a spectrum of fires while placed on a staging rack.

**NRC Staff's Evaluation of the DOE Standardized Canister Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design bases and design criteria of the DOE standardized canister and finds it acceptable because the maximum effective plastic strain as a predictor of ductile material (such as Type 316L stainless steel) failure criterion for evaluating whether loss of containment or breach of a canister has occurred (i) represents the unrecoverable portion of the true strain beyond the yield limit and (ii) is used in fracture mechanics as a standard industry practice (see Shah, et al., 2007aa). Further, the staff concludes in SER Section 2.1.1.4.3.3.1.1 that the applicant's use of maximum effective plastic strain is conservative and, therefore, acceptable.

The NRC staff also finds that the applicant's DOE standardized canister design criteria and design bases are acceptable because they (i) were derived from the PCSA, which identified the DOE standardized canister as an ITS SSC and appropriately linked the design criteria to the nuclear safety design bases and safety function; the NRC staff finds the applicant's identification of ITS SSCs acceptable, as documented in SER Section 2.1.1.6.3.1; and (ii) are consistent with the canister's intended safety function (e.g., maintaining containment integrity when subjected to structural and thermal challenges).

**Design Methods**

In SAR Section 1.5.1.3.1.2.6.1, the applicant provided the overall design methodology used for DOE standardized canister design. The applicant stated that the structural integrity of the DOE standardized canister is relied on to maintain containment for accidental events, such as drops and low-speed collisions during waste handling operations. The applicant stated that the DOE standardized canisters are designed in accordance with 1998 ASME Boiler and Pressure Vessel Code (ASME, 1998aa), which the applicant stated applies to the operating conditions at the GROA (SAR Section 1.5.1.3.1.2.8.1). Since the ASME B&PVC does not specifically address drop conditions, the applicant stated that alternative design methods, such as drop tests and finite element analyses, are used to evaluate the structural behavior of the canister when subject to a drop.

In SAR Section 1.5.1.3.1.2.6.1, the applicant referenced experimental drop tests and corresponding finite element analysis of drop test simulations. A number of full-scale 457-mm [18-in]-diameter standardized canisters were previously tested at the Sandia National Laboratory for the relevant structural challenges, as identified in SAR Table 1.5.1-26. In support of its design methodology, the applicant used these full-scale test results to validate its finite element analysis methodology. SAR Figures 1.5.1-23 through 1.5.1-28 showed the canister deformation obtained in the finite element analyses and the full-scale tests for three drop events of a 457-mm [18-in] diameter standardized canister. The applicant stated that the finite element analysis results of predicted deformations and strains were consistent with the full-scale test results.

**NRC Staff's Evaluation of the DOE Standardized Canister Design Methods**

The NRC staff evaluated the applicant's information on the design specifications and methodology for the DOE standardized canister and finds that the applicant's design methodology used for the DOE standardized canister is adequate because this methodology is consistent with the industry-accepted 1998 ASME Boiler and Pressure Vessel Code; application of this code to the GROA is evaluated further in Table 7-1. In addition, the applicant validated the finite element analytical method for prediction of the structural behavior of the canister by comparison with the physical drop tests results, which were consistent with the analytical predictions. Therefore, the NRC staff finds the applicant's finite element methodology acceptable for predicting the canister response to structural challenges.

**Design and Design Analyses**

The applicant provided the dimensions of the small- and large-diameter DOE standardized canister in SAR Figure 1.5.1-9. The applicant specified that Stainless Steel Type 316L, ASME SA–312 (UNS S31603), are used for the canister shell. The DOE standardized canister design includes a skirt along the circumferential edge on each end of the canister. The applicant stated that this skirt feature is important because it can absorb energy when subjected to an end drop. The applicant also stated that dished heads are located at each end of the canister and are fabricated from Stainless Steel Type 316L, ASME SA–240 (UNS S31600). The applicant stated that the stainless steel materials are annealed and pickled. The applicant further stated that low carbon stainless steel (as indicated by the letter "L" in the Type 316L designation) was selected due to its corrosion resistance.

In SAR Section 1.5.1.3.1.2.8.1, the applicant stated that the following code specifications in ASME (1997ab) apply to the DOE standardized canister design:

- ASME Boiler and Pressure Vessel Code, Section III, Division 3, for design, fabrication, and examination

- ASME Boiler and Pressure Vessel Code, Section V, Article 10, Appendix IV, 1995 Edition with 1997 addenda for leak testing

In SAR Table 1.5.1-27, the applicant presented information on the peak equivalent plastic strains occurring within the containment boundary for specific drop scenarios for the standardized 457-mm [18-in] and 610-mm [24-in]-diameter canisters. For the DOE standardized canisters, the applicant used a through-wall strain limit (i.e., the average strain across the wall thickness) of 48 percent as the failure criteria for the canister materials (Stainless Steel Type 316L, ASME SA–312 [UNS S31603]). The applicant stated that for the 0.6-m [2-ft] drop, 7-m [23-ft] drop, and the puncture drop events, the strains in the DOE standardized canister do not exceed the 48 percent through-wall true strain limit. In addition, SAR Table 1.5.1-27 showed that the midplane strains are less than half of the 48 percent limit for the Stainless Steel Type 316L, ASME SA–312 (UNS S31603) material for all drop events. Based on the finite element analysis results, the applicant concluded that the containment boundary for the 457-mm [18-in] and 610-mm [24-in]-diameter DOE standardized canisters remains intact post the drop events.

**NRC Staff's Evaluation of the DOE Standardized Canister Design and Design Analyses**

The NRC staff evaluated the information provided by the applicant on the design and design analyses for the DOE standardized canister and finds it acceptable because

- The applicant's choice of a low-carbon content stainless steel (i.e., Type 316L SS) for the DOE standardized canister is consistent with industry practice of selecting corrosion-resistant and high-ductile steels. The NRC staff also finds that the proposed design and design analyses for design, fabrication, examination, and leak testing are acceptable because they are based on the ASME Boiler and Pressure Vessel Code

(ASME, 1997ab) Section III, Division 3, which covers containment systems for transportation and storage packaging of spent fuel and high-level radioactive waste. The NRC staff finds this acceptable for use at the GROA, as further discussed in Table 7-1.

- The applicant's use of finite-element analyses to predict structural response to structural challenges is consistent with NRC guidance for storage of canisters containing SNF at NRC license facilities under 10 CFR Part 72, which are similar to the GROA preclosure facilities, as further discussed in Table 7-1.

- The material properties of the Stainless Steel Type 316L, ASME SA–312 (UNS S31603), including the true strain limit of 48 percent strain for drop events is consistent with the material mechanical properties.

2.1.1.7.3.9.3.2 High-Level Radiological Waste Canisters

According to the applicant, the proposed repository would receive HLW from four sources: (i) Hanford Waste Treatment and Immobilization Plant, (ii) Defense Waste Processing Facility at the Savannah River Site, (iii) Idaho National Laboratory, and (iv) West Valley Demonstration Project. The HLW canisters from these four sources were detailed in SAR Section 1.5.1.2.1. SAR Sections 1.5.1.2.1.1 and 1.5.1.2.1.2 provided physical characteristics for the HLW, as well as the HLW canisters. As stated in SAR Tables 1.5.1-15 and 1.5.1-16, the Hanford canister has a diameter of 610 mm [24 in], a length of 4,496 mm [177 in], and an approximate loaded weight of 4,037 kg [8,900 lb]. According to the applicant, the Savannah River Site and Idaho National Laboratory canisters have a diameter of 610 mm [24 in], a length of 2,997 mm [118 in], and an approximate loaded weight of 2,268 kg [5,000 lb]; and the West Valley canister has a diameter of 610 mm [24 in], a length of 2,997 mm [118 in], and an approximate loaded weight of 2,177 kg [4,800 lb]. The applicant indicated that all canisters were fabricated from austenitic Stainless Steel Type 304L (UNS S30400) (SAR Table 1.5.1-16). SAR Section 1.5.1.2.1.5 provided the design criteria and design bases for each type of HLW canister. The applicant stated that the HLW canisters were filled with a molten mixture of HLW and other constituents (e.g., silica sand), which were poured into the HLW canisters, and the canister was sealed once the waste solidified.

**Design Criteria and Design Bases**

The applicant stated that the safety function of the HLW canister is to provide containment of radioactive materials inside them. In SAR Section 1.5.1.2.1.5, the applicant provided the design criteria and design bases for the HLW canisters. The nuclear safety design bases and the design criteria for the HLW canisters were given in SAR Table 1.5.1-17; these were reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6.3.1.

The applicant stated in SAR Section 1.5.1.2.1.5 that the HLW canisters were designed prior to being classified as ITS, and the design considered neither fire nor DBGM–2 for the repository. However, based on the PCSA, the applicant classified HLW canisters as ITS, and the applicant stated that the HLW canister provides containment when it is subjected to GROA structural and thermal challenges. The applicant considered the potential structural challenges for the canister design by conducting full scale testing of these canisters, such as a drop of the canister or drop of a load onto the canister, side impact or collision, a drop of one HLW canister onto another HLW canister, and a drop of a DOE standardized canister onto a HLW canister. For all of these full-scale testing events, the applicant provided design criteria in terms of the maximum effective plastic strain that would result from a structural challenge, which the NRC staff reviewed and found to be acceptable in SER Section 2.1.1.4.3.3.1.1.

The applicant also stated that the HLW canister may also be subjected to thermal challenges (i.e., a spectrum of fires) and included a design criterion for fire events, which is reviewed and found to be acceptable in SER Section 2.1.1.4.3.3.1.3.

**NRC Staff's Evaluation of the HLW Canister Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design bases and design criteria of the HLW canister and finds that the maximum effective plastic strain as a predictor of ductile material (such as Type 316L stainless steel) failure criterion for evaluating whether loss of containment or breach of a canister has occurred is acceptable because (i) the maximum effective plastic strain used by the applicant is consistent with the physical material properties; (ii) the use of inelastic behavior of materials and the failure criterion as the maximum effective plastic strain is consistent with NRC guidance for storage of canisters containing SNF at NRC licensed facilities under 10 CFR Part 72 [NUREG–1536 (NRC, 2010ah)], which are similar to the GROA preclosure facilities, as further discussed in Table 7-1; and (iii) the acceptance criterion is based on full-scale testing of the designed canisters for structural and thermal challenges.

The NRC staff also finds that the applicant's evaluation of the existing HLW canisters for disposal is adequate because (i) the design bases and design criteria are consistent with the canister's intended safety function(s) (e.g., maintaining canister integrity during fires) and (ii) each HLW canister is evaluated in the PCSA for the GROA-specific conditions during operations and challenges from natural phenomena, including seismic events, and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.4.

**Design Methods**

The applicant provided in SAR Section 1.7.2.3.1 the HLW canister design methodology and the analyses basis for loss of containment. In SAR Section 1.7.2.3.1, the applicant stated that several full-scale vertical, top, and corner drop tests from a height of 7 m [23 ft] were performed to evaluate the structural design of these canisters. The applicant stated that for all tests, the HLW canister did not breach. The applicant then applied a Bayesian analysis methodology to estimate the mean and standard deviation of the conditional probability of canister failure given a drop (see SER 2.1.1.4.3.3.1.1 for additional details). The applicant's design methodology classified HLW canisters as ITS (SAR Section 1.5.1.2.1.5) and used the results of full-scale drop tests to estimate the failure probability of the HLW canisters subjected to structural challenges (SAR Section 1.7.2.3.1).

**NRC Staff's Evaluation of the HLW Canister Design Methods**

The NRC staff evaluated the information provided by the applicant on the design methodology for the HLW canister and finds that the applicant's proposed design methodology is acceptable because (i) results of the actual full-scale tests of the existing designs of HLW canisters showed no breach of the HLW canister; and (ii) the applicant used the failure probability of the HLW canister subjected to structural and thermal challenges (e.g., drops at various heights, thermal challenges, and seismic fragility assessment) to support the PCSA for the HLW canister handling operations at the GROA), which are reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.4.

**Design and Design Analyses**

In SAR Table 1.5.1-16, the applicant provided geometric details of the four HLW canisters from Hanford, Idaho National Laboratory, Savannah River, and West Valley. According to the applicant, the HLW canisters have a length of 300–450 cm [118–177 in], a diameter of 61 cm [24 in], and a shell thickness ranging from 0.34 to 0.95 cm [0.13 to 0.37 in]. The applicant stated that the four HLW canisters are constructed of an austenitic stainless steel (Type 304L Stainless Steel) and the HLW canisters are designed to the design codes and standards listed in SAR Table 1.5.1-18. The applicant stated that the canister welding and nondestructive weld evaluation are performed under the guidance of the 2001 ASME Boiler and Pressure Vessel Code (ASME, 2001aa). According to the applicant, the canister welding procedures follow the industry-accepted standards set forth by the 2001 ASME Boiler and Pressure Vessel Code, Section IX (ASME, 2001aa). The applicant also stated that all full penetration butt welds from the Hanford, Idaho National Laboratory, West Valley, and Savannah River Site canisters undergo a nondestructive evaluation examination, per 2001 ASME Boiler and Pressure Vessel Code, Section V (ASME, 2001aa).

**NRC Staff's Evaluation of the HLW Canister Design and Design Analyses**

The NRC staff evaluated the applicant's information on the design of HLW canisters and finds that the applicant's HLW canister design and design analyses is acceptable because (i) the material of construction (Type 304L Stainless Steel) is ductile (hence, impact tolerant) and corrosion resistant due to the low carbon content and (ii) the HLW canister welding, welding procedures, and nondestructive weld evaluations are consistent with the applicable industry codes and standards [2001 ASME Boiler and Pressure Vessel Code, Sections V and IX (ASME 2001aa)], which the NRC staff finds acceptable, as further discussed in Table 7-1.

2.1.1.7.3.9.3.3 Dual-Purpose Canister

In SAR Section 1.5.1.1.1.2.1.2, the applicant discussed the Dual-Purpose Canister (DPC). The applicant stated that the DPC is in use to store commercial SNF (CSNF) licensed under 10 CFR Part 72 at the utility sites and potentially to transport the SNF, under the provisions of 10 CFR Part 71, to the GROA. The applicant has not made a decision to transport SNF in DPCs from the utility sites to the repository. The applicant also stated that the DPC can be placed within a properly designed overpack for aging; however, the applicant also stated that the current DPC design has not been shown to be suitable for disposal. Therefore, the applicant stated that if it decides to transport SNF to the GROA in DPCs, SNF in DPCs would need to be repackaged at the repository site into a TAD canister for disposal at the repository prior to disposal.

**Design Criteria and Design Bases**

The applicant stated that the DPC safety function is to provide containment of radioactive materials. SAR Table 1.5.1-9 presented the nuclear safety design bases for the DPC.

On the basis of the PCSA in SAR Sections 1.6 to 1.9, the applicant classified the DPC as ITS; the NRC staff reviewed this determination in SER Section 2.1.1.6.3.1 and found it to be acceptable. The applicant considered, through numerical analysis, the following potential structural challenges for the DPC design: drop of the canister and/or a load onto the canister and side impact or collision. The applicant provided the respective design criteria in terms of the maximum effective plastic strain that would result from a structural challenge and whether it would meet the necessary reliability when compared to the canister's capacity curve. In addition, the applicant provided design criteria for the DPC to withstand thermal challenges at the GROA (e.g., fires) while contained within an overpack or a cask and within the CTM shield bell; these criteria were reviewed by the NRC staff and found to be acceptable in SER Section 2.1.1.4.3.3.1.3.

**NRC Staff's Evaluation of the DPC Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design bases and design criteria of the DPC and finds that the maximum effective plastic strain as a predictor of ductile material (such as Type 316L stainless steel) failure criterion for evaluating whether loss of containment or breach of a canister has occurred is acceptable because it (i) is based on numerical analysis for structural and thermal challenges; (ii) represents the unrecoverable portion of the true strain beyond the yield limit, consistent with the canister material properties; and (iii) is consistent with NRC guidance for storage of canisters containing SNF at NRC-licensed facilities under 10 CFR Part 72, which are similar to the applicable GROA preclosure facilities, as further discussed in Table 7-1 [NUREG–1536 (NRC, 2010ah)]. The NRC staff also finds that the design criteria and design bases the applicant used are adequate because they are consistent with the canisters' intended safety functions (e.g., maintaining canister integrity during fires), derived from the PCSA and evaluated by the NRC staff in SER Sections 2.1.1.4.3.4 and 2.1.1.7.6.3.1 and found to be acceptable. Also, the NRC staff finds the applicant's design criteria and design bases to be acceptable because the applicant analyzed the representative canister, which considers existing DPC canister designs for GROA-specific conditions. This analysis is reviewed by the NRC staff and found to be acceptable in SER Section 2.1.1.4.3.3.1.1.

**Design Methodology**

The applicant's design methodology for the DPC is identified in SAR Section 1.5.1.1.1.2.1.2. The applicant stated that structural analyses have been performed on the representative or generic canisters (BSC, 2008cp). The applicant derived generic canister geometrical and material properties based on typical DPC and naval canisters. These structural design methodology analyses focused on various canister drop scenarios at differing drop heights and orientations. The applicant used the results in quantifying an estimate of the passive reliability for a generic canister. The applicant stated that it applied finite-element analyses to model structural challenges to representative canisters within a class of canisters that encompasses TAD canisters, naval SNF canisters, and a variety of DPCs (SAR Section 1.7.2.3.1). The applicant stated that prior to the use of any DPC system (including associated overpacks) at the repository, additional analyses (e.g., structural, thermal, and criticality) would be performed, as appropriate to demonstrate compliance with GROA-specific design criteria and applicable nuclear safety design bases if a canister falls outside the values evaluated in the representative canister (SAR Section 1.5.1.1.1.2.1.2). The applicant also stated that analyses show that the local conditions at Yucca Mountain (e.g., temperature, rainfall, and tornado winds) are within the conditions specified in many DPC systems certified under 10 CFR Part 72.

**NRC Staff's Evaluation of the DPC Design Methods**

The NRC staff evaluated the information provided by the applicant on the design methodology for the DPC and finds that the design methodology to be used by the applicant for the DPC is acceptable because the structural analyses the applicant performed for a generic canister were based on geometrical and material properties representative of typical DPC and naval canisters.

The NRC staff also finds that the applicant's approach for evaluating the generic canister capacity to withstand possible repository structural challenges is acceptable because it is consistent with the approach used for the TAD canister, which the NRC staff determined to be acceptable in SER Section 2.1.1.7.3.9.2.

**Design and Design Analyses**

The applicant stated that, currently, DPC systems are licensed for storage at utility sites under 10 CFR Part 72 and for transportation under 10 CFR Part 71. The applicant stated in SAR Section 1.5.1.1.1.2.1.2 that if the selected DPC falls within the design envelope of the representative canister, then the structural analyses based on the representative canister (BSC, 2008cp) will be used to evaluate the structural performance of the DPC. The applicant stated in SAR Section 1.5.1.1.1.2.1.2 that if a DPC design falls outside of the representative canister design envelope, the applicant will perform additional analyses, prior to receipt of DPCs at the GROA, to determine whether the DPC meets GROA-specific nuclear safety design bases and criteria.

**NRC Staff's Evaluation of the DPC Design and Design Analyses**

The NRC staff evaluated the information provided by the applicant on the design and design analyses for the DPC and finds that the selected design approach is acceptable because the approach is consistent with the design approach used for the TAD, which the NRC determined was acceptable in SER Section 2.1.1.7.3.9.2. Like the TAD canister, a DPC would be certified under 10 CFR Part 71 and evaluated for the conditions at the GROA using the PCSA, as described by the applicant in SAR Sections 1.6 to 1.9, and reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.4. In SAR Section 1.5.1.1.1.2.1.2, the applicant stated that it would perform additional structural and criticality analyses, as appropriate, once a specific DPC type is selected for potential shipment of SNF to the GROA, if its design falls outside the evaluation of the generic canister. The NRC staff notes that for those DPCs falling outside the representative canister envelope, additional analyses are needed to determine whether the DPC meets GROA-specific nuclear safety design bases and criteria. Therefore, the NRC staff is proposing a condition on the Construction Authorization that prior to bringing any DPCs onsite, DOE must provide the necessary analysis for NRC review and approval that either (i) confirms that the current PCSA bounds the intended performance of the DPCs at the GROA or (ii) demonstrates, through the PCSA, that the DPCs can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

**Proposed Condition of Construction Authorization [10 CFR 63.32(a)]**

> DOE shall not, without prior NRC review and approval, accept DPCs at the repository.
>
> Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of the DPCs at the GROA or (ii) demonstrates, through the PCSA, that the DPCs can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

2.1.1.7.3.9.3.4 Naval Canister

The applicant stated that Naval SNF canisters are shipped to the repository in either naval short or naval long SNF canisters to accommodate different naval fuel assembly designs. SAR Figure 1.5.1-29 depicted a typical naval SNF canister. According to the applicant, the naval

SNF canister can be described as a circular cylinder with a bottom plate and a top shield plug. The applicant described that the bottom plate is 8.9 cm [3.5 in] thick, the top shield plug is 38.1 cm [15 in] thick, and the canister walls are 2.5 cm [1 in] thick. The applicant stated that the naval short SNF canister's maximum length is 475 cm [187 in] and the naval long SNF canister's maximum length is 538.5 cm [212 in]. The applicant also stated that the maximum outer diameter of the naval SNF canister is 167 cm [66.5 in]. The applicant stated that the maximum external dimensions ensure that the naval SNF canisters fit into the waste packages. The maximum design weight of the loaded long or short naval SNF canister is 44,452 kg [98,000 lb]. The applicant stated that the naval SNF canister is fabricated from a stainless steel that is similar to Stainless Steel Types 316 and 316L (SAR Section 1.5.1.4.1.2.1).

**Design Criteria and Design Bases**

The applicant specified that the safety function of the naval SNF canister is to provide containment of radioactive materials. In SAR Section 1.5.1.4.1.2.5, the applicant provided the design criteria and design bases for a representative canister that was used by the applicant to evaluate the naval canister (SAR Table 1.5.1-30) in the PCSA. The safety functions and reliability in the design basis and design criteria for the representative canister were reviewed and found to be acceptable by the NRC staff in SER Sections 2.1.1.4.3.3.1.1 and 2.1.1.6.3.2.8.6.

The applicant considered a drop of the canister, a drop of a load onto the canister, and a side impact or collision with an object or structure as potential structural challenges for the canister design. For these events, the applicant provided a design criterion in terms of the maximum effective plastic strain that would result from the structural challenge and whether it would meet the necessary reliability when compared to the canister's capacity curve. The applicant stated that it used a design criterion of designing the canister transfer machine (CTM) in accordance with the ASME–NOG–1–2004 (ASME, 2005aa) Type I (single-failure proof) cranes standards to minimize canister drop. The applicant further stated that the naval canister is designed such that the fire-induced failure hazard meets the necessary reliability when evaluated against a spectrum of fires while contained within a transportation cask, within a waste package, and within the CTM shield bell under additional design criteria (Table 1.5.1-30). The applicant stated that a breach of the naval canister (while in the transportation cask, CTM shield bell, or waste package) due to a fire is beyond a Category 2 event sequence (SAR Section 1.5.1.4.1.2.6.1).

**NRC Staff's Evaluation of the Naval Canister Design Criteria and Design Bases**

As documented in SER Section 2.1.1.4.3.3.1.1, the NRC staff finds the applicant's use of the representative canister for evaluating the naval canister appropriate. The NRC staff evaluated the information on the relationship between the design bases and design criteria of the naval canister and finds that the maximum effective plastic strain as a predictor of ductile material (such as Type 316L stainless steel) failure criterion for evaluating whether loss of containment or breach of a canister has occurred is acceptable because it represents the unrecoverable portion of the true strain beyond the yield limit, and as discussed in SER Section 2.1.1.4.3.3.1.1, the applicant's use of maximum effective plastic strain is conservative. The NRC staff's evaluation of the CTM, including the use of ASME–NOG–2004 (ASME, 2005aa) Type I (single-failure proof) cranes standard, is documented in SER Section 2.1.1.7.3.2.1, where the NRC staff finds that the design basis and design criterion for minimizing canister drop are adequate. In addition, the NRC staff concludes in SER Section 2.1.1.4.3.3.1.1 that the applicant's SNF canister failure probabilities, based on the drop analysis, are adequate. The NRC staff also finds that the design criteria and design bases the applicant used for the naval canister are adequate because the safety functions and reliability in the design basis and design criteria for the representative canister were reviewed and found to be acceptable by the NRC staff in SER Sections 2.1.1.4.3.3.1.1 and 2.1.1.6.3.2.8.6.

**Design Methodology**

The applicant's design methodology for the naval SNF canister reliability is determined using a representative canister, which was selected such that it encompasses TAD canisters, DPCs, and naval canisters. The applicant's design methodology evaluated the probability of a representative canister breach for structural and thermal challenges, including fire, an increase in temperature inside a surface facility due to a loss of HVAC cooling, seismic events, a flat bottom drop, collision with an object or structure, and the drop of an object on the canister (SAR Section 1.5.1.4.1.2.6.1). The applicant stated that finite element programs, such as ANSYS, ABAQUS/Explicit® (structural), ABAQUS/Standard® (thermal), and LS-DYNA™ were used to perform structural and thermal analyses of the canisters (SAR Sections 1.5.1.4.1.2.6.2 and 1.5.2.6.1.2).

**NRC Staff's Evaluation of the Naval Canister Design Methods**

The NRC staff documented its evaluation of the applicant's reliability analysis of naval SNF canister in SER Section 2.1.1.4.3.3.1.1, where the NRC staff finds that the applicant's SNF canister failure probabilities are adequate. As a part of this evaluation, the NRC staff evaluated the information provided by the applicant on the design methodology for the naval canister and finds that software programs such as ANSYS, ABAQUS/Explicit® (structural), ABAQUS/Standard® (thermal), and LS–DYNA for structural reliability assessment are appropriate because they are established, commercial finite element software and their usage is consistent with standard engineering practices for the types of analyses the applicant performed.

**Design and Design Analyses**

In SAR Section 1.5.1.4.1.2.8, the applicant stated that the naval canister is designed to the specifications of the 1998 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (ASME, 1998aa), for normal and accident conditions of storage and transportation. The applicant further stated that it used ANSI N14.6–93 (ANSI, 1993aa) to design the lifting features for GROA facilities for the naval SNF canister. The applicant also stated that all naval SNF canister outermost closure is leak tested at the time of canister closure, based on the guidelines of ANSI N14.5–97 (ANSI, 1998aa).

**NRC Staff's Evaluation of the Naval Canister Design and Design Analyses**

The NRC staff evaluated the information the applicant provided on the design and design analyses for the naval canister and finds that the cited codes and standards [ANSI N14.5–97 (ANSI, 1998aa)] to be used for the design, fabrication and testing of the naval SNF canister are appropriate because they are consistent with standard engineering practices in the nuclear industry. Specifically, Subsection NB addresses nuclear pressure vessel material design, fabrication, examination, testing, overpressure relief, marking, stamping, and preparation of reports. The NRC staff finds that this standard is acceptable for use, as proposed by the applicant, and further described in Table 7-1. The NRC staff's evaluation of the special lifting devices [designed to ANSI N14.6–93 (ANSI, 1993aa)] is documented in SER

Section 2.1.1.7.3.4.2 where the NRC staff finds that the design basis and design criteria for minimizing naval SNF canister drop are adequate.

2.1.1.7.3.9.3.5 Aging Overpack and Shielded Transfer Cask

Aging Overpack

In SAR Section 1.2.7.1.3.2, the applicant stated that two types of aging overpacks (AOs) would be used: (i) vertical overpacks for a TAD canister and a DPC and (ii) a horizontal aging module for a DPC. The applicant described that the vertical aging overpack is cylindrical and consists of a metal inner liner surrounded by reinforced concrete sidewalls and a steel outer shell, with a bolted lid on the top, which shields and protects the canister. The applicant stated that the concrete sidewall and the top of the vertical overpack are designed to shield and protect the canister against natural phenomena, such as tornadoes, airborne missiles, ambient-temperature extremes, and earthquakes. The applicant specified that the aging overpacks has a maximum fully loaded weight of 227 metric tons [250 tons], a maximum overpack diameter of 3.7 m [12 ft], and a maximum overpack height of 6.7 m [22 ft]. A drawing of a vertical aging overpack was shown in SAR Figure 1.2.7-6. SAR Section 1.2.7.1.3.2 described the aging overpack system. The horizontal aging module is a reinforced concrete, thick-walled, boxlike structure. The applicant stated that the minimum concrete wall shielding thickness is approximately 0.91 m [3 ft]. The applicant specified that the horizontal aging module has a maximum height of 6.40 m [21 ft], a maximum width of 2.6 m [8.5 ft], and a minimum length of 7.1 m [23.3 ft] with a minimum of 0.9 m [3 ft] concrete shielding. The horizontal aging module is loaded with the DPC at the aging pad. The DPC is inserted into the horizontal aging module cavity through a removable access door in a horizontal position. Once inside the cavity, the DPC is cradled by rails. The NRC staff evaluated the applicant's description of the AOs and found it to be acceptable, as described in SER Section 2.1.1.2.3.5.3.

**Design Criteria and Design Bases**

The applicant stated that the aging overpack system (i.e., either a vertical aging overpack or a horizontal aging module, as appropriate for the canister) protects the CSNF within TAD canisters and DPCs. The applicant stated that the aging overpack is a missile barrier and a radiation shield for the DPCs and TAD canisters within the AO (SAR Section 1.2.7.1.3.2).

The applicant provided the design bases and their relationship to the design criteria for the aging overpack system in SAR Table 1.2.7-1. The applicant stated that the main safety functions of an aging overpack system are (i) personnel protection against direct radiation exposure (horizontal and vertical AO), (ii) protection against structural collapse onto a waste container (horizontal), (iii) protection against sliding of an AO, and (iv) protection against tipover of an AO. The applicant also stated that it would impose a design criterion that the horizontal aging modules be designed in accordance with ACI 349–01/349R–01 (ACI, 2001aa) for (i) loads associated with impact or collision and (ii) loads and accelerations associated with a beyond DBGM–2 seismic event. In addition, the applicant stated that the vertical AO is designed for loads associated with drops and for preventing it from sliding into another AO during a beyond DBGM–2 seismic event. The applicant also stated in SAR Section 1.2.7.1.3.2.1 that the vertical AO must withstand a seismic event with horizontal and vertical peak ground accelerations (PGAs) of 96.52 ft/s$^2$ (3 g) without tipover and without exceeding the canister leakage rate (SAR Section 1.2.7.6.2). According to the applicant, the aging overpack serves the following functions: (i) providing stability (i.e., prevents tipover and cushions the canister for a drop or collision); and (ii) protecting the TAD canisters or DPCs from natural phenomena, so that they can maintain containment of radioactive materials.

**NRC Staff's Evaluation of the AO Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design criteria and design bases for the aging overpack and finds that the design criteria and design bases the applicant developed are appropriate because they accounted for a range of scenarios (e.g., structural collapse, sliding, tipover) derived from the PCSA. The NRC staff evaluated the applicant's analysis of the range of scenarios and resulting design and design criteria and found them to be acceptable, as described in SER Section 2.1.1.4.3.4 and 2.1.1.6.3.1. In addition, the design criteria and design bases are consistent with the intended safety functions of the overpack (e.g., remaining in an upright and freestanding position, post seismic event, with a PGA of 3g). The NRC staff finds the applicant's design criterion that the vertical AO must remain upright and free standing without exceeding the allowable leakage rate of the canister during and post seismic event with horizontal and vertical peak ground accelerations (PGAs) of 96.52 ft/s$^2$ (3 g) acceptable because it corresponds to a probability of exceedance of $10^{-6}$ per year, which was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.2.

**Design Methodology**

The applicant presented the AO design methods in SAR Section 1.2.7.6. In SAR Section 1.2.7.6.2, the applicant stated that before an aging overpack is used at the GROA, the aging overpack system is evaluated for normal handling, dead, thermal loads, and event sequence loads. The applicant indicated that the predicted stresses and leakage rates resulting from these structural and thermal challenges are compared to the allowable stresses in the design code and leakage rates, specified in ANSI N14.5–1997 (ANSI, 1998aa) to determine the acceptability of the aging overpack system. For example, the applicant expressed the maximum leakage rate limits of a $1.5 \times 10^{-12}$ fraction of canister free volume per second (normal) and $9.3 \times 10^{-10}$ fraction of canister free volume per second (off-normal) for a TAD in an AO. In addition, the applicant provided the cladding temperature limits for both normal and off-normal conditions in terms of the TAD canister specifications identified in SAR Section 1.5.1.1.1.2.6.1.1.

As part of the design methodology for the AO, the applicant stated in SAR Section 1.2.7 that the AO system's structural design will be evaluated using fragility assessments, as described in SAR Section 1.7. A corresponding structural analysis of an aging overpack containing an SNF canister was presented in BSC (2008cp). The applicant's structural analysis design methodology focused on impact events including a drop onto an unyielding ground surface and slapdown (subsequent impact) from an upright position. The applicant used these structural analysis results to estimate the failure probability for each of these impact events. The failure probabilities the applicant found for drops and collision in BSC (2008ac, Section 6.3.2) were determined to be $1.0 \times 10^{-8}$ or lower. The applicant, as outlined in BSC (2008ac, Table 6.3.7), used a higher conservative value of $1.0 \times 10^{-5}$ for the failure probability for events involving drops.

**NRC Staff's Evaluation of the AO Design Methods**

The NRC staff evaluated the information provided by the applicant on the design methodology for the aging overpack and finds that, for the TAD aging overpack, the allowable stress, the leakage rate, and cladding temperature limits the applicant specified are acceptable because these limits are consistent with the performance specifications for the vertical TAD aging overpack (DOE, 2008ag).

The NRC staff also reviewed the applicant's approach for evaluating the aging overpack's structural capacity as given in BSC (2008cp). The NRC staff finds that the applicant's structural analysis methodology is adequate because the applicant's use of nonlinear finite element analysis for modeling the drop (impact) is consistent with similar impact analyses for drop events involving SNF canisters (Shah, et al., 2007aa). The NRC staff further finds the applicant's use of a $1.0 \times 10^{-5}$ failure probability acceptable because it provides additional conservatism, as discussed in SER Section 2.1.1.4.3.3.1.1. The NRC staff also finds the maximum leakage rates for normal and off-normal conditions consistent with ANSI N14.5–1997 (ANSI, 1998aa), which the NRC staff finds to be applicable and acceptable for use at the GROA, as described in Table 7-1. TAD configurations during transfer operations involving the TAD and the AO or the waste package were reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.4.

**Design and Design Analyses**

The applicant provided general material specifications for the fabrication of the aging overpack system in SAR Section 1.2.7.1.3.2.1. The applicant stated that the vertical aging overpack is constructed of a metal liner and surrounded by reinforced concrete sidewalls and a concrete lid on top. In SAR Section 1.2.7.1, the applicant also stated that the concrete for the aging overpack would be in conformance with the specifications of ACI 349–01/349R–01 (ACI, 2001aa) and the reinforcing steel would be consistent with ASTM A706/A706M–06a (ASTM International, 2006ac) or ASTM A615/A615M–06a (ASTM International, 2006ad).

The applicant identified the design codes and standards it stated it would use for the aging overpack system design in SAR Section 1.2.7.8. Because the aging overpack system is classified as ITS, the applicant stated in SAR Section 1.2.7.9 that the overpack system is evaluated for normal handling loads, dead loads, thermal loads, and event sequence loads. It is also designed to withstand the natural phenomena loading parameters listed in SAR Table 1.2.2-1.

In SAR Section 1.2.7.8, the applicant stated that the design followed these codes and standards for the aging pads, concrete vertical AO, concrete horizontal aging modules, and reinforcing steel:

- ASCE/SEI 43–05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities (ASCE, 2005aa)

- ACI 349–01/349R–01, Code for Nuclear Safety Related Concrete Structures and Commentary (ACI, 2001aa)

- ANSI/ANS–6.4–1997, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, Appendix A (ANSI, 1997ab)

**NRC Staff's Evaluation of the AO Design and Design Analyses**

The NRC staff evaluated the information provided by the applicant on the design and design analyses for the aging overpack and finds that the applicant's proposed approach for design and design analyses is adequate because (i) the cited design codes and standards used for the design and construction of the aging overpack system are appropriate because they are consistent with standard engineering practices in the nuclear industry, and the NRC staff finds their use acceptable at the GROA, as proposed by the applicant, and further discussed in Table 7-1; (ii) the proposed materials are consistent with the cited codes and standards, which the NRC staff finds acceptable for use at the GROA, as further discussed in Table 7-1; and (iii) the loads considered in the design are consistent with those normal, dead, thermal, and event sequence loadings and with the site-specific characteristics. The NRC staff evaluated the applicant's exclusion of volcano activity hazards (e.g., volcanic ash blocking the AO vent leading to temperature increase inside the AO), weather-related hazards (e.g., tornado-generated missiles), and external fires impacting AOs, and found this to be acceptable, as documented in SER Sections 2.1.1.3.3.1.3.1, 2.1.1.3.3.1.3.2, and 2.1.1.3.3.1.3.5. The NRC staff also evaluated the applicant's reliability analysis of loss of containment and shielding due to structural challenges (e.g., vertical drop, side impact) for the AOs and found it be acceptable, as documented in SER Section 2.1.1.4.3.3.1.1. The NRC staff further evaluated the applicant's reliability analysis of loss of containment and shielding due to thermal challenges for the AOs and found it to be acceptable, as documented in SER Sections 2.1.1.4.3.3.1.3.

Shielded Transfer Cask

The applicant provided information on the shielded transfer casks in SAR Section 1.2.5.4. There were three different types of shielded transfer casks (STCs) proposed: (i) vertical STCs for use in the WHF for handling TAD canisters during loading and canister closure operations (e.g., drying and sealing), (ii) vertical STCs for handling DPCs during opening and unloading operations in the WHF, and (iii) horizontal STCs for moving horizontal DPCs in the horizontal aging modules at the AF to the WHF (SAR Section 1.2.5.4.1). Horizontal STCs would also be used for handling horizontal DPCs during opening and unloading operations in the WHF. A drawing of a representative vertical DPC STC is shown in SAR Figure 1.2.5-76. A drawing of a horizontal STC is shown in SAR Figure 1.2.5-78. STCs provide integral shielding, structural strength, and passive cooling functions (SAR Section 1.2.5.4.1.3). The applicant classified the STCs as ITS (SAR Section 1.2.5.4.3); this conclusion was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.6.3.1.

**Design Criteria and Design Bases**

The applicant provided the design bases and their relationship to the design criteria for the STC in SAR Table 1.2.5-3. The applicant stated that the safety functions of an STC system are (i) containment protection and (ii) protection against direction exposure to personnel. For the containment protection, the applicant imposed the design criteria that the cask and canister are designed such that canister reliability would be more than the specified design values. The applicant defined the reliability for containment as follows: the mean conditional probability of a breach of canister in a sealed cask resulting from a drop, a drop of a load onto the cask, or a side impact or collision shall be less than $1 \times 10^{-5}$ per drop and $1 \times 10^{-8}$ per impact, respectively. For protection against direct exposure, the applicant defined the direct exposure reliability as follows: the mean conditional probability of loss of cask gamma shielding resulting from a drop of cask, a drop of a load onto the cask, or a collision or side impact to a cask shall be less than or equal to $1 \times 10^{-5}$ per drop and $1 \times 10^{-8}$ per impact, respectively.

**NRC Staff's Evaluation of the STC Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design criteria and design bases for the STC and finds that the design criteria and design bases the applicant developed are appropriate because they accounted for the range of scenarios (e.g., drop of an STC or side impact with an STC) derived from the PCSA. The NRC staff evaluated the applicant's PCSA and the reliability analysis and found it acceptable, as described in SER Section 2.1.1.4.3.4.

**Design Methodology**

The applicant presented the STC design methodologies in SAR Section 1.2.5.4.6. The applicant stated that the design of STCs is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (ASME, 2004aa).

**NRC Staff's Evaluation of the STC Design Methods**

The NRC staff evaluated the information provided by the applicant on the design methodology for the STC and finds the applicant's use of ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (ASME, 2004aa) acceptable because Subsection NC was originally developed to address Class 2 components (components that are part of the important to safety core cooling systems) for nuclear power plants. It includes rules for the material, design, fabrication, welding, repair, examination, overpressure relief, marking stamping, acceptance standards, qualification and certification of NDE personnel, and reports by the Certificate Holder. The operations at the geological repository present a lower risk profile compared to the nuclear power plants in terms of pressure, temperature, and radioactivity. Therefore, the NRC staff finds the use of Subsection NC for the STC design acceptable. The acceptability of the ASME Boiler and Pressure Vessel Code, Section III (ASME, 2004aa) to the GROA is further discussed in Table 7-1.

**Design and Design Analyses**

The applicant stated that the materials of construction used in the design of the STCs would be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (ASME, 2004aa) (SAR Section 1.2.5.4.7). In addition, the applicant stated that, in order to limit the probability of a drop or tipover of an STC that results in a radiological release or criticality, a procedural safety control (PSC-6) is included in the WHF operating procedures to provide a warning that the loaded STC containing a TAD canister or a DPC have the lid secured in place with a minimum number of fasteners, such that the stress in the fastener from a drop or tipover is less than the yield strength of the material (SAR Section 1.2.5.1.4, Table 1.9-10).

**NRC Staff's Evaluation of the STC Design and Design Analyses**

The NRC staff evaluated the information provided by the applicant on the design and design analyses for the STC and finds that the applicant's proposed approach for design and design analyses is adequate because the cited design code [ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (ASME, 2004aa)] used for the design and construction of the STCs is appropriate because Subsection NC includes rules for the material, design, fabrication, welding, repair, examination, overpressure relief, marking stamping, acceptance standards, qualification and certification of NDE personnel, and reports by the Certificate Holder. Hence, the NRC staff finds the use of Subsection NC for the STC design acceptable. The acceptability of the ASME Boiler and Pressure Vessel Code, Section III (ASME, 2004aa) to the GROA is further discussed in Table 7-1.

Additionally the NRC staff finds the applicant's inclusion of a procedural safety control (PSC-6) of providing a warning to secure the lids on loaded STCs acceptable because it would reduce the probability of radioactivity release when operating STCs.

2.1.1.7.3.9.3.6 Transportation Cask

The applicant proposed to use transportation casks to transport different categories of waste forms (e.g., TAD, naval SNF, HLW, and DOE SNF canisters) and uncanistered commercial SNF to the repository (SAR Section 1.2.8.4.5.1). The applicant stated that the transportation casks, which are designed and certified under 10 CFR Part 71, are evaluated for compliance with the preclosure safety analysis requirements of 10 CFR Part 63 in SAR Sections 1.6 through 1.9, and were identified as ITS. The NRC staff reviewed and found to be acceptable the applicant's PCSA and identification of the transportation casks as ITS, as described in SER Sections 2.1.1.4.3.2.1 and 2.1.1.6.3.1. The applicant listed nuclear safety design bases and criteria for the transportation casks in SAR Tables 1.2.8-2 and 1.9-2 through 1.9-6.

**Design Criteria and Design Bases**

The applicant described the transportation cask in SAR Section 1.2.8.4.5 and listed the nuclear safety design bases for the transportation cask in SAR Table 1.2.8-2. The applicant stated that the transportation cask serves three safety functions at the repository during operations for removal of canisters and uncanistered SNF from the transportation cask for disposal: (i) to provide containment; (ii) to preclude lid contact with canisters from lid drops; and (iii) to protect personnel from direct exposure (i.e., shielding). The applicant stated that the transportation cask is designed to provide containment when the cask is subject to a drop or a low-speed impact and collision, and the cask containment is evaluated based on canister capacity. The applicant also stated that the geometry of the casks carrying HLW canisters is designed to preclude lid contact with the canisters following a drop of a cask lid and maintain shielding when subjected to a drop or a low-speed impact and collision. The applicant stated in SAR Section 1.2.8.4.5 that the transportation casks would be evaluated for the repository-specific structural and thermal challenges from natural phenomena and environmental conditions identified in SAR Table 1.2.2-1, including the DBGM–2 seismic events.

**NRC Staff's Evaluation of the Transportation Cask Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design bases and design criteria for the transportation cask and finds that the applicant's design criteria and design bases for the transportation cask are adequate because the design bases and design criteria (i) are derived from the PCSA, as reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.2.1; (ii) addresses the relevant safety functions of the transportation cask (i.e., containment of radioactive material from drops, impacts, and collisions; protection of personnel from direct radiation exposure); and (iii) addresses the structural and thermal challenges from natural phenomena, including the seismic events, specific to the GROA.

**Design Methods**

In SAR Section 1.2.8.4.5.6, the applicant provided the design methods used in the design of the transportation casks, including codes and standards. The applicant presented details of the design methodology used to estimate the transportation cask containment capacity to withstand repository structural and thermal challenges in SAR Sections 1.7.2.3.1 and 1.7.2.3.3, respectively, and for the loss of shielding in SAR Sections 1.7.2.3.2 and 1.7.2.3.4, respectively. In BSC (2008cp), the structural analyses were presented for a transportation cask containing a representative SNF canister. The applicant's structural analysis design methodology focused on different drop/impact conditions. The applicant used these structural analyses results to estimate the failure probability of loss of containment to be $10^{-8}$ or lower (BSC, 2008ac, Section 6.3.2)]. However, as outlined in BSC (2008ac, Table 6.3.7), the applicant used a higher conservative value of $10^{-5}$ for the failure probability of loss of containment of transportation cask, which was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.3.1.1.

**NRC Staff's Evaluation of the Transportation Cask Design Methods**

The NRC staff evaluated the applicant's information on the design methods for the transportation cask and finds that the design methods are adequate because the transportation casks are evaluated for structural and thermal challenges. As stated in SER Section 2.1.1.4.3.3.1.1., the NRC staff finds the applicant's use of an artificially high failure probability ($1.0 \times 10^{-5}$ instead of $1.0 \times 10^{-8}$) for its design evaluation to be acceptable because of the additional conservatism associated with the higher failure probability.

The NRC staff reviewed the applicant's approach for evaluating the transportation cask's structural capacity, as given in BSC (2008cp). The NRC staff finds that the applicant's use of the nonlinear finite element analysis for modeling the drop (impact) analyses in the structural analyses is appropriate because it is consistent with standard engineering practice for performing the nonlinear, transient impact analysis (Shah, et al., 2007aa).

**Design and Design Analyses**

The applicant stated that any transportation cask received at the repository would be certified by the NRC under 10 CFR Part 71, based on the codes and standards, materials of construction, and design load combinations used for NRC certification of transportation cask designs. The applicant evaluated a representative transportation cask for structural and thermal challenges at the GROA to demonstrate compliance with 10 CFR Part 63 regulatory requirements, as described in SAR Section 1.6 through 1.9 and 1.7.2.

**NRC Staff's Evaluation of the Transportation Cask Design and Design Analyses**

The NRC staff evaluated the applicant's information on the relationship between the design and design analyses for the transportation cask and finds that the applicant's evaluation of the performance of a representative transportation cask design certified under 10 CFR Part 71 is acceptable for use at the GROA because (i) performance of the cask was evaluated for structural and thermal challenges specific to the GROA; (ii) this evaluation was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.4.3.3.1.1; and (iii) applicability of the analyses of a representative cask to the transportation casks received at the repository will be verified by the applicant under the administrative programs described in SAR Section 5.10, prior to transportation cask use at the GROA.

**NRC Staff's Conclusion of the Canisters and Overpacks Design**

Based on the NRC staff's evaluation of the applicant's information on the canisters and overpacks designs described above, and the proposed conditions of construction authorization, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the description of the canisters and overpacks designs adequately (i) provides information on materials of construction, dimensions, proposed codes and standards, analytical and design methods; (ii) defines relationship between design criteria and performance objectives; and (iii) identifies the relationship between the design bases and the design criteria.

**2.1.1.7.3.10 Criticality Prevention and Shielding Systems**

This section contains the NRC staff's review of the design of ITS systems to prevent and control criticality and provide shielding. The applicant provided this information in SAR Sections 1.14, 1.2.1, 1.2.3 to 1.2.8, 1.9, and 1.10.3. The NRC staff's review focused on the applicant's design bases and design criteria, design methodology, and design analysis.

2.1.1.7.3.10.1 Criticality Prevention

The applicant provided design information for the ITS features for prevention and control of nuclear criticality. The objective of the NRC staff's review is to verify the design of criticality prevention and control features.

In SAR Section 1.14, the applicant described how its criticality safety program prevents and controls criticality during the preclosure period. The applicant stated that its criticality safety program includes the analysis and design of SSCs, which were performed in conjunction with the PCSA, to ensure that during normal operations and potential Category 1 and 2 event sequences, the calculated effective neutron multiplication factor, $k_{eff}$, does not exceed the design basis value of the Upper Subcritical Limit (USL). The applicant used a USL of 0.93 for CSNF and 0.89 for DOE SNF. This included an administrative margin of 0.05 (SAR Section 1.14.2.3.4.1). In SAR Section 1.7.5, the applicant stated that no Category 1 or 2 event sequences important to criticality were identified, which the NRC staff reviewed and found to be acceptable in SER Section 2.1.1.4.3.4.1. The applicant relied on the use of passive design features (e.g., physical barriers against introduction of moderation), engineered design features, and procedural safety controls to screen out criticality. The applicant listed the ITS SSCs and procedural safety controls (PSCs) relied on to prevent criticality in SAR Section 1.9.

The applicant described its criticality safety analysis process in SAR Section 1.14.2.2. The applicant's analysis of preclosure criticality prevention considered how the canister designs, facility designs and characteristics, as well as operations, affect criticality control parameters. The parameters considered important to criticality are waste form characteristics, moderation, neutron absorbers, geometry, neutron interaction, and neutron reflection. The applicant's criticality analyses evaluated changes to these parameters in sensitivity studies to provide input to the PCSA.

The applicant's program included criticality safety requirements, analysis process, and evaluation results based on the expected operations. The applicant stated that prior to the issuance of a license to receive and possess waste, the existing criticality safety design organization would be expanded to include operational components responsible for development and implementation of administrative practices, procedures, and training for nuclear criticality safety (SAR Section 1.14). This includes audits and assessments to ensure the criticality safety related ITS SSCs and PSCs are able to perform their safety function(s), as well as conducting evaluations to demonstrate that actual designs and fuel characteristics comply with the criticality safety requirements.

**Design Criteria and Design Bases**

The applicant classified all the canisters to be handled at the GROA as ITS. The applicant's criticality-related design basis provides containment, which prevents the introduction of a neutron moderator into the canisters. The applicant stated that, for the purposes of criticality safety, a moderator (e.g., water, oil) is assumed to be able to enter a breached canister (SAR Section 1.14.2.3.4.1).

The applicant stated that cranes and other lifting devices have design bases and design criteria to prevent canister breach, by reducing the likelihood of such incidents. Some cranes were classified as ITS because of design bases that help control moderators, such as limiting the mean probability of inadvertent introduction of an oil moderator into a canister following a canister breach. To meet this design basis, the applicant uses design criteria where cranes have double retention capability in the areas of the crane where leaked oil could enter a breached canister (SAR Table 1.2.4-4).

According to the applicant, if a canister is breached, water from the fire protection systems is one of the main sources that could introduce moderation into the canister. Thus, the applicant has design bases and design criteria in place to limit the probability of water being introduced from the fire protection systems into a breached canister.

To prevent a canister from being crushed by a closing slide gate or equipment shield door, the applicant uses a design criterion to provide for the force of the closing slide gates to be power limited so they are incapable of breaching a canister or sever the hoisting ropes causing a drop. Interlocks and obstruction sensors are also used to protect the slide gates from opening when the shield doors are not closed. The design bases and criteria were given in SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3 for all the slide gates used for handling SNF or HLW.

The applicant also included in SAR Table 1.2.4-4 the design bases and design criteria for DOE Canister Staging Racks to prevent criticality. The applicant specified that the DOE Canister Staging Racks would be designed in accordance with the applicable provisions of ANSI/AISC N690-1994 to limit the loss of spacing between the surface of adjacent DOE standardized canisters due to the spectrum of seismic events.

**NRC Staff's Evaluation of the Criticality Prevention Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the design bases and design criteria for criticality prevention and finds that the design bases and design criteria for preventing criticality are adequate because (i) the design bases and design criteria addressed the relevant safety functions for moderator exclusion relied on for criticality prevention (i.e., preventing damage to the canisters that could allow a moderator to enter the canister and providing the means to limit the presence of moderators, such as water and oil, if a canister were breached). The NRC staff evaluates the canisters and overpacks to be used in the GROA in SER Section 2.1.1.7.3.9, where their design was found to be acceptable; (ii) the method of preventing the introduction of moderators through the use of multiple barriers is consistent with ANSI/ANS–8.22–1997

(ANS, 1997ac), which NRC endorsed in Regulatory Guide 3.71 (NRC, 2010ai). The NRC staff evaluates the cranes and other lifting devices in SER Sections 2.1.1.7.3.4.1 and 2.1.1.7.3.4.2, where their design was found to be acceptable because of the consideration of breaches following a drop that could provide a pathway for moderator to be introduced into a canister; (iii) the design minimizes the introduction of fire protection water through the design of the fire protection systems. The NRC staff's evaluation of the fire protection systems is discussed in SER Section 2.1.1.7.3.8, where their design was found to be acceptable; (iv) the design limits the power of the gates and uses interlocks to prevent canister breach, in order to prevent the introduction of moderator; this is consistent with ANSI/ANS–8.22–1997 (ANS, 1997ac). The applicability of ANSI/ANS–8.22–1997 to the GROA is discussed in Table 7-1. The NRC staff evaluates ITS interlocks in SER Section 2.1.1.7.3.7, where their design was found to be acceptable; and (v) the criticality related design bases and design criteria for the DOE Canister Staging Racks were adequately defined because the only criticality related function of the DOE Canister Staging Racks is to control spacing to more than 30 cm, which is the value used in the design basis of the DOE Canister Staging Racks. The NRC staff evaluates the staging rack design in SER Section 2.1.1.7.3.4.3, where it is found to be acceptable.

**Design Methods**

The applicant's design methodology screened potential criticality events beyond Category 2 through controlling criticality parameters discussed in SAR Section 1.14.2.3.2. The applicant stated that criticality is prevented through a combination of ITS SSCs and PSCs. The applicant did not analyze potential criticality dose consequences, because criticality was excluded as beyond Category 2.

For dry handling in the surface and subsurface facilities, the applicant's design methodology relies on moderator control to prevent criticality. The moderator that is most common and available in significant quantities is water. The applicant controls moderator introduction by (i) keeping all moderators out of areas where canisters are handled and (ii) ensuring that the canisters are not breached in a drop or other accident to prevent a moderator from contacting the waste form. As part of its approach, the applicant's design methodology identifies SSCs as ITS if they are relied on to exclude a moderator from entering a breached canister or to prevent a canister breach.

In the WHF pool, the applicant's design methodology relies on neutron absorbers to control criticality. For wet handling operations, the applicant's design methodology applies 2,500 mg/L [0.02 lb/gal] of soluble boron enriched to 90 wt% B-10 as the criticality control parameter. The applicant selected the chemical form of the neutron absorber to be orthoboric acid ($H_3BO_3$), which is injected into the water in the pool and in the transportation cask and DPC fill water. To ensure the presence of enough enriched boron, the applicant developed PSC-9, which requires operators to check the boron concentration and enrichment periodically. In SAR Section 1.2.5.1.4, the applicant stated that the operating procedures would also require sampling following events that could significantly affect the concentration of the boron in the pool.

**NRC Staff's Evaluation of the Criticality Prevention Design Methods**

The NRC staff evaluated the applicant's information on the design methods for ITS SSCs used to prevent criticality, as discussed in SAR Section 1.14.2.3.2 and responses to the NRC staff RAIs. The NRC staff compared the applicant's design methodology for dry handling with ANSI/ANS–8.21–1995 and ANSI/ANS–8.22–1997 (ANS, 1995aa; ANS, 1997ac), which the

NRC endorsed in Regulatory Guide 3.71 (NRC, 2005ac). The NRC staff finds that the applicant's design methodology for dry handling to prevent criticality (e.g., excluding any moderator in the areas where canisters are handled and preventing canister breach) is adequate because it is consistent with ANSI/ANS–8.22–1997 (ANS, 1997ac), and is consistent with the standard criticality control practice in nuclear facilities. The NRC staff also finds that the applicant's use of soluble neutron absorbers for the WHF pool is acceptable because it is consistent with ANSI/ANS–8.14–2004 (ANS, 2004aa) and consistent with the standard criticality control practice in nuclear reactor spent fuel pools, which are similar to the WHF pool. These ANSI/ANS–8 nuclear criticality standards are applicable to the GROA, as discussed in Table 7-1, and are used consistent with the design criteria and design bases of YMRP (NRC, 2003aa). The NRC staff also reviewed PSC-9 and the WHF spent fuel pool in SER Section 2.1.1.3.3.2.6, where they were found to be adequately described and evaluated.

**Design and Design Analyses**

Code Validation

The applicant used the Monte Carlo N-Particle Transport (MCNP) code models and associated Evaluated Nuclear Data File (ENDF) V and VI neutron cross section libraries to determine the SNF $k_{eff}$. The ability of the models to calculate $k_{eff}$ was validated and documented in BSC (2008ce,cf; BSC, 2003ai; BSC, 2002ac).

For CSNF, the applicant specified the range of applicability (ROA) for six parameters represented in the benchmark experiments against which the model was checked in BSC (2008cf, Table 34). In this table, the applicant also provided the values and ranges for the CSNF models. Based on the MCNP results, the applicant determined the critical limit for CSNF is 0.988, which was rounded down to 0.98.

For the DOE SNF, the applicant documented its analysis of benchmark experiments in BSC (2008ce, BSC, 2003ai, BSC, 2002ac). In BSC (2003ai), the applicant recorded how it calculated critical limits for the different groups of DOE SNF. In BSC (2003ai, Table 6-43) the applicant summarized the calculated critical limit values and equations. The applicant calculated the critical limits using the $k_{eff}$s of the benchmarks that applied to the fuel type. In BSC (2002ac), the applicant described the benchmark experiments and provided tables containing the calculated $k_{eff}$s of the benchmarks and the $k_{eff}$s calculated by the MNCP models of the benchmarks for each configuration of the DOE SNF. The ROA analysis was also documented in BSC (2002ac), where the applicant concluded that the benchmarks apply to the different configurations of the DOE SNF. In BSC (2008ce), the applicant updated the material compositions it used. In BSC (2008ce, Table 7-2), the applicant listed the updated bias and bias uncertainty for the DOE SNF groups. Based on MCNP results, the applicant determined the critical limit for all DOE SNF is 0.948, which was rounded down to 0.94, as described in BSC (2008ba, Section 2.3.10). In SAR Section 1.14.2.3.4, the applicant subtracted an administrative margin of 0.05 from the critical limits to get a USL of 0.93 for CSNF and 0.89 for the DOE SNF.

Dry Handling

The applicant described the general characteristics of the canisters used in the GROA in SAR Section 1.5.1. Outside the WHF pool, the applicant relies on moderator and interaction control to prevent criticality. The applicant used MNCP models and the associated neutron cross-section libraries to determine whether the $k_{eff}$ of a modeled configuration of SNF exceeded the USL. Configurations with a $k_{eff}$ that exceeded the USL were considered critical.

The applicant relies on the moderator control provided by the TAD canisters' and DPCs' containment boundary for preventing a breach and subsequent moderator introduction, with canister internals providing defense in depth. In SAR Section 1.14.2.3.1.5, the applicant discussed the criticality potential of the DOE standardized SNF canisters. The applicant described the eight combinations of canister, basket, and representative fuel that it used to evaluate DOE SNF criticality in BSC (2008ba, Section 2.3.1.1.2). The DOE standardized SNF canisters' ITS containment boundaries are relied on to prevent criticality by providing moderator control (SAR Section 1.5.1.3.1.2.5.2). However, even with moderator control, the interaction of enough DOE SNF canisters that are close to each other can result in a criticality. BSC (2008ba, Figure 61) presented the results of an analysis that showed the change in $k_{eff}$ caused by changing the distance between an infinite array of the DOE SNF canisters containing the most reactive type of SNF. Based on these results, the applicant concluded that the minimum canister spacing that would ensure subcriticality is 30 cm [12 in]. The DOE canister staging racks are ITS steel structures in the CRCF that hold the HLW and DOE SNF canisters for staging purposes (SAR Section 1.2.4.2.2.1.3) and maintain the spacing between canisters greater than 60 cm [24 in], as outlined in BSC (2008ba, Section 2.3.1.3.4). The applicant's model of interaction between DOE SNF canisters was discussed in BSC (2008ba, Sections 2.3.1.3.4 and 2.3.2.3.4).

The HLW containers are used for the vitrified (glass) waste. The applicant stated that individual HLW canisters (i.e., canisters holding glass made from radioactive liquid solutions) are subcritical as per ANSI/ANS–8.1–1998 Table 1 (ANS, 2007aa) due to their low concentrations of fissile isotopes, as detailed in BSC (2008ba, Section 2.3.1.1.3).

For the naval SNF canisters, the applicant stated that criticality is considered to be controlled for the naval SNF canisters during the preclosure period because the probability of a naval canister being breached is beyond Category 2 (SAR Sections 1.5.1.4.1.2.5.2, 1.7.5.1, and SAR Table 1.7-7).

Wet Handling

In the WHF pool, another type of ITS staging rack is used. This is a submerged SNF staging rack to be used to hold PWR and BWR assemblies. The applicant uses control of spacing and control of boron concentration for subcriticality. The SNF assembly staging racks contain fixed neutron absorbers that the applicant stated were designed in accordance with ANSI/ANS–8.21–1995 (ANS, 1995aa) (SAR Section 1.2.5.2.2.1.3). The applicant provided details about the staging racks in BSC (2008ba, Sections 2.3.1.3.1 and 2.3.1.3.2). The fixed neutron absorber to be used in the staging racks are Boral (BSC, 2008ba). The applicant stated that it does not rely on the fixed absorber to control criticality. The applicant uses the non-ITS fixed absorber as defense in depth. The SNF canisters, such as the TAD canister, also contain non-ITS fixed absorbers that provide defense in depth against a criticality event.

The applicant used the MCNP software to model the configuration that would result from event sequences in which the staging racks are damaged by omitting the fixed neutron absorber, having the fuel pins in the most reactive spacing, and modeling the staging rack's flux traps as collapsed (decreasing spacing). The applicant found that this scenario required 30 percent of the soluble boron concentration proposed for the WHF pool to prevent criticality (SAR Section 1.14.2.3.2.2.4).

The applicant considered potential event sequences that would result in the interaction of a single assembly with the staging racks or shielded transfer casks containing TAD canisters or DPCs. The applicant's analysis found them to remain subcritical while crediting no more than 15 percent of the minimum stated soluble boron concentration (SAR Section 1.14.2.3.2.2.4). The applicant considered event sequences concerning drops and earthquakes during transfer operations into or out of the WHF pool that could modify the system geometry. The applicant stated that criticality could be prevented even with the canister baskets and the fixed neutron absorber omitted (SAR Section 1.14.2.3.2.2.4).

The applicant listed criticality-related PSCs in SAR Table 1.9-10, along with the basis for each of the PSCs. In the WHF, PSC-6 and PSC-9 are relied upon to prevent criticality. The applicant uses PSC–9 to control soluble absorber concentration through controlling operation of the boric acid makeup subsystem. The subsystem works by mixing dry boric acid with deionized water while agitating and heating the mixture solution. The solution is pumped into the pool to maintain the boron concentration. The water in the pool is sampled and analyzed on a regular basis to monitor boron concentration (SAR Section 1.2.5.3.2.2). Additionally, the applicant stated that it relies on PSC–6 to control neutron interaction by preventing assemblies from falling out of a cask if the cask tipped over into the pool.

**NRC Staff's Evaluation of the Criticality Prevention Design and Design Analyses**

The NRC staff evaluated the applicant's information on the criticality prevention design features for meeting each of the nuclear safety design bases and design criteria using the guidance in Regulatory Guide 3.71 (NRC, 2010ai).

In the area of code validation, the NRC staff evaluated the applicant's MCNP software validation information documented in BSC (2008ce,cf; BSC, 2003ai; BSC, 2002ac) and finds that use of the MCNP software is appropriate for criticality analysis because MCNP is used as a standard computer code in NRC guidance [NUREG–1536 (NRC,2010ah) Section 7.5.4.1 and NUREG–1617 (NRC, 2000aj) Section 6.5.3.3]. The NRC staff also finds the applicant's use of the ENDF V and VI neutron cross-section libraries to be appropriate because the data from these libraries are also incorporated in NRC guidance [NUREG–1536 (NRC,2010ah) Section 7.5.4.1]. Furthermore, the NRC staff finds that the applicant's use of the methodology for MCNP model validation specified in ANSI/ANS–8.1–1998 (ANS, 2007aa) and endorsed in Regulatory Guide 3.71 (NRC, 2010ai) is adequate because the applicant detailed the validation that showed the guidance of the standard was appropriately used. The NRC staff also finds the applicant's treatment of the bias and range of applicability (ROA) of the benchmarks acceptable because it is consistent with ANSI/ANS–8.1–1998 (ANS, 2007aa), which is applicable to the GROA, as discussed in Table 7-1.

While most of the parameters for CSNF are within the ROA of the benchmarks, the NRC staff finds some discrepancies exist between the CSNF models and the benchmarks. The discrepancies include different energy spectra and more soluble boron in the CSNF models than in any of the benchmarks. The NRC staff evaluated these differences and determines that the larger amount of B–10 in the CSNF models would absorb more of the thermal neutrons contributing to the model's energy spectrum because the models have more intermediate and fast neutrons than the benchmark experiments. Therefore, the NRC staff determines that the benchmarks provide acceptable validation of the applicant's CSNF models because the influence of the differences in the energy spectrum on reactivity are insignificant compared to the decrease in reactivity that the larger amount of B–10 causes.

The NRC staff finds that the use of an administrative margin of 0.05 is acceptable because it is consistent with standard criticality safety practice [NUREG–1617, Section 6.4.3, (NRC, 2000aj)], and the applicant's calculated USLs are acceptable for preclosure because this margin of safety was incorporated into those USLs.

For canister dry handling, the NRC staff finds that designing the canisters with a low probability of breach is an acceptable method of maintaining moderator control because it is consistent with ANSI/ANS–8.22–1997 (ANS, 1997ac). The NRC staff evaluated the applicant's modeling of a closely packed array of four DOE SNF canisters and finds that it is acceptable because the analysis used assumptions that resulted in an increase in the calculated $k_{eff}$ and the canisters remained subcritical even with an unrealistically conservative reflector (lead) and the most reactive type of DOE SNF. The NRC staff finds that the applicant's DOE canister staging rack design adequately controls interaction between DOE canisters, because the spacing between canisters will be greater than 60 cm [24 in], as outlined in BSC (2008ba, Section 2.3.1.3.4), whereas {30 cm [12 in]} is the minimum canister spacing that prevents criticality. The NRC staff evaluates the applicant's screening analyses for criticality-initiating events, including interaction, in SER Section 2.1.1.3.3.6, where they were found to be acceptable.

The NRC staff finds that the individual HLW canister design is subcritical because the fissile isotope concentrations listed in SAR Table 1.14-1 are far below the limits from ANSI/ANS–8.1–1998 (ANS, 2007aa, Table 1) and the lower fissile isotope concentrations in the HLW along with the conservative assumptions built into the limits in this standard provide an adequate margin of safety.

The NRC staff finds that criticality resulting from the interaction of multiple naval canisters is prevented by the IHF design because the canisters cannot fit next to each other, as discussed in SER Section 2.1.1.3.3.6. The NRC staff reviewed the applicant's event sequences for canister and cask handling operations at surface facilities (which includes naval canisters in the IHF) in SER Section 2.1.1.4.3.2.1.1, and event sequence quantification and categorization in SER Section 2.1.1.4.3.4, and found the applicant's conclusion that canister breaches are beyond Category 2 to be acceptable. The NRC staff also concluded in SER Section 2.1.1.3.3.2.6 that criticality-related initiating events for naval canisters at the IHF were appropriately excluded from the applicant's PCSA. The NRC staff's review of the naval canisters is in SER Section 2.1.1.7.3.9.4, where their design was found to be acceptable.

For the Wet Handling Facility, the NRC staff finds that the design and design analyses for the prevention of criticality are adequate because (i) the applicant used the design methods and practices provided in ANSI/AISC N690–1994 (AISC, 1994aa) to design the staging racks, as discussed in SER Section 2.1.1.7.3.4.3; (ii) the design analysis used the MCNP software that is consistent with NRC guidance [Section 7.5.4.1 of NUREG–1536 (NRC,2010ah) and Section 6.5.3.3 of NUREG–1617 (NRC, 2000aj)]; and (iii) the design analysis considered the relevant factors that affect criticality.

Additionally, the NRC staff performed a confirmatory calculation of the PWR staging racks for a Westinghouse 17 × 17 assembly using the SCALE 5.1 computer code. This calculation modeled a nominal case {75 percent Boral credit, 2,500 mg/L [0.02 lb/gal] of 90 wt% B-10, 51 mm [2 in] flux traps, and fresh fuel}, and the results indicate a subcritical condition for a Westinghouse 17 × 17 assembly, as is the case where the applicant modeled Boral as replaced with steel and the flux-traps were modeled as collapsed. The NRC staff finds that the applicant acceptably concluded that there are no Category 1 or Category 2 event sequences that would result in soluble boron dilution to a concentration insufficient to maintain subcriticality

(SAR Tables 1.7-13 and 1.7-14). Thus, the NRC staff concluded that the applicant acceptably excluded boron dilution as an initiating event (SAR Table 1.7-1), as discussed in SER Sections 2.1.1.3.3.2.5 and 2.1.1.3.3.2.6.

The fixed neutron absorbers in the staging racks are not ITS SSCs, but provide a defense-in-depth function. The NRC staff also evaluated the interaction of a single assembly with the staging racks or shielded transfer casks through confirmatory calculations using SCALE 5.1. The NRC staff's independent confirmatory analysis modeled both a Westinghouse 17 × 17 assembly and a B&W 15 × 15 assembly submerged in borated water with 2,500 mg/L [0.02 lb/gal] of boron enriched to 90 wt% B-10. The modeling results show that both models are subcritical.

The NRC staff finds that the applicant's approach for preventing criticality during Category 1 and 2 event sequences is adequate because the approach relies on the boron concentration and enrichment through the boron makeup system and PSC-9, a sufficient amount of boron in the WHF pool, the proposed administrative margin, and the applicant's conservative fresh fuel assumption for criticality modeling purposes. Further, the use of PSC–6 adds more margin to the operation by preventing assemblies from falling out of a cask if the cask tips over into the pool.

Based on the evaluation of the applicant's information on the criticality prevention described above, the NRC staff determines that the criticality design and design analyses are adequate because the criticality event is prevented in both the dry handling and wet handling operations. In addition, the NRC staff determines that the applicant's use of the criticality code (MCNP software) is adequate and the applicant's analyses are reasonably performed because (i) the analyses included the relevant factors that relate to criticality and (ii) the models and codes used were appropriate for the conditions analyzed. Therefore, the NRC staff concludes that the design features needed for criticality prevention during dry and wet handling of spent nuclear fuel at the GROA has been addressed adequately.

2.1.1.7.3.10.2 Shielding Systems

The applicant provided design information on the shielding features to be used at the GROA in SAR Sections 1.2.1 to 1.2.8, 1.9, and 1.10.3. Shielding features include concrete walls; floors and ceilings of the surface facilities (IHF, CRCF, WHF, and RF); shield doors; slide gates in concrete floors; CTMs; waste package trolleys; and penetration designs to allow items, such as piping, HVAC ducts, and electrical raceways, to pass through walls or floors. The applicant stated that the shielding features are designed to reduce worker dose from radiation sources, in conjunction with a program of controlled personnel access to, and occupancy of, restricted areas, to levels that are ALARA to comply with 10 CFR Part 20. Radiation sources can be found in transportation casks, waste canisters, aging overpacks, shielded transfer casks, and waste packages (SAR Figure 1.10-18). The applicant stated that the shielding equipment layout and design are consistent with the Regulatory Guide 8.8 Regulatory Position 2, which includes recommendations on minimizing worker time for maintenance and inspection, use of radiation damage-resistant materials in high radiation areas, use of radiation shielding, radiation monitoring systems, control of airborne contaminants and gaseous radiation sources, and a radiation protection program (SAR Section 5.11).

The applicant stated that the ITS shielding SSCs are those features that were credited in the PCSA for reducing the mean frequency of inadvertent exposure of personnel to below the Category 1 events sequence mean frequency, and include (i) shield doors, slide gates, transportation casks, and CTMs in the IHF, CRCF, WHF, and RF; (ii) intrasite operations, aging overpacks, and horizontal aging modules; and (iii) TEV subsurface operations. The applicant stated that the shielding features for the ALARA program are non-ITS. The NRC staff's evaluation of the applicant's shielding is documented in SER Section 2.1.1.6.3.2.3, where the NRC staff concludes that the applicant's PCSA adequately included consideration of suitable shielding.

**Design Criteria and Design Bases**

The applicant provided design bases of the shielding features and their relationship to the design criteria in SAR Sections 1.2.1 to 1.2.8. These features are relied upon to protect against direct exposure to personnel. Shielding design considerations provided the bases for the shielding evaluation of the various facility areas and the radiation zones established for the facility areas. The applicant stated that it designed concrete shielding in accordance with ANSI/ANS–6.4–2006 (ANS, 2006aa, Table 5.2) standard and ACI–349–01/349R–01 (ACI, 2001aa) code.

The applicant stated that the equipment and personnel shield doors are interlocked with associated equipment and personnel shield doors or transfer slide gates (complementary shielding) to prevent them from inadvertent opening when the complementary shielding is not closed, and the doors are interlocked to radiation monitors (SAR Table 1.2.4-4). Similarly, the slide gates are interlocked to prevent inadvertent opening unless the CTM is in place with its shield skirt lowered. The waste package and cask port slide gates are also interlocked to prevent inadvertent opening when complementary shielding is not closed. Transportation casks, aging overpacks, and horizontal aging modules are designed to withstand drops or impacts and collisions, as appropriate, to ensure that shielding remains intact. For the CTM, interlocks are used to prevent inadvertent opening of the slide gate and shield skirt. For the TEV, interlocks are used to prevent the front shield doors from opening during movement between the surface handling facility and emplacement drift turnouts. Additionally, the TEV shielded enclosure is constructed of nonflammable materials to limit radiation exposure from the waste package (SAR Table 1.3.3-6).

**NRC Staff's Evaluation of the Shielding Systems Design Criteria and Design Bases**

The NRC staff evaluated the applicant's information on the relationship between the design bases and design criteria for the shielding systems. On the basis of the review, the NRC staff finds that the design bases and criteria that the applicant proposed for use at the GROA are adequately identified for shielding design because (i) the design criteria and the design bases use standard industry guidance ANSI/ANS–6.4–2006 (ANS, 2006aa, Table 5.2), ACI–349–01/349R–01 (ACI, 2001aa), and NRC guidance in Regulatory Guide 8.8; (ii) design bases and criteria are derived from specific site facilities layout and operations (specific radiation zones are identified based on the radiation sources in the each area of the facilities); (iii) access controls are established to ensure that the exposure to radiation workers are within the ALARA requirements; (iv) the ITS interlocks are used to prevent inadvertent opening of doors with the use of complementary shielding; and (v) the use of a shielded enclosure for the TEV to limit radiation exposure to personnel.

**Design Methods**

The applicant stated that the design methodology for shielding systems identified the primary material as Type 04 concrete with a bulk density of 2.35 g/cm$^3$ [147 lb/ft$^3$]. This is based on

ANSI/ANS–6.4–2006 (ANS, 2006aa, Table 5.2). The applicant stated that the design of concrete used for shielding is in accordance with ACI–349–01/349R–01 (ACI, 2001aa).

The applicant's design methodology used radiation sources (summarized in SAR Figure 1.10-18) and bounding terms (described in SAR Section 1.10.3.4) to approximate the geometry and physical condition of sources in the various repository facilities. The applicant's design methodology then used the flux-to-dose-rate conversion factors taken from ANSI/ANS–6.1.1–1977 (ANS, 1977aa) to develop dose rates. The applicant stated that its design methodology includes performing shielding analyses using standard industry accepted methods and computer codes such as MCNP (Briesmeister, J.F., 1997aa) and SCALE (Oak Ridge National Laboratory, 2000aa). The applicant stated that the shielding source terms for the GROA facilities are based on bounding source terms (described in SAR Section 1.10.3.4). The applicant documented the shielding evaluation results for various areas and components in SAR Tables 1.10-35 through 1.10-46.

**NRC Staff's Evaluation of the Shielding Systems Design Methods**

The NRC staff evaluated the applicant's information on the design methodology for the shielding systems using the design recommendations of Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff finds that the applicant's design methodology for shielding ITS components is adequate because the proposed design methods use standard industry practice, as described in ANSI/ANS–6.4–2006 (ANS, 2006aa, Table 5.2); ACI–349–01/349R–01 (ACI, 2001aa); and ANSI/ANS–6.1.1–1977 (ANS, 1977aa). The applicant's use of ANSI/ANS–6.1.1–1977 (ANS, 1977aa) standard for flux-to-dose-rate conversion factors, and the use of this version of the standard, instead of the later version of the standard, ANSI/ANS–6.1.1–1991 (ANS, 1991aa), is evaluated by the NRC staff in SER Section 2.1.1.5.3.4 and found to be conservative and acceptable because it results in a higher estimate of personnel exposures than would be calculated from ANSI/ANS–6.1.1–1991. The NRC staff evaluates the overall shielding design methodology (same for both ITS and non-ITS shielding) in SER Section 2.1.1.8.3.3 and finds the methodology to be acceptable. Additionally, the NRC staff finds that the proposed design methodology is adequate because the applicant used the industry accepted methods and computer codes (MCNP and SCALE) appropriate for radiation types, sources, facility geometry, and materials at the GROA. Therefore, the NRC staff finds that the applicant's shielding design methodology is adequate and is consistent with the referenced standards and codes, which the NRC staff finds appropriate for use at the GROA, as further discussed in Table 7-1.

**Design and Design Analyses**

The applicant stated that the safety function of ITS shielding is to protect personnel from direct radiation exposure. The applicant discussed the analysis and design of the shielding features, including the calculation methodology, computer codes, and radiation sources, in SAR Sections 1.10.3.2, 1.10.3.3, and 1.10.3.4.

The applicant proposed numerous probable subjects of license specifications listed in SAR Table 5.10-1 to be incorporated into the limiting conditions for operations. A proposed probable subject of license specification related to the shielding involves the ITS radiation detectors and interlocks. This is intended to ensure that radiation detectors interlocked with ITS shield doors are operable to prevent inadvertent door opening if high radiation conditions from a waste package are present.

**NRC Staff's Evaluation of the Shielding Systems Design and Design Analyses**

The NRC staff evaluated the applicant's information on the design and design analyses for the shielding systems and finds that the design and design analyses for ITS SSCs for surface facilities are acceptable because they are based on industry-accepted approaches for radiation shielding and operational controls for limiting exposures. For instance, in SER Section 2.1.1.5.3.1, the staff finds the applicant's use of computer codes (e.g., SCALE and MCNP) acceptable because they are industry standards and have been used by the NRC staff in licensing activities of nuclear power plants and independent spent fuel installations [NUREG–1536 (NRC,2010ah) and NUREG–1617 (NRC, 2000aj)]. The codes are applicable to the GROA facilities, as further discussed in Table 7-1. In addition, the NRC staff finds the applicant's identification of a probable subject of license specification related to the ITS radiation detectors and interlocks acceptable because interlocking the shield doors based on the signal from the radiation detectors is consistent with the radiation protection practices at other nuclear facilities, and is acceptable for use at the GROA. Probable subjects of license specifications are discussed in SER Section 2.5.10.2.3.1. The NRC staff's evaluation of the applicant's shielding analyses and design, including the calculation methods, materials to be used for shielding, codes and standards, the computer codes, and assumptions used in the shielding analysis is in SER Section 2.1.1.8, where they are found to be acceptable.

**NRC Staff's Conclusion of the Criticality Prevention and Shielding Systems Design**

On the basis of the NRC staff's evaluation of the applicant's information on the design of the criticality prevention and shielding systems described previously, the NRC staff concludes, with reasonable assurance, that the applicable regulatory requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f) are satisfied. The NRC staff finds that the description of the criticality prevention and shielding systems designs adequately (i) provides information on materials of construction, dimensions, proposed codes and standards, and analytical and design methods; (ii) defines the relationship between design criteria and the performance objectives; and (iii) identifies the relationship between the design bases and the design criteria.

## 2.1.1.7.4 Evaluation Findings

The U.S. Nuclear Regulatory Commission staff has reviewed the Safety Analysis Report and other information submitted in support of the license application and has found, with reasonable assurance, that the applicable requirements of 10 CFR 63.21(c)(2), 63.21(c)(3), 63.112(f), and 63.112(e)(9) are satisfied, subject to the proposed conditions of construction authorization below. An adequate description and discussion of the design of structures, systems, and components important to safety for both the surface and subsurface geologic repository operations area has been provided for

(i) Materials of construction of the geologic repository operations area (including geologic media, general arrangement, and approximate dimensions), and codes and standards that DOE proposes to apply to the design and construction of the geologic repository operations area;

(ii) Dimensions, material properties, specifications, analytical and design methods used, along with any applicable codes and standards;

(iii) Design criteria used and their relationships to the preclosure and postclosure performance objectives for protection against radiation exposures and releases of radioactive material, numerical guides for design objectives, and identification of the design bases and their relation to the design criteria;

(iv) Explosion and fire detection systems and appropriate suppression systems.

**Proposed Conditions of Construction Authorization [10 CFR 63.32(a)]**

From SER Section 2.1.1.7.3.9.1

DOE shall not, without prior NRC review and approval, accept the following waste packages: (i) 5-DHLW/DOE long codisposal; (ii) 2-MCO/2-DHLW codisposal; and (iii) Naval Short.

DOE shall not, without prior NRC review and approval, accept the following canisters: (i) DHLW long; (ii) DOE long; and (iii) Naval Short.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these waste packages and canisters at the GROA or (ii) demonstrates, through the PCSA, that these waste packages and canisters can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

From SER Section 2.1.1.7.3.9.3.3

DOE shall not, without prior NRC review and approval, accept DPCs at the repository.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of the DPCs at the GROA or (ii) demonstrates, through the PCSA, that the DPCs can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.

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**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| American Concrete Institute, ACI 349–01, "*Code Requirements for Nuclear Safety Related Concrete Structures* (ACI 349–01) and *Commentary* (ACI 349R–01)"<br>(ACI, 2001aa)<br>- - - - -<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (CRCF, RF, and IHF)<br>2.1.1.7.3.1.2<br>Aging Facility<br>2.1.1.7.3.9.3.5<br>Aging Overpack<br>2.1.1.7.3.10.2<br>Shielding Systems | ACI 349–01 provides standards for the design and construction of nuclear safety-related concrete structures. Section R1.1 states that the code is applicable to radioactive waste repository structures. | ACI 349–01 is applicable to the design of the surface buildings at the GROA because it provides nuclear industry specifications for the design and construction of concrete structures at nuclear facilities, and is therefore appropriate for structures at the GROA designed to support, house, or protect safety class systems or component parts of nuclear safety class systems. |
| American Concrete Institute, ACI 350.3–01, "*Seismic Design of Liquid-Containing Concrete Structures* (ACI 350.3–01) and *Commentary* (350.3R–01)"<br>(ACI, 2001ab)<br>- - - - -<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (WHF) | ACI 350.3–01 provides standards for the seismic design of liquid containing concrete structures. | ACI 350.3–01 is applicable to the design of the surface buildings at the GROA because (i) it provides nuclear industry-accepted specifications for the seismic design of concrete structures containing fluid that are independent of a facility's type and size; and (ii) the code is specifically recommended in the updated version of ACI 349–01 (ACI 349–13) for the design and construction of tanks and reservoirs associated with safety-related nuclear structures. |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| American Gear Manufacturers Association, ANSI/AGMA 2001–C95, "*Fundamental Rating Factors and Calculation Methods for Involute Spur and Helical Gear Teeth*"<br>(American Gear Manufacturers Association, 2001aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.5.2<br>Site Transporter | ANSI/AGMA 2001–C95 provides a comprehensive method for rating the pitting resistance and bending strength of spur and helical involute gear pairs. It includes detailed discussions of factors influencing gear survival and calculation methods. | ANSI/AGMA 2001-C95 is applicable to the design of gears in equipment at the GROA because the methodology for evaluating gears is independent of the type of facility in which the gears are used. |
| American Institute of Steel Construction, ANSI/AISC 341–2002, "*Seismic Provisions for Structural Steel Buildings*"<br>(AISC, 2002aa)<br>- - - - -<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (IHF) | ANSI/AISC 341–2002 specifies provisions for the design of seismic force resisting systems of structural steel or composite structural steel/reinforced concrete. | ANSI/AISC 341–2002 is applicable to the design of the IHF steel structure column base connections because the ASCE/SEI 43–05 standard, which is applicable for the seismic design of structures at the GROA, as discussed in this Table, provides for the use of ANSI/AISC 341–2002. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| American Institute of Steel Construction,<br>ANSI/AISC N690–1994, "*Specification for Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*"<br>(AISC, 1994aa)<br>- - - - -<br>2.1.1.7.3.1.1.3.1<br>Surface Facilities<br>General Analysis and Design Procedures<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (CRCF, RF, and IHF)<br>2.1.1.7.3.4.3<br>Other Mechanical Structures<br>2.1.1.7.3.10.1<br>Criticality Prevention | ANSI/AISC N690–1994 provides standards for the design and construction of safety-related steel structures at nuclear facilities. | ANSI/AISC N690–1994 is applicable to GROA buildings because it provides nuclear industry-accepted specifications for the design and fabrication of steel structures at nuclear power plants and is, therefore, appropriate for design and fabrication of analogous structures at the GROA.<br><br>In addition to the bases discussed for ANSI/AISC N690–1994, the NRC Staff notes that the risk of radiological release at the GROA is smaller than at a nuclear power plant; and therefore, use of nuclear power plant references in the design of the GROA is conservative. |
| American National Standards Institute,<br>ANSI N14.30–1992, "*Nuclear Materials–Semi-Trailers Employed in the Highway Transport of Weight-Concentrated Radioactive Loads: Design, Fabrication, and Maintenance*"<br>(ANSI, 1992aa)<br>- - - - -<br>2.1.1.7.3.5.3<br>Cask Tractor and Cask Transfer Trailers | ANSI N14.30–1992 provides standards for the design, fabrication, testing, and maintenance of semi-trailers used in the highway transport of radioactive materials. | ANSI N14.30–1992 is directly applicable to all semi-trailers carrying radioactive loads traveling on highways and roads to various destinations, including to or at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| American National Standards Institute,<br>ANSI N14.5–97, "*Leakage Tests on Package for Shipment*"<br>(ANSI, 1998aa)<br>- - - - -<br>2.1.1.7.3.9.2<br>TAD<br>2.1.1.7.3.9.3.4<br>Naval Canister<br>2.1.1.7.3.9.3.5<br>Aging Overpack | ANSI N14.5–97 provides methods for demonstrating that Type B packages designed for the transport of radioactive material comply with the containment requirements 10 CFR Part 71. | ANSI N14.5–97 is applicable to the design of canisters used at the GROA for leak testing of canisters containing radioactive materials because the standards provide nuclear industry-accepted methods that are acceptable to the NRC staff for evaluating leakage of transportation packages to comply with 10 CFR Part 71. |
| American National Standards Institute,<br>ANSI N14.6–1993, "*American National Standards for Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More*"<br>(ANSI, 1993aa)<br>- - - - -<br>2.1.1.7.3.4.2<br>Special Lifting Devices | ANSI N14.6–1993 provides standards for the design, fabrication, maintenance, inspection, testing, and quality assurance aspects of the lifting devices for radioactive containers weighing more than 4,500 kg [10,000 lb]. | ANSI N14.6–1993 is applicable to devices used for lifting radioactive containers weighing more than 4,500 kg [10,000 lb], such as those proposed for use at the GROA. The standards provide nuclear industry-accepted methods for such lifting devices. |
| American National Standards Institute,<br>ANSI/ANS–6.4–97, "*Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants*"<br>(ANSI, 1997ab)<br>- - - - -<br>2.1.1.7.3.9.3.5<br>Aging Overpack | ANSI/ANS–6.4–97 provides methods and data for calculating the concrete thickness for adequate radiation shielding in nuclear power plants. Where possible, specific recommendations regarding radiation attenuation calculations, shielding design, and standards of documentation are made. It provides guidance to architect-engineers, utilities, and reactor vendors who are responsible for the shielding design of stationary nuclear plants. | ANSI/ANS–6.4–97 is relevant to concrete shielding design for the waste handling facilities in the GROA because this standard addresses (i) methods of analysis and the shielding input data appropriate to each method; (ii) applications of the analytical methods are given, including bulk transport, radiation heating, and streaming; and (iii) reflection problems that are applicable to nuclear facilities other than nuclear power plants. The NRC staff finds it appropriate for use at similar facilities at the GROA. |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| American Nuclear Society, ANSI/ANS–6.1.1–1977, "*Neutron and Gamma-Ray Flux-to-Dose-Rate Factors*" (ANS, 1977aa)<br>- - - - -<br>2.1.1.7.3.10.2<br>Shielding Systems | ANSI/ANS–6.1.1–1977 presents data recommended for computing biological dose rates due to neutron and gamma-ray radiation fields. Specifically, this standard is intended for use by shield designers to calculate whole-body dose rates to radiation workers and the general public. | ANSI/ANS–6.1.1–1977 is applicable to shielding design at the GROA because it is used to calculate effective dose equivalents from radioactive material independent of a particular type of facility. The guidance on neutron and gamma ray flux-to-dose-rate conversion factors is also appropriate for use at the GROA because it specifies biological dose rate factors for the radioactive materials to be handled at the GROA. |
| American Nuclear Society, ANSI/ANS–6.1.1–1991, "*American National Standard for Neutron and Gamma-Ray Fluence-to-Dose Factors*" (ANS, 1991aa)<br>- - - - -<br>2.1.1.7.3.10.2<br>Shielding Systems | ANSI/ANS–6.1.1–1991 presents data recommended for computing the biologically relevant dosimetric quantity in neutron and gamma-ray radiation fields. Specifically, this standard is intended for use by shield designers to calculate effective dose equivalent. Values are given for effective dose equivalent per unit fluence for neutron energies from 1 eV to 14 MeV and for gamma-ray energies from 0.01 to 12 MeV. | ANSI/ANS–6.1.1–1991 is applicable to shielding design at the GROA because it is used to calculate effective dose equivalents. The guidance on neutron and gamma-ray flux-to-dose-rate conversion factors is also appropriate for use at the GROA because it specifies biological dose rate factors for the radioactive materials to be handled at the GROA. |
| American Nuclear Society, ANSI/ANS–8.1–1998, "*Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors*" (ANS, 2007aa)<br>- - - - -<br>2.1.1.7.3.4.3<br>Other Mechanical Structures<br>2.1.1.7.3.10.1<br>Criticality Prevention | ANSI/ANS–8.1 is applicable to operations with fissionable materials outside nuclear reactors, except for the assembly of these materials under controlled conditions, such as in criticality experiments. | ANSI/ANS–8.1 is applicable to the repository because criticality is a potential hazard related to the transport, storage, and disposal of radioactive materials at the GROA, and this standard provides industry-accepted guidance for the handling of these materials for activities proposed at the GROA because they are similar to the operations with fissionable materials outside nuclear reactors that are the subject of the code. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| American Nuclear Society, ANSI/ANS–8.14–2004, "*Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors*"<br>(ANS, 2004aa)<br>- - - - -<br>2.1.1.7.3.10.1<br>Criticality Prevention | ANSI/ANS–8.14–2004 provides guidance for the use of soluble neutron absorbers for criticality control in facilities other than reactors. | ANSI/ANS–8.14–2004 is directly applicable to the soluble neutron absorbers used at the GROA for criticality control because it provides industry-accepted guidance for neutron absorber selection, system design and modifications, safety evaluations, and quality control programs. |
| American Nuclear Society, ANSI/ANS–8.22–1997, "*Nuclear Criticality Safety Based on Limiting and Controlling Moderators*"<br>(ANS, 1997ac)<br>- - - - -<br>2.1.1.7.3.10.1<br>Criticality Prevention | ANSI/ANS–8.22 provides guidance on how to limit and control moderators to achieve criticality safety in operations with fissile materials. | ANSI/ANS–8.22 is directly applicable to preventing the entry of moderator(s) into criticality control area(s) at the GROA because moderator control is an industry-accepted approach for criticality control at facilities handling nuclear waste, such as those at the GROA. |
| American Nuclear Society, ANSI/ANS–8.21–1995, "*Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*"<br>(ANS, 1995aa)<br>- - - - -<br>2.1.1.7.3.10.1<br>Criticality Prevention | ANSI/ANS–8.21–1995 provides guidance for the use of fixed neutron absorbers with fissionable material process equipment where such absorbers provide criticality safety control. | ANSI/ANS–8.21–1995 is directly applicable to the fixed-neutron absorbers for criticality control at the GROA because use of fixed-neutron absorbers for criticality control is an industry-accepted approach for criticality control for handling of nuclear waste at facilities analogous to those at the GROA. |
| American Nuclear Society, "*Design Guides for Radioactive Material Handling Facilities and Equipment,*" 1988<br>(Doman, 1988aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV | Design Guides for Radioactive Material Handling Facilities and Equipment provides guidance for the design and operation of facilities and equipment that primarily handle radioactive materials involved in examinations, reprocessing, fusion, fuel handling, and remote maintenance. | The Design Guides for Radioactive Material Handling Facilities and Equipment are directly applicable to the design and operation of the TEV at the GROA because they provide industry-accepted guidance for the design of handling equipment that would be used to handle spent nuclear fuel. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| American Nuclear Society, ANSI/ANS–6.4–2006,<br>“*Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants*”<br>(ANS, 2006aa)<br><br>- - - - -<br>2.1.1.7.3.10.1<br>Criticality Prevention<br>2.1.1.7.3.10.2<br>Shielding Systems | ANSI/ANS–6.4–2006 contains methods and data to calculate the appropriate concrete thickness for radiation shielding in nuclear power plants. Where possible, specific recommendations are made regarding radiation attenuation calculations, shielding design, and standards of documentation. The standard provides guidance to architect-engineers, utilities, and reactor vendors who are responsible for the shielding design of stationary nuclear plants. This standard does not consider sources of radiation other than those associated with nuclear power plants. | ANSI/ANS–6.4–2006 is relevant to concrete shielding design for the waste handling facilities at the GROA because this standard addresses (i) methods of analysis and the shielding input data appropriate to each method; and<br>(ii) applications of the analytical methods are given, including bulk transport, radiation heating, streaming, and (iii) reflection problems that are applicable to nuclear facilities other than nuclear power plants. The NRC staff finds it appropriate for use at similar facilities at the GROA. |
| American Petroleum Institute, API 526,<br>“*Flanged Steel Pressure Relief Valves*”<br>(API, 2002aa)<br>- - - - -<br>2.1.1.7.3.2.4<br>CCT | API 526 provides requirements for direct spring-loaded flanged steel pressure relief valves. | API 526 is applicable to equipment with pressure relief valves at the GROA because it provides industry-accepted requirements for the design of flanged steel pressure relief valves for pneumatic systems. The NRC staff finds this standard sufficient for its intended use in pneumatic systems at the GROA. |
| American Petroleum Institute, API 527,<br>“*Seat Tightness of Pressure Relief Valves*”<br>(API, 1991aa)<br>- - - - -<br>2.1.1.7.3.2.4<br>CCT | API 527 describes methods of determining the seat tightness of metal- and soft seated pressure relief valves, including those of conventional, bellows and pilot-operated designs. | API 527 is applicable because it provides industry-accepted guidance to determine the seat tightness for flanged steel pressure relief valves. The NRC staff finds this standard sufficient for its intended use in pneumatic systems at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| American Railway Engineering and Maintenance-of-Way Association, "*Manual for Railway Engineering*" (American Railway Engineering and Maintenance-of-Way Association, 2007aa)<br>- - - - -<br>2.1.1.7.3.5.4<br>Site Prime Movers | The Manual for Railway Engineering provides guidance for railway tracks design and layouts based on loads. | The Manual for Railway Engineering is applicable to the design of the railway tracks to be used by the site prime movers and the TEV at the GROA because it provides design criteria for railway systems based on the prospective load. Railways at the GROA will perform similar functions to the commercial and military railways for which this manual was developed, and; therefore, the NRC staff finds this manual appropriate for use at the GROA. |
| American Society of Civil Engineers, ASCE 4–98, "*Seismic Analysis of Safety-Related Nuclear Structures*"<br>(ASCE, 2000aa)<br>- - - - -<br>2.1.1.7.3.1.1.2<br>Surface Facilities<br>Design Methods<br>2.1.1.7.3.1.1.3.1<br>Surface Facilities<br>General Analysis and Design Procedures<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (IHF)<br>2.1.1.7.3.1.2<br>Aging Facility | ASCE 4–98 provides standards for performing analyses for the purpose of new structure design or existing structure evaluation to determine the reliability of structures under earthquake motions. The goal of this standard is to provide analysis parameters that are expected to produce seismic responses that have about the same probability of non-exceedance as the input. Specifications of input motions are provided. Analysis standards are given for modeling of structures, analysis of structures, soil-structure interaction modeling and analysis, input for subsystem seismic analysis, and special structures such as buried pipes and conduits, earth-retaining walls, above-ground vertical tanks, raceways, and seismic-isolated structures. | ASCE 4–98 provides methods for specifications of seismic input motions and seismic modeling and analysis of various types of structures.<br><br>ASCE 4–98 is applicable for the analysis of surface facilities at the GROA because it provides nuclear industry-accepted seismic modeling and analysis methods to evaluate above ground and underground structures. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| American Society of Civil Engineers, ASCE 7–98<br>"*Minimum Design Loads for Buildings and Other Structures*"<br>(ASCE, 2000ab)<br>- - - - -<br>2.1.1.7.3.1.1.1<br>Surface Facilities<br>General Design Criteria and Design Bases<br>2.1.1.7.3.1.1.3.1<br>Surface Facilities<br>General Analysis and Design Procedures<br>2.1.1.7.3.1.2<br>Aging Facility<br>2.1.1.7.3.9.2<br>TAD | ASCE 7–98 provides minimum load standards for the design of buildings and other structures that are subject to building code requirements. | ASCE 7–98 is applicable to the design of the surface buildings at the GROA because it provides industry-accepted minimum loads and load combinations for the design of buildings. This guidance has also been relied on by the NRC staff in the evaluation of fuel cycle facilities. These facilities have many analogous functions and design requirements to GROA facilities. Where applied at the GROA, the NRC staff finds this guidance is sufficient for its intended use in combination with other codes and standards for the GROA facilities, as proposed by the applicant. |
| American Society of Civil Engineers, ASCE 7–02<br>"*Minimum Design for Buildings and Other Structures*"<br>(ASCE, 2003aa)<br>- - - - -<br>2.1.1.7.3.9.2<br>TAD | ASCE 7–02 provides minimum load standards for the design of buildings and other structures that are subject to building code requirements. | ASCE 7–02 is applicable to the design of the surface buildings at the GROA because it provides industry-accepted minimum loads and load combinations for the design of buildings. This guidance has also been relied on by NRC staff in the evaluation of fuel cycle facilities. These facilities have many analogous functions and design requirements to GROA facilities. Where applied at the GROA, the NRC staff finds this guidance is sufficient for its intended use in combination with other codes and standards for the GROA facilities as proposed by the applicant. |
| American Society of Civil Engineers, ASCE/SEI 43–05, "*Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities*"<br>(ASCE, 2005aa)<br>- - - - -<br>2.1.1.7.3.1.1.3.1 | ASCE 43–05 is used in conjunction with ASCE 4–98 and provides criteria for seismic design of safety-related SSCs in a broad spectrum of nuclear facilities. | ASCE 43–05 is applicable to the surface facilities at the GROA because it provides nuclear industry-accepted seismic design criteria that can be adjusted to meet the performance objectives of the nuclear facility based a on risk-graded approach. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Surface Facilities<br>General Analysis and Design Procedures<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (CRCF and RF)<br>2.1.1.7.3.9.3.5<br>Aging Overpack | | |
| American Society of Heating, Refrigerating, and Air Conditioning Engineers, 2004 "*ASHRAE Handbook: Heating, Ventilating, and Air Conditioning Systems and Equipment*," 2004–Inch Pound Edition (ASHRAE, 2004aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | The 2004 ASHRAE Handbook provides guidance on designing and installing various air-conditioning and heating systems (e.g., central cooling and heating) and air handling equipment (e.g., fans, coils, and air cleaners). | The 2004 ASHRAE Handbook is applicable to the design of HVAC systems at the GROA because it provides industry-accepted guidance for heating, ventilation and air conditioning systems in industrial facilities. The NRC staff finds this guidance sufficient for the functions that the HVAC systems would perform at the GROA. |
| American Society of Heating, Refrigerating, and Air Conditioning Engineers, "*ASHRAE Handbook: Heating, Ventilating, and Air Conditioning Applications Design Guide for Department of Energy Nuclear Facilities*," 2007 (ASHRAE, 2007aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | The 2007 ASHRAE Handbook; Design Guide for the Department of Energy Nuclear Facilities provides design guidance for numerous HVAC equipment and systems for different applications at DOE facilities, (e.g., mines, power plants, nuclear facilities). The Design Guide (2007) was based on the ASHRAE 2004 handbook and prepared specifically for the DOE complex. | The 2007 ASHRAE Handbook is directly applicable to the design of HVAC systems at the GROA because the GROA is a DOE facility equipped with various heating, ventilation, and air conditioning systems described in this handbook. |
| American Society of Heating, Refrigerating, and Air Conditioning Engineers, ANSI/ASHRAE 52.1–1992, "*Gravimetric and Dust-Spot Procedures for Testing Air Cleaning Devices Used in General Ventilation for Removing Particulate Matter*" | ANSI/ASHRAE 52.1 provides procedures to measure air cleaning | ANSI/ASHRAE 52.1 is applicable to the design of HVAC systems at the GROA because it provides industry-accepted procedures for filter efficiency calculation for use in industrial facilities. The NRC staff finds this guidance sufficient for the |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| (ASHRAE, 1992aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | filter efficiency, and the ability of air-cleaning devices to remove dust as they become loaded. | functions that the HVAC systems would perform at the GROA. |
| American Society of Heating, Refrigerating, and Air Conditioning Engineer, "*ASHRAE Handbook: Fundamentals*," 2005, Inch-Pound Edition<br>(ASHRAE, 2005aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | The 2005 ASHRAE Handbook: Fundamentals, provides guidance on basic principles and data used for the heating, ventilating, refrigerating, and air conditioning systems. | The 2005 ASHRAE Handbook: Fundamentals is applicable to the design of HVAC systems at the GROA because it provides industry-accepted practices applicable to all heating, ventilating, refrigerating, and air conditioning systems in industrial facilities. The NRC staff finds this guidance sufficient for the functions that the HVAC systems would perform at the GROA. |
| American Society of Mechanical Engineers, "*ASME Boiler and Pressure Vessel Code,*" 2007, 2004, 2001, 1998 and 1997<br>(ASME, 2007aa; ASME, 2004aa; ASME, 1998aa; ASME, 1997ab)<br>- - - - -<br>2.1.1.7.3.2.4<br>CTT<br>2.1.1.7.3.9.1<br>Waste Package<br>2.1.1.7.3.9.2<br>TAD<br>2.1.1.7.3.9.3.1<br>DOE Standard Canister<br>2.1.1.7.3.9.3.2<br>HLW Canister<br>2.1.1.7.3.9.3.4<br>Naval Canister<br>2.1.1.7.3.9.3.5<br>Shielded Transfer Cask | The ASME B&PV Code contains standards used in the design and construction of boilers and pressure vessels (including nuclear and fossil power plants). | The ASME B&PV Code applies to the design of components at the GROA because it provides industry, including nuclear industry, accepted standards for pressure vessels in industrial facilities. The NRC staff finds this standard sufficient for its intended use at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/ Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| American Society of Mechanical Engineers, ASME B31.3–2004, "*Process Piping*" (ASME, 2004ab)<br>- - - - -<br>2.1.1.7.3.2.4<br>CTT | ASME B31.3–2004 provides standards for materials, components, design, fabrication, assembly, erection, examination, inspection, and testing of piping systems. | ASME B31.3–2004 is applicable to the design of process piping at the GROA because it provides standards for piping functions at industrial facilities. The functions of piping at the GROA are analogous to the piping functions in other industrial facilities. Therefore, the NRC staff finds this standard sufficient for its intended use at the GROA. |
| American Society of Mechanical Engineers, ASME N509–2002, "*Nuclear Power Plant Air-Cleaning Units and Components*" (ASME, 2003ab)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | ASME N509–2002 provides construction and testing design standards of units and components of high efficiency air and gas cleaning systems used in nuclear power plants. | ASME N509–2002 is applicable to the design of the HVAC systems at the GROA because it is a nuclear-grade standard and provides industry-accepted specifications for the design and fabrication of air and gas cleaning systems at nuclear power plants. The NRC staff finds that it is, therefore, appropriate for design and fabrication of analogous structures at the GROA.<br><br>In addition to the bases discussed for ASME N509–2002, the NRC Staff notes that the risk of radiological release at the GROA is smaller than at a nuclear power plant; and therefore, use of nuclear power plant references in the design of the GROA is conservative. |
| American Society of Mechanical Engineers, ASME NOG–1–2004 Code, "*Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)*" (ASME, 2005aa)<br>- - - - -<br>2.1.1.7.3.2.1<br>CTM<br>2.1.1.7.3.2.2<br>WPTT | ASME NOG–1–2004 provides standards for the construction of electric overhead and gantry multiple girder cranes with top running bridge, trolley, and crane components used at nuclear facilities. | ASME NOG–1–2004 is applicable to the design of cranes used at the GROA because it provides industry-accepted standards for safety-related overhead and gantry multiple girder cranes at nuclear power plants, and the NRC staff finds that it is, therefore, appropriate for use in the design and fabrication of analogous structures at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.2.3<br>SFTM<br>2.1.1.7.3.2.4<br>CTT<br>2.1.1.7.3.4.1<br>Crane Systems<br>2.1.1.7.3.4.2<br>Special Lifting Devices<br>2.1.1.7.3.4.3<br><br>Other Mechanical Structures<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.5.2<br>Site Transporter<br>2.1.1.7.3.5.3<br>Cask Tractor and Cask Transfer Trailers<br>2.1.1.7.3.5.4<br>Site Prime Movers<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls<br>2.1.1.7.3.9.3.4<br>Naval Canister | | |
| American Society of Mechanical Engineers, ASME–NUM–1–2004, "*Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type)*"<br>(ASME, 2005ac)<br>- - - - -<br>2.1.1.7.3.4.1<br>Crane Systems | ASME–NUM–1–2004 provides standards for construction of underhung cranes and top running bridge and gantry cranes with underhung trolleys, which are used in nuclear facilities. | ASME–NUM–1–2004 is applicable to the design of cranes at the GROA because it provides industry-accepted design and construction guidelines for safety-related underhung cranes at nuclear power plants and other spent fuel handling facilities and is, therefore, appropriate for design and fabrication of analogous structures at the GROA. |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| American Society of Mechanical Engineers, ASME AG–1–2003, ASME AG–1–2004 and ASME AG–1–2012, "*Code on Nuclear Air and Gas Treatment, Including the 2004 Addenda*" (ASME, 2004ac; ASME, 2012aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | ASME AG–1–2003,<br>ASME AG–1–2004, and<br>ASME AG–1–2012 provide standards for the design, fabrication, inspection, and testing of air cleaning and conditioning components used in engineering systems in nuclear facilities. | ASME AG–1–2003, ASME AG–1–2004, and ASME AG–1–2012 are applicable to the design of air cleaning and gas treatment systems at the GROA because they provide nuclear industry-accepted standards for these safety-related systems at nuclear power plants, and the NRC staff finds that they are, therefore, appropriate for use in the design and fabrication of analogous structures at the GROA.<br><br>In addition to the bases discussed for ASME AG–1–2003, ASME AG–1–2004, and ASME AG–1–2012, the NRC Staff notes that the risk of radiological release at the GROA is smaller than at a nuclear power plant; and therefore, use of nuclear power plant references in the design of the GROA is conservative. |
| American Society of Mechanical Engineers, ASME B16.34–2004. "*Valves—Flanged, Threaded, and Welding End*" (ASME, 2005ab)<br>- - - - -<br>2.1.1.7.3.2.4<br>CTT | ASME B16.34–2004 provides standards for pressure-temperature ratings, dimensions, tolerances, materials, nondestructive examination, testing, and marking for cast, forged, and fabricated flanged, threaded, and welding end and wafer or flangeless valves. | ASME B16.34–2004 is applicable to equipment with valves at the GROA because it provides industry-accepted standards for flanged, threaded and welding-end valves in industrial facilities. The functions of valves at the GROA are analogous to the valve functions in other industrial facilities. Therefore, the NRC staff finds the use of this standard sufficient for its intended use at the GROA. |
| American Welding Society, AWS D14.1/D14.1M, "*Specification for Welding of Industrial and Mill Cranes and Other Material Handling Equipment*" (American Welding Society, 2005aa)<br>- - - - - | AWS D14.1/D14.1M provides standards for the primary welds used in the manufacture of cranes for industrial, mill, power house, and nuclear facilities. | AWS D14.1/D14.1M is applicable to the design and fabrication of cranes and material handling equipment at the GROA because it provides industry-accepted standards for structural steel welds used in the fabrication of cranes and material handling at nuclear facilities, and the |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.5.2<br>Site Transporter | | NRC staff finds that it is, therefore, appropriate for use in the design and fabrication of analogous structures at the GROA. |
| Association of American Railroads, "*Manual of Standards and Recommended Practices. Section M*" (Association of American Railroads, 2004aa)<br>- - - - -<br>2.1.1.7.3.5.4<br>Site Prime Movers | Section M of the Manual of Standards and Recommended Practices provides standards for the design of locomotives, other rail-based transportation devices, and their components. | Section M of the Manual of Standards and Recommended Practices is applicable and sufficient to the design of the prime movers at the GROA because the site prime mover functions performed at the GROA are analogous to the rail-based transportation systems described in the Manual. |
| Crane Manufacturers Association of America, CMAA 70–2004, "*Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes*"<br>(Crane Manufacturers Association of America, 2004aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.5.2<br>Site Transporter | CMAA 70–2004 provides specifications for uniform quality and performance for purchasers, users and designers of top running bridge and gantry electric overhead cranes. | CMAA 70–2004 is applicable to the design of cranes at the GROA because it is recommended as part of ASME–NOG–1–2004, which applies to electric overhead and gantry multiple girder cranes with a top running bridge and trolley used at nuclear facilities. It also applies to the components of cranes at nuclear facilities. The cranes used at the GROA perform analogous functions to those at other nuclear facilities, and, therefore, the NRC staff finds this standard appropriate for use at the GROA. |
| Cummins Power Generation, "*Application Manual-Liquid Cooled Generator Sets*," 2004<br>(Cummins Power Generation, 2004aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | This Application Manual describes the electrical and mechanical design and equipment selection for stationary, liquid-cooled diesel and spark ignited engine generator sets. | This manual is applicable to the design of diesel generators at the GROA because it provides industry-accepted practices for the design, operation and maintenance of safety-related back-up power supply systems analogous to those proposed for use at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Department of Energy, DOE–HDBK–1169–2003, "*Nuclear Air Cleaning Handbook*"<br>(DOE, 2003ae)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | DOE–HDBK–1169–2003 provides guidance on systems and equipment to be used in DOE nuclear facilities for capturing and filtering radioactive aerosols and gases. | DOE–HDBK–1169–2003 is used by DOE when designing the nuclear air cleaning systems at DOE nuclear facilities, and is acceptable for the design of similar systems at the GROA because it adopts current industry-accepted ASME guidance for facilities that manage nuclear materials. |
| Industrial Truck Standards Development Foundation, ANSI/ITSDF B56.8, "*Safety Standard for Personnel and Burden Carriers*"<br>(Industrial Truck Standards Development Foundation, 2006aa)<br>- - - - -<br>2.1.1.7.3.5.3<br>Cask Tractor and Cask Transfer Trailers | ANSI/ITSDF (previously ASME) B56.8 provides standards for safe design, operation, and maintenance of powered personnel and burden carriers having three or more wheels, a maximum speed not exceeding 56 km/h [35 mph], and a load capacity not exceeding 4,536 kg [10,000 lb]. | ANSI/ITSDF B56.8 is applicable to the design of material and personnel handling equipment at the GROA because it applies to the safety aspects of the equipment necessary to satisfy the nuclear safety design criteria and is applicable to operator-controlled equipment proposed for use at the GROA. |
| Industrial Truck Standards Development Foundation, ANSI/ITSDF B56.9, "*Safety Standard for Operator Controlled Industrial Tow Tractors*"<br>(Industrial Truck Standards Development Foundation, 2006ab)<br>- - - - -<br>2.1.1.7.3.5.3<br>Cask Tractor and Cask Transfer Trailers | ANSI/ITSDF (previously ASME) B56.9 provides standards for the safe design, operation, and maintenance of operator controlled industrial tow tractors up to and including 66,750 N [15,000 lb] maximum rated drawbar pull. | ANSI/ITSDF B56.9 is applicable to the design of towing tractors at the GROA as it applies to the safety aspects of the equipment necessary to satisfy the nuclear safety design criteria; and is applicable to operator-controlled equipment. This standard is also conservative for use at the GROA because it is applied to the design of commercially available trailers that carry nuclear payloads traveling at interstate highway speeds, which are higher than the speed limit specified for the GROA site. |
| Institute of Electrical and Electronics Engineers, IEEE Std 1184–1994, "*IEEE Guide for the Selection and Sizing of Batteries for Uninterruptible Power Systems*"<br>(IEEE, 1995aa) | IEEE Std 1184–1994 provides characteristics of the various battery energy systems available for users to select the system best suited to their needs. | IEEE Std 1184–1994 is applicable to the design of battery energy systems for use at the GROA because it provides industry-accepted guidance for selection and sizing of DC electric power and battery-operated uninterruptible power supply) |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| - - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | | consistent with the engineering practice for the design of electrical power systems of nuclear facilities, including systems proposed for use at the GROA. This guide is endorsed in NRC Regulatory Guide 1.212 with clarifications by the NRC staff for use in conjunction with IEEE 485–1997 in the design of nuclear facilities, and the NRC staff finds that it is therefore acceptable for use at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 308–2001, "*IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations*"<br>(IEEE, 2001aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 308–2001 provides design and surveillance standards for the electrical power systems of nuclear power generating stations. | IEEE Std 308–2001 is applicable to the design of ITS electrical power systems and ITS Instrumentation and Controls at the GROA because it provides nuclear industry-accepted guidance for the design and construction of components used within electrical power and ITS Instrumentation and Controls systems that are required to power components that perform safety functions in nuclear facilities, including ones similar to those proposed for use at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 323–2003, "*IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*"<br>(IEEE, 2004aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 323–2003 provides standards for qualifying Class 1E equipment and interfaces used in nuclear power generating stations. | IEEE Std 323–2003 is applicable to the design of ITS electrical power and ITS Instrumentation and Controls systems at the GROA because it provides nuclear industry-accepted guidance for qualifying the electrical power systems and ITS Instrumentation and Controls that are required to power electrical and instrumentation and control components that perform safety functions in nuclear facilities, including ones similar to those proposed for use at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 336–2005, "*IEEE Guide for Installation, Inspection, and Testing for Class 1E Power, Instrumentation, and Control Equipment at Nuclear* | IEEE Std 336–2005 provides guidance for the pre-installation, installation, inspection, and testing of Class 1E power, instrumentation, | IEEE Std 336–2005 is applicable to the design of ITS electrical power, instrumentation, and control equipment and systems at the GROA because it provides nuclear industry-accepted guidance on |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| *Facilities*"<br>(IEEE, 2006aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | and control equipment and systems of a nuclear facility. | pre-installation, installation, inspection, and testing of these equipment and systems to ensure that they are capable of performing their safety functions in nuclear facilities, including the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 344–2004, "*IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*"<br>(IEEE, 2005aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 344–2004 provides recommended practices for establishing procedures that will yield data to demonstrate that the Class 1E equipment at nuclear power plants can meet performance requirements during and/or following a safe shutdown earthquake that was preceded by one or more operating basis earthquake events. | IEEE Std 344–2004 is applicable to the design of ITS electrical power and ITS Instrumentation and Control systems at the GROA because it provides nuclear industry-accepted guidance to establish tests, analyses, or experience based evaluations to demonstrate that the ITS electrical power and Instrumentation and Control systems will perform their intended safety function when subject to seismic events at nuclear power plants, and therefore, the NRC staff finds use of this standard appropriate for the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 379–2000, "*IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems*"<br>(IEEE, 2001ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 379–2000 provides guidance on the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power station safety systems. | IEEE Std 379–2000 is applicable to the design and evaluation of ITS electrical power, and instrumentation and control equipment and systems at the GROA because it provides nuclear industry-accepted guidance on the design of safety-related electrical power and instrumentation and control systems to enhance the reliability and availability of performing their safety functions at nuclear power plants, and the NRC staff finds this standard likewise appropriate for the design and evaluation of analogous systems proposed for the GROA. |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Institute of Electrical and Electronics Engineers, IEEE Std 383–2003, "*IEEE Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations*" (IEEE, 2004ab)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV | IEEE Std 383–2003 provides general standards, direction, and methods for qualifying Class 1E electric cables, field splices, factory splices, and factory rework for service in nuclear power stations. | IEEE Std 383–2003 is applicable to the design of ITS electrical power, and instrumentation and control equipment and systems at the GROA because it provides nuclear industry-accepted guidance for qualifying electric cables, field splices, and factory splices to ensure that they are capable of performing their safety functions in nuclear power plants, and the NRC staff finds that use of this standard is acceptable at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 384–1992, "*IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits*" (IEEE, 1998aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 384–1992 provides the standards for independence of the circuits and equipment comprising or associated with Class 1E systems that can be achieved by physical separation and electrical isolation of circuits and equipment that are redundant. | IEEE Std 384–1992 is applicable to the design of ITS electrical power, and instrumentation and control SSCs and supporting features at the GROA because it provides criteria and standards for establishing and maintaining the independence of circuits and equipment to ensure reliability and availability of the ITS SSCs. The applicant has proposed to use this code in concert with redundancy designs and other codes and standards, and the NRC staff finds this standard sufficient for use for GROA electrical systems. This standard is endorsed in NRC Regulatory Guide 1.75 and referenced for use in IEEE Standard 603. |
| Institute of Electrical and Electronics Engineers, IEEE Std 387–1995, "*IEEE Standard Criteria for Diesel Generator Units Applied As Standby Power Supplies for Nuclear Power Generating Stations*" (IEEE, 1996aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 387–1995 provides principal design criteria, factory production testing, qualification standards and site testing criteria for the application and testing of diesel-generator units as Class 1E standby power supplies in nuclear power stations. | IEEE Std 387–1995 is applicable to the design of the diesel-generators at the GROA because it provides industry-accepted guidance on design criteria, factory production testing, qualification specifications, and site testing for safety-related diesel generators in nuclear power generating stations. These systems are analogous to similar ones proposed for the GROA, and the NRC staff, therefore, finds this standard appropriate for use at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Institute of Electrical and Electronics Engineers, IEEE Std 446–1995, "*IEEE Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications*" (IEEE, 1996ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 446–1995 provides guidance on the uses, power sources, design, and maintenance of emergency and standby power systems. | IEEE Std 446–1995 is applicable to the design of the ITS emergency and standby power systems at the GROA because it provides industry-accepted guidance on selection and protection of power sources, switching equipment during fault conditions, system grounding, reliability, and maintenance for safety-related emergency and standby power systems in industrial facilities. Therefore the NRC staff finds that the use of this code for these standby systems is sufficient for use at the GROA. Moreover, the NRC staff notes that this standard has been employed to evaluate analogous functions at other nuclear and industrial facilities. |
| Institute of Electrical and Electronics Engineers, IEEE Std 450–2002, "*IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications*" (IEEE, 2003aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 450–2002 provides guidance on maintenance, test schedules, and testing procedures for optimizing the life and performance of, and replacement of, permanently installed, vented lead-acid storage batteries used for standby power applications. | IEEE Std 450–2002 is applicable to the maintenance and periodic testing of lead-acid storage batteries used for standby power applications at the GROA because it provides industry-accepted guidance on maintenance, testing and replacement of lead-acid batteries in industrial facilities. The NRC staff notes that this standard has been employed to evaluate analogous functions at other nuclear and industrial facilities. Therefore, the NRC staff finds that the use of this code for these standby systems is sufficient for use at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 484–2002, "*IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications*" (IEEE, 2003ab)<br>- - - - - | IEEE Std 484–2002 provides guidance on design practices and procedures for storage, location, mounting, ventilation, instrumentation, preassembly, | IEEE Std 484–2002 is applicable to the installation and implementation of lead-acid storage batteries used for standby power applications at the GROA because it provides industry-accepted design practices and procedures for storage, location, mounting, ventilation, instrumentation, preassembly, assembly, and charging of vented lead-acid |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.6<br>ITS Electrical Power Systems | assembly, and charging of vented lead-acid batteries. | batteries in industrial facilities. Therefore, the NRC staff finds that the use of this code for these standby systems is sufficient for use at the GROA. Moreover, the NRC staff notes that this standard has been employed to evaluate analogous functions at other nuclear and industrial facilities. |
| Institute of Electrical and Electronics Engineers, IEEE Std 485–1997, "*IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications*"<br>(IEEE, 1997ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 485–1997 provides guidance for defining the DC load and for sizing a lead-acid battery to supply that load for stationary battery applications. | IEEE Std 485–1997 is applicable to the design, sizing, and loading of lead-acid storage battery systems used for standby power applications at the GROA because it provides industry-accepted guidance on determining the DC load on and the size of lead-acid batteries in industrial facilities. Therefore, the NRC staff finds that the use of this code for these standby systems is sufficient for use at the GROA. Moreover, the NRC staff notes that this standard has been employed to evaluate analogous functions at other nuclear and industrial facilities. |
| Institute of Electrical and Electronics Engineers, IEEE Std 535–1986, "*IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations*"<br>(IEEE, 1986aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 535–1986 provides detailed qualification procedures for type testing Class 1E lead storage batteries for nuclear power generating stations. | IEEE Std 535–1986 is applicable to the qualification of lead-acid storage batteries used for standby power applications at the GROA because it provides nuclear industry-accepted guidance on qualification procedures for demonstrating that installed batteries will perform their intended safety functions at nuclear facilities, including the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Institute of Electrical and Electronics Engineers, IEEE Std 572–2006, "*IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations*"<br>(IEEE, 2006ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 572–2006 provides standards, direction, and methods for qualifying class IE connection assemblies for service in nuclear power generating stations. | IEEE Std 572–2006 is applicable to qualifying electrical connections, terminations, and environmental seal assemblies for the ITS SSCs at the GROA because it provides nuclear industry-accepted qualification guidance for connection assemblies to demonstrate the ability of the safety-related SSCs at nuclear power generating stations to perform their intended safety functions under applicable service conditions (including design basis events), reducing the risks of common cause failure of electrical connections, connection terminations, and environmental seal assemblies. The NRC staff finds that this standard is also applicable to the GROA, which contains similar ITS electrical systems. |
| Institute of Electrical and Electronics Engineers, IEEE Std 603–1998, "*IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations*"<br>(IEEE, 1998ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | IEEE Std 603–1998 provides design requirements for the power, instrumentation, and control portions of safety systems for nuclear power generating stations. | IEEE Std 603–1998 is applicable to the design of the ITS electrical power and ITS Instrumentation and Control systems at the GROA because it provides nuclear industry-accepted minimum functional and design standards for the power, instrumentation, and control portions of safety systems at nuclear power plants, and the NRC staff therefore finds acceptable the applicant's proposed for use of this standard at the GROA for similar electrical systems. |
| Institute of Electrical and Electronics Engineers, IEEE Std 650–2006, "*IEEE Standard for Qualification of Class 1E Static Battery Chargers and inverters for Nuclear Power Generating Stations*"<br>(IEEE, 2006ac)<br>- - - - - | IEEE Std 650–2006 provides methods for qualifying static battery chargers and inverters for Class 1E installations outside containment in nuclear power generating stations. | IEEE Std 650–2006 is applicable to the design of battery systems at the GROA because it provides nuclear industry-accepted methods for qualifying static battery chargers and inverters to ensure that the ITS SSCs that require continuous power supply will perform their intended safety functions at nuclear power plants, and therefore the NRC staff finds this standard appropriate for use for |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.6<br>ITS Electrical Power Systems | | similar electrical systems at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 741–1997, "*IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations*"<br>(IEEE, 1997aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 741–1997 provides criteria that establish protection standards for Class 1E power systems and equipment at nuclear power plants. | IEEE Std 741–1997 is applicable to the design of ITS electrical power systems at the GROA because it describes nuclear industry-accepted guidance for the provision of automatic protection features, testing, and surveillance standards for protection from electrical and mechanical damage for safety-related ITS equipment and components at nuclear power plants, and therefore the NRC staff finds use of this standard to be acceptable for similar electrical systems at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 835–1994, "*IEEE Standard Power Cable Ampacity Tables*"<br>(IEEE, 1994aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 835–1994 provides more than 3,000 ampacity tables for extruded dielectric power cables rated through 138 kV and laminar dielectric power cables rated through 500 kV. | IEEE Std 835–1994 is applicable to the design of electrical power systems at the GROA because it provides industry-accepted ampacity tables. The NRC staff finds the use of these ampacity tables for the GROA to be appropriate and analogous to their use in other nuclear or industrial facilities. |
| Institute of Electrical and Electronics Engineers, IEEE Std 944–1986, "*IEEE Recommended Practice for the Application and Testing of Uninterruptible Power Supplies for Power Generating Stations*"<br>(IEEE, 1986ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | IEEE Std 944–1986 provides application and performance specifications for a low-voltage uninterruptible power supply systems in nuclear and non-nuclear power generating stations. | IEEE Std 944–1986 is applicable to the design of the electric power systems at the GROA because it provides industry-accepted guidance for low-voltage uninterruptible power supply (UPS) subsystems for nuclear generating systems, and the NRC staff finds its use acceptable for analogous systems at the GROA. |
| Institute of Electrical and Electronics Engineers, IEEE Std 946–2004, "*IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations*"<br>(IEEE, 2005ab)<br>- - - - - | IEEE Std 946–2004 provides guidance for the design of the DC auxiliary power systems for nuclear and non-nuclear power generating stations. The guidance includes selection of the quantity and types | IEEE Std 946–2004 is applicable to the design of the electric power systems at the GROA because it provides nuclear industry-accepted guidance for auxiliary power systems at nuclear power plants. Therefore, the NRC staff finds that the use of this code for these standby systems is sufficient for |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.6<br>ITS Electrical Power Systems | of equipment, the equipment ratings, interconnections, instrumentation, control and protection for the ITS DC power subsystems to ensure that these subsystems will perform when needed | use at the GROA. Moreover, the NRC staff notes that this standard has been employed to evaluate analogous functions at other nuclear and industrial facilities. |
| International Code Council, "*International Building Code (IBC) 2000*"<br>(International Code Council, 2003aa)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | IBC 2000 provides standards for all general interior and exterior components and systems design and construction, including ventilation systems. | IBC 2000 is applicable to the design of HVAC ducts and duct supports at the GROA because it provides industry-accepted minimum building standards that the NRC staff finds are sufficient for the intended use of these ducts and duct supports at the GROA. |
| International Organization for Standardization, ISO 9001," *Quality Management Systems – Requirements*."<br>(International Organization for Standardization, 2008aa)<br>- - - - -<br>2.1.1.7.3.5.2<br>Site Transporter | ISO 9001 is a standardized quality management system used by organizations to achieve consistent results and improve processes. ISO 9001 certification means that an organization has met the requirements laid out in ISO9001 guidance. | ISO 9001 is an appropriate management system to apply to the design of the site transporter fuel tanks at the GROA because the ISO 9001 provides quality management guidance that fuel tanks have been properly manufactured to perform as designed (in this case, to be explosion-proof). |
| Mobley, R.K., "*Plant Engineers Handbook,*" 2001<br>(Mobley, R.K. 2001aa)<br>- - - - -<br>2.1.1.7.3.2.4<br>CTT | The Plant Engineers Handbook covers a wide variety of subjects (e.g., electrical systems, utility production, materials handling) that are applicable to operating and maintaining of industrial plants. | This handbook includes topics (e.g., pneumatic systems) applicable to a variety of industrial operations, including those proposed at GROA facilities. Therefore, the NRC staff finds the use of this handbook for pneumatic equipment safety practices, in concert with other codes and standards proposed by the applicant, sufficient for its intended use in the design of the CTT at the GROA. |
| National Electrical Manufacturers Association, ICS 7–2006, "*Industrial Control and Systems: Adjustable-Speed Drives*" | ICS 7–2006 provides guidance for drive converters, drives, drive systems, loop position and tension | ICS 7–2006 is applicable to the design of HVAC components at the GROA because it provides industry-accepted standards for adjustable speed |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| (NEMA, 2006ab)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | control systems, and wind and unwind drive systems. | drive systems used in industrial facilities, and the NRC staff finds it sufficient for analogous components proposed for the GROA. |
| National Electrical Manufacturers Association, NEMA MG–1 , "*Motors and Generators*"<br>(NEMA, 2006aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.5.2<br>Site Transporter | NEMA MG–1 provides guidance on performance, safety, testing, construction, and manufacture of alternating-current and direct-current motors and generators. | NEMA MG–1 is applicable to the design of the motors used in the TEV and the Site Transporter at the GROA because it is recommended for use in nuclear-industry applicable guidance (ASME NOG–1–2004) which NRC staff finds applicable and sufficient for the TEV and Site Transporter. |
| National Electrical Manufacturers Association, NEMA WC 51–2003, "*Ampacities of Cables Installed in Cable Trays*"<br>(NEMA, 2003aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | NEMA WC 51–2003 provides ampacity ratings for 600–15,000 volt solid dielectric cables installed in cable trays. | NEMA WC 51–2003 is applicable to the design of the ITS electric systems at the GROA because it provides industry-accepted ampacity tables for high voltage solid cables installed in cable trays in other nuclear facilities. The NRC staff finds the applicant's proposed use of these ampacity tables for the GROA to be analogous to their use in these other nuclear facilities and, therefore, acceptable. |
| National Fire Protection Association, NFPA 110, "*Standard for Emergency and Standby Power Systems*"<br>(NFPA, 2005ac)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | NFPA 110 provides standards for the performance of emergency and standby power systems that provide an alternate source of electrical power to loads in buildings and facilities in the event that the primary power source fails. | NFPA 110 is applicable to the design of standby power systems at the GROA because this standard has been used in the design of fire protection systems at other nuclear facilities such as nuclear power plants and fuel cycle facilities. The applicant has proposed to use this code in concert with other standards and codes, and the NRC staff finds the proposed use of this standard to be similar to that for other nuclear and industrial facilities for analogous systems at the GROA, and that it is, therefore, acceptable. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| National Fire Protection Association, NFPA 13, "*Standard for the Installation of Sprinkler Systems*," 2007 Edition<br>(NFPA, 2007ab)<br>- - - - -<br>2.1.1.7.3.8<br>Fire Protection Systems | NFPA 13 provides standards for the design and installation of automatic fire sprinkler systems. | NFPA 13 is used in concert with NFPA 72 for the applicant's design of the double interlock preaction fire protection systems at the GROA. Because these standards have been used to design fire protection systems at similar nuclear facilities (e.g., NPPs and Fuel Cycle Facilities), the NRC staff finds the use of these standards sufficient for their intended use at the GROA. |
| National Fire Protection Association, NFPA 70, "*National Electrical Code®*"<br>(NFPA, 2005ab)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | NFPA 70 provides guidance on the installation of (i) electrical conductors, equipment, and raceways; (ii) signaling and communications conductors, equipment, and raceways; and (iii) optical fiber cables and raceways that supply electricity for buildings, structures, parking lots, substations, and equipment at industrial facilities. | NFPA 70 applies to the design of the ITS electrical power standby systems and instrumentation and controls at the GROA because it provides industry-accepted guidance to prevent electrical fires at industrial facilities. The NRC staff finds that the installation of electrical conductors, equipment, and raceways; signaling and communications conductors, equipment, and raceways; and optical fiber cables and raceways at the GROA is similar to that for other nuclear and industrial facilities, and therefore the guidance is applicable to analogous systems at the GROA. |
| National Fire Protection Association, NFPA 72, "*National Fire Alarm Code*," 2007 Edition<br>(NFPA, 2006aa)<br>- - - - -<br>2.1.1.7.3.8<br>Fire Protection Systems | NFPA 72 provides guidance on the application, installation, location, performance, inspection, testing, and maintenance of fire alarm systems, supervising station alarm systems, public emergency alarm reporting systems, fire warning equipment and emergency communications systems (ECS), and their components. | NFPA 72 is used in concert with NFPA 13 by the applicant for the design of the double interlock preaction fire protection systems at the GROA. Because these standards have been used to design fire protection systems at similar nuclear facilities (e.g., NPPs and Fuel Cycle Facilities), the NRC staff finds use of these standards sufficient for their intended, similar use at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| **Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| Nuclear Regulatory Commission ,<br>NUREG–0800–Section 3.5.1.4, "*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—Missiles Generated by Natural Phenomena*,"<br>Rev. 3 (NRC, 2007al)<br>- - - - -<br>2.1.1.7.3.1.1.1<br>Surface Facilities<br>General Design Criteria and Design Bases | NUREG–0800–Section 3.5.1.4 provides guidance for evaluating potential hazards such as missiles generated by the design basis tornado and other natural phenomena that may affect a nuclear power plant. | NUREG–0800–Section 3.5.1.4 is applicable for evaluating the design of surface facilities at the GROA because it provides NRC-approved guidance for evaluating natural phenomena, including wind-generated missiles that may affect a nuclear power plant. Although this guidance was developed for nuclear power plants, it is applicable to the surface facilities at the GROA because it includes methodologies used to assess natural phenomena and extreme weather, which are independent of the type of facility being evaluated. |
| Nuclear Regulatory Commission ,<br>NUREG–0800–*Section 3.7.2, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant—Seismic System Analysis*"<br>(NRC, 2013ac)<br><br>- - - - -<br>2.1.1.7.3.1.1.2<br>Surface Facilities<br>Design Methods<br>2.1.1.7.3.1.1.3.1<br>Surface Facilities<br>General Analysis and Design Procedures<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (CRCF and RF) | NUREG–0800–Section 3.7.2 provides guidance for analyzing Category I safety-related structures for seismic events at a nuclear power plant. | NUREG–0800–Section 3.7.2 was developed for nuclear power plants, however, seismic system analyses are independent of the type of facility, and this guidance provides an NRC-approved approach to evaluating these potential hazards for nuclear power plants. The NRC staff finds this methodology equally applicable for use at the GROA. |
| Nuclear Regulatory Commission,<br>NUREG–0800–Section 3.8.4, "*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—Other Seismic Category I Structures*"<br>(NRC, 2013ad) | NUREG–0800–Section 3.8.4 provides methods for the design of all Seismic Category I structures and other safety-related structures at nuclear power plants (other than | NUREG–0800, Section 3.8.4 was developed for nuclear power plants; however, seismic system analyses are independent of the type of facility, and this guidance provides an NRC-approved approach to evaluating these potential hazards |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/ Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| - - - - -<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (WHF and IHF) | the containment and its interior structures). | for nuclear power plants. The NRC staff finds this methodology equally applicable for use at the GROA. |
| Nuclear Regulatory Commission, NUREG–0800, Section 2.4.3 NUREG–0800, Section 2.4.3, "*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—Probable Maximum Flood (PMF) on Streams and Rivers*" (NRC, 2007ak)<br>- - - - -<br>2.1.1.7.3.1.1.1<br>Surface Facilities<br>General Design Criteria and Design Bases | NUREG–0800–Section 2.4.3 provides guidance for evaluating the hydrological setting of a nuclear power plant site, as related to safety-related SSCs. | NUREG–0800, Section 2.4.3 is found by the NRC staff to be a suitable method to use for determining the PMF in the Yucca Mountain Review Plan Section 2.1.1.1 (NRC, 2003aa). |
| Nuclear Regulatory Commission, NUREG–0800, Section 3.8.5,"*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants–Foundations*" Rev. 3 (NRC, 2010ag) and Rev. 4 (NRC, 2013ag)<br>- - - - -<br>2.1.1.7.3.1.1.3.2<br>Surface Facilities<br>Facility-Specific Analysis and Design Procedures (CRCF, RF, WHF, and IHF) | NUREG–0800–Section 3.8.5 provides guidance for evaluating foundations (e.g., mat foundations, footing) used to support the safety-related SSCs at a nuclear power plant. | Section 3.8.5 of NUREG–0800 was initially developed by NRC to assess the structural characteristics of a nuclear power plant. However, the NRC finds that this guidance provides an NRC-approved approach to evaluating these structures that is equally applicable for evaluating analogous structures at the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 3.73, "*Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations*"<br>(NRC, 2003ae)<br>- - - - - | RG 3.73 provides guidance for site evaluations and Design Earthquake ground motion for Dry Cask Independent Spent Fuel Storage Installation (ISFSI) and Monitored Retrievable Storage Installations. | RG 3.73 is directly applicable to the design of surface facilities at the GROA because it provides NRC-approved guidance for seismic design of an ISFSI. The same operations and types of SSCs containing SNF (casks, canisters) would be in use at an ISFSI as at the GROA, and therefore the |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.1.1.1<br>Surface Facilities<br>General Design Criteria and Design Bases | | NRC staff finds this guidance acceptable for use at the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.100, "*Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,*" Rev. 2<br>(NRC, 1988aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | RG 1.100 provides NRC staff endorsement and clarification regarding IEEE standards on acceptable methods for use in the seismic qualification of electrical and active mechanical equipment, and on the functional qualification of active mechanical equipment for nuclear power plants. | RG 1.100 was developed for nuclear power plants; however, the NRC staff finds that this RG is similarly applicable to seismic qualification for analogous systems at the GROA, as the guidance provides an NRC-approved approach for evaluating the seismic qualification of safety-related systems. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.89, "*Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*," Rev. 1<br>(NRC, 1984aa)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems<br>2.1.1.7.3.7<br>ITS Instrumentation and Controls | RG 1.89 provides NRC staff endorsement and clarification regarding IEEE standards for qualifying electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident. | RG 1.89 is applicable to the design of electrical systems at the GROA facilities because it provides NRC-approved methods for qualifying electric and instrumentation and controls equipment important to safety at nuclear power plants. The electrical and instrumentation and control systems at the GROA perform analogous functions as at nuclear power plants and; therefore, the NRC staff finds this methodology equally sufficient for use at the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.102, "*Flood Protection for Nuclear Power Plants*"<br>(NRC, 1976ac)<br>- - - - -<br>2.1.1.7.3.1.3<br>Flood Control Features | RG 1.102 provides methods for designing flood protection structures and systems at nuclear power plants, which must withstand floods without the loss of safety-related functions. | RG 1.102 is applicable to the design of the surface facilities at the GROA because it provides NRC-approved criteria to protect safety-related equipment from floods. The NRC staff finds this guidance sufficient to design analogous safety-related equipment from flooding at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.76, "*Design Basis Tornado for Nuclear Power Plants*," Rev. 1<br>(NRC, 2007ai)<br>- - - - -<br>2.1.1.7.3.1.1.1<br>Surface Facilities<br>General Design Criteria and Design Bases | RG 1.76 provides guidance for use in selecting the design-basis tornado and design-basis tornado-generated missiles parameters for nuclear power plants based on site characteristics. | RG 1.76 is applicable to the design of surface facilities at the GROA because it provides an NRC-approved approach for evaluating design basis tornados and tornado-generated missiles that are independent of the facility type, and therefore the NRC staff finds that guidance appropriate for use at the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.91, "*Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants*"<br>Rev. 1 (NRC, 1978ac) and Rev. 2 (NRC, 2013af)<br>- - - - -<br>2.1.1.7.3.1.1.1<br>Surface Facilities<br>General Design Criteria and Design Bases | RG 1.91 provides methods for evaluating postulated explosions at nearby facilities and transportation routes near nuclear power plants. | RG 1.91 is applicable to the GROA facilities because the methods described in this RG are based on the total equivalent mass of explosives, and the NRC staff finds that they are facility-independent and applicable to the GROA. Also, the method for establishing the safe distances, based on a level of peak positive incident overpressure of 1 psi below which no significant damage to structures would be expected, is also applicable because the GROA facilities are similar to those at nuclear power plants. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.92 , "*Combining Modal Responses and Spatial Components in Seismic Response Analysis*," Rev. 1 (NRC, 1976ad) and Rev. 3 (NRC, 2012ac)<br>- - - - -<br>2.1.1.7.3.1.1.3.1<br>Surface Facilities<br>General Analysis and Design Procedures | RG 1.92 provides methods for combining modal responses and spatial components in seismic response analysis. | RG 1.92 is applicable to the GROA because the methods described in this RG are for combining seismic analysis component results and are independent of the facility type and size. |
| Nuclear Regulatory Commission Interim Staff Guidance (ISG) SFST–ISG–15: "*Materials Evaluation*"<br>(NRC, 2001ac) | SFST–ISG–15 provides guidance to the NRC staff to aid review of materials selected for use in dry cask storage systems and | SFST–ISG–15 is applicable to the evaluations of the design of the TAD proposed for use at the GROA in that it provides guidance to the NRC staff on how to assess the adequacy of the |

Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)

| **Code/Standard/NUREG/ Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER** | **Scope** | **Bases for Applicability to GROA** |
|---|---|---|
| - - - - -<br>2.1.1.7.3.9.2<br>TAD | radioactive material transportation packages. | materials (including weld integrity) selected to perform safety functions for a dry cask storage system or transportation packages, which share similar features with the TAD. |
| Nuclear Regulatory Commission,<br>SFPO-Interim Staff Guidance (ISG) –18, "*The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage*," Rev. 1<br>(NRC, 2008ae)<br>- - - - -<br><br>2.1.1.7.3.9.2<br>TAD | SFPO–ISG–18 provides methods for qualification of the final closure welds of austenitic stainless steel canisters. | ISG–18 is applicable to TAD canister closure welds because this ISG addresses welding flaws of sizes that could impair the weld structural strength or confinement capability and provides guidance that the NRC staff finds would not result in such flaws in canisters such as the TAD. |
| Nuclear Regulatory Commission, NUREG–0612, "*Control of Heavy Loads at Nuclear Power Plants*: *Resolution of Generic Technical Activity A-36*"<br>(NRC, 1980aa)<br>- - - - -<br>2.1.1.7.3.4.2<br>Special Lifting Devices | NUREG–0612 provides recommendations and actions to assure the safe handling of heavy loads (including spent fuel casks) at nuclear power plants. | NUREG–0612 is applicable to the design of the equipment for lifting at the GROA facilities because the types of loads (e.g., casks) evaluated in the NUREG are similar to the heavy loads that may be lifted at the GROA, and therefore the NRC staff finds this guidance appropriate for the GROA. |
| Nuclear Regulatory Commission, NUREG–1536, "*Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*"<br>(NRC, 2010ah)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC<br>2.1.1.7.3.9.2<br>TAD<br>2.1.1.7.3.9.3.1<br>DOE Standard Canister | NUREG–1536 provides guidance for reviewing applications for a Certificate of Compliance of a dry storage system for use at a Part 72 licensed facility. | NUREG–1536 is directly applicable to the design of GROA facilities because it provides NRC-approved review guidance for spent fuel storage systems, and analogous systems to those considered in the review plan will be employed at the GROA. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.9.3.2<br>HLW Canister<br>2.1.1.7.3.9.3.3<br>Dual Purpose Canister<br>2.1.1.7.3.10.1<br>Criticality Prevention<br>2.1.1.7.3.10.2<br>Shielding Systems | | |
| Nuclear Regulatory Commission, NUREG–1805, "*Fire Dynamics Tools (FDTs)–Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program*," Rev. 2<br>(NRC, 2004ac)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | NUREG–1805 addresses the technical bases for FDTs, which were derived from the principles developed in the Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering, National Fire Protection Association (NFPA) Fire Protection Handbook, and other fire science literature. The subject matter of this report covers many aspects of fire dynamics and contains descriptions of the most important fire processes. | NUREG–1805 is applicable for use at the GROA because, while some of the spreadsheet examples in the NUREG are specific to nuclear power plants, the NUREG has wide applications for various nuclear industrial settings, including the GROA. |
| Nuclear Regulatory Commission, Regulatory Guide 1.140, "*Design, Inspection, and Testing Criteria for Air Filtration and Absorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,*" Rev. 2<br>(NRC, 2001ad)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | RG 1.140 provides guidance and criteria for the design, inspection, and testing of air filtration and adsorption units installed in normal atmosphere cleanup systems of light-water-cooled nuclear power plants. | RG 1.140 was developed for nuclear power plants; however, the NRC staff finds that use of this guidance at the GROA is appropriate for the design of analogous components. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Nuclear Regulatory Commission, Regulatory Guide 1.189, "*Fire Protection for Nuclear Power Plants*," Rev. 2<br>(NRC, 2009ac)<br>- - - - -<br>2.1.1.7.3.8<br>Fire Protection Systems | RG 1.189 provides methods for designing a comprehensive fire protection system for nuclear power plants. | RG 1.189 is applicable to the GROA because numerous elements (e.g., fire hazard analysis, compensatory measures) outlined in this RG apply to fire protection programs in all nuclear facilities, and therefore the NRC staff finds it appropriate for use at the GROA. |
| Nuclear Regulatory Commission, Regulatory Guide 1.41, "*Preoperational Testing of Redundant On-Site Electric Power System to Verify Proper Load Group Assignments*"<br>(NRC, 1973ad)<br>- - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | RG 1.41 provides NRC staff endorsement and clarifications of IEEE standards providing guidance on methods for verifying the proper assignments of redundant load groups to on-site power sources. | RG 1.41 is applicable to the design of the electrical systems at the GROA because the principles outlined in this guidance are appropriate for analogous redundant loads supplied to the ITS electric equipment, including the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.52, "*Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants*," Rev. 3 (NRC, 2001ae) and Rev 4. (NRC, 2012ad)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | RG 1.52 provides guidance and criteria for the design, inspection, and testing of air filtration and iodine adsorption units of engineered-safety-feature (ESF) atmosphere cleanup systems in light-water-cooled nuclear power plants to meet the requirements of 10 CFR Part 50, Appendix A. | RG 1.52 is applicable to the design of the HVAC system at the GROA because it provides NRC-approved criteria to design, inspect, and test air filtration and adsorption units at nuclear power plants, and the NRC staff finds that the guidance is sufficient for analogous air filtration and adsorption systems that are proposed for use at the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 1.9,<br>"*Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants*," Rev. 3<br>(NRC, 1993ab) | RG 1.9 provides NRC staff endorsement and clarifications regarding IEEE standards providing guidance on the capacity, qualifications, and reliability and availability for design-basis events of safety-related diesel generators | RG 1.9 is applicable to the design of the electrical systems at the GROA because it provides NRC-approved guidance for design and testing of safety-related backup diesel generators employed at nuclear power plants. The GROA has similar emergency power needs and analogous emergency power systems. The NRC staff finds |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| - - - - -<br>2.1.1.7.3.6<br>ITS Electrical Power Systems | intended for use as onsite emergency power sources in nuclear power plants. | that the applicant's proposed use of IEEE Standard 387–1995 instead of IEEE Standard 387–1984 is appropriate because the proposed code is a revision to IEEE 387–1984, and it is applicable to the analogous electrical systems proposed at the GROA. |
| Nuclear Regulatory Commission, Regulatory Guide 3.18, "*Confinement Barriers and Systems for Fuel Reprocessing Plants*"<br>(NRC, 1974ab)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | RG 3.18 provides methods for establishing principal design criteria for confinement systems that will minimize the amount of radioactive material released to the environment or to areas normally occupied by personnel. | RG 3.18 is applicable to the GROA facilities because the principal design criteria are independent of the type and size of the nuclear facilities and the guidance is sufficient because the performance requirements and functions for the HVAC system at the GROA are analogous to those for the fuel reprocessing facilities described in RG 3.18. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 3.71, "*Nuclear Criticality Safety Standards for Fuel and Material Facilities*," Rev. 1 (NRC, 2005ac) and Rev. 2 (NRC, 2010ai)<br>- - - - -<br>2.1.1.7.3.9.2<br>TAD<br>2.1.1.7.3.10.1<br>Criticality Prevention | RG 3.71 provides guidance on criticality safety standards that the NRC has endorsed for use with nuclear fuels and material facilities. | RG 3.71 is applicable to the design of the surface facilities at the GROA because it provides NRC-approved criticality safety standards for all nuclear facilities handling spent nuclear fuels. Criticality safety control principles are common across different nuclear facility types, and therefore the NRC staff finds use of this guidance appropriate for the GROA. |
| Nuclear Regulatory Commission,<br>Regulatory Guide 8.8, "*Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable*," Rev. 3.<br>(NRC, 1978ab)<br>- - - - -<br>2.1.1.7.3.10.2<br>Shielding Systems | RG 8.8 provides guidance on the planning, designing, constructing, operating, and decommissioning of a light-water reactor (LWR) nuclear power station to meet the criterion that exposures of station personnel to radiation during routine operation of the station will be "as low as is reasonably achievable" (ALARA). | RG 8.8 would be applicable to the design of the GROA because the ALARA principle applies to all NRC-regulated facilities handling radioactive materials, including the GROA, pursuant to 10 CFR Part 20. The ALARA program proposed by the applicant is reviewed by the NRC staff in SER Section 2.1.1.8. |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| Nuclear Regulatory Commission, NUREG–1864, "*A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant*"<br>(NRC, 2006ab)<br>- - - - -<br>2.1.1.7.3.2.1<br>Aging Facility | NUREG–1864 documents the pilot probabilistic risk assessment (PRA) for a dry cask system at a boiling water reactor site. The methodology developed in this analysis can provide guidance for assessing the risk to the public and identifying the dominant contributors to that risk. | NUREG–1864 is applicable to the GROA in that the Aging Facility at the GROA is similar to a dry cask storage system at a nuclear power plant. Therefore, relevant information from this PRA study is appropriate for use in evaluating analogous dry cask storage systems at the GROA. |
| Nuclear Regulatory Commission, NUREG/CR–6865,"*Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Cask Storage Systems*"<br>(Luk, et al., 2005aa)<br>- - - - -<br>2.1.1.7.3.2.1<br>Aging Facility | NUREG/CR–6865 documents details of parametric analyses to characterize the sensitivity of a dry cask storage system's response (e.g., sliding displacements or tipping over) in the event of an earthquake. | NUREG/CR–6865 is applicable to the GROA in that the GROA Aging Facility is analogous to a dry cask storage system analyzed in the report. Information regarding cask response to seismic events from this NUREG/CR is, therefore, appropriate for use in the evaluation of the GROA. |
| Nuclear Regulatory Commission NUREG–1617, "*Standard Review Plan for Transportation Packages for Spent Fuel*"<br>(NRC, 2000aj)<br>- - - - -<br>2.1.1.7.3.10.1<br>Criticality Systems<br>2.1.1.7.3.10.2<br>Shielding Systems | NUREG–1617 is a standard review plan that provides guidance for the review and approval of applications for packages used to transport spent nuclear fuel under 10 CFR Part 71. | NUREG–1617 is applicable to GROA facilities and activities because it provides guidelines acceptable to the NRC staff on criticality analysis for transportation packages to comply with 10 CFR Part 71. Spent nuclear fuel being transported to the GROA must comply with 10 CFR Part 71, and use of this guidance for GROA operations is acceptable. |
| Nuclear Regulatory Commission NUREG–2152, "*Computational Fluid Dynamics Best Practices Guidelines for Dry Cask Applications*"<br>(NRC, 2013ae)<br>- - - - - | NUREG–2152 provides guidelines for undertaking numerical simulations using commercially available computational fluid dynamics (CFD) packages to evaluate the thermal response of dry casks storing spent nuclear fuel. | NUREG–2152 is applicable to the GROA facilities because the applicant used commercially available CFD packages (e.g., FLUENT) to evaluate the thermal performance of waste forms and waste containers used in Part 72 dry storage facilities, similar to those described in the guidance. Therefore, the NRC staff finds that use |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| 2.1.1.7.3.3<br>HVAC | | of this guidance is appropriate for the GROA. |
| Nuclear Regulatory Commission NUREG–1922, "*Computational Fluid Dynamics Analysis of Natural Circulation Flows in a Pressurized-Water Reactor Loop under Severe Accident Conditions*"<br>(NRC, 2010af)<br>- - - - -<br>2.1.1.7.3.3<br>HVAC | NUREG–1922 summarizes an updated analysis of severe accident natural circulation flows between the reactor upper plenum and the steam generator using CFD techniques. | NUREG–1922 is a study that endorses use of a computational fluid dynamics model also used by the applicant in thermal evaluations for waste forms at the GROA. The NRC staff finds that this NUREG is appropriate for use in evaluating circulation phenomena at the GROA because the NUREG's predicted flow models conform with data and known conditions at nuclear reactors, as described in the NUREG, providing model support for codes proposed for use by the applicant. |
| Rosaler, R.C., "*Standard Handbook of Plant Engineering*." 2nd edition, 1995<br>(Rosaler, R.C. 1995aa)<br>- - - - -<br>2.1.1.7.3.2.4<br>CTT<br>2.1.1.7.3.3.3<br>HVAC | The Standard Handbook of Plant Engineering covers aspects of plant engineering organization, operation, equipment selection, maintenance, safety, utilities, and other topics that support a wide variety of industrial operations. | This handbook addresses topics applicable to GROA operations, such as ventilation and material handling equipment that are commonly found in industrial settings. The NRC staff finds the use of this handbook for minimization of electrical hazards and for HVAC design, in concert with other codes and standards described in the application, sufficient for use at the GROA. |
| Society of Mining Engineers, American Institute of Mining, Metallurgical, and Petroleum Engineers, "*SME Mining Engineering Handbook*," Vols. 1 and 2<br>(Cummins and Given, 1973aa)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV | The SME Mining Engineering Handbook provides guidance and information on underground excavation and other mining activities, such as rock-breaking methods and underground development. | The SME Mining Engineering Handbook is applicable to the activities to be conducted in the subsurface of the GROA because it is an established industry standard for underground excavation that the NRC staff finds acceptable for use, in concert with other applicable codes and standards as proposed by the applicant, to ensure that the horsepower and traction power of the TEV are appropriate. |
| U.S. Department of the Army, U.S. Army Corps of Engineers, Engineer Manual, EM 1110–2–1913, "*Design and Construction of Levees*" | EM 1110–2–1913 provides guidance for the design and construction of earthen levees. | EM 1110–2–1913 is applicable to the design levees constructed using primarily earthen materials, such as the dikes and berms for floods |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| (U.S. Army Corps of Engineers. 2000aa)<br>- - - - -<br>2.1.1.7.3.1.3<br>Flood Control Features | | caused by the hydrologic events, like those evaluated at the GROA. Therefore, the NRC staff finds the applicant's proposed use of this guidance acceptable for designing the earthen levee flood control features at the GROA. |
| U.S. Department of the Army, U.S. Army Corps of Engineers, Engineer Manual, EM 1110–2–1601, "*Hydraulic Design of Flood Control Channels*"<br>(U.S. Army Corps of Engineers. 1994aa)<br>- - - - -<br>2.1.1.7.3.1.3<br>Flood Control Features | EM 1110–2–1601 provides guidance for the hydraulic design of flood control channels for flow in curved channels, flow at bridge piers, flow at confluences, and side drainage inlet structures. | EM 1110–2–1601 is applicable to the hydraulic design of flood control channels for flow in curved channels, flow at bridge piers, flow at confluences, and side drainage inlet structures, and the NRC staff finds use of this guidance acceptable for designing flood control features at the GROA. |
| U.S. Department of Transportation, Federal Highway Administration, Hydraulic Design Series No. 5, FHWA–NHI–01–020, "*Hydraulic Design of Highway Culverts*"<br>(Federal Highway Administration, 2005aa)<br>- - - - -<br>2.1.1.7.3.1.3<br>Flood Control Features | FHWA–NHI–01–020 provides guidance for the planning and hydraulic design of highway culverts and inlet improvements for culverts. | FHWA–NHI–01–020 is applicable to the design of the roads at the GROA because it provides guidance for hydraulic design of culverts and inlet improvements for culverts for various types of roadways, and the NRC staff finds it appropriate for use for such culverts at the GROA. |
| Underwriters Laboratories, UL 558, "*Industrial Trucks, Internal Combustion Engine Powered*"<br>Underwriters Laboratories. 1996aa)<br>- - - - -.<br>2.1.1.7.3.5.3<br>Cask Tractor and Cask Transfer Trailers | UL 558 provides standards for the fire safety of industrial trucks with internal-combustion engines, such as tractors, platform-lift trucks, fork-lift trucks, and other specialized vehicles for industrial use. | UL 558 is applicable to the design of transportation vehicles at the GROA because it provides industry-accepted guidance for design of internal combustion engines used at industrial facilities, and the NRC staff finds this guidance acceptable for analogous vehicles that are proposed for use at the GROA. |
| <u>ASTM International Standards</u><br><br>ASTM A240/A240M–06c–<br>*Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Applications* | ASTM standards are used to classify, evaluate, and provide specifications with regards to material properties (chemical, mechanical, and metallurgical) for a wide range of materials. These | ASTM standards are applicable for use at the GROA because they represent a means to ensure that materials used in construction at the GROA meet quality assurance, composition, and/or fabrication specifications. ASTM standards are considered best practice for |

**Table 7-1. Chapter 7 Codes/Standards/NUREGs/Regulatory Guides/ISGs/Technical Reference Applicability to GROA (continued)**

| Code/Standard/NUREG/<br>Regulatory Guide/ISG/Technical Reference<br>- - - - -<br>Sections Referenced in SER | Scope | Bases for Applicability to GROA |
|---|---|---|
| (ASTM International. 2006aa)<br><br>ASTM A 572/ASTM A 572M–04–*Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel*<br>(ASTM International. 2004ac)<br><br>ASTM A 276–*Standard Specification for Stainless Steel Bars and Shapes*<br>(ASTM International. 2006ab)<br><br>ASTM A 706/A706M–06a–*Standard Specification for Deformed and Plain Low-Alloy Steel Bars for Concrete Reinforcement*<br>(ASTM International. 2006ac)<br><br>ASTM A 615/ A615M–06a–*Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete Reinforcement*<br>(ASTM International. 2006ad)<br>- - - - -<br>2.1.1.7.3.5.1<br>TEV<br>2.1.1.7.3.9.2<br>TAD<br>2.1.1.7.3.9.3.5<br>Aging Overpack | standards are used for quality assurance to ensure that materials meet the requirements specified by the standards. | controlling quality of materials for industrial uses in the U.S, including numerous nuclear facilities. The NRC staff considers their use appropriate at the GROA for ITS structures and components where steel plates, high-strength steel, and concrete reinforcement bars would be used. |

# CHAPTER 8

## 2.1.1.8 Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences

### 2.1.1.8.1 Introduction

Safety Evaluation Report (SER) Section 2.1.1.8 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's ("DOE" or "applicant") Safety Analysis Report (SAR) (DOE, 2008ab, Section 1.10) and the Operational Radiation Protection Program (RPP) described in SAR Section 5.11. The objective of this review is to verify that the applicant's description of its proposed RPP reflects as low as is reasonably achievable (ALARA) considerations of maintaining the occupational doses to workers and doses to members of the public to as far below regulatory limits as is practical, consistent with the purpose for which the licensed activity is undertaken.

The applicant has described the ALARA policy, design, and operational work practices for the geologic repository operations area (GROA) relied upon to reduce doses to members of the public and occupational doses to workers. The applicant's policy considerations include its management commitment to maintain doses ALARA and the implementation of ALARA principles in the design process throughout the repository design and construction so that shielding design and structural loads are part of the design process. The applicant also described the facility shielding design used to meet the ALARA requirements for normal operations and Category 1 event sequences. The applicant's implementation of the ALARA principles into repository operations, including administrative controls to maintain doses ALARA and general operational guidelines, would be accomplished through its Operational RPP described in SAR Section 5.11.

### 2.1.1.8.2 Regulatory Requirements

The regulatory requirements applicable to this section are in 10 CFR 63.21(c)(6) and 10 CFR 63.111(a)(1), which require the GROA to meet the 10 CFR Part 20 ALARA requirements.

- 10 CFR 63.21(c)(6) requires the applicant to submit a description of its program for control and monitoring of radioactive effluents and occupational radiological exposures to maintain such effluents and exposures in accordance with the requirements of 10 CFR 63.111(a)(1).

- 10 CFR 20.1101(a) requires the applicant to develop, document, and implement an RPP commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR Part 20.

- 10 CFR 20.1101(b) requires the applicant to use, to the extent practical, procedures and engineering controls based upon sound engineering practices to achieve doses to members of the public and occupational doses that are ALARA.

- 10 CFR 20.1101(d) requires the applicant to establish a constraint on air emissions of radioactive material, excluding Radon-222 and its daughters, to the environment such that individual members of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent exceeding 0.1 mSv/year [10 mrem/year].

- 10 CFR 20.1201(a) requires the applicant to control the annual occupational dose to individual adults to the specified dose limits.

The NRC staff reviewed the applicant's ALARA section using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa, Section 2.1.1.8). In addition, the NRC staff used HLWRS–ISG–03 (NRC, 2007ac), which supplements the guidance in the YMRP. The acceptance criteria used in the staff's review are as follows:

- An adequate statement of management commitment to maintain exposures to workers and the public ALARA is provided.

- ALARA principles are adequately considered in geologic repository operations area (GROA) design.

- Proposed operations at the GROA adequately incorporate ALARA principles.

- The description of the RPP adequately addresses its organization, procedures, and implementation.

In addition to the YMRP, the NRC staff used other applicable NRC guidance, such as standard review plans, regulatory guides, and interim staff guidance. Often, this NRC guidance was written specifically for the regulatory oversight of nuclear power plants. The methodologies and conclusions in these documents are generally applicable to analogous activities proposed at the GROA (e.g., handling of spent nuclear fuel, criticality controls during storage of spent nuclear fuel, shield doors and interlocks for worker safety from direct radiation of spent nuclear fuel). The applicability of such NRC guidance is discussed in greater detail in the sections where the guidance was used as part of the application or the NRC staff's review.

### 2.1.1.8.3 Technical Review

In SAR Section 1.10, the applicant stated that the objective of its ALARA program is to keep doses to repository workers and the public ALARA and that the ALARA principles will be incorporated into the design, operations, maintenance, decommissioning, and dismantling activities. The applicant submitted a description of its RPP for control and monitoring of radioactive effluents and occupational radiological exposures to maintain these in accordance with the requirements of 10 CFR 63.111. The Operational RPP was described in SAR Section 5.11. As indicated in the introduction of SAR Section 5.11, the RPP will be available for NRC review in connection with the consideration of an updated application for a license to receive and possess spent nuclear fuel and high-level radioactive waste. A detailed RPP is not necessary for issuance of a construction authorization, because there are no radioactive materials in the applicant's possession. If the applicant submits an updated license application for a license to receive and possess material, the NRC staff would review that information.

The NRC staff's review of the applicant's program description for implementing ALARA principles, including its RPP, is discussed in the following sections.

#### 2.1.1.8.3.1 Management Commitment to Maintain Doses As Low As Is Reasonably Achievable

In SAR Section 1.10.1, the applicant described its management commitment to maintain doses ALARA. As a part of its management commitment, the applicant stated it will control worker doses and releases of radioactive materials to the environment below regulatory limits to include constraints on air emissions, with the exception of radon-222 and its daughters, as required in 10 CFR 20.1101(d). The applicant stated that its management will support the ALARA policy through direct communication, instruction, inspection, and audit of the workplace. As indicated in the SAR, aspects of the applicant's management commitment are the development of an ALARA program, implementation of an operational RPP, and personnel training.

According to SAR Section 1.10.1, the applicant stated that personnel will be made aware of the applicant's management commitment to ALARA through policy and instruction. Personnel, according to the applicant, will be instructed on their individual responsibilities related to ALARA implementation during operations. Personnel will receive training on radiation protection in accordance with 10 CFR 19.12. The applicant indicated that supervisors will be instructed to integrate appropriate radiation protection controls into work activities.

In the SAR, the applicant stated that during design and construction it will conduct ALARA-specific reviews to ensure the ALARA principles are incorporated in the design. The applicant also stated that it will conduct and document audits consistent with the recommendations of NRC Regulatory Guide 8.8 (NRC, 1978ab). The applicant stated that it will estimate occupational doses consistent with the recommendations of NRC Regulatory Guide 8.19 (NRC, 1979aa) and will use construction inspections to verify that shielding features are installed as designed.

For operations, the applicant's management commitment is to implement an operational ALARA program at the repository in accordance with NRC Regulatory Guide 8.8 (NRC, 1978ab, Position C.1). The applicant stated that the operational program will implement ALARA principles in policies and procedures, goals, and objectives for planning, design, and construction of modifications to operating facilities, operating activities, maintenance, housekeeping, decontamination, and dismantlement. During the decommissioning and dismantlement of the repository surface and subsurface nuclear facilities (SAR Section 1.10.1.3), the applicant stated that it will apply the ALARA principles by (i) reviewing prior radiation surveys to assess radiological conditions and (ii) performing visual inspections and radiation surveys to ensure that there are no unidentified radiation sources that might affect personnel exposures. The applicant stated that it will develop procedures for implementing ALARA principles in decommissioning and dismantlement activities. SER Section 2.1.3 provides the NRC staff's evaluation of the applicant's plans for permanent closure and decontamination or decontamination and dismantlement of the Yucca Mountain surface facilities.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's management commitment to maintain doses ALARA, using YMRP Section 2.1.1.8.3. The applicant's management commitment to implement radiation protection controls into all work activities is acceptable because the applicant stated that it will incorporate ALARA principles during design and construction, operations, and decontamination and decommissioning.  The applicant's radiation protection controls are consistent with the guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab) for design reviews and audits and NRC Regulatory Guide 8.19 (NRC, 1979aa) for estimating doses during the design process, as applicable, because it would ensure that occupational radiation exposures are minimized.

The NRC staff finds the approach in making personnel aware of the applicant's management commitment to ALARA principles through policy and instruction acceptable because it is consistent with NRC Regulatory Guide 8.10 (NRC, 1997ac).  The NRC staff finds the applicant's management commitment to provide appropriate radiation training to be acceptable because the applicant stated that it will provide radiation protection instruction to individuals who are likely to receive in a year an occupational dose exceeding 1 mSv [100 mrem], as required in 10 CFR 19.12.  Consistent with NRC Regulatory Guide 8.10 (NRC, 1997ac), the applicant stated that the training will be commensurate with the duties and responsibilities of training recipients and that workers will be periodically retrained in radiation protection procedures and techniques on the basis of job responsibility.

The applicant stated that it will implement an operational RPP that will follow the guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab, Position C.1), as applicable.  The applicant described its RPP in SAR Section 5.11.  The applicant stated in SAR Section 5.11 that it will use program policies and procedures to maintain radiation doses to workers and members of the public during operations to meet 10 CFR Parts 20 and 63 and maintain doses ALARA.  The NRC staff's evaluation of the RPP is provided in SER Section 2.1.1.8.3.5 of this chapter.

**NRC Staff's Conclusion**

The NRC staff concludes that the applicant has provided an adequate statement of management commitment to maintain doses to workers and the public ALARA.  The NRC staff finds with reasonable assurance that the applicant satisfies the requirements of 10 CFR 63.111(a)(1), relating to meeting the 10 CFR Part 20 ALARA requirements, with respect to the applicant's management commitment to maintain doses ALARA.

#### 2.1.1.8.3.2 Consideration of As Low As Is Reasonably Achievable Principles in Design and Modifications

In SAR Section 1.10.2, the applicant discussed the application of the ALARA principle into the design.  The applicant stated that its ALARA program is conducted through engineering procedures, training for engineering and design personnel, design reviews, cost-benefit analyses, audits, self-assessments of effectiveness, and a policy for its consistent application of ALARA principles in the design process.  According to the applicant, formal design criteria are used to implement ALARA design requirements.  The applicant's reviews during the early design process considered potential radiation exposure and contamination from normal operations and any Category 1 event sequences.  Implementation of the ALARA principle, as stated by the applicant, begins early in the design process and continues as an iterative process through detailed design for construction.  The applicant's assessments considered radiation workers, construction workers during staged operations, and members of the public.  The program focused on activities associated with higher potential doses so that greater reductions in worker and public doses could be realized.  An annual dose goal of 5 mSv [0.5 rem] was set by the applicant for an individual radiation worker.  The applicant applied ALARA principles to both collective and individual doses for radiation workers. The ALARA program considered estimated worker doses for both compliance demonstrations on the basis of minimized staffing levels, maximized source terms, facility annual throughput, and nominal conditions on the basis of more realistic assumptions for estimating annual-average doses. Although no Category 1 event sequences were identified, the applicant considered reducing doses for workers conducting recovery actions from potential event sequences.

The applicant described design objectives, considerations, and features for the facility layout and equipment design. The applicant discussed ALARA aspects for specific equipment, such as shield doors, shielded viewing windows, ventilation confinement, and radiation and airborne radioactivity monitoring. The applicant will apply access controls to high and very high radiation areas as well as restricted areas. The applicant will use radiation zone designations to identify the need for design features to maintain doses ALARA. Where contamination could occur, the applicant will incorporate design features to control the spread of contamination and facilitate maintenance and decommissioning.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's consideration of ALARA principles in design and modifications of the GROA using YMRP Section 2.1.1.8.3. The NRC staff finds that the applicant adequately considered ALARA objectives, principles, and philosophy in the repository design process, to the extent practical, because its design reviews consider good practices, such as

- Minimizing the time workers stay in radiation areas
- Incorporating remotely operated equipment to minimize worker doses
- Considering access and egress to work areas within the restricted area
- Placing and handling of equipment and shielding by remote operations
- Minimizing the potential for contamination, controlling the spread of contamination, and facilitating decontamination to limit doses during operations and decommissioning
- Segregating waste transfer areas from normally occupied areas
- Locating waste handling facilities and transfer routes away from locations accessible to members of the public
- Applying suitable methods to perform inspection of materials

The applicant stated in SAR Section 1.10.2 that it will locate radioactive material handling and storage facilities sufficiently away from the site boundary and from other onsite work areas to maintain doses ALARA. The applicant demonstrated in SAR Tables 1.8-28 and 1.8-36 that the facility design is sufficient to maintain doses to onsite members of the public in unrestricted areas below the limit specified in 10 CFR 20.1301(a)(2) for external sources. SER Section 2.1.1.5 provides additional details on the NRC evaluation of the applicant's demonstration of compliance with dose requirements. The applicant stated that it will control access to the restricted area and apply access controls to high and very high radiation areas in accordance with 10 CFR 20.1601 and 10 CFR 20.1602 by using the guidance in NRC Regulatory Guide 8.38 (NRC, 2006ac). Therefore, for these reasons, the NRC staff finds that the applicant adequately factored the ALARA principles into its facility design.

The applicant stated that it will conduct ALARA design reviews using multidisciplinary teams with experience in radiological safety, operations, and engineering backgrounds. The NRC staff finds this approach acceptable because using these multidisciplinary teams to conduct the reviews ensures radiological safety will be considered within the context of operation processes and nonradiological safety before the applicant decides to make potential modifications or improvements. This approach ensures that modifications would not adversely influence other components of the design. The NRC staff also compared the applicant's dose estimates for radiation workers during normal operations to the applicant's ALARA goal and determined that estimated doses exceeded the annual ALARA dose goal at several GROA facilities (BSC, 2008al). The applicant acknowledged situations when estimated doses did not meet the ALARA design goal and identified options for dose reduction (BSC, 2008bw). The applicant also assessed average worker doses when workers who perform similar tasks (operators, health physics technicians, or security) are rotated to different facilities. By accounting for work rotations, the applicant presented average worker doses (BSC, 2008al) that were below the annual ALARA dose goal of 5 mSv [0.5 rem]. Therefore, for these reasons, the NRC staff finds that the applicant has adequately factored the ALARA principle into its assessment of radiological consequences for radiation workers.

**NRC Staff's Conclusion**

The NRC staff concludes that the applicant has adequately considered ALARA principles into the design and modifications. The NRC staff finds with reasonable assurance that the applicant satisfies the requirements of 10 CFR 63.111(a)(1), relating to meeting the ALARA requirements in 10 CFR Part 20, with respect to considering ALARA principles in the design and modifications.

#### 2.1.1.8.3.3 Facility Shielding Design

In SAR Section 1.10.3, the applicant discussed the facility surface and subsurface shielding design objectives, criteria, and evaluation used to implement ALARA requirements for normal operations and Category 1 event sequences. The applicant stated that the objective is to design facility shielding to reduce dose rates from radiation sources such that worker doses are within the standards of 10 CFR Part 20 and are ALARA when combined with the program to control personnel access and occupancy at restricted areas. The applicant performed shielding evaluations to ensure that adequate space envelopes and structural loads are identified. According to the SAR, surface facility shielding will include concrete walls, floors, and ceilings; shielded viewing windows; slide gates; and shield doors. The applicant stated that it will adopt the concrete design used for shielding in accordance with ANSI/ANS-6.4-2006 (American Nuclear Society, 2006aa).

As part of the design objectives, the applicant provided its shielding design descriptions for individual facilities used in the shielding evaluation. The shielding design, according to the applicant, is based upon the various facility areas and the established radiation zones. The individual radiation zoning characteristics were presented in SAR Table 1.10-1, and specific area dose rate criteria used in the shielding evaluation were presented in SAR Table 1.10-2. The shielding design bases, according to the applicant, include worker occupancy time, external radiation sources, radiation effects on components, and bounding source terms.

According to the SAR, the primary material used for the shielding evaluation is Type 04 concrete with a bulk density of 2.35 g/cm$^3$ [147 lb/ft$^3$] based on ANSI/ANS-6.4-2006 (American Nuclear Society, 2006aa, Table 1). Other component materials used in the shielding evaluation, such as water in the Waste Handling Facility pool and other shielding features, were described in SAR Sections 1.2.3 to 1.2.8.

The applicant described its shielding evaluation methodology as follows:

- Radiation sources, summarized in SAR Figure 1.10-18, and bounding terms, described in SAR Section 1.10.3.4, are used to approximate the geometry and physical condition of sources in the various repository facilities.

- Flux-to-dose rate conversion factors taken from ANSI/ANS-6.1.1-1977 (American Nuclear Society, 1977aa) are used to develop dose rates. SER Section 2.1.1.5.3.1 evaluates the use of this standard as well as the updated 1991 version and concludes that the applicant's use of the 1977 standard is acceptable because it is based on conservative assumptions and results in an overestimation of personnel exposures, especially those that result from the neutron component of these exposures.

- Commonly accepted industry standard methods and codes such as Monte Carlo N-Particle and Standardized Computer Analysis for Licensing Evaluation are used to evaluate the basic design for the repository surface and subsurface facilities. The industry methods show that the shielding design will lower the dose rates from the various radiation sources to ensure appropriate protection of workers and the public.

The shielding evaluation for the various areas and components was summarized in SAR Tables 1.10-35 to 1.10-46. SAR Table 1.10-1 provides the shielding evaluation, including factors such as the radiation source, distance from the source to the shielding, shielding thickness, shielding material, and the radiation zones of each facility. The applicant designated radiation zones R1 through R5, which include unlimited occupancy through limited or no occupancy areas, respectively, including the dose rate range or limit of each zone. The applicant implemented ALARA principles through the combination of facility shielding design and the RPP.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's considerations of ALARA principles in the facility shielding design using NRC Regulatory Guide 8.8 (NRC, 1978ab, Position C.2). The NRC staff finds the applicant's shielding evaluations, which are based upon the shielding design objectives (SAR Section 1.10.3.1) and shielding design considerations (SAR Section 1.10.3.1.1), to be acceptable because they are based on the design recommendations of NRC Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff finds that in accordance with NRC Regulatory Guide 8.8 (NRC, 1978ab), the applicant's design objectives include providing shielding that will ensure that (i) personnel radiation doses are ALARA and within the limits of 10 CFR Part 20, (ii) worker access and occupancy times allow for normal operations, and (iii) minimum radiation damage occurs to equipment not designed for higher radiation fields. The design considerations that follow from these objectives include (i) providing shielding to reduce dose rates to levels consistent with the expected occupancy for personnel and equipment to conduct normal operations and (ii) providing shielding on the basis of bounding source terms applicable to the material that will be handled in each facility or location.

The NRC staff finds that the shielding design considerations, as discussed in SAR Section 1.10.3.1.1, adequately address reducing direct and scattered radiation exposure. The source terms used in the shielding evaluation, as described in SAR Section 1.8, are acceptable to establish the shielding design criteria as well as the radiation zoning areas because the applicant used bounding source terms. SER Section 2.1.1.5 provides the NRC staff's evaluation of the applicant's source term calculations, direct exposure calculations, and radiation dose calculations to workers and members of the public from airborne radionuclides, which the NRC staff finds acceptable.

**NRC Staff's Conclusion**

The NRC staff concludes that the facility shielding design will adequately implement the ALARA philosophy and guidance because the applicant will incorporate ALARA principles during design and construction, operations, and decontamination and decommissioning. The shielding design evaluation is appropriate because the applicant will follow the guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab) for design reviews and audits and NRC Regulatory Guide 8.19 (NRC, 1979aa) for estimating doses during the design process. The applicant also will implement an operational RPP that will follow the guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab, Position C.1). The RPP is evaluated in detail in SER Section 2.1.1.8.3.5. For the facility shielding design, the NRC staff finds, with reasonable assurance, that the applicant has satisfied the requirements of 10 CFR 63.111(a)(1), relating to meeting the ALARA requirements in 10 CFR Part 20.

#### 2.1.1.8.3.4 Incorporation of As Low As Is Reasonably Achievable Principles Into Proposed Operations at the Geologic Repository Operations Area

In SAR Section 1.10.4, the applicant has described its incorporation of ALARA principles into repository operations. This description included policies and procedures; monitoring and evaluation of worker doses, public doses, and area dose rates; oversight by an ALARA committee; establishing ALARA goals and administrative limits for workers; controlling worker access and equipment removal at restricted areas; reducing or preventing radioactive contamination; and monitoring and reducing radioactive waste production. The applicant stated that radiation protection training and personnel testing will be conducted for radiation workers before those individuals are allowed to begin work activities in restricted areas. The applicant included periodic retraining in its description. According to the SAR, individuals with job tasks outside of restricted areas are classified as onsite members of the public and will receive instruction on emergency procedures. The applicant stated that, during operations, it will apply job preplanning (for workers entering into radiation areas where significant doses could be received) to ensure that the work can be performed in a safe manner with personnel doses minimized in accordance with the ALARA principles. According to the SAR, the job preplanning includes dose estimates and debriefing sessions to capture ALARA good practices and lessons learned. On complex jobs, the applicant indicated that dry-run training will be utilized. The intent of dry-run training, according to the applicant, is to improve worker efficiency, minimize worker stay times, avoid unnecessary and potentially harmful actions, and minimize overall doses. The applicant indicated that localized areas with higher radiation levels will be identified and factored into work planning to minimize personnel doses, in accordance with the ALARA principles. Work planning will include surveys of radiation levels, contamination, and airborne material concentrations; consideration of remotely operated equipment use; consideration of data and experience attained in previous operations; and the potential for and response to off-normal occurrences.

According to the SAR, radiological work permits and written procedures will be used as administrative controls for operations and maintenance. The applicant stated that radiation areas will be designated and posted within restricted areas, and access to high and very high radiation areas will be controlled. The applicant indicated that ALARA reviews will be conducted before design changes and administrative control changes are approved. The applicant also stated that the ALARA program will address recovery actions from event sequences during operations and reviews of planned decommissioning and decontamination activities. Although no Category 1 event sequences have been identified for which recovery actions are preplanned, the applicant did consider reduction of worker doses for recovery from potential event sequences that the applicant described as off-normal events. SER Section 2.1.1.4 describes the NRC staff's evaluation of the applicant's categorization of event sequences.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of how the applicant would incorporate the ALARA principles into operations, using YMRP Section 2.1.1.8.3 and HLWRS–ISG–03 (NRC, 2007ac). Because the ALARA requirement applies during operations, SER Section 2.5.6 provides the NRC staff's evaluation of the applicant's plans for conducting normal activities, including maintenance; surveillance; and testing of structures, systems, and components. In SER Section 2.5.6, the NRC staff concludes that the applicant adequately described the plans for the conduct of normal activities, including maintenance, surveillance, and periodic testing that would be implemented before the applicant receives, possesses, processes, stores, or disposes high-level radioactive waste.

The NRC staff finds it acceptable that the applicant stated that it will incorporate ALARA guidance from Regulatory Guides 8.8 and 8.10 (NRC, 1978ab, 1997ac) into the repository processes and procedures. The applicant stated that it will apply ALARA principles to both individual and collective doses. Consistent with Regulatory Guide 8.8 (NRC, 1978ab) and common industry practice, the applicant considered tradeoffs between dose reduction alternatives and the potential hazards associated with these alternatives, such as the use of temporary shielding (which includes installation and removal) only if the total dose is reduced. According to the SAR, ALARA alternatives, based on operational experience and lessons learned from other nuclear facilities, will be incorporated into the applicable processes and procedures, consistent with Regulatory Guides 8.8 and 8.10 (NRC, 1978ab, 1997ac).

The NRC staff finds that the applicant's approach of using appropriate operational administrative controls, such as radiological work permits for sampling, inspection, maintenance, and calibration procedures, is acceptable because it is consistent with standard industry practice. The NRC staff also finds the applicant's use of preplanning and dry-run training for radiation workers who may enter into high radiation areas acceptable because it is consistent with the ALARA guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab).

The NRC staff finds the applicant's consideration of ALARA principles as a part of its review and approval process for issuing radiological work permits acceptable because the implementation of ALARA is tied to operational activities involving radiological exposure, thereby increasing the effectiveness of the applicant's ALARA program.

The NRC staff finds the applicant's work planning approach acceptable because it incorporates ALARA principles into the proposed operations, consistent with the ALARA guidance in Regulatory Guides 8.8 and 8.10 (NRC, 1978ab; 1997ac). Additionally, the NRC staff finds acceptable that the applicant's modifications to the proposed operations are reviewed to ensure that they do not adversely influence other aspects of area operations. The NRC staff finds the applicant's review of modifications also provides additional confidence that the applicant would execute an effective ALARA program.

The NRC staff finds that the applicant has adequately described how it will incorporate the ALARA principle into proposed operations. In accordance with the ALARA requirement in 10 CFR 20.1101(d), the applicant considered and evaluated the dose constraint on air emissions of radioactive material to the environment for public exposure with other preclosure objectives described in SAR Table 1.8-36. Because the applicant's preclosure safety analysis did not identify any Category 1 event sequences, a plan for recovery actions from the major types of Category 1 event sequences, including basic recovery steps and general radiation levels during recovery, is not necessary. However, the applicant acknowledged that ALARA principles would be factored into the review of any proposed recovery actions so that dose reduction measures would be included. The NRC staff finds this approach consistent with HLWRS–ISG–03 (NRC, 2007ac) and is therefore acceptable.

**NRC Staff's Conclusion**

The NRC staff concludes that the applicant has adequately incorporated ALARA principles into proposed operations. The NRC staff finds with reasonable assurance that the requirements of 10 CFR 63.111(a)(1) are satisfied. The operations at the geologic repository operations area, through permanent closure, will comply with the ALARA requirements in 10 CFR Part 20.

### 2.1.1.8.3.5 Radiation Protection Program

The applicant described its Operational RPP in SAR Section 5.11. The proposed RPP in SAR Section 5.11 described the policies and procedures and the program elements. Consistent with the guidance in HLWRS–ISG–03 (NRC, 2007ac), the NRC staff's review focused on (i) administrative organization; (ii) the descriptions of health physics equipment, instrumentation, and facilities; (iii) the description of policies and procedures for controlling access to radiation areas, description of procedures for the accountability and storage of radioactive material, and the radiation protection training programs; and (iv) the description of the program implementation. The applicant's description of program implementation is discussed throughout the following sections.

In accordance with 10 CFR 20.1101(c), the applicant stated that the RPP will require that a review and assessment be conducted at least annually to evaluate the adequacy of the program intent and its implementation. According to the SAR, the assessment will document program deficiencies and recommend corrective actions or improvements.

#### 2.1.1.8.3.5.1 Administrative Organization of the Radiation Protection Program

The applicant has described the RPP organization in SAR Section 5.11.1. The applicant stated that it will have the radiation protection and criticality safety (RPCS) program organization under the RPCS manager. The applicant indicated that the RPP organization will work independently of the operations and maintenance organizations. According to SAR Section 5.3.1.2 and the applicant's response to the NRC staff's request for additional information (DOE, 2009az), the

RPCS manager will report directly to the site operations manager and chief nuclear officer. The applicant stated that the RPCS manager will be responsible for developing and implementing the RPP as well as the program for nuclear criticality safety. SAR Section 5.3.2.1.7 addressed the qualifications of the RPCS manager. The applicant stated that it will use the guidance in ANSI/ANS-3.1-1993 (American Nuclear Society, 1993aa) for its radiation protection staffing requirements.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the administrative organization of its RPP using the guidance in YMRP Section 2.1.1.8.3; HLWRS–ISG–03 (NRC, 2007ac); NRC Regulatory Guide 1.8 (NRC, 2000ae); and NRC Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff finds that the RPP organization description has adequately defined the responsibilities of the RPCS manager. The applicant's description of the RPCS manager duties and authority are consistent with NRC Regulatory Guide 8.8 (NRC, 1978ab) because the RPCS manager is independent of operations and maintenance and has clear responsibility to implement the RPP program. The applicant stated that it will provide adequate staffing to support operations and will base the organizational staffing requirements on ANSI/ANS–3.1–1993 (American Nuclear Society, 1993aa). The NRC Regulatory Guide 1.8 (NRC, 2000ae) endorses ANSI/ANS–3.1–1993, with certain clarifications, additions, and exceptions. In SAR Section 5.3.2.1.7, the applicant provided the qualification requirements for the RPCS manager; minimum qualifications are a bachelor's degree in science, health physics, or engineering, with a combined 6 years of experience in the radiological protection aspects of nuclear facility design and operations, and 3 years of supervisory or management experience. In NRC Regulatory Guide 1.8 (NRC, 2000ae), the 3 years of nuclear power plant experience should be at a level requiring policy planning and decision-making related to the programmatic aspects of RPP as a whole. The description for supervisory or management experience specified in ANSI/ANS–3.1–1993 (American Nuclear Society, 1993aa, Section 6.3) includes policy planning and decision-making. Therefore, these qualification requirements are consistent with NRC Regulatory Guide 1.8. The applicant also stated that it will include radiological response personnel to support emergency response functions. These statements provide reasonable assurance that there will be adequate resources to maintain ALARA goals and objectives. The NRC staff finds acceptable that the applicant will review and assess the adequacy of the RPP program and the content at least annually in accordance with 10 CFR 20.1101(c).

2.1.1.8.3.5.2 Equipment, Instrumentation, and Facilities

As a part of the RPP, the applicant stated that it will describe and identify the equipment, instrumentation, and facilities used to support radiological monitoring, personnel protection, and contamination control. The RPP would describe the equipment to be used, including monitoring equipment and personnel protective equipment, as well as equipment to identify and mark access controlled areas. In SAR Section 5.11.2, the applicant described its plans to use instrumentation that is appropriate for the types, levels, and energies of radiation at the GROA and for the expected environmental conditions. The applicant stated that instrumentation will be periodically calibrated to the National Institute of Standards and Technology standards and routinely tested for operability. According to the SAR, the applicant will calibrate instruments and equipment used for quantitative measurements in accordance with NRC Regulatory Guide 8.6 (NRC, 1973ab); ANSI N323A-1997 (American Nuclear Society, 1997aa); and ANSI N323B-2003 (American Nuclear Society, 2003aa); as well as manufacturer recommendations. The applicant stated that surveys and monitoring would be conducted. In particular, personnel dosimeters would be evaluated by a processor holding a current accreditation from the National

Voluntary Laboratory Accreditation Program of the National Institute of Standards and Technology. The applicant also stated that it will provide a radiation protection organization with adequate facilities to effectively implement its RPP.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the radiation protection equipment, instrumentation, and facilities using the guidance in YMRP Section 2.1.1.8.3 and HLWRS–ISG–03 (NRC, 2007ac). The NRC staff finds that the applicant provided a high-level description of the type of protective equipment it will include in its RPP. The NRC staff finds that the applicant's description is adequate for construction authorization because it is consistent with the guidance provided in NUREG–1567 (NRC, 2000ab, Section 11.4.4.2), as applicable.

For the instrumentation, the applicant's description of how it will identify the types and quantities of instrumentation is acceptable because the applicant stated it will (i) consider the radiation types, levels, and energies and (ii) consider the environmental conditions and calibrate its instrumentation in accordance with NRC Regulatory Guide 8.6 (NRC, 1973ab); ANSI N323A–1997 (American Nuclear Society, 1997aa); ANSI N323B-2003 (American Nuclear Society, 2003aa); and manufacturer recommendations. The NRC staff finds the applicant's approach for selection and calibration of radiation protection instrumentation acceptable because it is consistent with the guidance in NUREG–0800 (NRC, 1981ae, Section 12.5), as applicable. The NRC staff finds acceptable that the applicant will conduct surveys and monitoring in accordance with the requirements in 10 CFR 20.1501 and 10 CFR 20.1502. The NRC staff also finds acceptable that the applicant will evaluate personnel dosimeters by a processor holding a current accreditation from the National Voluntary Laboratory Accreditation Program of the National Institute of Standards and Technology, as required by 10 CFR 20.1501(c). For area monitoring, the applicant provided in SAR Section 1.4.2 a high-level system description of the process and area monitoring equipment used to monitor effluents from the GROA release points. The applicant stated that this system will provide both historical and real-time information and will operate on a continuous basis. The NRC staff reviewed the applicant's statement that it would provide radiation protection facilities to implement the proposed RPP. The facilities include monitoring, access control, work areas, decontamination, storage, dosimetry, radiation protection records maintenance, and laboratory facilities. The NRC staff finds the applicant's approach acceptable because it is consistent with the guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab) and will support radiation protection operations, training, and assessments, consistent with the guidance in NUREG–1567 (NRC, 2000ab, Section 11.4.4.2) and NUREG–0800 (NRC, 1981ae, Section 12.5), as applicable.

2.1.1.8.3.5.3 Policies and Procedures Used for RPP Implementation

The applicant described the policies and procedures to be used to implement the RPP in SAR Section 5.11.3. The applicant stated that the RPP will be implemented through procedures and work controls that ensure that radiation protection measures are employed commensurate with the scope and extent of licensed activities for the protection of workers, the public, and the environment, as required by 10 CFR 63.21(c)(6). According to the SAR, the implementation of the RPP will establish that (i) radioactive material is controlled; (ii) potential for radioactive contamination of personnel, equipment, and areas is minimized; (iii) onsite generation of low-level radioactive waste and effluents is minimized; (iv) facilities, equipment, training, and qualified staff will be available to provide adequate radiation protection and safe radiological operations consistent with ALARA principles; and (v) individual and collective occupational and public doses are maintained below regulatory limits and are consistent with ALARA principles.

The applicant stated that it will develop the policies and procedures for the following program elements so that its program will meet 10 CFR Part 20 requirements:

- Radiation Surveys and Radiological Postings—The applicant stated that radiation survey policies and procedures will be developed to address the survey requirements of 10 CFR 20.1501, 10 CFR 20.1502, 10 CFR 20.1703, 10 CFR 20.1906, and 10 CFR 20.2101. According to the SAR, radiological postings will be in accordance with 10 CFR 20.1901 through 10 CFR 20.1903.

- Radiological Access Control and Onsite Dose—The applicant stated that the access control system will be developed to comply with 10 CFR 20.1601 and 10 CFR 20.1602. The applicant also stated that it will follow NRC Regulatory Guide 8.38 (NRC, 2006ac). According to the SAR, onsite dose requirements of 10 CFR 20.1201 through 10 CFR 20.1208 and 10 CFR 20.1301 will be met by identifying occupational dose monitoring practices in the RPP and a methodology to monitor dose limits for members of the public. The applicant stated that it will follow NRC Regulatory Guide 8.35 (NRC, 1992ac) for planned special exposures.

- Control of Radiological Material and Contamination—The applicant indicated that controls will be implemented to minimize the amount of material and equipment brought into areas and to control radioactive materials. According to the SAR, materials will be labeled and marked in accordance with 10 CFR 20.1904 and 10 CFR 20.1905. The applicant stated that it will meet the requirements of NRC Regulatory Guide 1.86 (NRC, 1974aa, Table 1) for determining whether materials and equipment can be released outside of restricted areas.

- Monitoring of External and Internal Dose—The applicant stated that the procedures and policies will be developed to meet the requirements of 10 CFR 20.1501(c), 10 CFR 20.1502, and 10 CFR 20.1204. The applicant indicated that it will follow NRC Regulatory Guide 8.34 (NRC, 1992ab) for monitoring methods and criteria for occupational doses. The applicant stated that it will follow NRC Regulatory Guide 8.9 (NRC, 1993aa) for internal dose monitoring. The applicant also stated that it will select dosimeters consistent with NRC Regulatory Guide 8.4 (NRC, 1973aa, Paragraphs C and C.1), and ANSI N322–1997 (American Nuclear Society, 1997ab). According to the SAR, the applicant will follow ANSI N42.20–2003 (American Nuclear Society, 2003ab) for the active personnel dose and dose rate warning system.

- Analysis of Airborne Radioactivity Sampling—The applicant stated that procedures will be developed to meet 10 CFR Part 20 requirements for surveys and measurements, and the applicant stated that it will follow the guidance in NRC Regulatory Guide 8.25 (NRC, 1992aa).

- Respiratory Protection—The applicant stated that it will include respiratory protection in accordance with 10 CFR 20.1701 through 10 CFR 20.1705 and will follow the guidance in NRC Regulatory Guide 8.15 (NRC, 1999ac).

- Radiation Protection Training—The applicant stated that its training will be consistent with the guidance in NRC Regulatory Guide 8.8 (NRC, 1978ab, Section C.2); NRC Regulatory Guide 8.27 (NRC, 1981aa); NRC Regulatory Guide 8.29 (NRC, 1996ac); and ASTM E 1168–95 (ASTM International, 1995aa).

- Notices to Workers—The applicant stated that it will post notices in accordance with 10 CFR 19.11 and 10 CFR 63.9(e)(1).

- Protection of the Pregnant Worker and Embryo/Fetus—The applicant stated that it will develop a program to implement the requirements of 10 CFR 20.1208 and will follow guidance in NRC Regulatory Guide 8.13 (NRC, 1999ab, Section C), and NRC Regulatory Guide 8.36 (NRC, 1992ad).

- Radiation Protection Records and Reports—The applicant stated that its program will address applicable requirements of 10 CFR 20.2101 through 10 CFR 20.2110, 10 CFR 20.2201 through 10 CFR 20.2206, and 10 CFR 19.13. The applicant also stated that it will follow the guidance in NRC Regulatory Guide 8.7 (NRC, 2005ab) and ANSI/HPS N13.6–1999 (American Nuclear Society, 1999aa).

- Environmental Radiological Monitoring—The applicant indicated that its environmental monitoring program will comply with 10 CFR 63.21(c)(6), 10 CFR 63.111(a)(2), 10 CFR 20.1101(d), 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR 20.1501, and 10 CFR 20.2001. The applicant also stated that it will follow NRC Regulatory Guide 1.21 (NRC, 2009aa).

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's description of the policies and procedures to be used to implement the RPP using the guidance in YMRP Section 2.1.1.8.3; HLWRS–ISG–03 (NRC, 2007ac); NUREG–1567 (NRC, 2000ab); and NUREG–0800 (NRC,1981ae). The NRC staff finds that the applicant's description of the policies and procedures is acceptable because it is consistent with commonly accepted programs and practices for radiation protection. The applicant adequately described the major program elements and included the applicable regulatory criteria and guidance for proper program implementation, consistent with NUREG–1567(NRC, 2000ab, Section 11.4.4, Table 11.2).

The NRC staff finds that the proposed RPP is commensurate with the scope of normal activities proposed for the GROA (e.g., the RPP includes policies and procedures for radiation surveys and postings, dose monitoring, radiation protection training, radiation protection records and reports).

The RPP also addresses (i) the administrative organization of the RPP; (ii) the descriptions of health physics equipment, facilities, and instruments; (iii) the description of policies and procedures for controlling access to the radiation area, description of procedures for the accountability and storage of radioactive material, and the radiation protection training programs; and (iv) the description of program implementation. The NRC staff also finds that the description of the RPP is consistent with the assumptions used in the PCSA consequence estimates, as reviewed in SER Section 2.1.1.5; the means to limit dose, as reviewed in SER Section 2.1.1.6; and the ALARA considerations, as reviewed in SER Section 2.1.1.8. Therefore, for these reasons, the applicant's policies and procedures in its RPP are acceptable and commensurate with the scope of normal activities proposed for the GROA.

**NRC Staff's Conclusion**

The NRC staff finds that the applicant adequately described the proposed RPP for the GROA, in accordance with 10 CFR 20.1101, as required by 10 CFR 63.21(c)(6).

### 2.1.1.8.4 Evaluation Findings

The NRC staff reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(6), and finds, with reasonable assurance, that the requirements of 10 CFR 63.111(a)(1) are satisfied. Based on the information provided, the NRC staff has reasonable assurance that the applicant will implement a RPP that will maintain occupational doses and public exposures below the applicable limits of 10 CFR Part 20. The operations at the GROA, through permanent closure, will comply with the ALARA requirements in 10 CFR Part 20.

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# CHAPTER 9

## 2.1.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes

### 2.1.2.1 Introduction

Safety Evaluation Report (SER) Section 2.1.2 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of Safety Analysis Report (SAR) (DOE, 2008ab, Section 1.11) as supplemented by the applicant's [U.S. Department of Energy (DOE)] responses to the NRC staff's requests for additional information (RAIs) (DOE, 2009ba, 2009bb). The objective of the review is to evaluate the feasibility and reasonableness of the applicant's retrieval plan and alternate storage by determining whether the repository design preserves the option of waste retrieval if retrieval becomes necessary.

In SAR Section 1.11, the applicant described its plans for retrieval and alternate storage of the radioactive wastes, including a discussion of compliance with the preclosure performance objectives. The applicant's description of its alternate storage plan identified a proposed alternate storage facility, including location, size, and storage operations. The applicant also provided a schedule for retrieval operations, should retrieval become necessary.

### 2.1.2.2 Regulatory Requirements

The regulatory requirements applicable to this section are in 10 CFR 63.21(c)(7) and 10 CFR 63.111(e):

- 10 CFR 63.21(c)(7) requires that the Safety Analysis Report include a description of plans for retrieval and alternate storage of the radioactive wastes, should retrieval be necessary.

- 10 CFR 63.111(e)(1) requires that the geologic repository operations area must be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and thereafter until the completion of the performance confirmation program. To satisfy this objective, the geologic repository operations area must be designed so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission.

- 10 CFR 63.111(e)(2) does not preclude decisions by the Commission to allow backfilling part, or all of, or permanent closure of the geologic repository operations area, before the end of the period of design for retrievability.

- 10 CFR 63.111(e)(3) defines a "reasonable schedule" for retrieval as one that would permit retrieval in about the same time as that required to construct the geologic repository operations area and emplace waste.

Finally, the retrieval operations are to be conducted in a manner consistent with the criteria for safety during preclosure operations governed by the requirements of 10 CFR 63.111(a), (b), and (c) these requirements provide (i) for protection against radiation exposures and releases of radioactive material, (ii) numerical guides for design objectives, (iii) for an adequate preclosure safety analysis demonstrating compliance with 10 CFR 63.111(a) and (b).

The NRC staff reviewed DOE's information using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa, Section 2.1.2). The relevant acceptance criteria are

- Adequate descriptions of plans for retrieval of waste packages are provided;

- The retrieval plan incorporates radiation safety, including as low as reasonably achievable (ALARA) considerations;

- The proposed description of alternate storage of retrieved radioactive wastes is reasonable; and

- A reasonable schedule for potential retrieval operations is provided.

### 2.1.2.3 Technical Review

The applicant proposed a preclosure period of 100 years, which includes construction of the geologic repository operations area, emplacement of waste in the underground facility, performance confirmation, and the 50-year retrievability period prescribed in 10 CFR 63.111(e) (SAR Section 2.2 and general information Section 1.1.2.1). The applicant's retrieval plan consists of maintaining access to waste packages in emplacement drifts throughout the preclosure period, such that waste packages could be retrieved, if necessary, by reversing the operational procedure used for waste emplacement. The applicant plans to accomplish this by (i) designing the ground support system in the access and ventilation mains and emplacement drifts to function for 100 years; (ii) developing a maintenance plan to test, inspect, and repair ground support as necessary to ensure functionality of the underground openings through a 100-year preclosure period; and (iii) designing the subsurface communication and transportation infrastructure to function through the preclosure period to support access for maintenance or equipment replacement as needed. The applicant also stated that if off-normal events (i.e., those outside the bounds of routine operations but within the range of analyzed conditions for SSCs), such as collapse of an emplacement drift section occurred, specialized procedures and equipment could be developed to restore access to waste packages. The applicant also identified an alternate storage facility location. The applicant did not propose the option of backfilling of emplacement drifts.

The NRC staff reviewed the applicant's description of its retrieval operations provided in SAR Section 1.11. Specifically, the NRC staff reviewed the applicant's waste retrieval plan to determine whether (i) waste packages could be retrieved during the period of potential waste retrieval by reversing the operational procedure for waste emplacement, (ii) the applicant identified a reasonable range of potential problems (off-normal scenarios) during retrieval, and (iii) the applicant described approaches for restoring access to waste packages from potential off-normal conditions without physical damage or overheating of the affected waste packages. The NRC staff also reviewed the applicant's retrieval operations schedule and description of alternate waste storage plans. The applicant's plan and the NRC staff's review and findings are summarized in the following sections.

#### 2.1.2.3.1 Waste Retrieval Plan

**Retrieval Under Normal Operations**

The applicant described its waste retrieval plan in SAR Section 1.11.1. In the plan, the applicant described the structures, systems, and components (SSCs) used for retrieval. The applicant would retrieve waste by performing emplacement operations in reverse, using the same SSCs used for emplacement. The SSCs relied upon are the transport and emplacement vehicle (TEV), invert structure and rails, electrical power system, communication system, and drift ventilation system. The applicant's plan includes maintaining access to the emplacement drifts and keeping the SSCs available throughout the preclosure period (DOE, 2009ba,bb).

The applicant described a monitoring and maintenance plan for the ground support system to keep the subsurface facility openings stable to permit access to the SSCs and waste packages. The applicant's monitoring plan for accessible openings (such as access mains and the North Ramp) consists of regular visual inspection of the openings by qualified personnel and use of a geotechnical instrumentation program to obtain measurements of drift convergence, ground support loads, and potential overstressed zones (DOE, 2009bb). The applicant indicated that, for the emplacement drifts and turnouts, it will use remotely operated equipment to inspect the openings to detect any indications of rockfall, drift deterioration, or instability and to measure drift convergence at locations selected on the basis of previous inspections (DOE, 2009bb). The applicant stated that every emplacement drift and turnout will be inspected over its entire length: once a year initially after waste emplacement but at a modified frequency subsequently. The applicant stated that subsequent inspection frequencies would use results of previous inspections and geologic mapping to support any changes because the frequency of monitoring is a key component of the monitoring program.

**NRC Staff's Evaluation**

The NRC staff used the guidance in the YMRP to determine whether the applicant adequately described the retrieval operations, including the equipment to be used. The NRC staff compared the emplacement operations to the retrieval operations and determined that these operations are the same except that during retrieval, the transport and emplacement vehicle (TEV) must climb a 2.5 percent grade when loaded with a waste package. During emplacement, the TEV is only loaded when descending. The NRC staff reviewed the TEV design to determine whether the TEV is designed to perform retrieval operations and whether the loading system (or propulsion duty cycle) is designed to climb a 2.5 percent grade when loaded with a waste package. The NRC staff's review of the TEV design in SER Section 2.1.1.7.3.5.1 finds that the TEV is designed to support waste package transportation and its drive system is designed to negotiate a 2.5 percent grade in both downward and upward directions when loaded with a waste package. As discussed in SER Section 2.1.1.2, the invert structure and rails, electrical power system, communication system, and drift ventilation system are designed to support retrieval operations. The NRC staff finds that the underground facility design along with the monitoring and maintenance programs would ensure accessibility to waste packages throughout the preclosure period.

The NRC staff also evaluated whether the applicant's plan to inspect the emplacement drifts and turnouts using remotely operated equipment is reasonable. In response to NRC staff's RAI, the applicant stated that it will inspect the entire length of every emplacement drift and turnout annually. After reviewing the applicant's RAI response (DOE, 2009bb), the NRC staff concludes that the applicant has provided sufficient spatial and temporal coverage of observations necessary to assess performance of the ground support systems. The applicant stated in DOE (2009bb) that it might change its inspection frequency if information gathered up to that point in time supports such a change. The applicant stated that the basis for change in the inspection frequency of ground support would be properly documented and supported as required by 10 CFR 63.44(c). The staff concludes that the applicant could adjust the temporal frequency of inspections as conditions change in accordance with the 10 CFR 64.44 process, provided the inspection is frequent enough to permit an assessment of the rate of any change in ground support conditions. The NRC staff's review of DOE's commitment to use the 10 CFR 63.44 process is documented in SER Section 2.5.10.1.3.1.1. As the applicant would document the basis for changes in the inspection frequency of ground support, the NRC staff finds that it can evaluate the effects of changes in the frequency of inspection, as needed. The NRC staff finds that the applicant's plan to inspect the emplacement drifts and turnouts once a year initially and modify the inspection frequency as necessary in accordance with 10 CFR 63.44 provides adequate temporal coverage of observations necessary to assess performance of the ground support systems.

**Retrieval Scenarios under Off-Normal Conditions**

The applicant postulated a number of off-normal events (i.e., conditions outside the bounds of routine operations but within the range of analyzed conditions) and evaluated strategies to recover from such events. Of these, the applicant identified two off-normal occurrences that could hinder access to waste packages during the preclosure period (BSC, 2007bw): derailment of a transport and emplacement vehicle (TEV) and rockfall resulting in rubble accumulation. The applicant used these two scenarios to encompass the range of potential structures, systems, and components (SSCs) failures that could affect access to waste packages during the preclosure period.

TEV derailment could result from damage to the invert structure or rail or from TEV malfunction. Recovery from such a derailment would involve isolating the affected area from radiation in adjacent areas, repairing damaged equipment, and lifting or pulling the TEV to the rail system. The second set of off-normal conditions related to rockfall was grouped together because of the similar operations needed to recover from such occurrences. Recovery actions include building a radiation barrier, removing rubble, and repairing ground support.

The applicant described the conceptual design of a multipurpose recovery vehicle (MRV) for recovering from potential off-normal occurrences. According to the applicant's description, the MRV design will be based on the TEV design. The NRC staff's review of the TEV is documented in SER Section 2.1.1.7. 3.5.1. The applicant indicated that the MRV will be a rail-based, remotely operated vehicle with hardware to support recovery operations. The MRV hardware includes (i) lights, cameras, and communication (potentially wireless) for remote, visual operation (teleoperation); (ii) batteries or tethered cables for loss-of-power conditions; and (iii) telescoping boom crane, manipulator arms and various attachments, winch, and rail clamps for remotely clearing rubble and for pulling a TEV or disassembling equipment. The applicant's plan relies on this concept of a multipurpose vehicle for restoring a derailed TEV or a collapsed emplacement drift to normal conditions.

The applicant identified three derailment conditions within emplacement drifts that encompass several severity levels of TEV failures and associated recovery plans.

(1) The minor failure case considered a derailment where the TEV retains full functionality. The applicant's recovery plan consisted of placing commercially available "rerailers" along the rail and using the MRV to drag the TEV back onto the rail.

(2) A more severe case considered a derailment that includes damage to the TEV drive system, the front shield doors in an open state, and the base plate fully extended and inoperable, thus providing no shielding protection. The applicant proposed to use winches, rail clamps, and rerailers to remove the base plate and pull the TEV onto the rails.

(3) The most severe scenario considered involved the repair of damaged rails near emplaced waste packages. In DOE (2009ba, Enclosure 3), the applicant described how a boom crane could be used to construct a temporary shield wall near the waste package such that workers could enter the emplacement drift and install new rails.

The applicant identified (BSC, 2008bt) three waste package failure modes that could result from overheating of a waste package buried in rubble.

(1) Loss of impact properties for the outer corrosion barrier (OCB) due to exposure to a temperature of 538 °C [1,000.4 °F] or higher.

(2) Creep rupture of minimum-strength weldment material due to an OCB temperature of 501 °C [933.8 °F] or higher.

(3) Pressure-induced rupture of the bottom lid of the OCB for minimum-strength material due to exposure to a temperature of 400 °C [752 °F] or higher near the center of the waste package lid.

The applicant analyzed the thermal effects on a waste package buried in rubble using the numerical code ANSYS and considering representative heat transfer parameters for the drift. The applicant analyzed a range of conditions to determine whether conditions such as collapse of the emplacement drift or rubble blockage of a ventilation conduit could interfere with retrieval operations (e.g., compromise of structural integrity of the waste package due to high temperatures) and concluded that no Category 1 or 2 event sequences (Category 1 and 2 event sequences are defined in 10 CFR 63.2) would interfere with retrieval. The applicant also concluded, on the basis of its calculations, that low-probability, beyond design bases conditions (off-normal conditions considered by the applicant) would not be likely to interfere with retrieval operations.

The applicant described the installation of a temporary shield wall, the design of the support structure, and the shape of the shield bricks such that direct radiation from the joints would be prevented. The applicant described the design and development of a multipurpose recovery vehicle (MRV) that would be needed for rubble removal. According to the applicant's plan, it would take approximately 8 years to design, develop, and build the MRV before retrieval can be initiated.

**NRC Staff's Evaluation**

The NRC staff used the guidance in the YMRP to perform a risk-informed, performance-based review of the applicant's proposed retrieval scenarios under degraded drift conditions and methodologies established for identifying and analyzing potential recovery problems for the various retrieval scenarios. The NRC staff notes that the combination of drift collapse and simultaneous loss of ventilation is an event sequence beyond Category 2 and would be considered unlikely. However, the staff reviewed the applicant's postulated off-normal conditions as a part of the review of DOE's responses to NRC staff's RAI.

For an off-normal condition involving rockfall, the NRC staff reviewed the applicant's evaluation of a scenario involving waste package burial in BSC (2008bt) and its response to staff's RAI in DOE (2009ba, Enclosure 4). The NRC staff reviewed the applicant's assumptions for representing a rockfall condition and the analytical approach to determining the thermal effects of rock rubble. The NRC staff finds the applicant's assumptions regarding the rockfall covering the waste package to be reasonable because they approximate the physical conditions after the postulated rockfall. The NRC staff finds the analytical approach the applicant used to be appropriate because the numerical code used (ANSYS) is a widely accepted code for performing such thermal analyses. The NRC staff reviewed the applicant's calculations taking into consideration any reliance on subsurface ventilation for cooling the buried waste packages and potential impacts on the retrieval schedule under off-normal conditions. The NRC staff reviewed the temperature profile of a waste package partially or completely buried in rubble and finds, on the basis of verifications against the information provided in BSC (2008bt), that the applicant's analyses considered appropriate failure modes of concern and the associated temperature limits as per specifications.

The NRC staff reviewed the thermal limits provided in DOE (2009ba) and finds that the impact and creep rupture limits were based on the ASME codes and the pressure-induced rupture limit was based on a calculation using the ANSYS code. The NRC staff finds these thermal limits for the failure modes (impact, creep rupture, and failure of the bottom lid due to over-pressurization) acceptable because they are based on standard codes and acceptable analytical methods as per standard engineering practice.

The NRC staff reviewed the calculations, and their bases, the applicant used to estimate the temperature of the waste package due to drift collapse and loss of ventilation. The applicant's calculations show that the waste package temperature would not exceed the thermal limits for failure. This determination is based on restoring ventilation and removing rubble from around waste packages within a 30-day period as stated by the applicant in SAR Section 1.3.1.2.4. The NRC staff finds the applicant's conclusion acceptable because a 30-day period allows sufficient time for rubble removal and restoration of the ventilation system when drift collapses are of limited extent and confined to small areas.

The NRC staff reviewed information provided by the applicant in BSC (2007bw) and BSC (2008ad), and as described further in responses to NRC staff's RAIs in DOE (2009ba) with respect to the installation of a temporary shield wall during recovery from postulated off-normal conditions to facilitate waste package access and retrievability. The NRC staff finds that this information is adequate for the evaluated off-normal conditions, which included drawings and descriptions of steps to be followed during the recovery process, because the NRC staff finds that the applicant's approach would ensure that retrievability is not precluded. Potential effects of off-normal events on waste retrievability are considered by the NRC staff in its risk-informed, performance-based review in light of the low likelihood of occurrence of the off-normal conditions postulated by DOE. The NRC staff's review and conclusions regarding the likelihood of drift collapse due to seismic events during the preclosure period are documented in SER Section 2.2.1.3.2. There, the NRC staff finds acceptable the applicant's conclusion that seismic ground motions strong enough to significantly damage an emplacement drift have a very low likelihood of occurring based on a review of the applicant's analyses. Given the underground facility design where the emplacement drifts would be designed to remain stable for 100 years and the applicant's plans to inspect, monitor, and maintain the emplacement drifts and invert structure during the preclosure period, the NRC staff finds the applicant's recovery plans under postulated off-normal conditions acceptable.

**NRC Staff's Conclusion**

On the basis of its review of the applicant's description of plans for retrieval and plans for monitoring drift convergence documented in SER Section 2.1.1.2, the NRC staff finds that (i) the applicant's description of the plan for maintaining access to waste packages in emplacement drifts through the preclosure period under normal operating conditions is acceptable because the applicant has provided sufficient information about the feasibility of retrieval plans under normal conditions; (ii) the applicant adequately identified retrieval scenarios under degraded drift conditions because the two off-normal scenarios the applicant analyzed bound the possible range of adverse conditions; (iii) the applicant's retrieval plan description under off-normal conditions is acceptable because it considered potential scenarios that could lead to rockfall and derailment, and the applicant performed analyses using accepted engineering models and codes; and (iv) the applicant's proposed solutions to address off-normal conditions are reasonable because they would be feasible and could be implemented within the proposed repository design concepts. The NRC staff finds that the applicant satisfied the requirements of 10 CFR 63.21(c)(7) and 10 CFR 63.111(e) for the waste retrieval plan.

### 2.1.2.3.2 Compliance With Preclosure Performance Objectives

**Preclosure Safety During Retrieval**

In SAR Section 1.11.1.3.1, the applicant discussed its approach to meeting the preclosure performance objectives of 10 CFR 63.111(a) and (b) and limiting radiation exposures during waste retrieval to be consistent with the preclosure safety analysis. The approach is to characterize event sequences, perform consequence analysis, and impose design requirements as explained in SAR Section 1.7.

In SAR Section 1.11.1.3.2, the applicant discussed, in general terms, how as low as is reasonably achievable (ALARA) concepts would be implemented. The applicant did not develop occupational dose limits for retrieval. However, the applicant stated that whatever radiation exposure considerations are applicable to emplacement operations would also apply to retrieval scenarios.

**NRC Staff's Evaluation**

According to SAR Section 1.11.1.3.1, the applicant did not identify any new event sequences for retrieval scenarios. The applicant's approach is based on the assumption that the same equipment and methods would be used for retrieval as in the emplacement operation. The NRC staff has reviewed the applicant's preclosure safety analyses for waste emplacement operations and finds them acceptable, as documented in SER Sections 2.1.1.4 and 2.1.1.5. As stated earlier, the results of the preclosure safety analyses conducted for the waste emplacement operations are deemed applicable to retrieval operations carried out in the reverse order. The applicant stated in SAR Section 1.11.3 that, when a decision to retrieve is made, it will submit additional details on its retrieval plan as needed, including dose calculations, which will be reviewed by the NRC staff.

The NRC staff's review of the operational Radiation Protection Program (RPP) is documented in SER Section 2.1.1.8, where the staff concludes that the applicant's RPP, including implementation of ALARA principles, is reasonable for preclosure operations. On the basis of the description of the RPP, including implementation of ALARA principles for the preclosure operations and the applicant's acknowledgment that similar radiation safety considerations would be applicable during retrieval, the staff concludes that the applicant's ALARA program would also be acceptable for retrieval operations. The NRC staff notes that, if retrieval is required, the applicant plans to implement radiation protection, including the ALARA program, consistent with the radiation protection guidelines current at the time of retrieval and that the applicant's plans would be subject to NRC staff's review and inspections.

**NRC Staff's Conclusion**

Based on the description of the applicant's proposed approach to meeting preclosure performance objectives [specified in 10 CFR 63.111(a) and (b)] for protection of workers and members of the public during repository operations and the applicant's commitment to apply similar safety standards during retrieval, the NRC staff finds that the applicant's safety aspects of the retrieval plan would be acceptable. The applicant satisfied the requirements of 10 CFR 63.21(c)(7) and 10 CFR 63.111(e) for compliance with preclosure performance objectives.

#### 2.1.2.3.3 Proposed Alternate Storage Plans

The applicant indicated in SAR Section 1.11.2 that facilities for handling and storage of retrieved waste packages could be sited in Midway Valley at approximately the location of surface waste handling and aging facilities for waste emplacement (SAR Figure 1.11-1). The applicant estimated that the alternate storage location can be developed to accommodate waste packages containing 70,000 metric tons of heavy metal. As the applicant explained, the facility could be developed to include equipment for unloading waste packages from the retrieval vehicle, transferring the waste packages into shielded long-term storage containers, and transporting the shielded containers to storage pads. The alternate storage location the applicant identified has been characterized for surface waste handling buildings and aging pads, as reviewed in SER Section 2.1.1.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information using the guidance in the YMRP. The NRC staff concludes that the description of surface facilities for handling and storing retrieved waste packages is reasonable because the areal extent of the identified alternate storage site is sufficient to accommodate the needed facilities as described in SAR Figure 1.11-1. The NRC staff finds that the site has the capacity to accommodate all the waste if needed. The NRC staff notes that the actual facility design would be provided for NRC staff's review if a decision to retrieve is made, at which time the amount of waste to be retrieved, the nature of the shielded storage containers, and the storage configuration could be determined.

**NRC Staff's Conclusion**

Based on the review of SAR Section 1.11.2, the NRC staff finds that the information presented in the SAR regarding alternate storage of retrieved waste is acceptable, as evaluated previously, to retrieve and handle disposed waste packages if waste retrieval becomes necessary. The applicant satisfied the requirements of 10 CFR 63.21(c)(7) and 10 CFR 63.111(e) for an alternate storage plan.

#### 2.1.2.3.4 Retrieval Operations Schedule

The applicant provided a conceptual retrieval timeline in SAR Figure 1.11-2 and BSC (2008ad, Section 4.2). According to the figure, after a decision is made to retrieve, initial evaluations would be made during the first 6 months. An additional 24 months are included in the applicant's schedule for developing a design and operational plan, and submitting a license amendment to NRC. The applicant's schedule includes 36 months for the NRC staff's review and a potential hearing and an additional 12 to 36 months for an NRC approval for the applicant's plan. Retrieval operations would take approximately 30 years following NRC approval.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's information using the guidance in the YMRP. The NRC staff notes that the retrieval operations schedule provided by the applicant is conceptual in nature. Actual conditions at the time of retrieval operations could potentially affect the schedule. For example, if the applicant relied on the availability of a multipurpose recovery vehicle (MRV) to recover from the postulated severe off-normal conditions, additional time may be required. In this regard, however, the NRC staff notes that some of the off-normal scenarios analyzed by the applicant are beyond Category 2. For example, a combination of potential drift collapse (presumably due to a low probability seismic event) resulting in waste packages being covered by substantial amounts of rock rubble for a considerable duration and a simultaneous loss of ventilation would be a beyond Category 2 event sequence. The NRC staff notes that designing, building, and procuring an MRV could potentially take longer than the time needed to exceed the thermal limits of rubble-covered waste packages [about 162 days according to SAR Section 1.3.5.3.2.1 and even earlier according to BSC (2008bt)].

The applicant stated that only limited off-normal occurrences would require mitigation during the retrieval process. No Category 1 or Category 2 event sequences were identified by the applicant for the preclosure period that would result in temperature-related problems or physical damage to the waste package and/or pallet that could interfere with retrieval as described in SAR Section 1.7. The NRC staff finds the applicant's following conclusions acceptable: (i) that postulated beyond design bases conditions (BSC, 2007bw; DOE, 2009ba,bb) would be excluded because of low probability (as provided by regulation in 10 CFR 63.2); (ii) that the underground facility design requirements described in SAR Sections 1.3.1 through 1.3.6 and supported by a maintenance plan to test, inspect, and repair ground support as necessary would ensure that accessibility to waste packages would be maintained throughout the preclosure period (as reviewed and documented in SER Section 2.1.1.2.3.3.1); and (iii) that the drift collapses resulting in significant amounts of rockfall require large seismic events, as reviewed and documented in SER Section 2.2.1.3.2, and which are also excluded from consideration in the preclosure safety analyses on the grounds of low probability [excluded features, events, and processes (SER Section 2.2.1.2.1, FEP 2.1.07.02.0A, Drift Collapse)]. The NRC staff finds that if rubble accumulation takes place due to other reasons than a seismic event, they will be limited to zones of weak rock conditions and would likely be a local phenomenon.

Because of the reasons enumerated previously, the NRC staff only considered Category 1 and Category 2 event sequences and excluded consideration of unlikely combinations of low probability events in evaluating the conceptual schedule presented by the applicant. The NRC staff finds the proposed conceptual schedule for potential retrieval operations to be comparable to that of the geologic repository operations area construction and waste emplacement.

**NRC Staff's Conclusion**

On the basis of the review of the information in the SAR and the responses to RAIs, the NRC staff concludes that the schedules related to the retrieval scenario presented by the applicant are reasonable for normal operations. The NRC staff also concludes that a significant impact on the overall schedule is not likely if the applicant decided to retrieve the entire inventory under a reasonable range of potential off-normal conditions postulated by the applicant. On the basis of the reasonableness of the assumptions and the conceptual details provided in the SAR and supporting documents, the NRC staff finds that the overall proposed schedule is achievable. The applicant satisfied the requirements of 10 CFR 63.21(c)(7) and 10 CFR 63.111(e) for the retrieval operations schedule because the proposed schedule is comparable to the overall schedule of geologic repository operations area construction and waste emplacement.

### 2.1.2.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application and finds, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(7) and 10 CFR 63.111(e) are satisfied because (i) the applicant adequately described its plans for retrieval and provided details of the geologic repository operations area design that preserves the option to retrieve any or all of the emplaced waste; (ii) radiation safety, including implementation of ALARA principles, is built into the retrieval concepts; (iii) alternate storage sites of sufficient capacity are identified; and (iv) a reasonable schedule for a potential retrieval scenario is provided.

### 2.1.2.5 References

BSC. 2008ad. "Concepts for Waste Retrieval and Alternate Storage of Radioactive Waste." 800–30R–HER0–00100–000. ML090770717. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2008bt. "Evaluation of an Event Sequence for Waste Package Burial." 800–K0C–WIS0–00600–000–00A. ML091540124. Las Vegas, Nevada: Bechtel SAIC Company.

BSC 2007bw. "Strategies for Recovery After an Off-Normal Event to the Waste Package Transport and Emplacement Vehicle." 800–30R–HE00–01800–000. Rev 000. ML091540122. Las Vegas, Nevada: Bechtel SAIC Company.

DOE 2009ba. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 1.11), Safety Evaluation Report Vol. 2, Chapter 2.1.2, Set 1." Letter (June 2) J.R. Williams to C. Jacobs (NRC). ML091540129. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

DOE. 2009bb. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 1.3.4, 1.3.5, and 1.11), Safety Evaluation Report Vol. 2, Chapter 2.1.2, Set 1." Letter (August 4) J.R. Williams to C. Jacobs (NRC). ML092170409. Washington, DC: U.S. Department of Energy, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: U.S. Nuclear Regulatory Commission.

# CHAPTER 10

## 2.1.3 Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities

### 2.1.3.1 Introduction

Safety Evaluation Report (SER) Section 2.1.3 provides the U.S Nuclear Regulatory Commission (NRC) staff's review of the Department of Energy's ("DOE" or "applicant") geologic repository operations area (GROA) design considerations and its plans to facilitate permanent closure and decontamination or the decontamination and dismantlement (PCDDD) of the GROA surface facilities. In conducting its review, the NRC staff evaluated the information in the DOE's Safety Analysis Report (SAR) Section 1.12 (DOE, 2008ab) related to design considerations to facilitate PCDDD and the applicant's plans for PCDDD. Additionally, the NRC staff reviewed the information the applicant provided in response to the NRC staff's request for additional information (RAI) (DOE, 2009ao).

Consistent with 10 CFR 63.21(a), the information provided by the applicant must be as complete as possible in light of the information that is reasonably available. In determining the acceptability of the applicant's information on permanent closure and decontamination, the NRC staff notes that the applicant's plans are prospective in nature. These plans will not reflect knowledge gained over the course of facility operation (e.g., detailed knowledge of the types, extent, and precise locations of contamination). Consistent with 10 CFR 63.51(a)(6), the applicant is required to submit an application to amend the license before permanent closure that shall include any substantial revisions of plans for PCDDD for NRC review and approval.

### 2.1.3.2 Regulatory Requirements

The regulatory requirements applicable to PCDDD are 10 CFR 63.21(c)(8) and 10 CFR 63.21(c)(22)(vi). The requirements in 10 CFR 63.21(c)(8) require that the applicant provide a description of design considerations intended to facilitate permanent closure and decontamination or decontamination and dismantlement of surface facilities. The requirements in 10 CFR 63.21(c)(22)(vi) require that the applicant include in its SAR information on its plans for the PCDDD of surface facilities.

The NRC staff reviewed the applicant's PCDDD information using the guidance and acceptance criteria in the Yucca Mountain Review Plan (YMRP) Section 2.1.3 (NRC, 2003aa). These criteria are:

- The license application describes and provides bases for features of the geologic repository operations area design that will facilitate permanent closure and decontamination or decontamination and dismantlement of surface facilities.

- The license application includes adequate preliminary plans for permanent closure and decontamination or decontamination and dismantlement of surface facilities.

In addition, the NRC staff used applicable guidance in NRC (2006aa, NUREG–1757), as appropriate, to support its review.

### 2.1.3.3 Technical Review

In SAR Section 1.12, the applicant provided information to address the regulatory requirements in 10 CFR 63.21(c)(8) and 10 CFR 63.21(c)(22)(vi). Specifically, in SAR Section 1.12.1, the applicant provided information on design considerations to facilitate PCDDD. In SAR Section 1.12.3, the applicant provided its plan to facilitate PCDDD. The applicant further stated that it will follow decommissioning program policies and guidance set forth in NUREG–1757 (NRC, 2006aa). In SAR Section 1.12, the applicant stated that its final plans for the decontamination and dismantlement of repository surface facilities in the GROA will be submitted to the NRC for review and approval. The applicant's planning timeline for decontamination or for decontamination and dismantlement is shown in SAR Figure 1.12-1.

The applicant did not discuss any plans for financial assurance in SAR Section 1.12. YMRP Section 2.1.3.2 states that the DOE is not required to provide a financial assurance plan in support of closure or decommissioning. Therefore, the lack of financial assurance plan is acceptable because the applicant is a federal agency and is not required to prepare such a plan.

The NRC staff's review of the design considerations that will facilitate PCDDD and the applicant's plans for PCDDD follows.

#### 2.1.3.3.1 Design Considerations That Will Facilitate Permanent Closure and Decontamination or Decontamination and Dismantlement

In accordance with 10 CFR 63.21(c)(8), the applicant described in SAR Section 1.12.1 the design considerations it will use to ensure that the design features of the GROA surface facilities will facilitate permanent closure and decontamination or decontamination and dismantlement of the facilities.

The applicant stated that it will evaluate and select those design features that facilitate PCDDD over competing alternatives where feasible and economical. The evaluation and selection of alternatives will be documented during the design process. The applicant provided a list of requirements and criteria that will be applied as the design progresses to ensure that design features to facilitate and support PCDDD will maintain radiation doses to the public and workers as low as is reasonably achievable (ALARA). Examples of the design considerations the applicant proposed to use to facilitate PCDDD are (i) selection of materials and processes to minimize waste production, (ii) use of a stainless-steel-lined wet handling pool with a leak-detection drainage system to minimize the contamination of concrete around the pool, (iii) incorporation of features to contain leaks and spills, and (iv) the incorporation of features that maintain occupational and public radiation doses ALARA during decommissioning. The applicant also provided examples of design and operational considerations, described in SAR Section 1.12.1, that it will use to minimize contamination, such as (i) minimizing the handling of uncanistered radioactive waste and (ii) use of Transportation Aging and Disposal (TAD) canisters to minimize the number of canisters to be opened. In its response to NRC staff's RAI, the applicant identified criteria that will ensure that the design considerations to facilitate PCDDD will be evaluated as the design progresses towards final design (DOE, 2009ao).

The applicant stated that during the design process structures systems and components will be reviewed for PCDDD considerations to ensure that features that support waste minimization and worker safety are incorporated and ALARA principles are considered in the design. The applicant further stated that evaluation and selection of alternatives for design features will be documented during the design process.

**NRC Staff's Evaluation**

The NRC staff used the guidance in YMRP Section 2.1.3 to determine whether the applicant adequately described in SAR Section 1.12.1 GROA design considerations that will facilitate PCDDD. The NRC staff also evaluated the information the applicant provided in its response to the NRC staff's request for additional information (RAI) (DOE, 2009ao). Specifically, the NRC staff reviewed the applicant's information on the design considerations it will use to facilitate PCDDD identified in SAR Section 1.12.1. Examples of the types of design considerations identified by the applicant in SAR Section 1.12.1 include (i) selection of materials and processes to minimize waste production, (ii) selection of materials and incorporation of features intended to ease decontamination and dismantlement such as in reinforced concrete structures that facilitate demolition techniques, and (iii) use of confinement systems to contain and minimize the spread of potential radioactive contamination generated during process operations and to isolate noncontaminated areas of surface facilities from potentially contaminated areas. The NRC staff also reviewed the applicant's design consideration information included in SAR Section 1.12.1. Examples of the types of design considerations identified by the applicant in SAR Section 1.12.1 include (i) use of TAD canisters to minimize the number of canisters to be opened and (ii) Storm water drainage diversion channels to protect the GROA from runoff from slopes above the facilities, thereby reducing the potential for this water to become contaminated. The NRC staff has determined that the design considerations identified in SAR Section 1.12 are adequate because they are the types of design considerations generally followed in the nuclear industry to reduce contamination and facilitate PCDDD. Additionally the NRC staff has determined that the applicant's design considerations are adequate because they are consistent with the guidance in the YMRP Section 2.1.3.

**NRC Staff's Conclusion**

On the basis of the evaluation discussed before, the NRC staff concludes that the applicant has acceptably described design considerations intended to facilitate PCDDD because the design considerations identified by the applicant in SAR Section 1.12.1 are the types of design considerations typically followed in the nuclear industry to reduce contamination and facilitate PCDDD. Therefore, the NRC staff concludes, with reasonable assurance, that the design considerations identified by the applicant to facilitate PCDDD meet the requirements of 10 CFR 63.21(c)(8).

### 2.1.3.3.2 Plans for Permanent Closure and Decontamination or Decontamination and Dismantlement

The applicant provided in SAR Section 1.12.3 information on its plans for PCDDD of the GROA surface facilities to address the requirements in 10 CFR 63.21(c)(22)(vi). The applicant described in SAR Section 1.12.3 the information it will collect and how it will maintain that information during the life of the facility. SAR Section 1.12 stated that the applicant will submit the final plans for the decontamination and dismantlement of the repository facilities in the GROA before permanent closure for NRC review and approval. The NRC staff's review of the applicant's plans for PCDDD is discussed as follows.

#### 2.1.3.3.2.1 Facility History

The applicant stated in SAR Section 1.12.3.1 that its plan will collect the following information on facility history: (i) the types of radioactive material received and processed at the GROA; (ii) the nature of the authorized use of radioactive materials at the GROA; (iii) the activities at the GROA that could have contributed to residual radioactive material being present at the GROA and the measures immediately taken to remove such contamination; (iv) the activities to be authorized under the license; (v) past authorized activities using licensed radioactive material at the site; (vi) activities involving radioactive material that could contribute to residual radioactivity being present at the site prior to the start of licensed operation; and (vii) previous decontamination, dismantlement, or residual activities at the site. The applicant also stated that the types of information collected by the plan will include information on the facility radiation monitoring information and records on contamination at the facility. The applicant further stated that this information will be available at the time of PCDDD because the applicant will create and maintain this information in accordance with its records management and document control process described in SAR Section 5.2.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the description of the types of information the applicant will collect on facility history using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff has determined that the applicant's plan for collecting information on facility history is adequate because the plan will collect information on past activities at the site, including operational activities that involve radioactive material. This information will facilitate PCDDD activities at the time of permanent closure. The NRC staff further determines that the applicant's plan for collecting information on facility history is adequate because it is comprehensive and follows the guidance provided in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document controls process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to facility history will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for collecting facility history information and for maintaining that facility history information is acceptable because it collects the types of information that will be needed to facilitate PCDDD. The NRC staff also concludes that the applicant's plan is acceptable because the applicant's use of its records management and document control process means that the information will be available at the time of permanent closure and decommissioning. Therefore, the NRC staff concludes with reasonable assurance that the information the applicant provided meets the requirements of 10 CFR 63.21(c)(22)(vi).

#### 2.1.3.3.2.2 Facility Description and Dose Modeling Evaluations

The applicant described, in SAR Section 1.12.3.2, its plan for collecting information related to the facility and the facility's environment that will be used to estimate doses to onsite and offsite populations during the time of PCDDD activities. Examples of the types of information that will be available to estimate doses include (i) a description of the GROA; (ii) a description of population distribution around the site; (iii) a summary of uses and potential uses in and around the site; (iv) site meteorological, geological, seismology and climatology conditions; and (v) descriptions of natural and water resources at the site.

The applicant also described in SAR Section 1.12.3.4 the types of information that will be used to develop radiological dose models. The applicant further stated that its dose modeling evaluations will be used to demonstrate that the total effective radiological dose equivalent to a critical group of individuals near the preclosure controlled area is consistent with ALARA principles and will not exceed regulatory requirements of radiological dose limits to the public and workers during PCDDD activity. The type of information the applicant will use to develop its radiological dose models include (i) source term information; (ii) a description of the exposure scenario; (iii) a description of the conceptual model of the site; (iv) identification, description, and justification of the mathematical models used; (v) a description of the analysis parameters; (vi) a discussion of the accuracy and quality control of the radiological dose modeling results; and (vii) input and output files or printouts, if a computer program is used. The applicant further stated that the information will be created and maintained in the applicant's records management and document control process described in SAR Section 5.2.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan describing the type of facility information it will collect and use to estimate radiological doses to onsite and offsite populations at the time of permanent closure and decommissioning, as well as the applicant's plan to develop its radiological dose models. This review was conducted using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff finds that the applicant's plans are adequate because they will collect the types of information, including, for example, information on the GROA facility, facility site information, populations, site meteorological conditions, source term information, exposure pathways and time estimates, and exposure groups that will enable the applicant to estimate radiological doses to onsite and offsite populations during permanent closure and decommissioning. The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document controls process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to facility description and dose modeling will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plans to collect and maintain information on the facility and its environs and to develop its radiological dose models are acceptable because they collect the types of information that will enable the applicant to evaluate radiological doses to onsite and offsite populations at permanent closure and facilitate decommissioning. The NRC staff also concludes that the applicant's plans are acceptable because the applicant's use of its records management and document control process means that the information will be available for permanent closure and decommissioning. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's plans for gathering information related to the facility and its environments and for radiological dose modeling meets the requirements of 10 CFR 63.21(c)(22)(vi).

### 2.1.3.3.2.3 Radiological Status of the Facility

The applicant described in SAR Section 1.12.3.3 how it will evaluate the radiological status of the facility and determine the anticipated magnitude of decontamination activities or of decontamination and dismantlement activities at the time of PCDDD. The information to perform these evaluations will be based on the facility operational records and data, radiological surveys and assessments, and safety and hazards analysis.

In SAR Sections 1.12.3.3.1 and 1.12.3.3.2, the applicant listed the information that will be available during PCDDD to evaluate the radiological status of the GROA surface facility, such as (i) a summary of the background levels used during scoping or characterization surveys; (ii) a list/description/location of structures, systems, and components that contain residual radioactive material exceeding site background levels; (iii) a summary of the radionuclides present at each location; (iv) the maximum and average radiation levels at the surface of each component; (v) a summary of the access control measures that may be implemented during remedial action, a description of the Radiation Protection Program (RPP), and the identification of the regulatory requirements that guide the program; (vi) a summary of the types and approximate quantities of contaminated materials at each location; and (vii) a scale drawing or map showing the location of contaminated systems and components.

In SAR Section 1.12.3.3.3, the applicant stated that the following information will be available during PCDDD to evaluate the radiological status of soil contamination: (i) a summary of the soil radioactive material background levels used during scoping or characterization surveys, (ii) a list/description of locations at the facility at which soil contains residual radioactive material exceeding site background levels, (iii) a summary of the radionuclides present at each location, (iv) the maximum and average contaminated soil at each location, (v) a summary of the access control measures that may be implemented during remedial action and a description of the RPP, (vi) a scale drawing/map showing the locations of radionuclide material contamination in soil, (vii) soil characteristics at each contaminated soil location, (viii) identification of the sources and quantities of uncontaminated materials from a nearby location that can be used to backfill excavations and reestablish area surfaces, (ix) grading and contouring considerations at each contaminated soil location, and (x) the depth of the soil contamination at each location.

In SAR Section 1.12.3.3.4, the applicant discussed its plans for addressing potential water contamination from process operations. On the basis of the site characterization, the applicant determined that there are no natural surface water bodies at the site (SAR Section 1.1.1.2). The applicant stated that it would develop storm water drainage diversion channels to protect the GROA from runoff from slopes above the facilities and keep storm water from becoming contaminated. The applicant stated it will provide two storm water detention impoundments and analyze the water for radioactive contamination in these impoundments. One impoundment will collect runoff from the North Portal pad operations area and the other impoundment will collect cooling tower blow down and nonradioactive wastewater. The applicant stated that it will collect data on radioactive contamination in the water in these impoundments.

The applicant also stated that it will create and maintain information related to the radiological status of the facility in accordance with its records management and document control process, described in SAR Section 5.2.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the description in SAR Section 1.12.3.3 of the applicant's plan to collect information pertaining to the site radiological status using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff reviewed the applicant's proposed process for collecting radiological contamination data for surface facilities' structures and buildings, systems and components, soil contamination, and potential water contamination from process operations. The NRC staff also reviewed the applicant's description of two storm water detention/impoundment ponds. One impoundment will collect runoff from the North Portal pad operations area, and the other impoundment will collect cooling water and nonradioactive wastewater. The applicant stated that it will collect data on radioactive contamination in the water in these impoundments. The NRC staff reviewed the applicant's proposed storm water drainage diversion channels and finds that the storm water drainage diversion channels are designed to protect the GROA from runoff from slopes above the surface facilities.

The NRC staff finds that the applicant's plan is adequate because it collects the types of information on radiological contamination of facility structures, systems, and components; potential water contamination from process operations; and soil contamination accumulated during GROA operation that will facilitate PCDDD. The NRC staff further finds that the applicant's plan is adequate because the plan collects the types of information on radiological contamination that is commonly used for the decommissioning of nuclear facilities, and the information collected is consistent with the guidance in NUREG-1757. The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document controls process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to the radiological status of the facility will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for collecting information on the radiological status of the GROA surface facility is acceptable because the plan collects the types of information on the radiological status of the facility needed to support permanent closure and decommissioning. The NRC staff also concludes that the applicant's plans are acceptable because the applicant's use of its records management and document control process means that the information will be available for permanent closure and decommissioning. Therefore, the NRC staff concludes, with reasonable assurance, that the information the applicant provided meets the requirements of 10 CFR 63.21(c)(22)(vi).

#### 2.1.3.3.2.4 Alternatives for Decommissioning

In SAR Section 1.12.3.5, the applicant stated that the decontamination and dismantlement of the facilities will be performed in a manner that will keep radiation doses to the workers and the public consistent with ALARA principles. Additionally, the applicant stated that it will evaluate alternative decontamination and dismantlement strategies. The applicant further stated that this evaluation will include information on (i) the effort required to decontaminate the facilities consistent with ALARA principles; (ii) the anticipated physical condition of the facilities, components, and structures over time; (iii) environmental impacts; and (iv) low-level radioactive waste disposal methods that meet regulatory requirements. The applicant also stated that records of alternative decontamination or dismantlement strategies will be created and maintained in accordance with its records management and document control process described in SAR Section 5.2.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan describing how the applicant will evaluate alternatives for decommissioning using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The applicant's plan describes the types of information that the applicant will gather to evaluate alternative decontamination and alternative strategies. The NRC staff finds that the applicant's plan is adequate because it collects the types of information commonly used in the nuclear industry when evaluating decommissioning alternatives. The NRC staff further finds that the applicant's plan for evaluating alternatives for decontamination and dismantlement is adequate because it is consistent with the guidance in NUREG–1757. The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document controls process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to alternatives for decommissioning will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for evaluating alternatives for decommissioning is acceptable because the applicant's plan considers the types of information needed to evaluate alternative decommissioning strategies for PCDDD. The NRC staff also concludes that the applicant's plans are acceptable because the applicant's use of its records management and document control process means that the information will be available for permanent closure and decommissioning. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's plan meets the requirements of 10 CFR 63.21(c)(22)(vi).

### 2.1.3.3.2.5 As Low As Is Reasonably Achievable Analyses

In SAR Section 1.12.3.6, the applicant's plan describes the scope of the as low as is reasonably achievable (ALARA) analyses and assessment it will perform to demonstrate that the decontamination and dismantlement plan dose goals for repository workers and members of the public are consistent with the ALARA principles. The applicant further stated that this assessment will address target residual radioactivity, planned remediation activities, and the decontamination and dismantlement guidelines to be deployed at the facility. Examples of the types of information that the applicant's plan will provide are (i) a description of the ALARA goals; (ii) a description of how the program will be implemented; (iii) a quantitative cost-benefit analysis and the assumptions, methods, and information used to estimate costs for lowering doses; and (iv) an evaluation that confirms that doses to the public and workers are consistent with the ALARA principles. The applicant stated that records associated with its ALARA analyses and assessment will be created and maintained in accordance with the applicant's records management and document control process described in SAR Section 5.2.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the description in SAR Section 1.12.3.6 of the applicant's plan for its ALARA analyses and assessment using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff evaluated the types of information the applicant will use to support its decontamination and dismantlement ALARA goals. The NRC staff finds that the applicant's plan to address ALARA is adequate because it gathers the types of information commonly used in the nuclear industry to achieve the ALARA objectives and the plan is consistent with the guidance in NUREG–1757 (NRC, 2006aa). The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document controls process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to ALARA analyses will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan to address ALARA analyses and assessment is acceptable because the plan gathers the types of information needed to address ALARA goals and objectives at the time of permanent closure and decommissioning. The NRC staff also concludes that the applicant's plans are acceptable because the applicant's use of its records management and document control process means that the information will be available for permanent closure and decommissioning. Therefore, the NRC staff concludes, with reasonable assurance, that the information the applicant provided meets the requirements of 10 CFR 63.21(c)(22)(vi).

### 2.1.3.3.2.6 Planned Decommissioning Activities

In SAR Section 1.12.3.7, the applicant described the information required to facilitate planned closure, decontamination, and dismantlement activities for GROA surface facilities and facilities that support the subsurface. The applicant's plan to address decommissioning activities collects information relating to contaminated structures, systems and components, and contaminated soil. The applicant's plan also collects information on the methods, procedures, schedules, and contractor resources that will be used to address dismantlement and decontamination of contaminated structures, systems and components, and contaminated soil. The applicant's plan does not include information on natural surface water bodies because there are no surface water bodies at the site. The applicant stated that the information collected will be used to facilitate decontamination and dismantlement activities.

The applicant's plan also includes a schedule that addresses the order in which remediation tasks will occur, the time required to perform remediation tasks, and the initiation and completion dates for the tasks. The applicant also stated that records associated with planned decommissioning activities will be created and maintained in accordance with records management and document control process described in SAR Section 5.2.1.

**NRC Staff's Evaluation**

The NRC staff reviewed the description in SAR Section 1.12.3.7 of the applicant's plan for decommissioning activities using the guidance in the YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The applicant's decommissioning plan addresses PCDDD activities related to the site as well as surface and subsurface facilities and structures, systems, and components. The applicant's plan also includes a schedule of remediation tasks. The NRC staff finds that the applicant's decommissioning plan is adequate because it collects the kind of information needed to facilitate dismantling and decontamination of surface and subsurface facilities at the time of decommissioning and is consistent with the guidance in NUREG–1757 (NRC, 2006aa). The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document controls process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to permanent decommissioning activities will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for addressing decommissioning activities is acceptable because the plan collects the types of information that will facilitate decontamination and dismantlement at the time of decommissioning. The NRC staff also concludes that the applicant's plan is acceptable because the applicant's use of its records management and document control process means that the information will be available for permanent closure and decommissioning. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's plan for decommissioning activities meets the requirements of 10 CFR 63.21(c)(22)(vi).

#### 2.1.3.3.2.7 Project Management and Organization

The applicant described in SAR Section 1.12.3.8 its plan for project management and organization. The applicant stated that the plan will create a management organization responsible for conducting activities associated with the closure, decontamination, and dismantlement of the facility. The applicant's plan includes information on the management organization, the responsibilities of each project unit, and the management reporting hierarchy. The applicant's plan also includes information on decontamination and dismantlement task management, including a description of how tasks will be managed, evaluated, and approved. The applicant's plan also includes a description of management positions and qualifications as well as annual or periodic training on radiation safety that will be provided to each employee.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan for project management and organization described in SAR Section 1.12.3.8 using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff evaluated the applicant's management and organization plan for PCDDD and finds that the plan is adequate because the plan describes a project management organization and structure, including identifying management responsibilities, positions, qualifications, and task management and training, capable of managing PCDDD activities that will facilitate PCDDD. The NRC staff also finds that the applicant's plan is adequate because it follows the guidance provided in NUREG–1757 (NRC, 2006aa).

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for project management and organization is acceptable because it describes a management and organization structure needed for PCDDD activity. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's project management and organization plan meets the requirements of 10 CFR 63.21(c)(22)(vi).

#### 2.1.3.3.2.8 Health and Safety Program During Decommissioning

The applicant described in SAR Section 1.12.3.9 its plan for a radiological health and safety program during decommissioning. The applicant's plan describes how the radiological health and safety program will be implemented during decommissioning to comply with 10 CFR Part 20. The applicant stated that the preclosure Operational Radiation Protection Plan (ORPP) will be modified to address PCDDD activities. The modified ORPP will include information on workplace air sampling, respiratory protection, internal exposure determination, external dose determination, ALARA principles, a contamination control program, radiation protection instrument use, nuclear criticality safety and radiation protection audits, inspections, and a record-keeping program.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan for developing a health and safety program during decommissioning described in SAR Section 1.12.3.9 using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff has reviewed the applicant's preclosure ORPP in SER Section 2.1.1.8 and found it acceptable. The NRC staff finds that the applicant's plan for a health and safety program during decommissioning is adequate because it modifies the applicant's preclosure ORPP and facilitates the integration of health and safety information relating to decommissioning activities with the applicant's preclosure ORPP. The NRC staff also finds that the plan is adequate because the types of information that will be integrated into the applicant's preclosure ORPP are the types of information required to implement a radiological health and safety program.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's radiological health and safety plan is acceptable because the plan will be integrated with the applicant's preclosure ORPP and contains the types of information needed to protect radiological health and safety during PCDDD activities. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's plan for its radiological health and safety program meets the requirements of 10 CFR 63.21(c)(22)(vi).

**2.1.3.3.2.9 Environmental Monitoring and Control Program**

The applicant described in SAR Section 1.12.3.10 its plan for an environmental monitoring and control program. The applicant stated that its plan for environmental monitoring and control program during decontamination and dismantlement includes descriptions of (i) the ALARA goals and implementation plans for effluent control; (ii) the procedures, engineering controls, and process controls to maintain doses consistent with ALARA principles; and (iii) the ALARA reviews and reports to management. The applicant also stated that its Environmental Radiological Monitoring Program (SAR Section 5.11.3.1) will be evaluated and revised to measure and record potential impacts to the site environment during closure and during decontamination and dismantlement.

**NRC Staff's Evaluation**

The NRC staff evaluated the applicant's plan for an environmental monitoring and control program described in SAR Section 1.12.3.10 using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The applicant's environmental monitoring and control program includes a description of ALARA goals, reviews, and reports to management as well as a description of engineering and process controls to maintain radiation doses consistent with ALARA principles. The NRC staff finds the applicant's plan is adequate because it uses standard industry practices and engineering and process controls for an environmental monitoring program. The NRC staff also finds that the applicant's plan is adequate because it is consistent with the guidance in NUREG–1757 (NRC, 2006aa).

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's environmental monitoring and control plan is acceptable because it utilizes standard industry practices and controls for an environmental monitoring program. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's environmental monitoring and control plan meets the requirements of 10 CFR 63.21(c)(22)(vi).

### 2.1.3.3.2.10 Radioactive Waste Management Program

The applicant described in SAR Section 1.12.3.11 its plan for a radioactive waste management program. In SAR Section 1.12.3.11, the applicant provided information on its radioactive waste management program, including information on the management of low-level radioactive waste (LLW) generated through planned closure, decontamination, and dismantlement activities. The applicant also provided preliminary estimates of the volume of LLW expected to be generated annually during operation of the facility in SAR Table 1.4.5-1.

The applicant further stated in SAR Section 1.12.3.11.1 that the preliminary estimate of the volume of low-level waste generated during the repository closure phase is approximately 3,500 $m^3$ [123,620 $ft^3$] after treatment. The applicant listed the following information that will be used to update estimates of the types and quantities of LLW that may be generated during PCDDD activities: (i) the types of LLW expected to be generated; (ii) estimated volume of each solid LLW type; (iii) radionuclides, including the estimated activity of each radionuclide in each estimated solid LLW type; (iv) volumes of Class A, Class B, and Class C solid LLW that will be generated; (v) description of how and where each of the solid LLW will be stored at the GROA prior to shipment for disposal; and (vi) description of how each of the solid LLW will potentially be treated and packaged to meet site disposal acceptance criteria.

In SAR Section 1.12.3.11.2, the applicant provided information on its plans for minimizing the quantities of LLW and for disposing of that LLW. These plans include (i) a description of how volumetrically contaminated material will be managed; (ii) a description of how contaminated soil or other loose solid LLW will be prevented from being redisbursed after exhumation and collection; (iii) a description of the waste volume reduction techniques to be used to minimize the amount of waste requiring burial; (iv) the name and location of the disposal facility intended to be used for each solid low-level radioactive waste type; and (v) a description of the methods intended to be used to package and transport each waste type to its designated disposal facility. The applicant stated in SAR Section 1.12.3.11 that the previous and other information necessary at the time of updating plans for management of LLW, in connection with PCDDD activities, will be available through the applicant's records management and document control program.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan for a radioactive waste management program described in SAR Section 1.12.3.11 using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff finds that the applicant's plan is adequate because it (i) provides preliminary estimates of the volume of waste that will be generated through closure and decommissioning activities, (ii) identifies the appropriate types of information needed to update the types and quantities of LLW to facilitate closure and decommissioning activities, and (iii) identifies the appropriate types of information necessary to plan for disposing of LLW generated through PCDD activities. The NRC staff also finds that the applicant's plan is adequate because the applicant's use of its records management and document control process, which the staff evaluates and finds acceptable in SER Section 2.5.1.4.2, demonstrates that the information collected relating to a LLW management program will be available at the time of permanent closure and decommissioning.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for a LLW management program is acceptable because it provides the types of information needed for managing LLW generated through closure and PCDDD activities. The NRC staff also finds that the applicant's plan is acceptable because the applicant's use of its records management and document control process means that the information will be available for permanent closure and decommissioning. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's plan for a LLW management program meets the requirements of 10 CFR 63.21(c)(22)(vi).

**2.1.3.3.2.11 Facility Radiation Surveys**

The applicant described in SAR Section 1.12.3.13 its plan for obtaining radiological information necessary to facilitate PCDDD. The applicant stated that this information will be obtained from (i) historical records gathered during the preoperational and operational period of the facility, (ii) characterization surveys performed during planning for decontamination and dismantlement, (iii) routine and special radiological surveys performed during decontamination and dismantlement, and (iv) final radiological surveys in support of license termination.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan for obtaining radiological information that will facilitate PCDDD using the guidance in YMRP Section 2.1.3 and NUREG–1757 (NRC, 2006aa). The NRC staff finds that the applicant's plan for obtaining radiological information necessary to facilitate PCDDD is adequate because (i) the applicant will be collecting the types of radiological survey information needed to support PCDDD activities and license termination, (ii) the radiological survey data will be available for PCDDD activity, and (iii) the applicant followed the guidance in YMRP Section 2.1.3.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's plan for obtaining radiological information to support PCDDD activities is acceptable because it provides the types of information needed to support the control of radiation dose to workers and the public during PCDDD activity and license termination. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's plan for obtaining radiological information meets the requirements of 10 CFR 63.21(c)(22)(vi).

**2.1.3.3.2.12 Quality Assurance Program**

The applicant described in SAR Section 1.12.3.12 its plan for a Quality Assurance (QA) program to facilitate PCDDD. The applicant also stated that this QA program will be integrated with the applicant's preclosure QA program. As described in the SAR, the applicant's QA program to facilitate decontamination and dismantlement will include descriptions of (i) the organization responsible for implementing the QA program; (ii) how QA activities, documents, and measuring/test equipment will be controlled; (iii) how conditions adverse to quality will be corrected; (iv) the QA records that will be maintained; and (v) the audits and surveillance that will be performed as part of the QA program.

**NRC Staff's Evaluation**

The NRC staff reviewed the applicant's plan for its quality assurance program during PCDDD activities using the guidance in YMRP Section 2.1.3. The NRC staff reviewed the information in SAR Section 1.12.3.12 to determine whether the applicant has adequately described a plan for the QA program (i.e., the type of information that will be required to facilitate decommissioning, with respect to QA) that will be used for PCDDD. The NRC staff has determined that the applicant's proposed QA plan for PCDDD includes a description of a QA organization, QA activity controls, QA program documentation, QA records, and audits and surveillance. The NRC staff finds that the applicant's plan for a QA program is adequate because (i) the applicant's QA program would contain features commonly found in an effective QA program, (ii) the applicant has adequately described how the program will operate, (iii) the applicant has described how it will integrate its PCDDD QA program with its preclosure QA program, and (iv) the applicant has followed the guidance in YMRP Section 2.1.3.

**NRC Staff's Conclusion**

On the basis of the NRC staff's evaluation, the NRC staff concludes that the applicant's QA program is acceptable because it addresses the types of information needed for a QA program and will be integrated with the applicant's preclosure QA program. Therefore, the NRC staff concludes, with reasonable assurance, that the applicant's QA program meets the requirements of 10 CFR 63.21(c)(22)(vi).

## 2.1.3.4 Evaluation Findings

The NRC staff reviewed and evaluated the applicant's SAR and other supporting information submitted in support of the license application and has found, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(8) are satisfied because the applicant's plan describes the functions of design considerations that will facilitate permanent closure and decontamination or decontamination and dismantlement of surface facilities. The NRC staff also reviewed the SAR and other supporting information submitted in support of the license application and has found, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(22)(vi) are satisfied because the applicant has provided adequate plans for permanent closure and decontamination or decontamination and dismantlement of surface facilities.

## 2.1.3.5 References

DOE. 2009ao. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 2, Chapter 2.1.3, Set 1; (Safety Analysis Report Section 1.12.1)." Letter (March 4) J.R. Williams to C. Jacobs (NRC). ML090690434, ML090690439. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

DOE. 2008ab. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 0.1. ML081560400. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

NRC. 2006aa. NUREG–1757, "Consolidated Decommissioning Guidance." Vols. 1 and 2. ML063000243, ML14093A221. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: U.S. Nuclear Regulatory Commission.

# CHAPTER 11

# CONCLUSIONS

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the U.S. Department of Energy's ("DOE" or the "applicant") Safety Analysis Report (SAR), Chapter 1: Repository Safety Before Permanent Closure and the other information submitted in support of its license application and has found that DOE submitted applicable information required by 10 CFR 63.21. The NRC staff has also found, with reasonable assurance, that subject to proposed conditions of construction authorization, DOE's design of the proposed geologic repository operations area (GROA) and preclosure safety analysis complies with the preclosure performance objectives at 10 CFR 63.111 and the requirements for preclosure safety analysis of the GROA at 10 CFR 63.112.

**Proposed Conditions of Construction Authorization**:

(1) Within 90 days of issuance of construction authorization, DOE must confirm its site characterization information and related analyses in the SAR submitted in accordance with 63.21(c)(1) continue to be accurate with respect to (i) site boundaries, (ii) man-made features, (iii) previous land use, (iv) existing structures and facilities, and (v) potential exposure to residual radioactivity. DOE must provide to the NRC written notification when its confirmatory analysis is complete. This notification must include, for NRC staff's verification, a copy of DOE's confirmatory analysis. (SER Sections 2.1.1.1.3.1 and 2.1.1.1.3.9)

(2) DOE shall not, without prior NRC review and approval, accept DOE spent nuclear fuel (SNF) in multicanister overpacks (MCOs) or commercial mixed oxide (MOX) fuel.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these MCOs and MOX fuel at the GROA or (ii) demonstrates, through the PCSA, that MCOs and MOX fuel can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112. (SER Section 2.1.1.2.3.6.1)

(3) DOE shall provide the NRC staff written notification that the agreements for the six flight restrictions and operational constraints that DOE credits in its frequency analysis (SAR Section 1.6.3.4.1) are in place before commencement of construction to confirm that the technical bases for exclusion of aircraft crash hazards at the GROA from the PCSA that DOE provided in accordance with 10 CFR 63.112(d) remain valid. These restrictions and operational constraints are (i) prohibiting fixed-wing flights below 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (ii) 1,000 overflight limit per year above 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (iii) overflights are limited to straight and level flights (i.e., maneuvering is not permitted); (iv) carrying ordnance is prohibited within 9 km [5.6 mi] of the North Portal; (v) electronic jamming activities are prohibited within 9 km [5.6 mi] of the North Portal; and (vi) helicopters are not permitted within 0.8 km [0.5 mi] of facilities that process, stage, or age waste forms. (SER Section 2.1.1.3.3.1.3.3)

(4) DOE shall not, without prior NRC review and approval, take or implement any exception to the IEEE Standards 308–2001, 384–1992, 379–2000, and 603–1998 in the design of the ITS safety interlock subsystems.

Any amendment request must include the design basis for the use of the exception(s), including the ability of structures, systems, and components to perform their intended safety functions assuming the occurrence of event sequences in accordance with 10 CFR 63.112(e)(8).
(SER Section 2.1.1.6.3.2.8.2.1)

(5) DOE shall not, without prior NRC review and approval, accept the following waste packages: (i) 5-DHLW/DOE long codisposal; (ii) 2-MCO/2-DHLW codisposal; and (iii) Naval Short.

DOE shall not, without prior NRC review and approval, accept the following canisters: (i) DHLW long; (ii) DOE long; and (iii) Naval Short .

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these waste packages and canisters at the GROA or (ii) demonstrates, through the PCSA, that these waste packages and canisters can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.
(SER Section 2.1.1.7.3.9.1)

(6) DOE shall not, without prior NRC review and approval, accept DPCs at the repository.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of the DPCs at the GROA or (ii) demonstrates, through the PCSA, that the DPCs can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112.
(SER Section 2.1.1.7.3.9.3.3)

# CHAPTER 12

## Glossary

This glossary is provided for information and is not exhaustive. The glossary provides explanations for the terms shown in italics.

***absorption***: The process of taking up by capillary, osmotic, solvent, or chemical action of molecules (e.g., absorption of gas by water) as distinguished from *adsorption*.

***abstracted model***: A *model* that reproduces, or *bounds*, the essential elements of a more detailed *process model* and captures *uncertainty* and v*ariability* in what is often, but not always, a simplified or idealized form. See *abstraction*.

***abstraction***: Representation of the essential components of a *process model* into a form suitable for use in a *total system performance assessment*. A *model abstraction* is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and *uncertainty* analyses.

***adsorb***: To collect a gas, liquid, or dissolved substance on a surface as a condensed layer.

***adsorption***: The adhesion by chemical or physical forces of molecules or ions (as of gases or liquids) to the surface of solid bodies. For example, the transfer of *solute* mass, such as *radionuclides*, in groundwater to the solid geologic surfaces with which it comes in contact. The term *sorption* is sometimes used interchangeably with this term.

***advection***: The process in which *solutes*, particles, or molecules are transported by the motion of flowing fluid.

***aging***: The retention of *commercial spent nuclear fuel* on the surface in *dry storage* to reduce its thermal output as necessary to meet proposed repository thermal management goals.

***aging overpack***: A cask specifically designed for aging spent nuclear fuel. Transport, aging, and disposal canisters and dual-purpose canisters would be placed in aging overpacks for aging on the aging pad.

***airborne mass loading***: The amount of fine particulates resuspending above a surface deposit, generally expressed as mass per unit volume of air.

***aleatory uncertainty***: An *uncertainty* associated with the chance of occurrence of a feature, event, or process of a physical system or the environment such as the timing of a volcanic event. Also referred to as irreducible *uncertainty* because no amount of knowledge will determine whether or not a chance event will or will not occur. See also *epistemic uncertainty*.

***Alloy 22***: A nickel-based, *corrosion*-resistant alloy containing approximately 22 weight percent chromium, 13 weight percent molybdenum, and 3 weight percent tungsten as major alloying elements. This alloy is used as the outer container material in the U.S. Department of Energy's waste package design.

***alluvial, alluvial fan***: Pertaining to the process of moving sediment by running water (see *alluvium*). An *alluvial fan* is a wedge-shaped (fan-shaped in plan view) sedimentary deposit of *alluvium* formed at the base of a slope in arid regions.

***alluvium***: Detrital (sedimentary) deposits made by flowing surface water on river beds, flood plains, and *alluvial fans*. It does not include subaqueous sediments of seas and lakes.

***alternative***: In the context of system analysis, plausible interpretations or designs that use assumptions other than those used in the base case, which could also be applicable or reasonable given the available scientific information. When propagated through a quantitative tool such as performance assessment, *alternative* interpretations can illustrate the significance of the *uncertainty* in the base case interpretation chosen to represent the system's probable behavior.

***ambient***: Undisturbed, natural conditions, such as ambient temperature caused by *climate* or natural subsurface thermal gradients, and other surrounding conditions.

***anisotropy***: Variation in physical properties when measured in different directions. For example, in layered rock, *permeability* is often greater within the horizontal layers than across the horizontal layers.

***annual frequency***: The number of occurrences of an event in 1 year.

***aqueous***: Pertaining to water, such as *aqueous* phase, *aqueous* species, or *aqueous* transport.

***ash***: Fragments of volcanic rock that are broken from *magma* and/or country rock during an explosive volcanic eruption to less than 2 mm [0.08 in] in diameter. See also *tephra* and *pyroclastic*.

***ash flow tuff***: A type of volcanic rock formed by the deposition and accumulation of dominantly ash-size particles during an explosive eruption. *Ash flows* (also called *pyroclastic* flows) commonly result from eruptions of more viscous, silica-rich *magma* such as *rhyolite*. *Ash flow tuff* forms the *host horizons* for the proposed geologic repository at Yucca Mountain. See also *tuff* and *welded tuff*.

***basalt***: A common type of *igneous* rock (and/or low-viscosity *magma*) that forms black, rubbly-to-smooth-surfaced lavas and black-to-red *tephra* deposits.

***borosilicate glass***: A predominantly noncrystalline, relatively homogenous glass formed by melting silica and boric oxide together with other constituents such as alkali oxides. *Borosilicate glass* is a high-level radioactive waste material in which boron takes the place of the lime used in ordinary glass mixtures.

***boundary condition***: For a *model*, the establishment of a set condition for a given *variable*, often at the geometric edge of the *model*. An example is using a specified groundwater flux for *net infiltration* as a *boundary condition* for an *unsaturated zone flow model*.

***bound***: An analysis or selection of *parameter* values that yields limiting results, such that any actual result is certain to exceed these limits only with an extremely small likelihood.

***breach***:  A penetration in the waste package caused by failure of the outer and inner containers or barriers that allows the *spent nuclear fuel* or the high-level radioactive waste to be exposed to the external environment and may eventually permit *radionuclide* release.

***burnup***:  A measure of nuclear reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission, or as the amount of energy produced per unit weight of fuel.

***burnup credit***:  The concept of taking credit for the reduction in reactivity (ability to undergo fission) due to fuel irradiation.  The reduction in reactivity is due to the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides.

***caldera***:  A volcanic depression in the Earth's surface more than 1 km [0.7 mi] wide, formed by the collapse of the upper crust into an evacuated *magma chamber* during or after a large volcanic eruption.  Many *calderas* resulting from the explosive eruption of large amounts of *rhyolite magma* are several tens of kilometers [up to 20 mi] wide.

***calibration***:  (1) Comparison of *model* results with actual data or observations, and adjusting *model parameters* to increase the precision and/or accuracy of *model* results compared to actual data or observations.  (2) For tools used for field or lab measurements, the process of taking instrument readings on standards known to produce a certain response, to check the accuracy and precision of the instrument.

***canister***:  An unshielded cylindrical metal receptacle that facilitates handling, transportation, storage, and/or disposal of high-level radioactive waste.  It may serve as (i) a pour mold and container for vitrified high-level radioactive waste; (ii) a container for loose or damaged fuel rods, nonfuel components and assemblies, and other debris containing *radionuclides*; or (iii) a container that provides *radionuclide* confinement.  *Canisters* are used in combination with specialized overpacks that provide structural support, shielding, or confinement for storage, transportation, and emplacement.  Overpacks used for transportation are usually referred to as transportation *casks*; those used for emplacement in a proposed repository are referred to as waste containers.

***carbon steel***:  A steel made with carbon up to about 2 weight percent and only residual quantities of other elements.  *Carbon steel* is a tough but ductile and malleable material that is used in some components in the U.S. Department of Energy's design of the engineered barrier system.

***cask***:  (1) A heavily shielded container used for the *dry storage* or shipment (or both) of radioactive materials such as *spent nuclear fuel* or other high-level radioactive waste.  *Casks* are often made from lead, concrete, or steel.  Casks must meet regulatory requirements and are not intended for long-term disposal in a proposed repository.  (2) A heavily shielded container that the U.S. Department of Energy would use to transfer *canisters* between waste handling facilities at the proposed repository.

***Category 1 event sequences***:  Those event sequences that are expected to occur one or more times before permanent closure of a proposed geologic repository operations area.

***Category 2 event sequences***:  Event sequences other than Category 1 event sequences that have at least one chance in 10,000 of occurring before permanent closure.

***cinder cone***:  A steep, conical hill formed by the accumulation of *ash* and coarser erupted material (*tephra*) around a volcanic vent.  Synonymous with *scoria cone*.

***cladding***:  The metal outer sheath of a fuel rod generally made of a zirconium alloy or *stainless steel*, intended to protect the uranium dioxide pellets, which are the nuclear fuel, from *dissolution* by exposure to high-temperature water under operating conditions in a reactor.

***climate***:  Weather conditions, including temperature, wind velocity, precipitation, and other factors, that prevail in a region.

***climate states***:  Representations of *climate* conditions.

***colloid***:  As applied to *radionuclide migration*, *colloids* are large molecules or very small particles, having at least one dimension with the size range of $10^{-6}$ to $10^{-3}$ mm [$10^{-8}$ to $10^{-5}$ in] that are suspended in a solvent.  *Colloids* in groundwater arise from clay minerals, organic materials, or (in the context of a proposed geologic repository) from *corrosion* of engineered materials.

***commercial spent nuclear fuel***:  Nuclear fuel rods, forming a fuel assembly, that have been removed from a nuclear power plant after reaching the specified *burnup*.

***common cause failure****:*  Two or more failures that result from a single event or circumstance.

***conceptual model***:  A set of qualitative assumptions used to describe a system or subsystem for a given purpose.  Assumptions for the *model* are compatible with one another and fit the existing data within the context of the given purpose of the *model*.

***conduit***:  A *pathway* along which *magma* rises to the surface during a volcanic eruption. *Conduits* are usually cylindrical and flare upwards toward the surface *vent*.  *Conduits* are near-surface *features* and develop along *dikes*, focusing *magma flow* from the longer and possibly narrower *dike* to the vent.

***consequence***:  A measurable or calculated outcome of an event or process that, when combined with the *probability* of occurrence, gives a measurement of *risk*.

***conservative***:  A condition of an analysis or a *parameter* value such that its use provides a pessimistic result, which is worse than the actual result expected.

***corrosion***:  The deterioration of a material, usually a metal, as a result of a chemical or electrochemical reaction with its environment.  *Corrosion* includes, but is not limited to, general *corrosion*, microbially influenced *corrosion*, localized *corrosion*, *galvanic corrosion*, and stress corrosion cracking.

***coupled processes***:  A representation of the interrelationships between *processes* such that the effects of variation in one process are accurately propagated among all interrelated *processes*.

***criticality***:  The condition in which a fissile material sustains a chain reaction.  It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle.  The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.

***criticality accident***: The accidental production of a self-sustaining or divergent neutron chain reaction resulting in the release of energy.

***design concept***: An idea of how to design and operate the aboveground and belowground portions of a proposed repository.

***diffusion***: (1) The spreading or dissemination of a substance caused by concentration gradients. (2) The gradual mixing of the molecules of two or more substances because of random thermal motion.

***diffusive transport***: *Diffusive transport* is the process in which substances carried in groundwater move through the subsurface by means of *diffusion* because of a concentration gradient.

***dike***: A tabular, generally vertical body of *igneous* rock that cuts across the *structure* of adjacent rocks. *Dikes transport* molten rock (*magma*) from depth to an erupting volcano, but not all *dikes* feed an eruption.

***dimensionality***: Modeling in one, two, or three dimensions.

***direct exposure***: The manner in which an individual receives dose from being in close proximity to a source of radiation. *Direct exposures* present an external dose *pathway*.

***dispersion (hydrodynamic dispersion)***: (1) The tendency of a *solute* to spread out from the path it is expected to follow if only the bulk motion of the flowing fluid were to move it. The tortuous path the *solute* follows through openings (pores and *fractures*) causes part of the dispersion effect in the rock. (2) The macroscopic outcome of the actual movement of individual *solute* particles through a porous medium. Dispersion dilutes *solutes*, including *radionuclides*, in groundwater.

***disruptive event***: An unlikely, off-normal event that, in the case of the proposed repository at Yucca Mountain, could include *volcanic* activity, *seismic* activity, and nuclear *criticality*. *Disruptive events* alter the normal or likely behavior of the system.

***dissolution***: Dissolving a substance in a solvent.

***distribution***: In a *total system performance assessment*, the overall scatter of values for a specific set of numbers (e.g., *corrosion* rates, values used for a particular *parameter*, dose results). A term used synonymously with *frequency distribution* or *probability distribution* function. *Distributions* have *structures* that are the *probability* that a given value occurs in the set.

***docketing***: Docketing is the acceptance of a document for placement in a docket. A docket is the information collection that constitutes the record of agency review of a license application or administrative action.

***drift***: From mining terminology, a horizontal or sub-horizontal underground passage. In the proposed Yucca Mountain repository design, *drifts* include excavations for emplacement of waste canisters (*emplacement drifts*) and access (access mains).

***drift degradation***: The progressive accumulation of rock rubble in a *drift* created by weakening and collapse of *drift* walls in response to stress from heating or earthquakes.

***drip shield***: A metallic *structure* placed along the extension of the *emplacement drifts* and above the *waste packages* to prevent *seepage* water from directly dripping onto the *waste package* outer surface. The *drip shield* may also prevent the *drift* ceiling rocks (e.g., due to *drift* spallation) from falling on the *waste package*.

***dry storage***: Storage of *spent nuclear fuel* without immersion of the fuel in water for cooling or shielding; it involves the encapsulation of spent fuel in a steel cylinder that might be in a concrete or massive steel *cask* or *structure*.

***dual-purpose canister***: A canister suitable for storing (in a storage facility) and shipping (in a transportation cask) commercial spent nuclear fuel assemblies.

***effective porosity***: The fraction of a porous medium volume available for fluid flow and/or *solute* storage, as in the saturated zone. *Effective porosity* is less than or equal to the total void space (*porosity*).

***empirical***: Reliance on observation or experimentation rather than on a theoretical understanding of fundamental *processes*.

***emplacement drift***: See *drift*.

***enrichment***: The act of increasing the concentration of fissile isotopes from their value in natural uranium. The *enrichment* (typically reported in atom percent) is a characteristic of nuclear fuel.

***eolian***: Relating to *processes* caused by near-surface winds.

***epistemic uncertainty***: A *variability* that is due to a lack of knowledge of quantities or *processes* of the system or the environment. Can also be referred to as reducible *uncertainty* because the state of knowledge about the exact value of a quantity or process can increase through testing and data collection. See also *aleatory uncertainty.*

***equilibrium (chemical)***: The state of a chemical system in which the *phases* do not undergo any spontaneous change in properties or proportions with time; a dynamic balance.

***events***: In a *total system performance assessment*, (1) occurrences of phenomena that have a specific starting time and, usually, a duration shorter than the time being simulated in a *model* or (2) uncertain occurrences of phenomena that take place within a short time relative to the time frame of the *model*.

***event tree****:* A modeling tool that illustrates the logical sequence of *events* that follow an initiating event.

***expected annual dose***: The average annual radiological dose calculated for the *reasonably maximally exposed individual*, which includes the likelihood of the individual receiving a dose from all relevant exposure *scenarios*.

***expert elicitation***: A formal, highly *structured*, and well-documented process whereby expert judgments, usually of multiple experts, are obtained.

***Exploratory Studies Facility***: An underground laboratory at Yucca Mountain that includes a 7.9-km [4.9-mi] main loop (tunnel); a 2.8-km [1.75-mi] cross *drift*; and a research alcove system constructed for performing underground studies during site characterization.

***extrusive (extrusion)****:* In relation to *igneous* activity, an event where *magma* erupts at the surface. An extrusion is the deposit formed by an extrusive event. See also *intrusive*.

***failure***: The loss of ability of a structure, system, or component to perform its intended safety function or operate as specified.

***fault (geologic)***: A planar or gently curved *fracture* across which there has been displacement of rocks or sediment parallel to the *fracture* surface.

***fault tree***: A graphical logic model that depicts the combinations of events that result in the occurrence of an undesired event.

***features***: Physical, chemical, thermal, or temporal characteristics of the site or proposed repository system at Yucca Mountain. For the purposes of screening *features*, *events*, and *processes* for the *total system performance assessment*, a feature is defined to be an object, *structure*, or condition that has a potential to affect disposal system performance.

***fissure***: In relation to *igneous* activity, a *fissure* is an elongated vent or line of *vents*, formed when a *dike* breaks to the surface to start a *volcanic eruption*.

***flow***: The movement of a fluid such as air, water, or *magma*. *Flow* and *transport* are *processes* that can move *radionuclides* from the proposed repository to the *receptor* group location.

***flow pathway***: The subsurface course that water or a *solute* (and dissolved material) would follow in a given groundwater velocity field, governed principally by the hydraulic gradient.

***fluvial***: Processes related to the downslope movement of water in streams and rivers on the Earth's surface.

***fracture***: A planar discontinuity in rock along which loss of cohesion has occurred. It is often caused by the same stresses that cause folding and faulting. A *fracture* along which there has been displacement of the sides relative to one another is called a *fault*. A *fracture* along which no appreciable movement has occurred is called a joint. *Fracture*s may act as paths for fast groundwater movement.

***fragility****: Fragility* of a *structure, system, or component* is defined as the conditional *probability* of its failure, given a value of the ground motion, or response *parameter*, such as stress, bending moment, and spectral acceleration.

***frequency***: The number of occurrences of an observed or predicted event during a specific time period.

***galvanic***: Pertains to an electrochemical process in which two dissimilar electronic conductors are in contact with each other and with an electrolyte, or in which two similar electronic conductors are in contact with each other and with dissimilar electrolytes.

***geochemical***: The *distribution* and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere; the movement of the elements in nature on the basis of their properties.

***geophysics (geophysical survey; geophysical magnetic survey)****:* Study of the physical properties of rocks and sediment and interpretation of data derived from measurements made. Properties commonly measured are the velocity of sound (*seismic* waves) in rocks, density, and magnetic character. A program of measurements made on a series of rocks is usually termed a survey.

***groundwater***: Water that is below the land surface and in a saturated zone.

***half-life***: The time required for a radioactive substance to lose half of its activity due to *radioactive decay*. At the end of one *half-life*, 50 percent of the original radioactive material has decayed.

***heterogeneity***: The condition of being composed of parts or elements of different kinds. A condition in which the value of a *parameter* varies over the space an entity occupies, such as the area around the proposed repository, or with the passage of time.

***host horizon, host rock***: The rocks in which the proposed Yucca Mountain geologic repository are intended to be mined.

***human failure event***: An event in a human reliability analysis (safety assessment) in which a human performs an unsafe action or error, which can then initiate or propagate an accident sequence.

***human reliability analysis***: Human reliability analysis evaluates the potential for, and mechanisms of, human errors that may affect the safety of operations in the proposed geologic repository operations area, including consideration of human reliability as it relates to design and programs such as training of personnel.

***hydrologic***: Pertaining to the properties, *distribution*, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

***igneous***: (1) An activity or process related to the formation and movement of *magma*, either in the subsurface (*intrusive*) or on the surface (*extrusive*, or *volcanic*). (2) A type of rock that has formed from a molten, or partially molten, material, or *magma*.

***infiltration***: The process of water entering the soil at the ground surface. *Infiltration* becomes percolation when water has moved below the depth at which evaporation or *transpiration* can return it to the atmosphere. See also *net infiltration*.

***intrusive (intrusion)****:* In relation to *igneous* activity, an event where *magma* approaches the surface but does not break through in an eruption (or the unerupted *magma* during an *igneous* event). An intrusion is the solidified rock formed below the surface by an *intrusive* event. See also *extrusive*.

***invert***:  A constructed surface that would provide a level *drift* floor and enable emplacement and support of the *waste packages*.

***license application***:  An application from the U.S. Department of Energy to the U.S. Nuclear Regulatory Commission for a license to construct and operate the proposed repository at Yucca Mountain.

***lithophysal***:  Containing lithophysae, which are holes in *tuff* and other volcanic rocks.  One way lithophysae are created is by the accumulation of volcanic gases during the formation of the *tuff*.

***magma***:  Molten or partially molten rock that is naturally occurring and is generated within the Earth.  *Magma* may contain crystals along with dissolved gasses.

***mathematical model***:  A mathematical description of a *conceptual model*.

***matrix (geology)***:  In general terms, rock material and its pore space.  For Yucca Mountain, the rock is conceptually divided into matrix and *fractures*; the matrix is the portion of rock between *fractures*.  The pore space in the matrix can be referred to as the primary *porosity*, as opposed to the pore space in *fractures* that can be referred to as secondary *porosity*.

***matrix diffusion***:  The process by which molecular or ionic *solutes*, such as *radionuclides* in groundwater, move from areas of higher concentration to areas of lower concentration.  For the proposed Yucca Mountain repository, this process refers to the movement of *radionuclides* by *diffusion* between the *fracture* and *matrix* continua.

***matrix permeability***:  The capability of the *matrix* to transmit fluid.

***mean (statistical)***:  For a statistical data set, the sum of the values divided by the number of items in the set.  The arithmetic average, sometimes referred to as expected value.

***mechanical disruption***:  Damage to the *drip shield* or *waste package* because of external forces.

***median (statistical)***:  A value such that one-half of the observations are less than the value and one-half are greater than the value.

***meteorology***:  The study of climatic conditions such as precipitation, wind, temperature, and relative humidity.

***microbe***:  An organism too small to be viewed with the unaided eye.  Examples of *microbes* are bacteria, protozoa, and some fungi and algae.

***migration***:  *Radionuclide* movement from one location to another within the engineered barrier system or the environment.

***mineralogical***:  Of or relating to the chemical and physical properties of minerals, their occurrence, and their classification.

***mode (statistical)***:  A statistic for a set of data values that represents the value that occurs most frequently in that set.

***model***: A depiction of a system, phenomenon, or process, including any hypotheses required to describe the system or explain the phenomenon or process.

***model support***: A process used to gain confidence in the reasonableness of *model* results through comparison with outputs from detailed process-level *models* and/or *empirical* observations such as laboratory tests, field investigations, and *natural analogues*.

***natural analogues***: Naturally occurring, observable *features, events,* or *processes*, equivalent to those that might affect the repository in the future. These provide insights on similar *features, events, or processes* that are required to be examined for the proposed Yucca Mountain repository system. An example might be a *dike* in an existing volcanic system, or a fault that affects similar rocks to those at the repository, both occurring near the repository site or directly relatable to it.

***near-field***: The area and conditions within the proposed repository including the *drifts* and *waste packages* and the rock immediately surrounding the *drifts*. The *near-field* is the region in and around the proposed repository where the excavation of the proposed repository *drifts* and the emplacement of waste have significantly impacted the natural *hydrologic* system.

***net infiltration***: The downward flux of infiltrating water that escapes below the zone of evapotranspiration. The bottom of the zone of evapotranspiration generally coincides with the lowermost extent of plant roots.

***nominal scenario class***: The *scenario*, or set of related *scenarios*, that describes the expected or nominal behavior of the natural system as perturbed only by the presence of the proposed repository at Yucca Mountain. The nominal *scenarios* contain all likely *features, events*, and *processes* that have been retained for analysis.

***nuclear criticality safety***: Protection against the consequences of a criticality accident, preferably by prevention of the accident.

***numerical model***: An approximate representation of a *mathematical model* that is constructed using a numerical description method such as finite volumes, finite differences, or finite elements. A *numerical model* is typically represented by a series of program statements that are executed on a computer.

***occupational dose***: The dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. *Occupational dose* does not include doses received from background radiation, from any medical administration the individual has received, from exposure to individuals who were administered radioactive material and released under 10 CFR 35.75, from voluntary participation in medical research programs, or as a member of the public (10 CFR 20.1003, "*Occupational dose*").

***oxidation***: A *corrosion* reaction in which the corroded metal forms an oxide, usually applied to reaction with a gas containing elemental oxygen, such as air.

***parameter***: Data, or values, such as those that are input to computer codes for a *total system performance assessment* calculation.

***patch***: In the U.S. Department of Energy modeling of *waste package corrosion*, a *patch* is the minimal surface area of the *waste package* over which general corrosion occurs, as opposed to localized corrosion in *pits*.

***pathway***: A potential route by which *radionuclides* might reach the accessible environment and pose a threat to humans. For example, *direct exposure* is a human external *pathway*, and inhalation and ingestion are human internal *pathway*s.

***permeability***: A measure of the ease with which a fluid such as water or air moves through a rock, soil, or sediment.

***phase***: A physically homogeneous and distinct portion of a material system, such as the gaseous, liquid, and solid *phases* of a substance. In liquids and solids, single *phases* may coexist.

***phase stability***: A measure of the ability of a particular *phase* to remain without transformation.

***pit*** (in material science): A small cavity formed in a solid as a result of localized corrosion.

***Pliocene***: The epoch of geologic time from ~ 5 to ~ 2.5 million years ago.

***porosity***: The ratio of the volume occupied by openings, or voids, in a soil or rock, to the total volume of the soil or rock. Porosity is expressed as a decimal fraction or as a percentage.

***probabilistic***: Based on or subject to *probability*.

***probability***: The chance that an outcome will occur from the full set of possible outcomes. Knowledge of the exact *probability* of an event is usually limited by the inability to know, or compile, the complete set of possible outcomes over time or space.

***probability distribution***: The set of outcomes (values) and their corresponding probabilities for a random *variable*. See *distribution*.

***processes***: Phenomena and activities that have gradual, continuous interactions with the system being modeled.

***process model***: A depiction or representation of a *process*, along with any hypotheses required to describe or to explain the *process*.

***pyroclastic***: In relation to *igneous volcanic* activity, this describes fragments or fragmental rocks and deposits produced by explosive eruptions, where the *magma* is ripped apart during the release of gas and/or by interaction with surface and near-surface water.

***qualitative human reliability analysis***: Human reliability analysis tasks that include (1) identification of human failure events and unsafe actions; (2) identification of important factors influencing human performance; and (3) selection of appropriate human reliability analysis quantification method(s), if considered necessary.

***Quaternary****:* The period of geologic time from about 2 million years ago to the present day.

***radiation worker***:  A proposed geologic repository operations area worker within the controlled area boundary, with assigned duties that involve exposure to radiation or radioactive material, and who receives an occupational dose, as defined in 10 CFR Part 20.

***radiation protection program***:  A program for controlling and monitoring radioactive effluents and occupational radiological exposures in order to maintain such effluents and exposures in accordance with the requirements of 10 CFR 63.111 ("Performance objectives for the proposed geologic repository operations area through permanent closure").

***radioactive decay***:  The process in which one *radionuclide* spontaneously transforms into one or more different *radionuclides*, which are called daughter *radionuclides*.

***radioactivity***:  The property possessed by some elements (such as uranium) of spontaneously emitting energy in the form of radiation as a result of the decay (or disintegration) of an unstable atom.  *Radioactivity* is also the term used to describe the rate at which radioactive material emits radiation.

***radiolysis***:  Chemical decomposition by the action of radiation.

***radionuclide***:  An unstable isotope of an element that decays or disintegrates spontaneously, thereby emitting radiation.  Approximately 5,000 natural and artificial radioisotopes have been identified.

***range (statistical)***:  The numerical difference between the highest and lowest value in any set.

***reasonably maximally exposed individual***:  A hypothetical person meeting the criteria of 10 CFR 63.312.

***receptor***:  An individual for whom radiological doses are calculated or measured.

***reliability***:  The *probability* that the item will perform its intended function(s), under specified operating conditions for a specified period of time.

***repository footprint***:  The outline of the outermost locations of where the waste is proposed to be emplaced in the proposed Yucca Mountain repository.

***retardation***:  Slowing or stopping *radionuclide* movement in groundwater by mechanisms that include *sorption* of *radionuclides*, *diffusion* into *rock matrix* pores and microfractures, and trapping of particles in small pore spaces or dead ends of microfractures.

***rhyolite****:*  A common type of *igneous* rock that forms light-colored, rough blocky surfaced lavas and white-grayish-yellow *tephra* deposits.  A common fragment type is pumice.  Rhyolitic *magma* has a high viscosity, and the resulting lava flows are usually quite short and thick. It more frequently erupts explosively from the volcano and forms *ash-flow tuffs*.

***risk***:  The *probability* that an undesirable event will occur, multiplied by the *consequences* of the undesirable event.

***risk assessment***:  An evaluation of potential *consequences* or hazards that might be the outcome of an action, including the likelihood that the action might occur.  This assessment focuses on potential negative impacts on human health or the environment.

***risk-informed, performance-based***: A regulatory approach in which *risk* insights, engineering analysis and judgments, and performance history are used to (i) focus attention on the most important activities; (ii) establish objective criteria on the basis of *risk* insights for evaluating performance; (iii) develop measurable or calculable *parameters* for monitoring system and licensee performance; and (iv) focus on the results as the primary basis for regulatory decision making.

***rockfall***: In terms of the proposed Yucca Mountain repository, the release of *fracture*-bounded blocks of rock from the *drift* wall, usually in response to an earthquake.

***rock matrix***: See *matrix*.

***runoff***: Lateral movement of water at the ground surface, such as down steep hillslopes or along channels, that is not able to infiltrate at a specified location.

***scenario***: A well-defined, connected sequence of *features, events*, and *processes* that can be thought of as an outline of a possible future condition of the proposed repository system. Scenarios can be undisturbed, in which case the performance would be the expected, or nominal, behavior for the system. Scenarios can also be disturbed, if altered by *disruptive events* such as human intrusion or natural phenomena such as *volcanism* or nuclear *criticality*.

***scenario class***: A set of related *scenarios* sharing sufficient similarities that can usefully be aggregated for screening or analysis. The number and breadth of *scenario classes* depend on the resolution at which scenarios have been defined.

***scoria; scoria cone****:* Scoria is the *basalt*ic equivalent of pumice, a frothy material due to gas expansion in the *magma*. For scoria cone, see *cinder cone*.

***seepage***: Water dripping into a *drift.* This usage is specific to Yucca Mountain.

***seismic***: Pertaining to, characteristic of, or produced by earthquakes or Earth vibrations.

***seismic hazard curve****:* A graph showing the ground motion *parameter* of interest, such as peak ground acceleration, peak ground velocity, or spectral acceleration at a given *frequency*, plotted as a function of its annual *probability* of exceedance.

***seismic performance****: Seismic performance* of *structures, systems, and components* refers to their ability to perform intended safety functions during a *seismic* event, expressed as the annual *probability* of exceeding a specified limit condition (stress, displacement, or collapse). This is also referred to as the *probability* of failure, or *probability* of unacceptable performance, $P_F$.

***sill****:* A tabular, generally flat-lying body of *intrusive igneous* rock that lies along (is concordant with) the *structure* of adjacent rocks. *Sills* are part of the *transport* system for molten rock (*magma*) rising from depth to the surface. See also *dike*.

***sorb***: To undergo a process of *sorption*.

***solute***: A substance that is dissolved in a solution (e.g., radioactive waste dissolved in groundwater).

***sorption***: The binding, on a microscopic scale, of one substance to another. *Sorption* is a term that includes both *adsorption* and *absorption* and refers to the binding of dissolved *radionuclides* onto geologic solids or *waste package* materials by means of close-range chemical or physical forces. *Sorption* is a function of the chemistry of the radioisotopes, the fluid in which they are carried, and the material they encounter along the *flow* path.

***sorption coefficient ($K_d$)***: A numerical means to represent how strongly one substance *sorbs* to another.

***source term***: Types and amounts of *radionuclides* that are the source of a potential release.

***spatial variability***: A measure of how a property, such as rock *permeability*, varies at different locations in an object such as a rock formation.

***speciation***: The existence of the elements, such as *radionuclides*, in different molecular forms in the *aqueous phase*.

***spent nuclear fuel***: Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.

***stainless steel:*** A class of iron-base alloys containing a minimum of approximately 10 percent chromium to provide *corrosion* resistance in a wide variety of environments.

***stratigraphy***: The branch of geology that deals with the definition and interpretation of rock strata; the conditions of their formation, character, arrangement, sequence, age, and *distribution*; and especially their correlation by the use of fossils and other means of identification. See *stratum*.

***stratum (*plural *strata)***: A layer of rock or soil with geologic characteristics that differ from the layers above or below it.

***structure***: In geology, the geometric arrangement of rocks, or geologic *features* (or areas of interest) such as folds and *faults*. Includes *features* such as *fractures* created by faulting, and joints caused by various *processes*, including those associated with the heating of rock. For engineering usage, see *structures, systems, and components*.

***structures, systems, and components***: A *structure* is an element, or a collection of elements, that provides support or enclosure, such as a building, *aging* pad, or *drip shield*. A *system* is a collection of components, such as piping; cable trays; *conduits*; or heating, ventilation, and air conditioning equipment that are assembled to perform a function. A *component* is an item of mechanical or electrical equipment, such as a *canister* transfer machine, transport and emplacement vehicle, pump, valve, or relay.

***tectonic***: Pertaining to geologic *features* or *events* created by deformation of the Earth's crust.

***tephra***: A collective term for all clastic (fragmental) materials ejected from a volcano during an eruption and *transported* through the air.

***thermal chemical***: Of or pertaining to the effect of heat on chemical conditions and reactions.

***thermohydrologic***: Of or pertaining to changes in groundwater movement due to the effects of changes in temperature.

***thermal mechanical***: Of or pertaining to changes in mechanical properties from effects of changes in temperature.

***total system performance assessment***: A *risk assessment* that quantitatively estimates how the proposed Yucca Mountain repository system will perform in the future under the influence of specific *features, events*, and *processes*, incorporating *uncertainty* in the *models* and *uncertainty* and *variability* of the data.

***transparency***: The ease of understanding the process by which a study was carried out, which assumptions are driving the results, how they were arrived at, and the rigor of the analyses leading to the results. A logical structure ensures completeness and facilitates in-depth review of the relevant issues. *Transparency* is achieved when a reader or reviewer has a clear picture of what was done in the analysis, why it was done, and the outcome.

***transpiration***: The removal of water from the ground by vegetation (roots).

***transport***: A process that allows substances such as contaminants, *radionuclides*, or *colloids*, to be carried in a fluid from one location to another. Transport *processes* include the physical mechanisms of *advection*, convection, *diffusion*, and *dispersion* and are influenced by the chemical mechanisms of *sorption*, leaching, precipitation, *dissolution*, and complexation.

***transportation, aging, and disposal canister (TAD)***: A standardized canister developed by DOE for commercial spent nuclear fuel storage at a utility site, as well as transportation to, and aging/disposal at, the proposed Yucca Mountain repository.

***tuff***: A general term for volcanic rocks that formed from fragmented *magma* and fragments of other rocks, and that erupted from a volcanic vent, flowed away from the vent as a suspension of solids and hot gases, or fell from the eruption cloud and consolidated at the location of deposition. *Tuff* is the most abundant type of rock at the proposed Yucca Mountain repository site. *Welded tuff* is one type.

***uncertainty***: How much a calculated or measured value varies from the unknown true value. See also *aleatory uncertainty* and *epistemic uncertainty.*

***unsaturated zone***: The zone between the land surface and the regional water table. Generally, fluid pressure in this zone is less than atmospheric pressure, and some of the voids may contain air or other gases at atmospheric pressure. Beneath flooded areas or in perched water bodies, the fluid pressure locally may be greater than atmospheric.

***unsaturated zone flow***: The movement of water in the *unsaturated zone*, as driven by capillary, viscous, gravitational, inertial, and evaporative forces.

***vadose zone***: Synonymous with *unsaturated zone*.

***variable***: A nonunique property or attribute used to represent the *parameters* or unknowns in an equation or formula.

***variably saturated zone***: Synonymous with *unsaturated zone*.

***variability (statistical)***:  A measure of how a quantity varies over time or space.

***vent (geology)****:*  The point on the Earth's surface at which *magma* extrudes to form a volcanic eruption.  May include geologic deposits or structures associated with the *vent*.

***volcanic, volcanic activity, volcanism***:  Pertaining to extrusive *igneous* activity.

***wash***:  In relation to landforms (geomorphology), a streambed, dry or running, usually in a semi-arid or arid environment.

***waste package***:  The waste form and any containers, shielding, packing, and other absorbent materials immediately surrounding an individual waste container.

***watershed***:  Used to indicate an area of land from which all water falling as precipitation would *flow* toward a single point.  *Watershed* is also sometimes used for drainage area (i.e., the area drained by a single stream-river system, including the adjacent ridges and hillslopes).  The upstream boundaries of *watersheds* are the high points (ridges, etc.) that separate two drainage areas.

***welded tuff****:*  A *tuff* deposited under conditions where the particles that make up the rock remain sufficiently hot to weld or sinter together.  In contrast to nonwelded *tuff*, *welded tuff* is denser, less porous, and more likely to be *fractured* (which increases *permeability*).

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11. ABSTRACT (200 words or less)

Volume 2, "Repository Safety Before Permanent Closure," of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the U.S. Department of Energy's (DOE) Safety Analysis Report (SAR), entitled, "Repository Safety Before Permanent Closure," provided by DOE on June 3, 2008, as updated by DOE on February 19, 2009. In its application, DOE seeks authorization from the Commission to construct a repository for high-level radioactive waste at Yucca Mountain, Nevada. The NRC staff also reviewed information DOE provided in response to the NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, SER Volume 2 documents the results of the NRC staff's evaluation to determine whether the proposed repository design complies with the performance objectives and requirements that apply before the repository is permanently closed. Based on its review, and subject to the proposed conditions of Construction Authorization documented in Volume 2 of this SER, the NRC staff finds, with reasonable assurance, that DOE has demonstrated compliance with the NRC regulatory requirements for preclosure safety. This includes "Performance objectives for the geologic repository operations area through permanent closure" in 10 CFR 63.111, "Requirements for preclosure safety analysis of the geologic repository operations area" in 10 CFR 63.112, and "Preclosure Public Health and Environmental Standards" in 10 CFR Part 63, Subpart K.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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