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NUREG-2108

Technical Evaluation Report on the Content of the U.S. Department of Energy's Yucca Mountain Repository License Application

Preclosure Volume: Repository Safety Before Permanent Closure

Office of Nuclear Material Safety and Safeguards

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ABSTRACT

This "Technical Evaluation Report on the Content of the U.S. Department of Energy's Yucca Mountain Repository License Application—Preclosure Volume: Repository Safety Before Permanent Closure" presents the U.S. Nuclear Regulatory Commission (NRC) staff's review of the preclosure design and operations that the U.S. Department of Energy (DOE) described and provided in its Safety Analysis Report (SAR), dated June 3, 2008, as updated on February 19, 2009. The NRC staff also reviewed information DOE provided in response to NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, this report provides information on the NRC staff's evaluation of (i) Site Description as it Pertains to Preclosure Safety Analysis; (ii) Description of Structures, Systems, Components, Equipment, and Operational Process Activities; (iii) Identification of Hazards and Initiating Events; (iv) Identification of Event Sequences; (v) Consequence Analyses; (vi) Identification of Structures, Systems, and Components Important to Safety, Safety Controls, and Measures to Ensure Availability of the Safety Systems; (vii) Design of Structures, Systems, and Components Important to Safety and Safety Controls; (viii) As Low As Reasonably Achievable for Category 1 Sequences; (ix) Plans for Retrieval and Alternate Storage of Radioactive Wastes; and (x) Permanent Closure and Decontamination, or Decontamination and Dismantlement (PCDDD) of Surface Facilities.

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[1]In this Technical Evaluation Report (TER), the section numbering used within the volume is based on the Yucca Mountain Review Plan (YMRP). [NRC. 2003. "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: NRC.] The U.S. Nuclear Regulation Commission (NRC) staff used the YMRP to guide its review of information the U.S. Department of Energy (DOE) provided in its Safety Analysis Report (SAR).

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EXECUTIVE SUMMARY

Background

After docketing the U.S. Department of Energy (DOE) license application seeking a construction authorization for the proposed repository at Yucca Mountain, Nevada, the U.S. Nuclear Regulatory Commission (NRC) staff began documenting its safety review in a Safety Evaluation Report. On March 3, 2010, DOE filed a motion with the Atomic Safety and Licensing Board seeking to withdraw its license application to develop a repository at Yucca Mountain, Nevada. In June 2010, the Board denied the DOE motion. To date, petitions asking the Commission to reverse or uphold this decision are pending before the Commission.

On October 1, 2010, the NRC staff began orderly closure of its Yucca Mountain activities. As part of orderly closure, the NRC staff prepared this technical evaluation report (TER), a knowledge management document. This document captures the NRC staff's technical assessment of information presented in DOE's Safety Analysis Report (SAR), dated June 3, 2008, as amended, and supporting information. The TER describes the NRC staff's technical evaluation of the DOE SAR and, in particular, this document (TER Preclosure Volume) provides technical insights on the expected performance of the geologic repository operations area (GROA) during the period of operations (i.e., prior to permanent closure or preclosure period). The TER was developed using the regulations at 10 CFR Part 63 and guidance in the Yucca Mountain Review Plan (YMRP). The TER does not, however, include conclusions as to whether or not DOE satisfies the Commission's regulations.

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. These regulations prescribe requirements governing the licensing (including issuance of a construction authorization) of DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository operations area sited, constructed, or operated at Yucca Mountain, Nevada. Under 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, and a period of operations. The period of operations includes the time during which emplacement would occur; any subsequent period before permanent closure during which the emplaced wastes are retrievable; and permanent closure. In addition, the regulations at 10 CFR Part 63 represent a risk-informed, performance-based (RIPB) approach to the review of geological disposal. The RIPB approach uses risk information to focus the review to areas most significant to safety or performance. Therefore, the TER includes discussions regarding how the staff used risk information in its review. This technical evaluation report presents information on the NRC staff's assessment of the SAR DOE provided on June 3, 2008, as updated on February 19, 2009.[1] The NRC staff also reviewed information DOE provided in response to NRC staff's requests for additional information, and other information that DOE provided related to the SAR. In conducting its review of DOE's SAR, the NRC staff was guided by the Yucca Mountain Review Plan (YMRP).[2]

Technical Evaluation of the Geologic Repository Operations Area

NRC evaluates the design of the geologic repository operations area (GROA) with respect to the types and amounts of radioactive material described in the SAR to be received and

[1]DOE. 2009. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.
[2]NRC. 2003. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC

possessed in the GROA. Specifically, NRC staff evaluation of the GROA operations considers whether

- The GROA design and operations maintain radiation exposures to as low as is reasonably practical.

- The GROA is designed to limit radiation exposures during normal operations and during any event or Category 1 event sequence expected to occur at least once before permanent closure, and for less likely events or Category 2 event sequences (those having at least 1 chance in 10,000 of occurring before permanent closure).

- Potential radiation exposures during operations are estimated using a particular type of quantitative analysis called a preclosure safety analysis (PCSA).

- The GROA is designed to permit implementation of a performance confirmation program for monitoring, laboratory and field testing, and onsite experiments. This program provides data from measurements of (i) actual conditions encountered during the operational period and (ii) natural and engineered systems and components required for repository operation to ensure these are functioning as intended.

- The GROA is designed to preserve the option of waste retrieval throughout the period during which wastes are emplaced and thereafter until the completion of the performance confirmation program and review of information obtained from this program.

Site Description as it Pertains to Preclosure Safety Analysis

The PCSA includes a description of the proposed repository site and the GROA location and design. The site description includes information on geology, hydrology (surface and groundwater), geochemistry, geomechanical properties and conditions of soil and rock, and meteorology and climatic conditions. The GROA is that part of the proposed repository, including both surface and subsurface areas, where waste is handled. DOE provided relevant data about the site and surrounding region, to the extent necessary, to identify naturally occurring and human-induced hazards to the GROA.

NRC staff reviewed DOE's site description and notes, based on the information presently available, that DOE reasonably described the site to support the PCSA and design of the GROA. The DOE identification of naturally occurring and human-induced hazards at the GROA is reasonable for use in the preclosure safety analysis (PCSA).

DOE stated it would continue to monitor such items as rock and soil properties and conditions to confirm the performance of the design of the GROA (e.g., surface buildings, storage pads, drifts). As part of the detailed design process, DOE should confirm soil strength properties associated with the load-bearing pressure of surface building foundations for the construction of safety-relevant surface facilities. As part of the performance confirmation program, DOE should confirm the mechanical properties of the lithophysal and nonlithophysal rock encountered in the repository block.

Descriptions of Structures, Systems, Components, Equipment, and Operational Process Activities

The PCSA includes a general description of structures, systems, and components (SSCs); equipment; and process activities at the GROA. DOE provided information about the GROA to perform the PCSA and to design the GROA, including (i) location of both surface and subsurface facilities and their functions; (ii) design of SSCs and safety controls; (iii) civil and structural systems; (iv) mechanical systems; (v) electrical power systems; (vi) heating, ventilation, and air conditioning (HVAC) systems; (vii) radiation and radiological monitoring systems; (viii) types of radioactive waste; (ix) waste containers; (x) instrumentation and communication systems; and (xi) facility operational processes.

In reviewing the SAR and other information submitted in support of the SAR, the NRC staff notes that DOE's descriptions of SSCs, equipment, and process activities are reasonable. DOE provided information on geologic media, general arrangement, approximate dimensions, and codes and standards for the GROA design. DOE described and discussed the design of various SSCs and of the engineered barrier system, including dimensions, material properties, specifications, and analytical and design methods used along with applicable codes and standards. DOE explained the functions, operations, and design of the GROA for use in its PCSA.

As part of its design process, DOE stated it would provide additional details as designs were finalized for items such as electrical power located in nonaccessible areas, fire and smoke detection systems, remotely operated vehicles, and monitoring and maintenance of openings and rails in the emplacement drifts.

Identification of Hazards and Initiating Events

Naturally occurring and human-induced hazards at the GROA are identified and systematically analyzed by DOE. DOE used different approaches to identify initiating events that could lead to event sequences based on the specific nature of external (both naturally occurring and human induced) and internal (operational) hazards. DOE developed a list of 89 external hazards on the basis of past licensing experience of other nuclear facilities. DOE also conducted a separate study to identify hazards specific to the subsurface facilities at the proposed Yucca Mountain repository. DOE evaluated these hazards using criteria of applicability, frequency and rate of occurrence, and associated consequences at the repository. On the basis of the screening analysis, DOE concluded that seismic events (earthquakes) and loss of offsite power are the two key external initiating events that have the greatest potential for initiating an event sequence at the repository during the preclosure period.

To identify operation-related hazards (internal initiating events), DOE evaluated equipment failure and human performance data by applying engineering analysis. DOE screened the internal initiating events from further analysis by determining that (i) the event could not occur, (ii) the initiating event was bounded by another event, or (iii) the probability of occurrence is too low to become a credible hazard. In the case of internal hazards, the NRC staff conducted an audit review selecting several internal hazards and corresponding initiating events from each type of internal hazard on the basis of their risk potential. The NRC staff reviewed each of the following areas: (i) drops and collisions, (ii) fire-related initiating events, (iii) internal flooding, (iv) criticality-related initiating events, (v) subsurface-related initiating events, and (vi) human failures.

The NRC staff notes that DOE's hazard and initiating event identification is reasonable because DOE's approach (i) is based on NRC guidance and standard industry practice and (ii) used reasonable methodologies, data, and rationale to screen and quantify events. In particular, DOE used the system information and the information related to the operating environments to construct a fault tree model to screen out initiating events at subsurface facilities. The NRC staff also notes that DOE treated dependencies reasonably and used reasonable quantification methods for specific human failure events.

DOE stated it would provide procedures and specific equipment to limit the probability of certain events, such as controls to restrict the airspace near Yucca Mountain and interlock functions in the canister transfer machine. DOE should confirm that its human reliability analyses (e.g., task analyses) identified potential vulnerabilities for the repository facilities and associated activities as part of the detailed design process

Identification of Event Sequences

DOE developed event sequences on the basis of initiating events and the failure of associated SSCs that could lead to radiological dose to the public or workers. DOE analyzed three main categories of event sequences: (i) internal events initiated by random component failure or human error, (ii) seismically initiated events, and (iii) fire-initiated events from "local fires" affecting the waste form in specific areas and "large fires" that propagate through the facility.

Event sequences might result in exposure to workers either directly with no mitigation or because of loss or degradation of shielding, filtered and unfiltered radionuclide release, and releases from criticality consequences. On the basis of the reliability of SSCs, interlock and control systems, and procedural safety controls, DOE determined that there are no Category 1 event sequences. Limited Category 2 event sequences involved filtered and unfiltered releases from breach of canisters, spent nuclear fuel assemblies, and transportation casks with uncanistered spent nuclear fuel; these resulted mostly from operational event sequences, and some were from seismic event sequences. DOE determined most event sequences were beyond Category 2 for which a consequence analysis is not performed due to the low probability of occurrence for the event sequence.

The NRC staff notes that DOE's event sequence development and categorization are reasonable because DOE's overall methodologies and modeling approaches for event sequence analysis are consistent with standard practice in probabilistic risk analysis and NRC guidance. In particular, the passive reliability of canisters and casks under structural and thermal challenges are well supported, DOE justified the reliability of HVAC systems, and DOE used a reasonable methodology for evaluating the seismic fragility of the SSCs.

DOE stated it would conduct further activities to confirm the identification of event sequences, such as verification that final equipment designs continue to support the basis for event identification and further analyses of drop and tipover scenarios for transportation casks.
As part of the detailed design process, DOE should confirm the basis for the identification of event sequences (i) using structural analysis of any design changes related to surface facilities, (ii) quantifying probabilities used in the fault tree models (e.g., interlocks), and (iii) verifying that the exposure time of containers is consistent with the exposure time used in the PSCA for event sequence quantification and categorization.

Consequence Analysis

DOE identified and described individuals and locations for the purpose of estimating potential radiation exposures. DOE defined the radiation workers as those who are qualified and trained as radiation workers and who will receive occupational doses in performing their duties. Within the preclosure controlled area, referred to as the onsite areas, DOE defined an onsite member of the public as any individual not receiving an occupational dose in performing duties. Offsite public is defined as individuals located at or beyond the site boundary of the preclosure controlled area. DOE performed dose calculations for radiation workers and the onsite and offsite members of the public. Because DOE did not identify any Category 1 event sequences, the potential radiation worker dose calculation included only events related to normal operations.

The NRC staff reviewed the SAR, and other information DOE submitted in support of its SAR, and notes that DOE used a reasonable methodology and input parameters for dose calculations for workers and members of the public from normal operations and for members of the public from a single Category 2 event.

Identification of Structures, Systems, and Components Important to Safety, Safety Controls, and Measures To Ensure Availability of the Safety Systems

The PCSA includes an analysis to identify SSCs important to safety (ITS) and measures taken to ensure the availability of safety systems. DOE determined a SSC is ITS if it satisfies one or more of the following four DOE criteria: (i) reduce the frequency of an event sequence from Category 1 to Category 2, (ii) reduce the frequency of an event sequence from Category 2 to beyond Category 2, (iii) reduce the aggregated dose of Category 1 event sequences by reducing the event sequence mean frequency, or (iv) perform dose mitigation or criticality safety control functions. For example, DOE indicated that it will rely on HVAC systems to limit airborne radioactive contamination by controlling airflow from areas of low contamination potential to areas of high contamination potential, and DOE determined that the shielding features, including shield doors and slide gates in the surface facilities, are ITS because they were credited in the PCSA for reducing the mean frequency of inadvertent exposure of personnel to below the mean frequency of the Category 1 event sequences.

In addressing the criticality concern, DOE conducted a separate criticality-initiating event and event sequence analysis for the preclosure operations. DOE evaluated seven parameters to determine whether they should be controlled to prevent criticality during the preclosure period. Through this analysis, DOE identified the SSCs relied on to maintain subcriticality by preventing moderator from contacting the fissile materials as ITS.

On the basis of its evaluation of DOE's PCSA, the SAR, and other information submitted in support of DOE's SAR, the NRC staff notes that DOE's PCSA is appropriate for its intended purpose. DOE reasonably developed nuclear safety design bases for ITS SSCs and procedural safety controls because they are determined using the PCSA process including hazard assessment, event sequence categorization, and consequence analysis. DOE stated that it would develop a reliability-centered inspection, testing, and maintenance program for the ITS SSCs.

As part of the detailed design process, DOE should confirm that the identification of ITS components, the associated nuclear safety design bases for the ITS components, and the assumptions regarding passive and active systems relied on to screen out initiating events are consistent with the design.

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

DOE described and discussed the design, including dimensions, materials properties, specifications, and analytical and design methods along with applicable codes and standards. Additionally, DOE provided a description and discussion of the GROA design, including (i) the relationship between design criteria and GROA performance and (ii) the design bases and their relation to the design criteria.

For the surface structural and civil facilities, DOE provided design information on the ITS surface waste-handling facilities, the aging facility, and flood control features. In particular, DOE (i) determined that seismic loading bounds the design of the ITS surface buildings; (ii) analyzed the structural integrity of the aging facility to protect the ITS SSCs from external events, such as earthquakes, extreme winds, and tornado winds, and protect against aging overpack tipover and sliding; and (iii) performed probable maximum flood and flood inundation analyses for the proposed flood control features and provided design information for these features. DOE also provided design and design analyses information on the ITS mechanical systems, HVAC systems, transportation systems for moving wastes, electrical power systems, instrumentation and control systems, fire protection systems, and waste packages and canister systems to be used in the GROA.

The NRC staff's review focused on DOE's information related to design of the ITS SSCs and safety controls and notes that DOE's design of the SSCs and safety controls is reasonable. DOE provided information relative to the codes and standards for GROA design and construction and the design bases and their relationship to DOE's proposed design criteria. The design methodologies, design analysis, and design are appropriately supported by reasonable technical bases and are consistent with established industry practices. The NRC staff also notes that the design information describes the relationship between the proposed design criteria and the GROA performance, and the relationship between the design bases and the design criteria.

As part of the detailed design process, DOE stated it would conduct additional analyses that will provide further information on and evaluation of design parameters and assumptions. The NRC staff notes that this information could be used to confirm that more refined soil properties and detailed designs are consistent with DOE's currently estimated demand-to-capacity ratios for the structural integrity of surface structures (DOE, 2009ev) (TER Section 2.1.1.7.3.1.1). As part of the detailed design process, DOE should (i) evaluate the effect of soil–structure interaction on the response of aging pad prior to excavation, to confirm the demand-to-capacity ratio estimated for the aging pad (TER Section 2.1.1.7.3.1.2); (ii) confirm the coefficient of friction between concrete pad and aging cask, and between concrete pad and horizontal aging module (TER Section 2.1.1.7.3.1.2); and (iii) confirm that the reliabilities for the types/manufacturing-specifications of ITS electrical power system, ITS I&C, and ITS interlock equipment procured for use in the GROA are consistent with the PCSA and final designs (TER Sections 2.1.1.7.3.6 and 2.1.1.7.3.7).

As Low As Is Reasonably Achievable for Category 1 Sequences

DOE provided a description of the proposed Radiation Protection Program (RPP) to demonstrate that the RPP reflects ALARA considerations for maintaining the occupational doses to workers and doses to members of the public to as low as is reasonably practical, consistent with the purpose for which the licensed activity is undertaken. The description included (i) the administrative organization of the RPP; (ii) health physics equipment, facilities, and instruments; (iii) policies and procedures for controlling access to the radiation area; (iv) procedures for the accountability and storage of radioactive material; (v) radiation protection training programs; and (vi) program implementation.

The NRC staff reviewed DOE's RPP information and other information submitted in support of the SAR and notes that DOE's RPP will maintain occupational doses and public exposures as low as practicable, consistent with the proposed activities. The NRC staff also notes that the GROA operations, through permanent closure, are consistent with ALARA principles. DOE stated it would provide a detailed RPP when it becomes available and prior to the receipt of radioactive waste at the GROA.

Plans for Retrieval and Alternate Storage of Radioactive Wastes

DOE provided a description of plans for retrieval and alternate storage of the radioactive wastes should retrieval be necessary. DOE, in its description of its alternate storage plan, identified a proposed alternate storage site for a facility, including the location, size, and storage operations. DOE also provided a schedule for retrieval operations. DOE's retrieval plan consists of maintaining access to waste packages in emplacement drifts through the preclosure period, such that waste packages could be retrieved, if necessary, by reversing the operational procedure used for waste emplacement.

The NRC staff reviewed DOE's plans for retrieval under expected conditions of normal operations and postulated off-normal conditions and notes the retrieval plans are reasonable. DOE's retrieval plans show that the proposed concepts for retrieval are reasonably feasible on the basis of (i) current knowledge of the site and design, (ii) a generalized set of postulated retrieval scenarios, and (iii) currently available technology and equipment. The staff notes that the GROA has been designed to preserve the option to retrieve any or all of the emplaced waste on a reasonable schedule.

Permanent Closure and Decontamination

DOE described the design considerations that are intended to facilitate permanent closure and decontamination or decontamination and dismantlement (PCDDD) of surface facilities, including information on plans for PCDDD of surface facilities. DOE provided design considerations to facilitate PCDDD and a planning timeline for decontamination and dismantlement. On the basis of the staff's review of the SAR and other information DOE provided in support of the SAR, the design considerations that are intended to facilitate PCDDD of surface facilities are reasonably described and reasonable plans for PCDDD are provided.

Summary Conclusions

The NRC staff reviewed the SAR and other information DOE submitted in support of its SAR and notes that DOE's (i) preclosure safety analysis (PCSA), which includes consideration of the design of the proposed geologic repository operations area (GROA) and activities associated with the period of operations, is reasonable and (ii) identification of structures, systems, and components (SSCs) important to safety (ITS) is reasonable. In addition, DOE has designed the GROA to preserve the option to retrieve any or all of the emplaced waste on a reasonable schedule and enable a performance confirmation program. The NRC staff also notes that the design considerations that are intended to facilitate PCDDD of surface facilities are reasonably described.

DOE stated it would evaluate additional design details and conduct analyses to confirm the safety functions of structures, systems, and components (SSCs) important to safety (ITS) are consistent with what was presented in the SAR. DOE should confirm the basis for (i) identification of event sequences, (ii) identification of ITS and non-ITS components, and (iii) assumptions regarding passive and active systems relied on to screen out initiating events.

Abbreviations and Acronyms	
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
AWS	American Welding Society
BDBGM	beyond design basis ground motion
BWR	boiling water reactor
CCC	Center Control Center
CCCF	central control center facility
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
ESD	event sequence diagram
ESF	exploratory studies facility
FAA	Federal Aviation Administration
FDH	fault displacement hazard
FE	finite element
GROA	geologic repository operations area

Abbreviations and Acronyms (continued)	
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
HMI	human–machine interface
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
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
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
MRVs	maintenance and repair vehicles
MTHM	metric tons of heavy metal
NARA	Nuclear Action Reliability Assessment
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
non-ITS	not-important to safety
NRC	U.S. Nuclear Regulatory Commission
NTS	Nevada Test Site
NTTR	Nevada Test and Training Range
OCB	outer corrosion barrier
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
PFDHA	probabilistic fault displacement hazard analysis
PGA	peak ground acceleration
PGV	peak ground velocity
PLCs	programmable logic controllers
PMF	probable maximum flood

Abbreviations and Acronyms (continued)	
PMP	probable maximum precipitation
PRA	probabilistic risk 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
RIPB	risk-informed, performance-based
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
SFTM	spent fuel transfer machine
SNF	spent nuclear fuel
SONET	Synchronous Optical NETwork
SPM	site prime mover
SSCs	structures, systems, and components
SSI	Soil–structure interaction
STC	shielded transfer cask
TAD	transportation, aging, and disposal
TEDE	total effective dose equivalent
TER	technical evaluation report
TEV	transport and emplacement vehicle
TNT	trinitrotoluene
TSPA	total system performance assessment
UHS	uniform hazard spectras
UPS	uninterruptible power supply
USL	upper subcritical limit
V_p	compression wave velocity
V_s	shear wave velocity
WHF	wet handling facility
WPTT	waste package transfer trolley
X/Q	atmospheric dispersion coefficients
YMRP	Yucca Mountain Review Plan

INTRODUCTION

After docketing the U.S. Department of Energy (DOE) license application seeking a construction authorization for the proposed repository at Yucca Mountain, Nevada, the U.S. Nuclear Regulatory Commission (NRC) staff began documenting its safety review in a Safety Evaluation Report. On March 3, 2010, DOE filed a motion with the Atomic Safety and Licensing Board seeking to withdraw its license application to develop a repository at Yucca Mountain, Nevada. In June 2010, the Board denied the DOE motion. To date, petitions asking the Commission to reverse or uphold this decision are pending before the Commission.

On October 1, 2010, the NRC staff began orderly closure of its Yucca Mountain activities. As part of orderly closure, the NRC staff prepared this technical evaluation report (TER), a knowledge management document. This document captures the NRC staff's technical assessment of information presented in DOE's Safety Analysis Report (SAR), dated June 3, 2008, as amended, and supporting information. The TER describes the staff's technical evaluation of the DOE SAR and, in particular, this document (TER Preclosure Volume) provides technical insights on the expected performance of the geologic repository operations area (GROA) during the period of operations (i.e., prior to permanent closure or preclosure period). The TER was developed using the regulations at 10 CFR Part 63 and guidance in the Yucca Mountain Review Plan (YMRP). The TER does not, however, include conclusions as to whether or not DOE satisfies the Commission's regulations.

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. These regulations prescribe requirements governing the licensing (including issuance of a construction authorization) of DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository operations area sited, constructed, or operated at Yucca Mountain, Nevada. Under 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, and a period of operations. The period of operations includes the time during which emplacement would occur; any subsequent period before permanent closure during which the emplaced wastes are retrievable, and permanent closure. In addition, the regulations at 10 CFR Part 63 represent a risk-informed, performance-based (RIPB) approach to the review of geological disposal. The RIPB approach uses risk information to focus the review to areas most significant to safety or performance. Therefore, the TER includes discussions regarding how the staff used risk information in its review. This technical evaluation report presents information on the NRC staff's assessment of the SAR DOE provided on June 3, 2008, as updated on February 19, 2009.[1] The NRC staff also reviewed information DOE provided in response to NRC staff's requests for additional information, and other information that DOE provided related to the SAR. In conducting its review of DOE's SAR, the NRC staff was guided by the Yucca Mountain Review Plan (YMRP).[2]

Risk-Informed, Performance-Based Review

The Preclosure Safety Analysis (PCSA) quantifies GROA performance as a means of estimating radiation exposures. The PCSA is a systematic analysis that answers three questions to define a risk: (i) What can go wrong?, (ii) How likely is it?, and (iii) What are the consequences? DOE's PCSA includes a number of evaluations such as (i) identification of

[1]DOE. 2009. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.
[2]NRC. 2003. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC

hazards and initiating event sequences; (ii) development and categorization of event sequences; (iii) failure mode and reliability assessments of structures, systems, and components (SSCs); and (iv) SSCs' fragility assessments. The NRC staff used risk information throughout the technical evaluation to focus the evaluation on significant items that affect preclosure performance. YMRP Section 2.1.1 provides guidance as to how the NRC staff should apply risk information throughout the review of the PCSA.

DOE's Preclosure Safety Analysis (PCSA)

To answer the three risk-related questions, DOE conducted a PCSA to systematically examine the site and design and the potential hazards, initiating events, and event sequences and their radiological safety consequences. The analysis identified structures, systems and components (SSCs) that are important to safety (ITS). DOE included the PCSA as an integral part of its repository design process. The NRC staff's technical evaluation considered whether (i) DOE's PCSA contains sufficient information to estimate radiation exposures and (ii) a clear link is established between design bases and design criteria. The focus of the review in various TER Preclosure Volume sections is summarized next.

<u>Site Description as It Pertains to Preclosure Safety Analysis (TER Section 2.1.1.1)</u>

This TER section provides the NRC staff's evaluations of the site description information that DOE used to (i) identify natural and human-induced hazards, emphasizing those features, events, and processes that might affect the GROA design and (ii) conduct the PCSA. The NRC staff evaluates features (i.e., site rocks, sediments and soils, rock fractures and faults, landforms, surface and groundwater quantities and flow processes, chemistry of the rocks and water, earthquake frequency and magnitude, volcanic and climatic history, weather conditions, topography and land-use boundaries, the current and future population, and any natural or human-made sources of radiation) that can influence GROA design and may be important parameters for PCSA consideration.

<u>Description of Structures, Systems, Components, Equipment, and Operational Process Activities (TER Section 2.1.1.2)</u>

This TER section provides the NRC staff's evaluations of DOE's descriptions of SSCs, safety controls, equipment, and operational process activities in surface and subsurface facilities. More specifically, the NRC staff evaluates the descriptions of

- Civil and structural systems
- Mechanical systems
- Electrical power systems
- Heating, ventilation, and air conditioning systems
- Radiation/radiological monitoring systems
- Types of radioactive waste
- Waste containers
- Operation of the facilities

In the evaluations, the NRC staff emphasizes the ITS SSCs and risk-significant operation processes involving radioactive waste handling.

Identification of Hazards and Initiating Events (TER Section 2.1.1.3)

This TER section provides the NRC staff's evaluations of DOE's identification of hazards and initiating events that may lead to an event sequence at the repository facilities during the preclosure period. Identification of hazards and initiating events begins with systematic examination of the site, the design of the facilities, and the operations to be conducted at these facilities. This TER section assesses the probability of the potential hazards taking into account a range of uncertainties associated with the data that support the probability estimation. The estimated probability of the initiating events is used to analyze the associated event sequences.

Identification of Event Sequences (TER Section 2.1.1.4)

This TER section provides the NRC staff's evaluations of the information DOE used for identification of event sequences relevant to preclosure safety. The NRC staff evaluates the technical basis for developing, quantifying, and categorizing event sequences. The review in this section 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. This TER section also assesses the methodology for event sequence development and categorization and reliability of SSCs.

Consequence Analysis (TER Section 2.1.1.5)

This TER section provides the NRC staff's evaluation of the consequence analysis DOE conducted to support its PCSA. The NRC staff focuses its review on the

- Dose calculation methodology
- Atmospheric dispersion determination
- Assumptions and input parameters
- Source terms
- Methodology for the worker and public dose determination
- PCSA dose estimates

Identification of Structures, Systems, and Components Important to Safety, Safety Controls, and Measures To Ensure Availability of the Safety Systems (TER Section 2.1.1.6)

This TER section provides the NRC staff's evaluations of DOE's identification of important to safety (ITS) structures, systems, and components (SSCs) and procedural safety controls for reducing event sequences or mitigating dose consequences. In addition, the NRC staff evaluates DOE's consideration of the following aspects in its PCSA:

- The means to limit the radioactive material concentration in air
- The means to limit the time required to perform work near the radioactive materials
- Shielding protection
- Monitoring and controlling dispersal of radioactive contamination
- Access control to high radiation or airborne radioactivity areas
- Criticality control and prevention
- Radiation alarms
- Ability of ITS SSCs to perform their intended safety functions
- Fire detection and suppression

- Radioactive waste and effluent controls
- The means to provide timely and reliable emergency power
- Redundant systems
- ITS SSCs inspection, testing, and maintenance

This TER section assesses the criteria DOE developed for identification of ITS SSCs and procedural safety controls. The NRC staff evaluates the nuclear safety design basis requirements DOE developed for the ITS SSCs from the PCSA event sequence analyses.

Design of Structures, Systems, and Components Important to Safety and Safety Controls (TER Section 2.1.1.7)

This TER section provides the NRC staff's evaluations of the information DOE provided for the design, construction, and operation of the SSCs designated as important to safety (ITS). The evaluations include (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. The NRC staff also evaluates the relationship between the proposed design criteria and the GROA performance.

Radiation Protection Program

As Low As Is Reasonably Achievable for Category 1 Sequences (TER Section 2.1.1.8)

This TER section provides the NRC staff's evaluations of the DOE-proposed Radiation Protection Program (RPP) to confirm that the RPP reflects as low as is reasonably achievable (ALARA) considerations for occupational doses to workers and doses to members of the public, consistent with the purpose for which the licensed activity is undertaken. The RPP description included (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 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 implementation of the program.

Retrieval of Wastes

Plans for Retrieval and Alternate Storage of Radioactive Wastes (TER Section 2.1.2)

This TER section provides the NRC staff's evaluation of DOE's (i) plans for retrieval and alternate storage of the radioactive wastes should retrieval be necessary and (ii) design of the GROA to preserve the option of waste retrieval. The NRC staff's evaluation of DOE's retrieval plans is performed in the context of (i) current knowledge of the site and design, (ii) a generalized set of postulated retrieval scenarios, and (iii) currently available technology and equipment. The NRC staff also evaluates whether the GROA has been designed to preserve the option to retrieve any or all of the emplaced waste on a reasonable schedule. DOE's description of an alternate storage plan that identifies a proposed alternate storage site for a facility, including the location, size, and storage operations, is also evaluated in this section.

Permanent Closure

Permanent Closure and Decontamination (TER Section 2.1.3)

In this TER section, the NRC staff evaluates DOE's plans for permanent closure and decontamination or decontamination and dismantlement (PCDDD) of the surface facilities. The NRC staff also evaluates the design considerations to facilitate PCDDD and PCDDD plans including facility history, dose modeling, facility radiological status, alternatives for decommissioning, ALARA, planned decommissioning activities, project management and organization, health and safety program for PCDDD, environmental monitoring and control program, radioactive waste management program, radiation surveys, and quality assurance program.

CHAPTER 1

2.1.1.1 Site Description as It Pertains to Preclosure Safety Analysis

2.1.1.1.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the Yucca Mountain site description as it pertains to the preclosure safety analysis (PCSA) and design of the geologic repository operations area (GROA). The objective of the review is to verify that the site description is reasonable for identifying the natural and human-induced hazards that might affect the design of the GROA and the PCSA. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.1 (DOE, 2008ab) and information DOE provided in response to NRC staff's requests for additional information (RAIs) (DOE, 2009ab,ap–au,bf,bg,eh–ej).

The site description includes information on those features, events, and processes that might affect the design and performance of the surface and subsurface GROA facilities. DOE described the site's natural features including its rocks, sediments and soils, rock fractures and faults, landforms, surface and groundwater quantities and flow processes, chemistry of the rocks and water, earthquake frequency and magnitude, volcanic and climatic history, weather conditions, topography and land-use boundaries, the current and future population, and any natural or man-made sources of radiation.

2.1.1.1.2 Evaluation Criteria

The regulatory requirements for site description as they pertain to the preclosure period are set forth in 10 CFR 63.21(c)(1)(i–iii) and 10 CFR 63.112(c). 10 CFR 63.21(c) requires that the SAR describe the Yucca Mountain site, with appropriate attention to those features, events, and processes of the site that might affect design of the GROA and performance of the geologic repository.

10 CFR 63.21 requires the information to include the GROA location, geology, hydrology (both surface and groundwater), geochemistry, geomechanical properties and conditions of soil and rock, and meteorology and climatic conditions. 10 CFR 63.112(c) requires DOE to include in its SAR data pertaining to the Yucca Mountain site and surrounding region, to the extent necessary, used to identify naturally occurring and human-induced hazards at the GROA.

10 CFR 63.132(a) requires DOE to include a continuing program of conducting surveillance, measurement, testing, and geologic mapping during repository construction and operation to ensure that geotechnical and design parameters are confirmed.

In terms of the repository boundaries, the boundary of the preclosure *controlled area* and the *restricted area* must be consistent with the definitions of these terms in 10 CFR 20.1003. The *general environment* must be consistent with the definition of this term in 10 CFR 63.202.

The NRC staff reviewed DOE's site information using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.1 (NRC, 2003aa). The relevant acceptance criteria follow:

- The DOE license application contains a description of the site geography adequate to permit evaluation of the PCSA and the GROA design.

- The DOE license application contains a description of the regional demography adequate to permit evaluation of the PCSA and the GROA design.

- The DOE license application contains a description of the local meteorology and regional climatology adequate to permit evaluation of the PCSA and the GROA design.

- The DOE license application contains sufficient local and regional hydrological information to support evaluation of the PCSA and the GROA design.

- The DOE license application contains descriptions of the site geology and seismology adequate to permit evaluation of the PCSA and the GROA design.

- The DOE license application contains descriptions of the historical regional igneous activity adequate to permit evaluation of the PCSA and the GROA design.

- The DOE license application provides analysis of site geomorphology adequate to permit evaluation of the PCSA and the GROA design.

- The DOE license application contains site-sufficient geochemical information to support evaluation of the PCSA and the GROA design.

- The DOE 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 also used additional guidelines, such as NRC standard review plans and regulatory guides, when applicable. These additional guidance documents are discussed in the relevant sections that follow.

2.1.1.1.3 Technical Evaluation

The NRC staff organized its evaluation of DOE's site description generally following the YMRP outline. NRC staff focused on assessing the site information used to conduct the PCSA and support the design of the GROA.

2.1.1.1.3.1 Site Geography

In SAR Section 1.1.1, DOE provided the site geographic information used to describe the location of the GROA with respect to the site boundary and prominent natural features that may be significant to the PCSA. DOE also provided site geographic information used to identify naturally occurring and human-induced hazards. DOE's information on the natural features included elevations and drainage pathways as they relate to the GROA. Locations and activities of man-made features included federal and military facilities, civilian and military airports, roads, railroads, and potentially hazardous commercial operations and manufacturing centers outside of the controlled area. The planned man-made features within the controlled area, described in SAR Sections 1.2 through 1.4, are discussed and evaluated as part of the NRC staff's PCSA and design reviews in Technical Evaluation Report (TER) Chapters 2.1.1.2, 2.1.1.3, 2.1.1.4, and 2.1.1.7.

Repository Boundaries

The repository is located in Nye County, Nevada. The site boundary of the preclosure controlled area (also known as "proposed land withdrawal area") is that area which DOE will control. DOE also described the general environment, and the protected and restricted areas of the GROA. SAR Figure 1.1-1 depicts the site boundary, preclosure controlled area, general environment, and location of the GROA. In response to NRC staff RAIs regarding specific descriptions of the boundaries of the entire GROA and the preclosure controlled area, DOE provided a legal 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 U.S. Department of Energy (DOE) Enclosures 5 and 6 (2009au). In DOE Enclosures 7 and 8 (2009au), DOE stated that it would update its controlled area boundary depicted in SAR Figure 1.1-1 to exclude the patented mining claim acreage at the Lathrop Wells cinder cone (U.S. Patent 27-83-0002). This mining claim is on approximately 0.8 km^2 [200 acres] of private property and is not identified on maps of the proposed land withdrawal area. There are three controlled access points to the surface GROA. The layout and phased development of the surface GROA are depicted in SAR Figures 1.1-2 and 1.1-3, respectively. DOE stated it will construct a physical barrier, as defined at 10 CFR 73.2**,** for the protected area (the area where waste will be handled) of the surface GROA (SAR Section 1.1.1.1.4). The physical barrier for the protected area is discussed in SAR General Information Section 3.5.

NRC Staff Evaluation: NRC staff reviewed DOE's description of the repository boundaries in SAR Section 1.1.1.1, as supplemented in the RAI responses, using the guidance in the YMRP. NRC staff also used professional experience and knowledge of the Yucca Mountain site gained from onsite field investigations. SAR Figures 1.1-1, 1.1-2, and 1.1-3 delineate the boundaries and are of sufficient detail and scale to evaluate the site boundary and the preclosure controlled area, access points, the general environment, and the representation of the surface GROA, including its phased development. NRC staff compared the coordinates of the GROA provided in DOE Enclosures 5 and 6 (2009au) with the SAR figures and verified they are consistent. On the basis of this information, the site boundary description is reasonable for use in the PCSA and GROA design.

Natural Features

In SAR Section 1.1.1.2, DOE described the natural features within the preclosure controlled area. The preclosure controlled area is depicted 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 shown in SAR Figure 1.1-5. Using information from the Final Environmental Impact Statement (DOE, 2002aa), DOE concluded that there are no perennial or natural surface water features, including wetlands, on the Yucca Mountain site.

NRC Staff Evaluation: The NRC staff reviewed DOE's descriptions of the natural features in SAR Section 1.1.1.2 using the guidance in the YMRP. NRC staff also used publicly available maps (Carr, et al., 1996aa; Day, et al., 1998aa,ab; Potter, et al., 2002aa; Slate, et al., 1999aa), satellite images, and professional knowledge of the Yucca Mountain site to verify DOE's information regarding the natural features. On the basis of NRC staff examination of these geologic and topographic maps (Carr, et al., 1996aa; Potter, et al., 2002aa; Slate, et al., 1999aa; U.S. Geological Survey, 1961aa,ab,ac) and staff observations during onsite field investigations, NRC staff notes that there are no perennial or natural surface water features at Yucca Mountain. Descriptions of natural features are reasonably defined and the maps used to depict this information are of reasonable scale and detail to permit evaluation of the site topography and surface water drainage patterns. On the basis of this information, the description of the natural features is reasonable to permit evaluation of these features in the PCSA and GROA design.

Man-Made Features

In SAR Section 1.1.1.3, DOE described the existing man-made features and facilities located outside of the Yucca Mountain site and, in particular, within the abutting Nevada Test Site (NTS) to the east. 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 may impact 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. On the basis of DOE's analysis of hazard-initiating events due to industrial/military events (BSC, 2008an) using the Nevada Test and Training Range Chart (National Imagery and Mapping Agency, 2001aa), DOE concluded that there are no active, commercial railroad lines, passenger or freight, within 32 km [20 mi] of the surface GROA [a distance determined to be safe from the effects of explosions, fires, or toxic releases associated with transportation accidents (BSC, 2008an)].

NRC Staff Evaluation: NRC staff reviewed the descriptions of man-made features outside of the Yucca Mountain repository site provided in SAR Section 1.1.1.3 using the guidance in the YMRP. NRC staff reviewed publicly available maps (National Imagery and Mapping Agency, 2001aa; U.S. Department of Transportation, 2009aa; U.S. Geological Survey, 1961aa,ab,ac) and satellite images of the site to independently verify DOE's information. NRC staff determined the patented mining claim should be excluded from the controlled area, as discussed previously. The description of this mining claim (SAR Section 1.1.1.3.6.3) and its map location (SAR Figure 1.1-6) are reasonable. DOE's information on other man-made features is consistent with the publicly available information. The man-made features are reasonably defined and the maps in SAR Figures 1.1-6 through 1.1-10 used to depict this information are of appropriate scale and detail to permit evaluation of these features. NRC staff independently verified the location of NTS facilities indicated in SAR Figure 1.1-6 by comparing the SAR figure to satellite images (Dubreuilh, et al., 2008aa) and other available maps (National Imagery and Mapping Agency, 2001aa; U.S. Department of Transportation, 2009aa). On the basis of this information, NRC staff notes that the figures are comprehensive and of sufficient detail to permit evaluation of the location and potential impacts of man-made features and facilities relative to the Yucca Mountain GROA and the preclosure controlled area.

2.1.1.1.3.2 Regional Demography

DOE described the regional demography in SAR Section 1.1.2. DOE used this information to determine the location of the real members of the public and to support a PCSA and design of the GROA. DOE provided the population distribution in the demographic study area it established based upon NRC Regulatory Guide 4.2 (NRC, 1976aa). DOE also described the population locations, regional population centers, and population projections for the preclosure operations period (2017-2067) under consideration.

Demographic Study Area

DOE used census data from the U.S. Census Bureau (2000aa) along with supplemental data from the states of Nevada and California (BSC, 2003ah) 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.

DOE established the demographic study area following the guidance in NRC Regulatory Guide 4.2 Section 2.1 (NRC, 1976aa). 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 is divided into study area grid cells, for which DOE 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 is the baseline population distribution within the 84-km [52-mi] grid and that DOE used for population projection estimates for the 50-year operation of the repository.

There were no permanent residents identified within 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 Evaluation: The NRC staff reviewed DOE's demographic data and its methodology to establish the demographic study area using the guidance in the YMRP and NRC Regulatory Guide 4.2 (NRC, 1976aa). The NRC staff performed independent calculations to confirm the DOE population estimates {2003 population distribution within 84 km [52 mi] of the GROA using the latest Nevada County population estimates from 2001–2007 (Nevada Small Business Development Center, 2008aa)}. NRC staff's results are comparable to those of DOE's baseline 2003 population distribution data presented in the SAR. The NRC staff thus notes that DOE's estimate is reasonable The NRC staff also compared 2003 population distribution within 84 km [52 mi] of the GROA using the most recent U.S. Census Bureau data (2008aa) with that of DOE's baseline population distribution data and notes DOE's data estimate is higher, which is conservative. DOE used applicable census data, and the distribution estimates are reasonable, as confirmed by the NRC staff's independent confirmatory calculations.

Population Centers

In SAR Section 1.1.2.2, DOE 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 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 2000.

NRC Staff Evaluation: The NRC staff reviewed DOE's information pertaining to population centers near the Yucca Mountain Repository using the guidance in NRC Regulatory Guide 4.2 (NRC, 1976aa). NRC staff verified DOE's information using independent sources of information, including state and federal census data (U.S. Census Bureau, 2000aa), and notes that DOE reasonably identified the population centers of interest. DOE followed the guidance provided in NRC Regulatory Guide 4.2 (NRC, 1976aa) because DOE identified all significant population centers within the demographic study area.

Population Projections

DOE estimated the population distribution projections 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 by 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 of 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).

DOE based 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, rates that decrease at an increasing rate through 2040, and rates decreasing at lower rates for the next decades. On the basis of these decreasing population rates, an assumed constant average decline rate 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 an assumed no-change was applied from 2060–2067 on the basis of the assumption that no decline in population is expected beyond 2060.

DOE also estimated projected populations in Nye and Clark Counties due to construction and operation of the proposed repository and 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 considered the midpoint of the 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, DOE did not perform projection estimates for these areas.

NRC Staff Evaluation: The NRC staff reviewed DOE's data, assumptions, and methodology used for the population distribution projections within 84 km [52 mi] of the GROA using the guidance in NRC Regulatory Guide 4.2 (NRC, 1976aa). NRC staff also performed independent confirmatory comparisons to estimate the population projections using Nevada County Population Estimates from 2001–2007 (Nevada Small Business Development Center, 2008aa) and Nevada County Population Projections from 2008 to 2028 (Nevada Small Business Development Center, 2008ab). DOE did not address transient population estimates as recommended in NRC Regulatory Guide 4.2 (NRC, 1976aa). However, the NRC staff does not consider this to be significant, because the transient population estimate is a small number compared to the resident population. The NRC staff's estimated population projection results are comparable to DOE’s presented population distributions within 84 km [52 mi] of the GROA, and therefore DOE’s results are reasonable. DOE’s growth rate assumptions are reasonable because they are based on state and county information, which is applicable to these types of studies. On the basis of this information, DOE’s population distribution projections are reasonable for use in the PCSA and GROA design.

2.1.1.1.3.3 Local Meteorology and Regional Climatology

DOE 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 used to develop design bases for structures, systems, and components (SSCs) at the site. Atmospheric conditions, such as atmospheric stability categories, average windspeeds, and prevailing wind direction, are also described in SAR Section 1.1.3. DOE used this information to evaluate the consequences of airborne radionuclide transport in hypothetical preclosure release scenarios.

<u>Data Summaries and Collection Techniques</u>

DOE set up 12 meteorological monitoring stations to characterize the site meteorological conditions. DOE stated that it used NRC Regulatory Guide 1.23, Section C (NRC, 2007aa) and earlier versions to design and operate the monitoring stations with respect to wind, temperature, humidity, and precipitation measurements. DOE also collected the meteorological data in accordance with its quality assurance project procedures. The stations, located throughout the GROA, include a 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 windspeed and direction, temperature, humidity, and precipitation. The information collected from 1994–2006 was provided in DOE’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. DOE described the sensors used (BSC, 2007bs) and described how these sensors meet the accuracy and performance specifications of NRC Regulatory Guide 1.23 (NRC, 2007aa). DOE also described the data reduction techniques used to calculate atmospheric stability and classify windspeed characteristics according to atmospheric stability class, and described how these techniques meet the specifications of NRC Regulatory Guide 1.23 (NRC, 2007aa).

NRC Staff Evaluation: The NRC staff reviewed DOE’s data summaries and collection techniques using the guidance in the YMRP and NRC Regulatory Guide 1.23 (NRC, 2007aa). The NRC staff compared DOE’s system accuracy requirements for wind, temperature, humidity, and precipitation measurements summarized in SAR Table 1.1-9 with guidance in NRC Regulatory Guide 1.23 Section C (NRC, 2007aa) and notes that the collection techniques were based on reasonable methods and DOE’s reported system accuracy requirements for these parameters are consistent with NRC guidance.

The NRC staff compared DOE’s description, in BSC Section 4.2 (2007bs), of the methods used to determine atmospheric stability and joint frequency distributions of windspeed and direction with guidance in NRC Regulatory Guide 1.23 Section 2.2 (NRC, 2007aa) and notes that DOE’s methods are consistent with NRC guidance. The NRC staff also examined the locations of the tower and precipitation sites (SAR Figure 1.1-12) and regional sites (SAR Figure 1.1-13) and determined that (i) these are located such that each type of the primary geomorphic features of Yucca Mountain (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. Therefore, DOE's monitoring site locations provide meteorological data representative of the Yucca Mountain site consistent with guidance in NRC Regulatory Guide 1.23 Section C (NRC, 2007aa).

Annual and Probable Maximum Precipitation

DOE summarized the site precipitation data used 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. DOE included site-specific precipitation data summaries for each precipitation station, described by month 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 provided the maximum 24-hour precipitation totals for September 21 through 22, 2007, which DOE 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].

The probable maximum precipitation information is used to determine the flood hazards within the GROA. Following guidance for nuclear power plants specified in NUREG–0800 Section 2.4.3 (NRC, 1987aa), DOE used a National Oceanic and Atmospheric Administration procedure (Hansen, et al., 1977aa) to estimate probable maximum precipitation. Hansen, et al., Chapter 4 (1977aa) 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., Figures 4.5, 4.7, and 4.9 (1977aa). This procedure uses historical records from meteorological stations across the Great Basin, including several stations in southern Nevada, and does not use site observations. In SAR Section 1.1.4.3.1, DOE 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. DOE used these observed maximum values as a design basis for calculating loading on structures. New snow falling at 0 °C [32 °F] has a water content of approximately 20 percent, such that 15 cm [6 in] of fresh snowfall represents approximately 3 cm [1.2 in] of water equivalent or 0.3 kPa [6.2 lbf/ft^2]. DOE also considered a design basis volcanic ash live load of 1 kPa [21 lbf/ft^2]. The design basis value of volcanic ash live load is larger than the observed snowfall value and regional snowfall loads of 0.24 kPa [5 lbf/ft^2] that the American Society of Civil Engineers provided in ASCE 7-05, Figure 7-1 (2005ab).

NRC Staff Evaluation: The NRC staff reviewed DOE's precipitation information using the guidance in the YMRP and NUREG–0800. DOE provided reasonable information on the annual amount and forms of observed precipitation using reasonable methods because (i) DOE provided peak hourly and daily precipitation rates and described seasonal and interannual variation in precipitation, (ii) the data collection techniques were based on reasonable methods, and (iii) the monitoring locations are consistent with NRC regulatory guidance.

The NRC staff independently confirmed that DOE'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 factors from the corresponding Figures 4.5, 4.7, and 4.9 (Hansen, et al., 1977aa) and multiplying them together to obtain 34 cm [13.2 in], consistent with DOE's estimate for the basin encompassing the North Portal pad. DOE reasonably estimated the probable maximum precipitation because the methodology that DOE used is consistent with the regulatory guidance for nuclear power plants specified in NUREG–0800 Section 2.4.3. The NRC staff notes that DOE reasonably evaluated the design basis snow load because DOE (i) used a snowfall comparable to recommended guidelines and (ii) used a design basis load based upon much larger loads from volcanic ash.

Severe Weather

The DOE's assessment of severe weather was generally based on regional information or regulatory guidance, with DOE's assessment of extreme straight-line winds based on local wind data. This information was provided in SAR Section 1.1.3.6.

Tornadoes

DOE described tornadoes as infrequent and weak in the Yucca Mountain region because of generally dry weather conditions and unfavorable terrain conditions, but determined that meteorological conditions favorable for tornado formation could exist at the site on rare occasions and reported that three tornadoes have been observed in Nye County (SAR Section 1.1.3.6.1).

DOE followed procedures described in NRC Regulatory Guide 1.76 (NRC, 1976ab) to develop design basis tornado characteristics, except that, as recommended in NRC Regulatory Guide 1.76, DOE used more extreme tornado characteristics to describe the design basis tornado. DOE used information in NUREG/CR–4461 (Ramsdell and Andrews, 1986aa) to develop the alternative design basis tornado characteristics. DOE established design basis tornado parameters including a windspeed 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). For comparison, NRC Regulatory Guide 1.76 Table 1 (NRC, 1976ab) recommends using a windspeed of 257 km/hr [160 mph], a pressure drop of 4.1 kPa [0.6 psi], and a rate of pressure drop of 1.4 kPa/s [0.2 psi/s] for design basis tornado parameters for nuclear power plants in Region III, which includes the contiguous United States west of the continental divide where the GROA is located.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and notes that DOE reasonably characterized tornado hazards at the site. DOE used local site data to determine that tornadoes have been observed within Nye County. The NRC staff compared the design basis tornado parameters DOE used to design facilities in the GROA with the current regulatory guidance for nuclear power plants in the general geographic region and notes that the design basis tornado parameters are reasonable for their intended purpose, because DOE's parameters represent a more conservative condition than the NRC guidance, NRC Regulatory Guide 1.76 (NRC, 1976ab), recommends.

Extreme Winds

DOE considered hazards arising from extreme winds by using a probabilistic approach applied to measured windspeeds to estimate a maximum 3-second gust straight wind of 193 km/hr [120 mph] at a million-year recurrence interval. DOE used windspeeds measured at the Site 1 tower located approximately 1 km [0.6 mi] south of the North Portal to characterize extreme winds for this analysis (SAR Section 1.6.3.4.4). DOE considered wind hazards from hurricanes, concluding that these are not expected because of the distance between the nearest ocean, approximately 362 km [225 mi] southwest of the proposed repository, and the Yucca Mountain site. A cyclone requires sustained winds of at least 119 km/hr [74 mph] to be classified as a hurricane and at least 249 km/hr [155 mph] to be classified in the most intense hurricane category. DOE concluded that the design basis tornado {including a windspeed of 304 km/hr [189 mph]} represents a conservative bound for the effects of extreme winds (including hurricane winds).

NRC Staff Evaluation: The NRC staff evaluated DOE's assessment of hazards related to extreme winds in TER Section 2.1.1.3.3.1.2.3 using the guidance in the YMRP and notes that DOE used reasonable methods to derive extreme winds from site-measured windspeeds. Because DOE used appropriate methods to collect the windspeed data, DOE reasonably derived extreme winds from site-measured windspeeds. The NRC staff compared DOE's estimated potential for hurricane winds with the design basis tornado windspeed of 304 km/hr [189 mph] and notes that the design basis tornado bounds the extreme wind effects from hurricanes expected at the site location. On the basis of this information, DOE reasonably assessed extreme winds because its evaluation was comprehensive and it appropriately considered site measurements, tornadoes, and hurricanes in addressing the effects of extreme winds.

Lightning

DOE considered lightning strikes in evaluating external hazards that might initiate an event sequence at the repository. DOE estimated strike frequencies using annual cloud-to-ground lightning observations from 1991 through 1996 collected at the NTS and warm-season cloud-to-ground lightning data in the vicinity of the NTS from 1993 through 2000 (SAR Section 1.1.3.6.2). The Air Resources Laboratory and Special Operations and Research Division of the National Oceanic and Atmospheric Administration 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. DOE indicated that these observations are generally consistent with other estimates for southern Nevada. On the basis of this information, DOE determined that a direct lightning strike is an initiating event at the repository (SAR Section 1.6.3.4.6) and DOE provided lightning protection in the design of facilities in the GROA.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and notes that lightning strikes are a potential initiating event at the repository because the available lightning data, gathered from a credible source consistent with regional estimates, suggest that dozens of lightning strikes are expected within the GROA over the assumed 100-year preclosure period. DOE's statement to provide lightning protection in the design of the GROA facilities is reasonable as discussed in TER Chapter 2.1.1.3, where the NRC staff notes that the special design features DOE proposes to install on ITS SSCs are reasonable because they have been designed following industry standard codes and NRC guidance.

Hurricanes

In addition to extreme winds from hurricanes as discussed previously, DOE considered external flooding as a result of hurricanes in evaluating external hazards. DOE identified Santa Monica Bay, approximately 360 km [225 mi] southwest of the proposed repository, as the nearest body of water susceptible to hurricane-induced external flooding and concluded that this body of water did not pose a credible external flooding threat.

NRC Staff Evaluation: The NRC staff notes that hurricane-induced external flooding does not provide a credible external flooding threat due to the distance of the site from the nearest body of water.

Other Severe Weather

DOE also considered sandstorms and dust storms at the site. DOE determined that sandstorms would be unlikely because a windspeed of greater than 40 km/hr [25 mph]—rare at the site—would be needed to initiate them. However, DOE considered potential safety consequences of sandstorms and dust storms, concluding that sandstorms and dust storms do not form a credible means for causing a loss of cooling capability because of the design of the cooling water supply.

NRC Staff Evaluation: The NRC staff notes that DOE reasonably characterized sandstorms and dust storms at the site because DOE used appropriate site information to determine the frequency and magnitude of windspeeds at the site. Consistent with industry assumptions regarding windspeed, a windspeed of greater than 40 km/hr [25 mph] is needed to initiate sand and dust storms. NRC staff also notes that DOE's information is reasonable to support the PCSA and GROA design because the level of detail in DOE's description is commensurate with its low importance to safety.

2.1.1.1.3.4 Regional and Local Surface and Groundwater Hydrology

DOE investigated the regional and local surface and groundwater hydrology to support its evaluation of the PCSA and the GROA design. 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, DOE estimated the probable maximum flood resulting from the probable maximum precipitation (see TER Section 2.1.1.1.3.3) to determine the design bases for the flood protection structures DOE plans to construct to protect the GROA from runoff and debris flows. NRC staff evaluated DOE's hydrological information presented in SAR Section 1.1.4.

Surface and Groundwater Hydrologic Features

DOE provided information pertaining to the regional and local surface and groundwater hydrology. DOE 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 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. DOE 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, Section 3.4.3 (2004bj). DOE 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 Yucca Mountain surface facilities are situated on the east side of Exile Hill in Midway Valley at the eastern margin of Yucca Mountain. The natural drainage channels near the Yucca Mountain site were shown 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), DOE determined that the pooling or ponding of large quantities of water on the surface is limited and aging pads are graded to prevent the pooling of water.

The elevation of the surface GROA is 1,120 m [3,675 ft] above the sea level, whereas the water table is approximately 390 m [1,280 ft] below the surface GROA (SAR Section 1.1.4.2.3), which means the unsaturated zone is 390 m [1,280 ft] thick. Perched water (entrapped water) has been identified in several boreholes (SAR Figure 1.1-56). DOE stated that the perched water bodies do not represent obstacles to repository design, because they are located 100 to 200 m [328 to 656 ft] below the repository horizon.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of surface and groundwater hydrologic features using the guidance in the YMRP. The regional flood history is reasonably described because DOE conducted a comprehensive literature review of the history of drainage system patterns, paleo and historical surface water flow conditions, and historical flood occurrences and flood discharges in the Yucca Mountain area. NRC staff notes that the description of the hydrology reasonably identifies hydrologic features (unsaturated zone, saturated zone, flash flood, and perched water) relevant to the PCSA and GROA design.

<u>Probable Maximum Flood</u>

In SAR Section 1.1.4.3, DOE provided a flood inundation analysis. DOE analyzed probable maximum flooding to assess the performance of preventive measures such as dikes and channels around critical facilities to control run on. DOE's analyses considered a surface facility design with and without a flood-inundation mitigation measure. DOE used the U.S. Army Corps of Engineers HEC-1 (Version 4.0) software program, an industry-standard code for event-based rainfall-runoff analysis, to predict surface runoffs and channel discharge resulting from the probable maximum precipitation event (BSC, 2007br). DOE used the U.S. Army Corps of Engineers HEC-RAS (Version 2.1) software program, also an industry-standard code for calculating water surface profiles, to conduct flood inundation analysis. More specifically, DOE used HEC-RAS to determine the magnitude and duration of runoff that would occur during the HEC-1 predicted probable maximum flood event.

The inputs to HEC-1 and HEC-RAS include topography, probable maximum precipitation, and hydraulic properties for each subarea and channel. The subareas DOE used in the HEC-1 probable maximum flood analysis are depicted in BSC Figure 6-1 (2007br). DOE used a 0.6-m [2-ft] elevation contour map to produce a digital elevation model of the study area. DOE obtained the length, slope, and channel dimensions using topographic data for natural channels and engineering drawings for man-made channels.

Subarea Hydraulic Properties

DOE used a unit hydrograph method to develop a runoff hydrograph. Determination of a unit hydrograph requires subarea size and time of concentration. DOE used a U.S. Bureau of Reclamation empirical formula to calculate the time of concentration for each subarea. Among common formulas available, the U.S. Bureau of Reclamation empirical formula gives the smallest time of concentration, as identified in BSC Section 6.1.4 (2007br) and, thus, the largest peak flow. DOE assumed a uniform infiltration rate of 38.1 mm/hr [1.5 in/hr], which is lower than the lowest infiltration values obtained from the *in-situ* infiltration tests conducted in the surrounding area (lower infiltration leads to greater runoff), as described in BSC Section 6.1.4 (2007br). NRC staff recognizes that further lowering the assumed infiltration rate of 38.1 mm/hr [1.5 in/hr] would not significantly increase the runoff rate according to the Manning's equation.

DOE used a bulking factor to account for increased flow depths caused by the presence of entrained air, debris, and sediment load. DOE used a bulking factor of 10 percent. In other words, DOE increased the peak discharges by 10 percent in its probable maximum flood analyses. A literature review by DOE suggested that flow bulking may not be a significant factor affecting probable maximum floods, as outlined in BSC Section 6.1.4 (2007br), because of the large peak flow rates associated with the latter.

NRC Staff Evaluation: The NRC staff examined the division of subareas using the guidance in the YMRP and NRC Sections 2.4.1, 2.4.2, and 2.4.3 (1987aa) and notes that subareas were divided in such a way that allowed the basin drainage characteristics to be captured. The resolution of the elevation contour of 0.6 m [2 ft] is suitable for characterizing the topography and for delineating the watershed, its subareas, and the channel flow paths. NRC staff notes that (i) the subarea properties DOE used in its HEC-1 model and the assumptions are reasonable because they are based on applicable data and (ii) DOE applied a standard approach in developing a runoff hydrograph for HEC-1 and DOE assumed a reasonable infiltration rate for all subareas. On the basis of the range of values DOE considered and the surface and geologic conditions specific to the GROA, DOE's selected bulking factor is reasonable for probable maximum flooding analysis at the GROA.

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 HEC-1 and HEC-RAS modeling. In a sensitivity study described in BSC Section 6.1.5 (2007br), DOE considered the range of Manning's coefficient for three flow conditions: clear water flow, high sediment transport, and mudflow. DOE 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 DOE provided in DOE (2009bf).

For probable maximum flood analysis, DOE considered the amount of clear water runoff that would be large enough so that mudflow condition is unlikely to develop. Results of DOE'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, only resulted in an insignificant increase of 0.15 m [0.5 ft] (DOE, 2009bf). Therefore, DOE used a Manning's coefficient of 0.09 in its probable maximum flood analysis, corresponding to the high sediment transport flow condition.

NRC Staff Evaluation: The NRC staff reviewed the procedure DOE used to determine the Manning's coefficient of 0.09 for the probable maximum flood analysis. The NRC staff notes that the value used is reasonable because it was selected on the basis of applicable literature survey and sensitivity studies.

Probable Maximum Flood Rate

The maximum probable flood peak flow rate resulting from DOE's HEC-1 model is 1,564 m^3/s [55,240 cfs], as shown in BSC Table 7-1 (2007br).

NRC Staff Evaluation: The NRC staff recognizes that this peak flow rate is about 20 percent higher than the maximum local probable flood peak flow rate predicted by Bullard (1986aa), who computed flood potentials for 11 small drainage basins on Yucca Mountain for clear water flows. Assuming a bulking factor of 10 percent, the NRC staff notes that the simulated probable maximum flood peak flow rate is still higher than that Bullard (1986aa) predicted. NRC staff reviewed the application of the analysis of the probable maximum flood from a flood hazard perspective in TER Chapter 2.1.1.7. DOE reasonably analyzed the probable maximum flood because (i) the software codes and methodology DOE used follow professional practice in hydrological engineering; (ii) the techniques and assumptions DOE used to derive the channel properties for HEC-1 and HEC-RAS analyses are reasonable; and (iii) DOE used reasonable input data for probable maximum flood simulation on the basis of site geographic and probable maximum precipitation data that NRC staff notes are reasonable in TER Sections 2.1.1.1.3.1 and 2.1.1.1.3.3. The staff notes that the probable maximum flood DOE's HEC-1 model simulated is comparable with other independent studies.

Application of Probable Maximum Flood Analyses

DOE summarized the estimated peak probable maximum flood flows from subareas and concentration points, as shown in BSC Table 7-1 (2007br), and flood inundation results for man-made channel segments, as shown in BSC Tables 7-2 to 7-4 (2007br). In the no-mitigation case, DOE assumed that the planned facilities upstream were not constructed and the flood control measures were not implemented. In SAR Figure 1.1-57, DOE 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.

DOE's calculations indicated that water would not overflow the South Portal pad or the planned North Construction Portal during a nonmitigated probable maximum flood event. To protect the surface GROA from inundation, DOE showed in SAR Figure 1.2.2-7 that ITS structures, North Portal, and Aging Facility areas are protected by engineered features such as dikes, and drainage and diversion channels.

NRC Staff Evaluation: DOE addressed 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. The NRC staff evaluated the effects of proposed changes to natural drainage and flood control features in TER Chapter 2.1.1.7 and notes that DOE's probable maximum flood analysis reasonably demonstrates that ITS structures will not be subject to flood inundation.

2.1.1.1.3.5 Site Geologic Conditions, Seismology and Seismic Site Response, Geotechnical Engineering Conditions, and Fault Displacement Hazard Analysis

DOE provided information related to site geology and seismology used to support the PCSA and the GROA design in SAR Section 1.1.5 and supplemental information. This information was also used to identify naturally occurring hazards. 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 (FDHs).

2.1.1.1.3.5.1 Site Geologic Conditions

In SAR Section 1.5, DOE provided the geologic information from its site characterization investigations that was used to support PCSA and GROA design. In SAR Section 1.1.5.1, DOE 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 will be placed in the underground (subsurface) GROA. DOE 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), likely to affect the GROA mechanical and hydrologic properties and conditions. The NRC staff review and evaluation of the Yucca Mountain site geologic conditions are described in the following subsections on the geology of the subsurface GROA and the geology of the surface GROA.

2.1.1.1.3.5.1.1 Geology of the Subsurface Geologic Repository Operations Area

The subsurface GROA is composed entirely of rocks called tuff. The tuff is layered, 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 the topics of stratigraphy and structural geology.

<u>Stratigraphy of the Subsurface GROA</u>

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 (solidified erupted ash, with minor lava flows) that originated from large explosive volcanoes to the north. 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 was 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.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of stratigraphy of the subsurface GROA, including DOE's stratigraphical studies conducted in the Yucca Mountain region (BSC, 2004bi; Sawyer, et al., 1994aa). DOE 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). NRC staff notes that DOE has described the age of the rocks, rock layer stacking order, and thickness variations of the volcanic rocks and that these descriptions are consistent with 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 necessary to understand 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). NRC staff notes the principal geologic deformation features and processes that might affect the volcanic rocks at Yucca Mountain during the preclosure period are faulting and fracturing.

Faulting at the Subsurface GROA

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 response to NRC staff RAIs (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.

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, 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 1.8 million years ago) and have the potential for significant future movement.

DOE also described intrablock faults, which are defined as less continuous (i.e., shorter length) and having smaller displacement than the block-bounding faults. DOE classified the Sundance and Ghost Dance faults, located within the subsurface GROA, as intrablock faults. DOE observed additional small-displacement faults and shear fractures in the ESF and ECRB.

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 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 NRC staff RAIs, DOE described significant fault displacements and how to determine such displacements (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. DOE described the displacement on these two faults during the Quaternary Period, which was well established through trenching field studies.

The standoff (also called “setback”) distance determination is dependent upon the characterization of the main fault and splays and their displacements, width of fault damage zones, and attendant zones of influence (DOE, 2009as,bf). Setback distance from this significant Quaternary fault is a prerequisite for completing the subsurface GROA design. In particular, the location of the westernmost endpoints of emplacement drifts (and therefore, the location and length of emplacement drifts) is dependent upon the location of the west access main, which DOE stated will also be 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 that this information on which its estimates are based will be confirmed during excavation of the openings.

However, DOE also stated that a standoff distance of 60 m [197 ft] from Quaternary block-bounding faults with potential for significant displacement also provides a safety margin from preclosure FDHs. This is based on an analysis of displacement and stress adjacent to an active fault for displacements up to 1,000 mm [3.3 ft]. The largest mean preclosure displacement on the Solitario Canyon fault is 320 mm [1 ft] with an annual exceedance probability of 10^{-5} (BSC, 2003aj). In addition to the hazard of direct fault displacement, DOE 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] standoff 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).

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). Because the access mains are subject to the 60-m [197-ft] standoff design control parameter 01-05, and because waste emplacement is typically 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 setback and predict location of intersections in adjacent drifts [DOE Section 1.2.3.1 (2009as); SAR Table 5.10-3]. The identification and characterization of these faults are important commitments. For PCSA, DOE introduced the term “fault damage zone”: a zone of faulting and fracturing that can represent a potential hazard for breaching a waste package. A narrow damage zone is one in which shear displacement is the primary hazard; a wide damage zone is one in which rock blocks may be disaggregated and/or large enough to pose a rockfall hazard (DOE, 2009as,bf).

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 deemed them not credible. Furthermore, narrow faults with observed total displacement of 2 m [6.7 ft] or less are estimated to have an annual probability of exceedance (APE) of less than 10^{-8} for future displacements of 3 m [10 ft] and are beyond Category 2. NRC staff evaluates the probabilistic FDH in TER Section 2.1.1.1.3.5.5.

NRC Staff Evaluation: NRC staff reviewed DOE’s description of faulting of the subsurface GROA in SAR Section 1.1.5.1 and responses to RAIs using the guidance in the YMRP. NRC staff also used its professional experience and knowledge gained on the Yucca Mountain site 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). NRC staff notes that DOE’s six cross sections covering the entire length and width of the subsurface GROA reasonably depicted stratigraphic layering and faults at a detailed scale of 1:14,400 with a vertical exaggeration of 2. At this scale, the cross sections depict the individual subdivided members and zones of the rock formations, and enable the NRC staff to evaluate 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.

Because DOE extended the geological cross sections beyond major faults and provided reasonable supplemental explanations on faulting in the subsurface in response to NRC staff RAIs, DOE’s representation and interpretation of the spatial relationship between the major block-bounding faults that influence GROA design are reasonable. NRC staff notes that DOE provided information on the subsurface GROA structural geology (DOE, 2009ar,as,bg).

DOE’s evaluation of block-bounding fault displacement and setback, provided in response to RAIs (DOE, 2009as,bf), is reasonable because DOE used a suitable method to estimate the shortest distance from a waste package to a fault or fault zone.

DOE stated that it will confirm actual fault locations and characteristics (including fault width and displacement), for faults such as the Solitario Canyon fault, in accordance with Design Control Parameter (DCP) 01-05; DOE Section 1.3.1 (2009bf); and SAR Section 1.3.4.2.2. For intrablock faults that are unnamed or undiscovered, such faults will be confirmed from field data to be collected from near-horizontal borings during construction [DOE Section 1.3.1, No. 3 (2009as); DOE Section 1.3.2 (2009bf); SAR Table 1.9-10, Procedural Safety Control (PSC)-25]. DOE also specified a setback distance measurement process for aging pads from the Bow Ridge fault, as outlined in DOE Section 1.1.3 (2009bf). DOE would then use this new information to be collected during the performance confirmation program and construction period to confirm the design of the subsurface GROA structures as it pertains to setback distances from faults. NRC staff notes that these approaches are appropriate and will ensure that (i) the known Solitario Canyon fault that might be encountered during construction and (ii) unknown intrablock faults that are encountered during construction will be appropriately considered for setback. Also, DOE recognized the characterization of intrablock faults as "probably needing administrative control" (SAR Table 5.10-3).

For reasons stated previously, NRC staff notes that both (i) the analysis of fault displacement and rockfall hazard (BSC, 2003aa) DOE 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) DOE's assessment of the significance of fault displacement in PCSA are appropriate.

Slickenlines, also known as slickensides, are physical marks (e.g., grooves, streaks of minerals; parallel and linear or curvilinear) on a fault surface. According to DOE, any known or unknown faults encountered during construction (as DOE proposes in Design Control 01-05 and PSC-25) are evidence of and a means to measure the total magnitude and direction of displacement of the fault (i.e., the actual amount and vector of slip). Evidence of slickenlines, wherever observed, can be used to ensure the optimization of direct measurements of fault displacement (i.e., amount of slip) when performing the geologic mapping of the subsurface GROA during planned performance confirmation activities and construction. The staff notes that not using the orientation of slickenlines when determining the amount of fault offset, as described in DOE Enclosure 2 (2009bg), is likely to underestimate displacement in all cases except where slickenlines are parallel to fault dip. The small number of slickenlines reported during site characterization activities implies that this will not affect the results of the Probabilistic Fault Displacement Hazard Analysis (PFDHA), but their inclusion could enhance the fault displacement database.

On the basis of the NRC staff review as described previously, NRC staff notes that DOE provided reasonable information on the subsurface GROA structural geology and faulting interpretations (DOE, 2009ar,bf,bg).

Fracture Characteristics at the Subsurface GROA

In SAR Section 1.1.5.1.3, DOE characterized fractures of the rocks in the subsurface GROA. In SAR Section 1.1.5.1.3.3, DOE stated 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. DOE discussed fracture formation and assessed its characteristics, including orientation, dip angle, length, spacing, and connectivity.

DOE 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 it measured in different RHH zones. DOE considered the hazard from fractures 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 NRC staff in TER Sections 2.1.1.1.3 and 2.1.1.1.4.

DOE's description of fractures was based mainly 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 process of pyroclastic flow deposits. These fractures are longer fractures as compared to the 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]. DOE characterized the middle nonlithophysal zone as a network of long, relatively closely spaced fractures that DOE separated into four sets on the basis of orientation: two sets are subvertical with northwest-striking and northeast-striking orientations, the third set strikes to the northwest with a moderate dip, and the fourth set is northwest striking and shallowly dipping.

For fractures in the lower nonlithophysal zone along the ECRB cross drift, DOE identified three steeply dipping sets, with the most prominent striking northwest. DOE also identified a northwest-striking, shallowly dipping set among the lower nonlithophysal zone fractures. DOE 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.

DOE 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 DOE, the intensity of long fractures {greater than 1 m [3.3 ft]} correlates to rock type, but not to proximity to faults. However, for shorter fractures, DOE 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] away from the fault. 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 is partly dependent on the stratigraphic interval, such that nonwelded rocks are characterized by sharp faults and smaller zones of influence than welded rocks. Also, DOE noted that small, discontinuous faults interact with the fracture network and suggested that the preexisting weakness represented by the fracture network distributes strain as shear along multiple fractures. Such shear is manifest as thin selvages of tectonic breccia and slip lineations on fracture surfaces.

NRC Staff Evaluation: NRC staff reviewed DOE's information on fractures using the guidance in the YMRP and NRC staff's professional experience and knowledge of the site gained through independent analyses. 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). DOE's description of fracture orientations in the RHH intervals is generally consistent with staff independent analyses, and the NRC staff notes that DOE's fracture-orientation information is reasonable to support its use in the PCSA and the GROA design. For example, the results of staff fracture analyses show that the prevailing fracture orientations in each of the four Topopah Spring Tuff in which the emplacement drifts would be excavated (Smart, et al., 2006aa) are within DOE's range of azimuths on which to align the proposed emplacement drifts (i.e., between 60 and 105°), as identified in DOE Enclosure 4 (2009as) and SAR Section 1.3.4.2.3.

The NRC staff analyses show that DOE's characterization of fracture networks for use in the GROA design reflected several sampling biases (Smart, et al., 2006aa). DOE's interpretation of fracture spacing and connectivity may not reasonably capture the uncertainties of these parameters. The NRC staff analyses show that DOE overestimated fracture spacing at Yucca Mountain because of sampling biases (Smart, et al., 2006aa). Fracture spacing and connectivity are considerations for design of ground support systems for safety during operations and are relevant to postclosure analyses of drift degradation, rockfall, and seepage

(NRC, 2004ab, 2005aa; Ofoegbu, et al., 2007aa). Nevertheless, uncertainties resulting from the differences in average values the NRC staff and DOE derived may be constrained or reconciled by data-reduction and statistical methods applied to existing data and data to be collected during construction (NRC, 1999aa, 2005aa; Smart, et al., 2006aa). DOE's characterization of the fracture data from the RHH included the recognition of natural variability within a stratum and between strata when applied to (i) design and performance bases for the ground support system (rock spallation) and (ii) rockfall assessments and what the optimum azimuth of the drifts should be. DOE stated that the fracture parameters input to design and performance will reflect the actual field observations to be made during construction of underground openings .

2.1.1.1.3.5.1.2 Geology of the Surface Geologic Repository Operations Area

In SAR Section 1.1.5.1.4, DOE 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. However, DOE identified the upper portion of ramps and ventilation shafts that are located on Yucca Mountain and Exile Hill and are connected to the subsurface GROA as part of the surface GROA. They are excavated entirely in volcanic tuff. Therefore, the staff notes that characteristics of faults and fractures of the tuff of the subsurface GROA discussed in the preceding section influence parts of the surface GROA constructed in tuff. Characterization of alluvium and rock properties and conditions at the surface GROA is necessary for the design of facilities and their foundations and cut and fill slopes. DOE 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.

<u>Stratigraphy of the Surface GROA</u>

In SAR Section 1.1.5.1.4, DOE characterized the near-surface stratigraphy using geologic mapping, boreholes, test pits, trenches, and geophysical investigations (BSC, 2002aa; SNL, 2008af). DOE determined that the surface GROA is underlain in ascending order by volcanic rocks of the Tiva Canyon Tuff, the post-Tiva Canyon Tuff bedded tuff, the pre-Rainier Mesa Tuff bedded tuff, and the Rainier Mesa Tuff. These volcanic rocks are 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 Evaluation: NRC staff reviewed DOE's stratigraphic information and notes DOE's general description and thicknesses of the bedrock formations and alluvium are reasonable because they are consistent with NRC staff's field observations and independent studies (Waiting, et al., 2007aa; NRC, 1999aa). NRC staff further evaluated the properties, variations, and thicknesses of the volcanic rocks and alluvium in TER Section 2.1.1.1.3.5.4 and notes that DOE's stratigraphic description of the surface GROA is reasonable to use in the PCSA and design of the surface GROA.

<u>Structural Geology of the Surface GROA</u>

The dominant structural features of the volcanic rocks relevant to the surface GROA facilities design, construction, and operation consist of tilted and faulted rock layers. DOE 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 referred to as 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, DOE also interpreted several 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.

DOE revised its understanding of the Bow Ridge fault position, length, orientation, and displacement on the basis of five of its 2006–2007 boring data and indicated that it relocated the trace of that fault by about 100 m [330 ft] to the east (Orrell, 2007aa; SAR Figure 1.1-59). However, DOE stated that the fault location is consistent with the locations of previously identified faults. DOE 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 further assessed for their potential hazards.

NRC Staff Evaluation: NRC staff reviewed DOE's faulting information and notes that DOE's geological map of the surface GROA area, at a scale of approximately 1:15,000 (SAR Figure 1.1-64), provided reasonable geological detail, but did not cover the entire area of the surface GROA. However, the revised geological map DOE submitted was reasonable because it covered the whole surface GROA at a detailed scale of 1:12,000, showed the topographic and surface geological features in relation to the major surface facilities, and showed the relocated Bow Ridge Fault, as outlined in DOE (2009at) and revised by DOE (2009bg).

DOE's 3 geological cross sections in SAR Figures 1.1-65 to 1.1-67 covered only 20 to 35 percent of the width of the surface GROA and showed a number of interpreted subsurface faults inconsistent with the number of fault traces shown on the geological map (SAR Figure 1.1-64). The NRC staff notes that the new and revised six interpreted geological cross sections (DOE, 2009at) DOE provided cover the main footprints of the surface facilities, but did not cover the entire width of the surface GROA. For example, the two east-west cross sections (SE-SE' and SD-SD') covering the two northern aging pads (17P and 17R) stop short of Midway Valley Fault to the east, and cross section SE-SE' stops short of reaching a Bow Ridge Fault splay to the west. Although the revised geological cross sections still show some inconsistencies and are too short to provide a view and interpretation of the geology underneath the immediate footprint of some facilities or aging pads, the total information provided by DOE is reasonable because DOE compensated for its incomplete graphical representations of the surface GROA structural geology by providing explanations and justifications on its geological map and cross sections. In addition, the NRC staff notes that DOE's information is reasonable because DOE's graphical representations also included limitations regarding the methods used, the assumptions made, and the associated uncertainties on DOE's structural geology interpretations.

NRC staff notes that DOE explained (i) how 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 further assessed for their potential hazards to facilities ITS and Important to Waste Isolation during initial construction activities; (ii) its criteria and technical basis used to select standoff distance to relocate Aging Pads 17P and 17R to about 100 m [330 ft] east of the reinterpreted location of the Bow Ridge fault, classified as a Quaternary fault of significant displacement; and (iii) the base of the alluvium-colluvium on top of the bedrock is not faulted, taking into account the uncertainties in the methods used to assess faults vertical offsets.

DOE provided information on the potential effect of faults and, specifically, the Bow Ridge Fault, which exhibits clear field evidence of Quaternary displacement. Using the displacement approach and the earthquake approach (methods described in TER Section 2.1.1.1.3.5.5), DOE stated that the Bow Ridge Fault would have no impact on ground motion at the surface GROA structures (DOE, 2009as). DOE stated that this lack of impact on ground motion is further enhanced by the fact that relocation of the Bow Ridge Fault slightly increases the distance from the fault to the ground motion calculation point. Thus this would result in a slightly lower calculated ground motion. DOE's explanation of a lower ground motion calculation is reasonable because as the distance increases between the area where the Bow Ridge Fault was relocated and the surface GROA facilities, the calculated ground motion would decrease as a function of the distance.

NRC staff notes that DOE provided reasonable explanations on its geological map and cross sections and on its faulting interpretations, including their limitations, assumptions, and associated uncertainties. To further address these faulting uncertainties, DOE stated that if buried or unknown faults were encountered in the course of excavating for foundations, faults would be further investigated to define the associated hazard in accordance with the preclosure methodology (DOE, 2009bf). DOE's approach for further fault investigation is reasonable because it follows standard engineering practice to confirm or modify the design of a facility as new geotechnical and geological information becomes available during excavation and drilling activities to be conducted at the time of construction. Therefore, DOE's information on the surface GROA structural geology is reasonable and can be used to further evaluate the basis for foundation designs and assess potential seismic and FDHs and risks during the preclosure period.

2.1.1.1.3.5.2 Seismology and Probabilistic Seismic Hazard Analysis (PSHA)

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, DOE provided its description of site seismology. DOE 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's responses to the NRC staff RAIs in DOE Enclosure 19 (2009ab) and DOE Enclosures 6, 7, and 8 (2009aq) and the references cited therein.

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

1. DOE conducted an expert elicitation in the late 1990s to develop a PSHA for Yucca Mountain. This assessment included a PFDHA that is discussed in TER Section 2.1.1.1.3.5.5 (CRWMS M&O, 1998aa). 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., Section 5 (1996aa).

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

3. DOE modified 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.

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

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

PSHA—Methodology

DOE 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). DOE stated that its PSHA methodology followed the guidance of the DOE-NRC-Electric Power Research Institute (EPRI)-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa). In SAR Section 2.2.2.1.1.1, DOE concluded that the methodology used for the PSHA expert elicitation is consistent with NRC expert elicitation guidance, which is described in NUREG–1563 (NRC, 1996aa).

To conduct the PSHA, DOE 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 relevant 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 random and unknown 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 DOE, the U.S. Geological Survey, and Risk Engineering Inc. (1998aa), which 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.

According to DOE-NRC-EPRI-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa), the basic elements of the PSHA process 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. According to the Budnitz, et al. (1997aa) 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.

NRC Staff Evaluation: The NRC staff reviewed DOE's PSHA methodology described in SAR Sections 1.1.5.2.4 and 2.2.2.1.1 using the guidance provided in the YMRP and NUREG–1563. NRC staff also evaluated DOE's PSHA development to determine whether it included the four basic elements described in Budnitz, et al. (1997aa). In addition, NRC staff observed all expert elicitation meetings and reviewed summary reports of those meetings as they were produced. On the basis of this information, including the evaluation with respect to Budnitz, et al. (1997aa) and NRC staff's direct observations of the expert elicitation process, the NRC staff notes that DOE's elicitation for the PSHA is consistent with the framework for conducting an expert elicitation described in NUREG–1563. Because DOE used NUREG–1563 or an equivalent procedure, DOE's implementation of the PSHA expert elicitation is reasonable to develop estimates of seismic hazards for use in the PCSA and GROA design.

<u>PSHA—Input Data and Interpretations</u>

During the expert elicitation, DOE's seismic source teams considered a range of information from many resources including DOE, the U.S. Geological Survey, project-specific Yucca Mountain studies, and information published in the scientific literature. 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). DOE also supported the PSHA with a broad range of data, process models, empirical models, and seismological 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 DOE with sufficient representation of the seismic hazard for use in the PCSA and GROA design.

DOE 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 DOE'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 Evaluation: The NRC staff reviewed DOE's PSHA input data and interpretations as described in SAR Sections 1.1.5.2 and 2.2.2.1.1. NRC staff notes that DOE reasonably developed the geological, geophysical, and seismological information necessary to support the expert elicitation. NRC staff's evaluations in NUREG–1762 (NRC, 2005aa) indicated that DOE's information was consistent with site conditions at Yucca Mountain. The NRC staff has first-hand knowledge of the geology and seismic characteristics of the Yucca Mountain region, which includes 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 notes 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 is reasonable because it is consistent with NRC guidance in NRC Regulatory Guide 1.165 (NRC, 1997ab) and NRC Regulatory Guide 3.73 (NRC, 2003ae).

The NRC staff also reviewed additional geological, geophysical, and seismological information (e.g., Wernicke, et al., 2004aa) discovered since the elicitation was completed and published by DOE in 1998. On the basis of its detailed understanding of the Yucca Mountain geology, the staff notes that new geological and seismological information, with the exception of overly conservative information on large ground motions at low annual exceedance probabilities as described next in the conditioning of ground motion hazard, would not substantially alter the PSHA results.

Conditioning of Ground Motion Hazard

In SAR Section 1.1.5.2.5.1, DOE provided the conditioning of ground motion hazard at the reference bedrock outcrop where the PSHA was developed. Since PSHA completion in 1998, several studies and reports, including ones from NRC staff (NRC, 1999aa), the Nuclear Waste Technical Review Board Panel on Natural System and Panel on Engineered Systems (Corradini, 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. 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 (Kana, et al., 1991aa).

The overly conservative earthquake ground motions arose in DOE’s study because the seismic hazard curves are constructed as unbounded lognormal distributions. In past practice, probabilistic seismic hazard curves were used to estimate ground motions with annual exceedance probability to 10^{-4} or 10^{-5} (typical annual exceedance probability values for nuclear power plant design and safe shutdown earthquakes).

For Yucca Mountain, however, the seismic hazard curves are extrapolated to estimate ground motions with annual exceedance probabilities as low as 10^{-8}. At these low probabilities, the seismic hazard estimates are driven by the tails of the untruncated Gaussian distributions 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 reconcile 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. Rather than reconvene the PSHA expert elicitation and redo the PSHA, DOE chose to treat the issue as part of the ground response analysis.

Accordingly, DOE’s second step in developing ground motion inputs for design and PCSA and postclosure assessment, after the development of the PSHA, was to condition the ground motion hazard. This second step included information on the level of extreme ground motion that is consistent with the geological setting of Yucca Mountain. Conditioning of ground motion hazard is a unique study developed for the Yucca Mountain project.

Methods for PSHA Results Conditioning

DOE used two 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 shear-strain-threshold distribution is then related to the distribution of horizontal PGV through ground motion site-response modeling.

To develop the shear strain threshold distribution, laboratory rock mechanics data, corroborated by numerical modeling, are used. The shear-strain levels to initiate unobserved stress-induced failure of lithophysal deposition of the Topopah Spring Tuff are derived. The site-response calculation uses the random vibration theory (RVT)-based equivalent-linear model (SAR Section 2.1.1.1.3.5.3.1) to compute the mean motions strains for the deaggregation earthquakes that dominate the contribution of ground motion hazard of the specified APE. As DOE discussed (BSC, 2008bl), this approach has been (i) generalized to other than horizontal PGV; (ii) modified to use the inferred shear-strain threshold at the repository waste emplacement level to determine the level of ground motion not experienced at the reference rock outcrop, rather than at the waste emplacement level; (iii) refined to include variability in shear strain levels and integration over the entire hazard curve; and (iv) updated to incorporate additional geotechnical data on site tuff and alluvium properties in the site-response part of the approach.

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 from laboratory experiments) and interpretations. It is used in the RVT method for point sources to develop distributions of PGV and peak ground acceleration (PGA) at the reference bedrock outcrop. The extreme stress drop is characterized by a lognormal distribution with a median value of 400 bars and σ_{ln} of 0.6 (mean of 480 bars). This distribution is discretized to three values of 150, 400, and 1,100 bars with the weighting factors of 0.2, 0.6, and 0.2, respectively. This distribution is mapped into a distribution of extreme ground motion for the reference bedrock outcrop through the RVT site-response modeling. However, as discussed in SAR Section 1.1.5.2.5.1 and BSC (2008bl), DOE conducted conditioning using, in series, the shear-strain-threshold and extreme-stress-drop methods.

NRC Staff Evaluation: The NRC staff reviewed DOE's methods for conditioning of PSHA results in SAR Section 1.1.5.2.5.1 and DOE's responses to NRC RAIs (DOE, 2009aq). DOE's methods for conditioning the PSHA results on the basis of the shear-strain-threshold and extreme-stress-drop methods provide reasonable scientific and engineering bases to determine the physical limitations of the Yucca Mountain rock. Therefore, DOE's methods are reasonable for conditioning the PSHA results.

Results of PSHA Conditioning

The unconditioned hazard curve, which is the 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 shear-strain-threshold conditioning has a marginal impact as compared to the extreme-stress-drop approach. For example, for an APE of 10^{-8} the shear-strain-threshold-conditioned PGV hazard is reduced from 1,200 to about 1,100 cm/sec [433 in/sec] or about 10 percent; the stress-drop-conditioned PGV hazard is reduced from 1,200 to about 480 cm/sec [189 in/sec] or about 60 percent, as outlined in BSC Section A4.5.1 (2008bl). DOE found that the combined conditioning has minimal impact to ground motions of DBGM-1, DBGM-2, and beyond DBGM, which are for APEs of 10^{-3} to 10^{-4}. In contrast, for APEs of 10^{-5}, 10^{-6}, 10^{-7}, and 10^{-8}, the impact is greater (SAR Section 1.1.5.2.5.1). SAR Figures 1.1-79 and 1.1-80 compared the unconditioned and conditioned PGA and PGV mean hazard curves for the reference bedrock outcrop.

The four workshop proceedings described in BSC Appendix A (2008bl) conducted to develop the expert judgment were well documented in presentations, discussions, and assessments. The stress drop data from the United States and other countries were used in the expert judgment. The parameter variability involved in the empirical ground motion attenuation relation and numerical simulations, which the experts relied on, was included. 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 receiver) were considered in the modeling to map the stress drop into ground motion distribution.

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

NRC Staff Evaluation: The NRC staff notes that (i) the conditioned hazard curves are reasonable for GROA design and PCSA because the conditioning methods follow the basic mechanical, material, and seismological principles that are applicable and (ii) 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 greater than those expected in the Yucca Mountain region. DOE's assumptions that the tectonic setting and therefore the stress drops of earthquakes from the existing faults at Yucca Mountain are not going to change significantly over the next million years are reasonable, as they are in accordance with the basic studies of tectonics in the Yucca Mountain region, which provide the basis for the conditioning at very low APE.

2.1.1.1.3.5.3 Seismic Site Response Modeling

DOE 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. DOE provided its models of ground motions used to develop seismic inputs to preclosure design.

To address the effects of earthquakes at the site over long periods of time, DOE 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 review of DOE's information and analyses within these four topical areas follows.

2.1.1.1.3.5.3.1 Site-Response Modeling Methodology

Overall Approach to Site-Response Modeling

In SAR Section 1.1.5.2.5.2, DOE 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 understanding and 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 design of earthquake-resistant facilities). DOE used the site-specific ground motion curves that are consistent with the conditioned PSHA ground motion hazard curves.

DOE used 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 APE). There are five approaches (1, 2A, 2B, 3, and 4) in increasing order of accuracy. Approach 4 requires site-specific soil attenuation relations, which are usually not available because of lack of observational data. DOE adopted Approach 3 for preclosure site-response analyses. 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 the model uncertainty in the site-response inputs (SAR Section 1.1.5.2.5.3).

NRC Staff Evaluation: The NRC staff reviewed DOE's overall approach to site-response modeling using the guidance of NUREG/CR–6728 and the YMRP. DOE reasonably chose Approach 3 from NUREG/CR–6728 because it is the most accurate method available for the GROA and is recommended by NUREG/CR–6728. The NRC staff notes that the two frequency ranges (1–2 and 5–10 Hz) used in the calculations of input control motions are reasonable because they conform to NRC guidance provided in NRC Regulatory Guide 1.165, Appendix C (NRC, 1997ab).

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

DOE relied on an RVT-based point-source equivalent-linear site response model to perform the site response calculation in the adopted Approach 3 discussed previously. 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 DOE's ground motion report, BSC Section 6.1.1 (2004aj). The point-source model assumes the source is small enough and can be approximated as a point. The important aspects of this model, which DOE validated as described next, 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). The NRC staff review focused on the applicability and accuracy of the model to develop earthquake ground motion input for PCSA.

Model Validation

DOE conducted a series of validation studies (BSC, 2004aj, 2008bl) to justify the applicability of the RVT-based point-source one-dimensional equivalent-linear model, which is much simplified when compared with a multidimensional, finite-source, nonlinear model. The major issues related to the justification follow.

Point Source vs. Finite Source

As part of the model validation of a point source in the site-response modeling, DOE showed that the point-source models produce ground motion response spectra (5 percent damped pseudo-absolute response spectra) which are in reasonable agreement with observed data, as outlined in BSC Section 7 (2008bl). DOE also included the evaluation of modeling variability and model bias. DOE's criterion for "reasonable agreement" was that the standard error of the residuals (the difference of the logarithms of the observed and predicted response spectra) be 0.5 or smaller. This number was adopted from the observed ground motion variation in many attenuation models (Abrahamson and Shedlock, 1997aa).

NRC Staff Evaluation: NRC staff reviewed DOE's justification for using a point-source model. The NRC staff notes that (i) the DOE approach used a published defensible "reasonable agreement" criterion and (ii) the standard error of the residuals is a reasonable criterion for strong ground motion modeling because it is commonly used for strong motion attenuation modeling (Abrahamson and Shedlock, 1997aa). The results of the point-source ground motion prediction are reasonable because DOE demonstrated that DOE point-source models produced ground motion response spectra that are in agreement with observed data based on the previous "reasonable agreement" criterion.

Equivalent Linear vs. Nonlinear

DOE used an equivalent-linear site-response model, rather than a nonlinear model. Laboratory testing and field observations have shown that soils and rocks exhibit nonlinear behavior under large applied shear loading, in which the shear modulus decreases with increasing strain accompanied with an increase in material damping. The equivalent-linear approach models this nonlinear behavior assuming the nonlinear soil response can be approximated with a linear relation over a limited range of the model variables. In this approach, the variations of shear modulus and damping ratio with shear strain are prescribed through modulus reduction and damping curves.

In the validation study for the one-dimensional equivalent-linear model, described in EPRI Appendix 6.B (1993ab), the appropriateness of one-dimensional site response analysis was evaluated and the RVT equivalent-linear modeling approach and the fully nonlinear approaches were compared. The study focused on the differences between approaches and how the equivalent linear and nonlinear models match the observed data. EPRI's validation showed that the equivalent-linear approach has a tendency to overpredict PGA for large events. Other researchers, such as Assimaki, et al. (2008aa), have confirmed this tendency.

NRC Staff Evaluation: DOE's equivalent-linear model is a reasonable one to use because (i) it has similar accuracy in predicting observations when compared with the nonlinear model and (ii) the equivalent-linear approach tends to overpredict the observed ground motions, which is conservative for safety purposes.

One-Dimensional vs. Two- or Three-Dimensional Modeling

DOE used a one-dimensional site response model to account for two- and three-dimensional effects. Therefore, two- and three-dimensional effects, such as topographic amplification and lateral variability in dynamic material properties, are not explicitly addressed in the site response model. DOE stated that two- and three-dimensional modeling are not necessary, because validation studies have shown that simple one-dimensional models accommodate the significant and stable features of the site that dominate the site response (Silva, et al., 1996aa).

To improve the one-dimensional model, DOE randomized soil properties over the site as a way to accommodate lateral variations in soil properties. Secondly, DOE developed several one-dimensional profiles with multiple alluvium thicknesses. For example, DOE used 60 randomized velocity profiles and 4 alluvium thickness values {9, 21, 30, and 60 m [30, 70, 100, and 200 ft]} to model the area northeast of the fault splay in the surface GROA. The resulting hazard curves derived from all the various combinations of parameters were then averaged with relative weights to reflect site parametric epistemic uncertainty, as described in BSC Section 6.4.5 (2008bl).

DOE noted in BSC Section 7.4 (2004aj) that although the two-dimensional effects are not explicitly included in the one-dimensional model, the two- and three-dimensional effects are implicitly included in the modeling by using control motions determined from PSHA. When the ground motion experts developed the attenuation relationships, as identified in CRWMS M&O Section 6 (1998aa), U.S. Geological Survey's strong motion database was used. This database includes records from sites (one-third rock sites and two-thirds soil sites) that have been subjected to two- and three-dimensional ground motion effects.

NRC Staff Evaluation: In response to an NRC staff RAI, DOE further supported the validity of a one-dimensional model for the Yucca Mountain region, by describing that the velocity and the impedance contrasts are generally small laterally (DOE, 2009as). Staff considers small lateral contrast of velocity and impedance to be a condition that favors use of a one-dimensional model because it means that the lateral effects will not significantly amplify or deamplify the ground motions.

NRC staff notes that the one-dimensional equivalent-linear model is reasonable. Staff recognizes that the Gilroy #2 validation shown in BSC Figure 7 (2004aj) provides assurance that two-dimensional and three-dimensional effects would not be significant at Yucca Mountain and that they can be accommodated by DOE's approach of parameter variability and randomization of velocity profiles with multiple alluvial thicknesses.

Model Validity at High Shear-Strain Condition

A crucial test of validity of the equivalent-linear model is at high shear-strain conditions. The RVT equivalent-linear site response model was tested against observed data from the Kobe-Port Island site strong-motion recording that experienced high shear strains (about 1 percent, strong enough to induce soil failure) in the upper 9 to 21 m [30 to 70 ft] of soil. The magnitude of this 1995 Japanese earthquake was 6.9 with PGA of 0.3 g. RASCALS and SHAKE codes were used for equivalent-linear modeling and TESS and SUMDES for nonlinear modeling, as described in BSC Section 7.3.5 (2004aj). Both models, equivalent linear and nonlinear, overpredicted the observed PGA. The equivalent-linear model overpredicted observations by about 20 percent. The shear strains (1 percent) for the shallow

layers at Kobe-Port Island exceeded those shear strains (0.7 percent) achieved in the Yucca Mountain evaluation (BSC Section 7.3.5.4.2 (2004aj).

NRC Staff Evaluation: The NRC staff notes that the general agreement of response spectra between observed and predicted values by the equivalent-linear and nonlinear codes shows the RASCALS code reasonably models the strain conditions, including high shear strain. The

RASCALS code, which DOE used for its site-response modeling, is reasonable for use at low and high shear-strain conditions.

Comparison of Combined Source, Path, and Site-Response Modeling to Observed Data

DOE performed combined model validation to confirm that the model is appropriate for all the previously discussed approximations taken together (Silva, et al., 1996aa). The RVT point-source modeling combined with the equivalent-linear site-response modeling was used to calculate the response spectra for 16 earthquakes at 502 sites. These response spectra were compared to the actual recordings of ground motion at these sites. These comparisons show that the model bias over all the sites is slightly positive for frequencies greater than 10 Hz and is near zero from 1 to 10 Hz.

NRC Staff Evaluation: The staff reviewed the comparison of the modeling results to the data set in Silva, et al. (1996aa) and notes that this information supports the applicability of the model for Yucca Mountain site-response calculations. The validation covered a variety of site conditions, ground motion levels, low and high strain levels, source distances, and source magnitudes that are applicable to Yucca Mountain conditions.

Model Applicability to Yucca Mountain

In the model validation described previously, most of the strong motion data were derived from California earthquake records. The only data set from the Yucca Mountain region is the 1992 magnitude 5.7 Little Skull Mountain earthquake. DOE stated in BSC Section 7.4 (2004aj) that a key factor in the applicability of the site-response model using data from other areas is the level of peak mean shear-strain reached in the site materials. DOE also noted potential two-dimensional effects associated with the dipping alluvial-rock interface, which are not explicitly included in the one-dimensional model.

DOE summarized the shear strain from EPRI's (1993aa) validation study and compared it with DOE's Yucca Mountain modeling in BSC Tables 7-29 and 7-30 (2004aj). The mean strains in the Yucca Mountain analyses for APE values of 5×10^{-4} to 10^{-6} exceed the values in EPRI's (1993aa) study, but for APE of 10^{-7} they are at the same levels as the Kobe-Port Island earthquake.

NRC Staff Evaluation: The NRC staff reviewed the site-response model, an RVT-based point-source model combined with the one-dimensional equivalent-linear site-response approximations. DOE 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 DOE 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 conducted independent calculations using velocity profiles and material properties similar to DOE's calculations using 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 amplification factor as a function of frequency has a high value ranging between 2 and 3,

which confirms DOE's results shown in BSC Figures 6.5.2-1a to 3d (2008bl). An independent calculation Gonzalez, et al. (2004aa) performed resulted in the same high amplification factor ranging from 2 to 3.

The NRC staff notes that DOE justified these simplifications and approximations through validation results, which showed the model predictions having near-zero bias and low variability compared with observations. The model parameter uncertainties and the geotechnical data, such as the material dynamic properties' uncertainties and the velocity profiles, were reasonably incorporated in the model.

DOE provided evidence from independent researchers, described in BSC Section 6.2.5 (2008bl), that strong two-dimensional or three-dimensional effects involving conditions of deep basins or outside basin sources at low frequencies (≤ 0.5 Hz) are not significant at Yucca Mountain. Geotechnical data collected at Yucca Mountain (BSC, 2002aa) did not indicate significant two-dimensional or three-dimensional velocity variations that would require more than the one-dimensional approached used by DOE.

NRC staff notes that the model results combined with the one-dimensional equivalent linear approximations showed reasonable agreement or slight overprediction with the results of complicated models (Silva, et al., 1996aa). The validation spanned the ranges of magnitudes, ground motion levels, source distances, and shear strains compatible with the seismic sources and ground motions of Yucca Mountain.

The limitation involving the equivalent-linear approximation is related to the appropriate levels of cyclic shear strain. DOE validation shows the reasonableness of the approximation at large shear strain (see Model Validity at High Shear-Strain Condition). Therefore, the one-dimensional limitation is not a significant concern at Yucca Mountain and the RVT-based model is applicable and reasonable for the Yucca Mountain site.

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

As part of site characterization activities, DOE collected geotechnical and geophysical data across the GROA and in the repository block. These data, described in SAR Section 1.1.5.3, were used to develop the necessary inputs for the seismic site-response modeling. DOE'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 velocities; (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 tests. 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, DOE described the methodology and site characterization studies used to develop this information.

To develop depth to the base of the alluvium and the V_S, V_P, and density profiles for the surface GROA, DOE collected data from 89 exploratory boreholes and surface wave survey lines across the site. DOE used several standard methods to obtain the data: conventional downhole logs, including gamma ray logs to obtain density information; downhole suspension surveys; and 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 DOE's alluvium thickness calculations that are important because the acoustic contrast between alluvium and bedrock and the overall thickness of the alluvium have the greatest influence on the site response, (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 DOE's one-dimensional RASCALS site-response models.

Alluvium Thickness Calculations

DOE identified alluvium thickness as an important factor in developing a suite of representative profiles used in its one-dimensional site-response models. DOE 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. In response to an NRC staff RAI, DOE provided all the information from the boreholes drilled during the 2006–2007 campaign in DOE (2009ap).

NRC Staff Evaluation: NRC staff reviewed DOE's information on alluvium thickness by checking the modeled alluvium thicknesses the contour map (SAR Figure 1.1-130) provided against recorded alluvium thickness from selected borehole logs, including those from the 2006–2007 campaign that DOE did not use to develop the contour map. NRC staff notes that the contour map reasonably represents the observed alluvium thickness for most of the GROA surface facility sites. Discrepancies between DOE'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 DOE's seismic velocity profiles for the surface GROA, these uncertainties in alluvium thickness are bounded by DOE'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 DOE's alluvium thickness map indicated. By developing the model with the thickest possible {in this case 61 m [200 ft]} alluvium, DOE 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, DOE has sufficient information on alluvium thickness to develop representative soil profiles for its one-dimensional site-response models.

Shear Wave Velocity

V_s and V_p 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 a greater amplification of the seismic energy as it passes through the strata. As DOE described in SAR Section 1.1.5.3.1.3.1, V_S were obtained from a range of techniques including SASW, downhole seismic velocity surveys, suspension logging surveys, sonic velocity logging, and vertical seismic profiling. Of these, DOE 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 Evaluation: NRC staff evaluated DOE's information in the SAR and supporting references, including the use and application of the SASW methodology to acquire much of the V_S data used in the site-response calculations. NRC staff also reviewed DOE's site data and information collected prior to 2005, as documented in Gonzalez, et al. (2004aa). DOE's use of the SASW methodology is reasonable 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 deems sufficient to characterize the full range of V_S for the site. Because the data collection methods and the spatial coverage were reasonable, DOE has collected sufficient information on the V_s of the rocks and alluvium at Yucca Mountain to develop reasonable site-response models.

Compression Waves

DOE described development of V_P information used for its site-response models in SAR Section 1.1.5.2.7.2 and in DOE's supplemental ground motion input document (BSC, 2008bl). According to DOE, 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.

NRC Staff Evaluation: NRC staff evaluated DOE's information in the SAR and supporting documentation including DOE (2009aq). Although DOE made direct measurements of V_P at only 15 boreholes, staff deems DOE's approach to use interpolated V_P values in combination with other geotechnical information (e.g., Poisson's ratio) to be reasonable, because of the well-established theoretical relationships between V_s, V_P, and Poisson's ratio. In addition, DOE's sensitivity analyses (DOE, 2009aq) showed that the seismic hazard at the surface GROA and in the repository are relatively insensitive to uncertainties in Poisson's ratio or V_P. NRC staff thus notes that DOE has collected sufficient information on the V_p velocity of the rocks and alluvium at Yucca Mountain to develop reasonable site-response models.

Density

The bulk density is important because it influences the site-response modeling, especially damping. DOE determined the bulk density of the rocks in the repository and alluvium beneath the surface GROA using both field and laboratory measurements as described in BSC (2002aa).

NRC Staff Evaluation: NRC staff evaluated DOE's use of gamma-gamma measurements and core samples to determine density and notes that their use was reasonable because they are in accordance with standard industry practice. NRC staff also compared DOE'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 information provided in SNL (2008af) is consistent with DOE's initial data, confirming that the values used in the SAR are reasonable.

NRC staff notes that DOE used reasonable density values in its site-response models. NRC staff also notes that the use of averaged single values is appropriate because DOE demonstrated through sensitivity studies that seismic site response is insensitive to the range of measured bulk densities at the site. DOE's information on the bulk density of the rocks and soil at Yucca Mountain is sufficient to develop site-response models.

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 DOE's response to staff's RAI (DOE, 2009aq). On the basis of the available velocity data and site geology, DOE developed 13 base case velocity profiles for the surface GROA to capture the variability and uncertainty of the site. To capture the randomness of the site response, DOE used each base case profile as the basis for stochastically generating 60 randomized profiles that remain consistent with 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, DOE enveloped the site-specific hazard results to develop a single hazard curve for the entire surface GROA.

NRC Staff Evaluation: NRC staff evaluated DOE's information in DOE (2009aq) by performing independent calculations of the one-dimensional linear equivalent site response modeling. These confirmatory calculations focused on 26 borehole-specific lithologic profiles throughout the GROA using the SHAKE2000 code. Mean transfer functions based on the individual profiles for each of the additional 26 boreholes are bounded by DOE's site response model. These results are also consistent with the NRC staff's earlier evaluation of DOE'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 DOE'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 DOE's distribution. Because of these results, the NRC staff notes DOE's conclusion (DOE, 2009aq) that the site hazard curves are conservative is reasonable, because they are based on an envelope of the individual site-specific hazard curves. Therefore, DOE developed reasonable information and an appropriate approach to develop velocity profiles for seismic site-response models that are representative of site conditions.

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, DOE developed four separate velocity profiles to represent a central "stiff" zone and three relatively "softer" zones. Akin to the methodology for the surface GROA, DOE developed a suite of site-response models that were combined to produce representative hazard curves for the repository block.

NRC Staff Evaluation: NRC staff evaluated DOE's information in the SAR and supporting documents by performing independent calculations of the one-dimensional linear equivalent site response modeling. DOE's seismic velocity profiles for the subsurface GROA (repository block) are reasonable because the approach used to develop velocity profiles for the repository block parallels the approach DOE used for the surface GROA, which was reasonable (see previous NRC Staff Evaluation under the heading "Seismic Velocity Profiles for Surface GROA"). Thus, the information and approach DOE used are reasonable to develop velocity profiles for seismic site-response models, models that NRC staff notes are also representative of site conditions in the repository block.

2.1.1.1.3.5.3.3 Geotechnical Information Used to Develop Dynamic Material Properties

DOE provided information on the dynamic properties of the site materials 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 needed 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 site response analysis. DOE derived these values from experiments conducted over the past two decades. DOE 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

DOE 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 Section 6.4.4 (2008bl)]. Additional information needed for reduced shear modulus and damping values for each rock layer present at the site is available in BSC (2004aj) and SNL (2008af). Reduced shear modulus and damping values are necessary for each rock layer present in the site. BSC Section 6.2.4 (2004aj) 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 Evaluation: The NRC staff reviewed the information provided in the SAR and relevant documents using the guidance in the YMRP. DOE tested samples for alluvium from the surface facilities area and for tuff from the repository block in the laboratory to determine the normalized shear modulus and damping ratio curves at different shear strain levels. Samples of tuff were tested from the range of tuff strata at Yucca Mountain, including the repository horizon. As a result, NRC staff notes that DOE reasonably characterized the range of dynamic material properties at the site. 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 DOE's quality assurance program, as outlined in BSC Section 6.4.4.2 (2008bl), results from qualified tests (SNL, 2008af) from the same area corroborate the curves developed in BSC (2008bl).

NRC staff also notes that DOE used reasonable 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. DOE used industry standard guidance provided in EPRI (1993ab), which suggested using these properties to model the behavior of the geologic units to estimate the ground motion at the surface. Results obtained for both normalized shear modulus and damping ratio reasonably represent the characteristics of both alluvium and rock at the repository area for a shear strain up to 0.1 percent. 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). Therefore, DOE's use of this "type curve" shape is reasonable to represent both tuff and alluvium response.

DOE recognized that significant uncertainty exists in both curves on the basis of the scatter of the experimental data. DOE used two sets of mean normalized shear modulus and damping ratio curves developed for both tuff and alluvium to bound the uncertainty. DOE's approach is reasonable because it fully accounts for the uncertainty. DOE extended the curves for both alluvium and rock at shear strain larger than 0.1 percent. This extension was conducted using the curve for cohesionless soil as a guide in addition to engineering judgment. The cohesionless soil curve of EPRI (1993aa) reasonably represents the data trend.

2.1.1.1.3.5.3.4 Development of Seismic Design Inputs

DOE'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. DOE provided site-specific hazard curves, design response spectra, time histories, and strain-compatible soil properties that are used to calculate the potential seismic hazards at the GROA. For preclosure, DOE used NUREG/CR–6728 to develop these ground motion inputs.

<u>The Design Ground Motions</u>

DOE developed site-specific hazard curves for various combinations of velocity profile, dynamic material property curves, and alluvium thickness (SAR Section 1.1.5.2.5.3). The hazard curves represent epistemic uncertainties in velocity and dynamic properties that are averaged using weighting factors on the basis of the likelihood of the 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 by enveloping. For the repository block, hazard results for the two velocity profiles (northeast and south of the Exile Hill Fault splay) were enveloped. This process incorporates epistemic uncertainty and aleatory variability in hazard curve development.

Design-response spectra (5 percent damped) for the surface GROA and repository block were developed from the location-specific mean uniform hazard spectra (UHS) with spectral holes smoothed out and extrapolated on the basis of the linear trend in log(SA)/log(period) at period of 10 seconds (SA is the spectral acceleration). Design spectra for damping (D) values of 0.5, 1, 2, 3, 7, 10, 15, and 20 percent were assessed for the surface GROA using the linear relations Idriss (1993aa) developed between spectral ratio and Ln(D) to adjust the 5 percent damped spectra to other damping values, as outlined in BSC Section 6.5.2.3.2 (2008bl). Design time histories that are spectrally matched to the 5 percent damped design spectra were used to develop the site-specific coefficients in the Idriss (1993aa) relation to ensure that the damped spectra were consistent with the hazard-consistent design spectra to which the time histories were matched, as identified in BSC Section 6.5.2.3.2 (2008bl).

Earthquake time histories were developed to ensure that their response spectra closely match the design spectra of given APEs (SAR Section 1.1.5.2.5.5). Seed-strong-ground-motion recordings were chosen from the time history database provided in NUREG/CR–6728. The spectral matching criteria of NUREG/CR–6728 defined the closeness of matching and ensured that no gaps in the power spectral density will occur over the significant frequency range.

Strain-compatible soil properties were determined for the surface GROA that are consistent with the design spectra and time histories of given APEs. These properties, such as V_s and V_p velocities and associated damping, were determined during the development of the site response amplification factors and used in the soil–structure interaction analyses. The strain-compatible properties have to be consistent with APE, the amplitude of ground motion, and the epistemic uncertainty and aleatory variability in the site dynamic material properties.

NRC Staff Evaluation: NRC staff evaluated the methods to develop site-specific ground motion outputs, which included (i) site-specific hazard curves, (ii) design-response spectra (5 percent damped), (iii) earthquake time histories, and (iv) strain-compatible soil properties. DOE followed the recommended fully probabilistic Approach 3 (McGuire, et al., 2001aa) to develop the site-specific hazard curves for the surface and subsurface GROA for the horizontal motions. DOE 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 ground motions these hazard curves predicted are reasonable because they were developed with inputs to the validated site-response model, incorporated epistemic and aleatory uncertainties, and incorporated the recommended approach of McGuire, et al. (2001aa). The staff notes that those final ground motion results are conservatively high at low APE ($<10^{-6}$) compared with the available worldwide strong motion data. For example, the PGA value of 4 g at APE of 10^{-8} in BSC Figure 6.5.2-34 (2008bl) is higher than any observed data.

Ground Motion Inputs for Surface GROA and Subsurface GROA

DOE provided ground motion inputs developed for the surface GROA and repository block in SAR Section 1.1.5.2.6. For the surface GROA, 52 combinations of site properties were evaluated in the site-response modeling (SAR Section 1.1.5.2.6.1). These combinations were from two base case velocity profiles (south and northeast of the Exile Hill Fault splay), two base case sets of dynamic material property curves for tuff and alluvium separately, four values of alluvium thickness northeast of the fault splay, and three values of alluvium thickness south of the fault splay. Each combination incorporated aleatory variability by averaging the amplification factors from 60 randomized velocity profiles and dynamic material property curves.

The seven combinations of site-specific alluvium and tuff hazard curves were combined into two sets, the northeast and south fault splay sets. The four and three combinations of hazard curves for four and three alluvium thicknesses were enveloped separately for south and northeast of the fault splay. These two sets of hazard curves were enveloped again to produce mean horizontal and vertical hazard curves for the entire surface GROA (BSC, 2008bl). The final mean horizontal and vertical hazard curves for PGA; SA at 0.05, 0.1, 0.2, 0.5, 1.0, 2.0, and 3.3 seconds; and PGV were provided in BSC Figures 6.5.2-34 to 6.5.2-42 (2008bl) for the surface facility area and BSC Figures 6.5.3-9 to 6.5.3-16 (2008bl) for the repository block.

The data for these plots are in DTN: MO0801HCUHSSFA.001, as discussed in BSC Section 6.5.2.2 (2008bl).

The horizontal and vertical UHS for the surface GROA, described in BSC Figures 6.5.2-43 to 6.5.2-49 (2008bl) for 5 percent damping and APEs of 10^{-3}, 5×10^{-4}, 10^{-4}, 10^{-5}, 2×10^{-6}, and 10^{-7}, were calculated from the final hazard curves. The UHS for the repository block at APEs of 10^{-3}, $5{\times}10^{-4}$, 10^{-4}, 10^{-5}, 10^{-6}, 10^{-7}, and 10^{-8} are shown in BSC Figures 6.5.3-19 to 6.5.3-25 (2008bl). The design spectra were the extrapolation of UHS to a period of 10 seconds following the linear trend in log (SA)/log (period) between 2 and 3.3 seconds. For design spectra with damping other than 5 percent, the spectral ratio method of Idriss (1993aa) was used. The final surface-area facility design spectra with multiple dampings were shown in BSC Figures 6.5.2-60 to 6.5.2-65 (2008bl) and summarized in SAR Figures 1.1-90 and 1.1-91 for 5 percent damping only. The design spectra for repository block were shown in BSC Figures 6.5.3-26 to 6.5.3-28 (2008bl).

The newly developed design spectra (BSC, 2008bl) differ from the 2004 version (BSC, 2004aj) because (i) the 2004 version used velocity data from an area southwest of the Exile Hill Fault splay; (ii) the 2004 version used Approach 2B, not Approach 3; and (iii) the site-response control motion in the 2004 version was not conditioned (SAR Section 1.1.5.2.6.1).

Seed-strong-motion recordings were chosen from the McGuire, et al. (2001aa) time history database to provide the phase and duration characteristics in the design spectral matching process for producing the time histories. The design spectra at APE of 10^{-3}, 5×10^{-4}, and 10^{-4}, and identified in BSC Figures 6.5.2-60 to 6.5.2-65 (2008bl) for the surface facility area, were matched with the five sets of three-component time histories selected for the GROA, as outlined in BSC Figures 6.5.2-86 to 6.5.2-95 (2008bl). The spectrally matched acceleration, velocity, and displacement time histories and the match details were shown in BSC Figures 6.5.2-96 to 6.5.2-230 (2008bl) for the surface-area facility. The design spectra, described in BSC Figures 6.5.3-26 to 6.5.3-28 (2008bl) for the repository block at APEs of 10^{-3}, 5×10^{-4}, and 10^{-4} were matched with one set of time histories. The matched time histories and match details were shown in BSC Figures 6.5.3-49 to 6.5.3-77 (2008bl). The repository block time histories for postclosure analyses were developed differently for AFEs of 10^{-5}, 10^{-6}, and 10^{-7} (SAR Section 1.1.5.2.6.2), where 17 sets of time histories were developed. One horizontal component of each seed time history was scaled according to the PGV from site-response modeling; the other two components were scaled to maintain the intercomponent variability of the seed-time-history (SAR Section 1.1.5.2.6.2).

The final results for strain-compatible material properties (V_S, S-wave damping, V_P, and P-wave damping) were shown in BSC Figures 6.5.2-231 to 6.5.2-258 (2008bl) for APE of 10^{-3}, BSC Figures 6.5.2-259 to 6.5.2-286 (2008bl) for APE of 5×10^{-4}, and BSC Figures 6.5.2-287 to 6.5.2-314 (2008bl) for APE of 10^{-4}.

NRC Staff Evaluation: The NRC staff notes that the processes and procedures DOE used to develop site-specific hazard curves, UHS and site design response-spectra, time histories, and strain-compatible soil properties are reasonable for the PCSA and the GROA design for the following reasons. The DOE development of the hazard curves for the surface-area facility and repository block using an averaging process to account for the data (velocity profiles and dynamic material properties) and site-response model uncertainties and an enveloping process to accommodate the alluvium thickness change (spatial variability) is reasonable based on standard industry practice. Also DOE followed the recommended routine procedures in engineering seismology (McGuire, et al., 2001aa) in producing UHS, design spectra, and spectrally matched time histories for different APEs and dampings as the preclosure ground motion inputs. DOE developed the strain-compatible soil properties using standard methods used in the industry. Although the newly developed design spectra (BSC, 2008bl) differ from the 2004 version (BSC, 2004aj), both versions are reasonable because both have been developed using industry standard methods and are conservative.

2.1.1.1.3.5.4 Site Geotechnical Conditions and Stability of Subsurface Materials

DOE described, in SAR Section 1.1.5.3, the geomechanical properties and conditions of the repository site for design and PCSA of the GROA. DOE described the types and geometrical configuration of subsurface materials at the site and mechanical properties of the materials that are needed for evaluating the stability of subsurface materials and potential effects on the performance of proposed SSCs of the GROA for use in PCSA. On the basis of DOE's information in SAR Section 1.1.5.3, 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 facility site, (ii) stability of subsurface materials at the surface facility site, and (iii) site geotechnical conditions at the subsurface GROA.

Types and Geometrical Configuration of Subsurface Materials at the Surface GROA

DOE provided information pertaining to the nature and geometrical configuration of subsurface materials at the surface facility site. DOE 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. DOE concluded that the surface facility site is underlain by Quaternary alluvium and colluvium of up to 61 m [200 ft] thick, which overlies a sequence of volcanic tuff, as shown in SNL Table 6.2-1 (2008af). DOE stated that the tuff is much stronger than the alluvium and poses no constraints on site development because tuff deformation is much smaller than the alluvium deformation.

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. DOE determined the variation of the alluvium thickness through geologic analysis of site-specific borehole data as described in SNL (2008af). DOE in its soils engineering report (BSC, 2007bq) 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. DOE did not provide information regarding the special relationship between the cemented and noncemented alluvia. 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 Evaluation: The NRC staff reviewed DOE's information pertaining to the types and geometrical configuration of subsurface materials at the surface facility site using guidance in the YMRP. DOE's information from test pits and cutting samples from boreholes regarding the types and geometrical configuration of subsurface materials at the surface facility is reasonable for use in performing engineering evaluations of the performance of surface facility structures because the information was obtained through investigations conducted at the site using a combination of surface and subsurface geologic and geophysical techniques that are commonly used in the industry for geological and geophysical investigation and are consistent with NRC

Regulatory Guide 1.132, p. 1.132-8 (NRC, 2003ag). DOE's information provides sufficient data on site geology to permit evaluation of the PCSA and the GROA design.

Stability of Subsurface Materials at the Surface GROA

Shear Strength of the Alluvium

DOE provided information on the stability of the subsurface materials at the surface GROA in SAR Sections 1.1.5.3.2.3 and 1.1.5.3.2.4. DOE'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, DOE provided information pertaining to the bearing capacity of the alluvium that is the allowable foundation bearing pressure on alluvium, which DOE calculated using estimates of the shear strength of the alluvium.

To estimate the shear strength of the alluvium, DOE estimated the relative density of the alluvium using a combination of *in-situ* and laboratory testing and used the relative density in empirical relationships to estimate shear strength as described in its soil engineering report, BSC Appendix I (2002ab). First, DOE determined bulk density of the alluvium through geophysical measurements in seven boreholes (BSC 2002aa,ab) and water-replacement and sand-cone density tests in test pits. Second, DOE used samples from the sand-cone and water-replacement density test locations in the test pits to determine maximum and minimum density indices in the laboratory. DOE used these density indices to calculate relative density of the alluvium. DOE calculated the relative density of the alluvium using this procedure in seven test pits shown in SAR Figure 1.1-129 at depths below the ground surface in the range of 1.2 to 5.8 m [4 to 19 ft] covering a small part of the footprint of the surface facility structures, as shown in BSC Attachment I (2002ab) and BSC Figure 6-11 (2007bq).

DOE used the procedure described in the previous paragraph to obtain 22 measurements of relative density of the alluvium in seven test pits, which indicate a minimum relative density of 25 at a depth of 2.4 m [8 ft] and a maximum of 120 at a depth of 5.8 m [19 ft], as outlined in BSC Attachment I (2002ab). On the basis of this information, the degree of compaction of the alluvium could range from "loose" to "very dense," which encompasses four of the five categories Lambe and Whitman, p. 31 (1969aa) used to describe the potential range of compaction of granular materials. DOE concluded that there is no correlation between relative density of the alluvium and depth at the site, on the basis of plotting the 22 measurements against depth. DOE determined a mean relative density of 68 and standard deviation of 21 for the alluvium by averaging the measurements. As described in DOE's soils engineering report, BSC Appendix I (2002ab), DOE used the relative density data from test pits with empirical relationships between relative density and shear strength to define the shear strength of the alluvium. For the relative densities ranging from 25 to 120, the angle of internal friction estimated from the empirical relationship ranged from 33 to 52°. DOE selected an average internal friction angle of 39° with zero cohesion (SAR Section 1.1.5.3.2.4) to represent the shear strength of the alluvium at the site.

DOE's information indicated the minimum footprint width dimension for proposed surface facility structures varies from approximately 34.7 m [114 ft] for an aging pad (SAR Section 1.2.7) to approximately 119.5 m [392 ft] for a Canister Receipt and Closure Facility (SAR Section 1.2.4). On the basis of literature information [e.g., Bowles, p. 93 (1996aa)], the zone of influence of foundation loadings can extend to a depth of 1.5B to 2B below the foundation base, where B is the minimum dimension of the foundation footprint. Therefore, the footprint dimensions of surface facility foundations implies a zone of influence of foundation loading up to 70 m [230 ft] below an aging pad or 240 m [787 ft] below a Canister Receipt and Closure Facility. As DOE provided, the thickness of the alluvium under the surface facility structures varies from 9.1 to 61 m [30 to 200 ft.]. Therefore, the potential zone of influence of the foundation loading could encompass the entire alluvium thickness, indicating the shear strength of the alluvium needs to be defined for the entire alluvium depth for assessing the stability of subsurface materials at the surface facility site. Furthermore, DOE indicated the alluvium is variably cemented, but did not provide information regarding spatial relationships between the cemented and noncemented alluvium.

In response to an NRC staff RAI, DOE stated that (i) the derived soil classifications and gradation for the alluvium from borehole cuttings do not change with depth or laterally across the site, as outlined in DOE Enclosure 1 (2009aq), and (ii) the effects of cementation were not accounted for in the alluvium shear strength in the SAR, but were implicitly included in the field measurements of V_s that DOE used to assess potential settlement, as described in DOE Enclosure 1 (2009aq) and BSC (2007bq). In response to the NRC staff questions regarding the basis for DOE's assumption that uniformity of the derived soil classification and gradation of the alluvium would imply uniformity of the alluvium shear strength, DOE stated in DOE Enclosure 1 (2009eh) that (i) uniformity of soil classification does not imply uniform shear strength and (ii) although the alluvium is laterally discontinuous and layered when taken as a whole at the Midway Valley scale, the material can be interpreted as homogeneous when taken at the scale of ITS structures at the surface facilities. However, DOE did not provide any quantitative information (measurements) to substantiate the assumption that the alluvium can be treated as homogeneous at the scale of the surface facilities structures though it is laterally discontinuous at the Midway Valley scale.

DOE's estimation to determine shear strength parameters using relative density data as described in its soils engineering report, BSC Table I-17 (2002ab), indicated the value of the internal friction angle (shear strength) for the alluvium could be in the range of 33–52°, but DOE described the shear strength of the alluvium using a mean shear strength value of 39°. DOE stated in DOE Enclosure 1 (2009bg) the use of this value is appropriate and conservative because, at the scale of the ITS mat foundations, the geotechnical behavior of the alluvium has average characteristics over the very large volume of material. Furthermore, the subsequent 9 measurements of relative density of the alluvium indicate a minimum relative density of 60 at a depth of 5.8 m [19 ft] and a maximum of 102 at depths of 1.2 and 3.6 m [4 and 12 ft] (SAR Section 1.1.5.3.2.3).

NRC Staff Evaluation: The NRC staff reviewed the information on the stability of the subsurface materials at the surface GROA using guidance in the YMRP. The NRC staff notes that DOE used the *in-situ* and laboratory tests and standard procedures, which are generally reasonable and used in the geotechnical engineering profession, to determine geotechnical parameters of the alluvium. Therefore, in this regard, DOE's information provides reasonable data on site geology to permit evaluation of the PCSA and the GROA design.

However, the limited range of depth of DOE's relative density data {1.2–5.8 m [4–19 ft] below the ground surface} causes an undefined uncertainty in the shear strength value of 39° for alluvium, considering the thickness of the alluvium {9.1–61 m [30–200 ft]} within the zone of influence of potential foundation loadings. Also, the lateral coverage of relative density measurements, taken from a limited number of test pits, does not encompass the footprints of the proposed surface facility structures. DOE should perform measurements of the alluvium that can be reliably correlated with relative density at test locations, distributed systematically over the surface facility site and through the zone of influence of the foundation loadings, to confirm the range of internal friction angles of the alluvium and associated uncertainty in defining the shear strength of the alluvium. Furthermore, DOE should conduct investigation to determine the spatial relationships between cemented and noncemented alluvium to assess the likelihood of the foundation of an ITS structure being partially constructed over mostly cemented alluvium and partially over mostly noncemented alluvium.

Therefore, DOE should conduct confirmatory measurements and associated evaluation for the shear strength of the alluvium within the zone of influence of foundation loadings of the surface GROA facilities, including uncertainty reflected in the variation of internal friction angles from 33 to 52°.

Compressibility of Low-Density Tuff

Because the thickness of the alluvium below proposed surface facility structures is generally smaller than the zone of influence of potential foundation loadings (i.e., 1.5B to 2B as discussed previously), the mechanical behavior (e.g., compressibility) of volcanic tuff that underlies the alluvium could be important for assessing the performance of surface facility structures. SAR Section 1.1.5.3.2.6.2.1.1 indicates the presence of low-density bedded tuff that potentially could affect the engineering performance of the structures if the low-density tuff is more compressible than the overlying alluvium.

In response to NRC staff RAIs, DOE stated in DOE Enclosure 1 (2009aq) that laboratory and field test results show the low-density tuffs and other tuffs directly underlying the alluvium at the surface facility site have V_s equal to or greater than the V_s of the alluvium, which indicate the low-density tuff is less compressible than the alluvium.

NRC Staff Evaluation: The NRC staff notes that the compressibility of the low-density tuff underlying the alluvium at the surface facilities will not have any significant effect on the stability of subsurface materials underlying the proposed surface facility structures, because DOE's laboratory and field testing demonstrate that the low-density tuff is less compressible than the alluvium. The NRC staff also notes that the test data support DOE's evaluation of the compressibility of the low-density tuff and it has been reasonably characterized because DOE used standard methods to conduct laboratory and field testing to determine the compressibility of low-density tuff.

Allowable Bearing Pressure and Settlement of Foundations of Surface GROA Facilities

DOE provided information pertaining to allowable bearing pressure for the foundations of the surface facility structures in its soils engineering reports in BSC Appendix B (2007bq). DOE 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 Figure B6–2 (2007bq); (ii) square and strip footings with settlement limited to 12.7 mm [0.5 in], as shown in BSC Figures B6–7, B6–8, and B7–2 (2007bq); and (iii) square and strip footings with settlement limited to 25.4 mm [1.0 in], as outlined in BSC Figures B6–13 and B7–1 (2007bq). DOE'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). This condition does not consider the distributed localized shear failure of the subsurface materials, as shown in Bowles Figure 4-3 (1996aa). Conditions (ii) and (iii) determined potential limits on foundation loading without causing excessive settlement due to distributed 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, gave values of allowable bearing pressure that decreased as the footing width increased but approached a minimum value for large footing widths. Condition (ii) gives a value of approximately 239 kPa [5 ksf] for a footing width up to 9.1 m [30 ft], and condition (iii) gives approximately 479 kPa [10 ksf] for a footing width up to 9.1 m [30 ft].

DOE stated in BSC Table 2-2 (2007dg) that for mat foundations, the recommended maximum allowable soil bearing pressure is 479 kPa [10 ksf] for normal loading conditions and 2,394 kPa [50 ksf] for extreme loads such as seismic load conditions. In response to an NRC staff RAI (DOE, 2009ei), DOE provided (i) Table 1, presenting new results of total and differential settlements for the static loading conditions (normal load) for all 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 Figure B6-2 (2007dg) 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 Figure 2 (2009ei), limiting the settlement to 50 mm [2 in] for large mat foundations, DOE recommended an allowable bearing pressure of 479 kPa [10 ksf] for normal loading conditions. DOE provided the rationale for a settlement limit not exceeding 50 mm [2 in] for large mat foundations of ITS structures. For extreme loading, DOE recommended an allowable bearing capacity of 2,394 kPa [50 ksf] from DOE Figure 1 (2009ei) on the basis of rotational shear failure of the foundation material criterion. This extreme loading condition allowable bearing pressure is about five times the allowable bearing pressure for the normal loading condition. DOE Table 1 (2009ei) lists the average foundation pressure for various 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].

NRC Staff Evaluation: The NRC staff reviewed DOE’s information on allowable bearing pressure for surface GROA facility foundations using guidance in the YMRP. DOE’s analysis methods used for calculating allowable bearing pressure are reasonable because they are commonly used in the geotechnical engineering profession For footings up to 9.1 m [30 ft] wide, the allowable bearing pressure in BSC Figures B7-1 and B7-2 (2007bq) was controlled by settlement criterion of 25.4 and 12.7 mm [1.0 and 0.5-in], respectively. The bearing pressure that would result in the predetermined settlement criterion was calculated from empirical relationships in literature, using relative densities measured at the site, as described in Terzaghi, et al., Section 50.2 (1996aa).

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 criterion does not control the design of large mat foundations. DOE calculated short-term, or elastic, settlement of mat foundation for (i) a range of uniform loads of 144, 239, and 335 kPa [3, 5, and 7 ksf]; (ii) 36.6-m [120-ft]-thick alluvium layer of uniform thickness; and (iii) elastic modulus of soil calculated from V_s data. BSC Table B7–1 and p. B–45 (2007dg) presented calculated settlements at the center and corners of the mat foundation. The methodology used to calculate elastic settlement for normal load condition is reasonable. However, in addition to elastic settlement, settlement due to distributed localized shear in the alluvium could occur. The magnitude of the additional settlement could be controlled through limiting the allowable bearing pressure as described earlier [e.g., DOE's conditions (ii) and (iii) calculations using the criterion settlement].

In DOE's response to an NRC staff RAI, DOE Figure 2 (2009ei) presented a revised evaluation of allowable bearing pressure on the basis of limiting the settlement to 50 mm [2 in] and provided a reasonable rationale for limiting the settlement to 50 mm [2 in] for large mat foundations. The methodology used in the calculations, on the basis of empirical relations developed from case histories of performance of large mat foundations on granular soils, is reasonable to the NRC staff. Therefore, DOE's recommendation of allowable bearing pressure of 479 kPa [10 ksf] for normal loading is reasonable subject to further consideration of the uncertainty associated with the shear strength and relative density data used in the analysis, as discussed previously (under the previous heading "Shear Strength of the Alluvium"). Furthermore, DOE chose an allowable bearing pressure for the extreme loading condition on the basis of a calculation considering rotational shear failure of the foundation material that does not include a limit on foundation settlement [DOE's condition (i) calculation].

To use a value of bearing capacity that does not include a limit on foundation settlement, DOE assumed in DOE Figure 1 (2009ei) and BSC Figure B6–2 (2007bq) that the foundations would not undergo excessive settlement when subjected to the applicable loading. DOE used this chart to limit the maximum bearing pressure under the mat foundation in the extreme loads situation {e.g., seismic events to 2,394 kPa [50 ksf]} without providing a reasonable basis for not considering the potential settlement.

DOE designed the mat foundation, considering design-basis seismic loads, using an FE method where the alluvium under the mat foundation was modeled as soil spring. The analysis yielded deflection of the mat and resulting soil-spring reaction (bearing pressure) at nodal points of the FE mesh of the mat foundation under the seismic loading condition. The maximum bearing pressure at any nodal point of the mat FE model was limited to the allowable bearing pressure. The FE modeling analysis used to design mat foundation represents the alluvium as a spring (soil spring using shear modulus calculated from shear wave velocity). Representing the alluvium as a spring assumes that the alluvium will respond linearly and will undergo small deformation under the imposed loading. TER Section 2.1.1.7.3.1.1 presents NRC staff's evaluation of DOE's mat foundation design by FE method. Note that the high bearing pressure under the seismic load condition is only at a few locations (FE nodes) and less than 2,394 kPa [50 ksf]. In the final design of mat foundations, DOE should address consideration of settlement in recommending the maximum allowable bearing pressure under the seismic loading condition.

In designing the mat foundation for the normal loading condition, DOE's recommended maximum allowable bearing pressure is 479 kPa [10 ksf]. In designing the mat foundation for a design basis seismic event (extreme loading condition), DOE's recommended maximum allowable bearing pressure is 2,394 kPa [50 ksf]. In the NRC staff's judgment, the recommended pressures for the alluvium fall within a reasonable range of maximum allowable bearing pressure. However, as part of the detailed process, DOE should confirm that the foundation material's settlement response would be linear in the range of stresses considered in the design and the settlement the mat foundation would experience during a design basis seismic event would not affect the mat foundation's safety function.

Stability of Slopes

The proposed layout of aging pads in SAR Figure 1.1-129 indicated excavations for the aging pads could expose the alluvium in a cut slope up to approximately 10 m [33 ft] high.

Also, transportation routes that link the aging pads to other surface facility structures (SAR Figure 1.2.7-2) could involve cut-and-fill slopes. In response to an NRC staff RAI on an assessment of the stability of the slopes under seismic conditions, DOE stated it would evaluate the stability of the slopes as part of the detailed design [DOE Enclosure 2 (2009aq) and DOE Enclosure 2 (2009eh)]. DOE 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 be steeper than 2:1 (ratio of horizontal to vertical dimensions). DOE provided a stability analysis in DOE Enclosure 1 (2009ej) that indicated 2:1 slopes in the alluvium will be stable under DBGM-2 seismic loading. The analysis considered shear strength on the basis of a friction angle of 39°, but did not consider the effects of uncertainties in shear strength of the alluvium [e.g., a friction angle in the range of 33° through 52°, as indicated in the soils engineering report, BSC Table I-17 (2002ab)].

NRC Staff Evaluation: The NRC staff reviewed DOE's information on stability of the cut-and-fill slopes using guidance in the YMRP. For the purpose of the PCSA, DOE's information is reasonable to assess the engineering design and performance of the slopes at the surface facilities because (i) cut-and-fill slopes will not be steeper than 2:1 (ratio of horizontal to vertical dimensions) and (ii) the friction angle used in the stability analysis is within the range of values indicated in the soils engineering report.

DOE stated it would provide an assessment of the stability of slopes that considers the effects of uncertainties regarding the shear strength of alluvium as part of the detailed design process (DOE, 2009aq,eh,ej). As part of the detailed design process, DOE should confirm the stability of slopes under applicable seismic loading conditions using an approach that accounts for uncertainties in the shear strength of alluvium [such as friction angle of 33–52° as indicated in the soils engineering report, BSC Table I–17 (2002ab)].

Geotechnical Conditions at the Subsurface GROA

DOE provided information pertaining to geotechnical conditions at the subsurface GROA in SAR Sections 1.1.5.3 and 2.3.4.4.2.1. DOE detailed the type and configuration of repository host materials and parameters describing properties of the materials needed for an engineering analysis.

RHH Materials

In SAR Section 1.1.5.3.1.1, DOE stated that the repository emplacement areas will be located approximately 300–400 m [984–1,312 ft] below the ground surface within several subunits of the crystal-poor member of the Topopah Spring Tuff (SAR Figure 2.3.4-21). DOE stated 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 (Tptpll) (SAR Figure 2.3.4-22). DOE stated 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.

DOE 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 DOE, 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 Tptpll unit is heavily fractured with small-scale {lengths smaller than 1-m [3.3-ft]} fractures. DOE stated 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 Evaluation: The NRC staff reviewed DOE's information on RHH materials using guidance in the YMRP. DOE's description of the type of materials that constitute the repository horizon was based on geologic studies performed at the site using standard techniques. On the basis of the geologic studies, DOE described the expected locations of stratigraphic contacts and estimated the percentage occurrence of each rock type within the repository horizon. DOE's information provides sufficient data on site geology to permit evaluation of the PCSA and the GROA design.

Mechanical Properties of Nonlithophysal Rocks

To characterize the mechanical properties of nonlithophysal rock, DOE (i) determined geometric and surface properties (i.e., dip, dip direction, trace length, spacing, end terminations, roughness, filling, and offset) of fractures through detailed-line surveys and full-periphery geologic mapping of the ESF and ECRB cross drift (SAR Section 1.1.5.3.1.2.1); (ii) tested intact rock specimens using unconfined and triaxial compression tests (SAR Section 1.1.5.3.1.2.2.1) and tested fracture surfaces using direct shear and rotary shear tests (SAR Section 1.1.5.3.1.2.1); and (iii) used well-established empirical rock mass classification systems (SAR Section 1.1.5.3.1.2.1) to determine rock mass quality designations and calculate values of rock-mass strength and stiffness parameters, as described in SAR Table 1.1-82 and BSC Table 6-76 (2007be). SAR Table 1.1-82 summarized the rock-mass strength and stiffness of the nonlithophysal rock units.

The summary of strength and stiffness of nonlithophysal rock mass provided in SAR Table 1.1-82 was based on fracture data from the ESF, and DOE used an established procedure, as shown in BSC Section 6.4.4.2 (2007be), to determine rock mass strength and stiffness parameters from the fracture data. DOE's information provided an engineering characterization of the nonlithophysal rock units that DOE encountered in the ESF tunnel. DOE assumed that the nonlithophysal units encountered in the ESF tunnel are mechanically representative of nonlithophysal rock within the repository block.

NRC Staff Evaluation: NRC staff reviewed the information on the mechanical properties of the nonlithophysal rocks using guidance in the YMRP. DOE provided mechanical properties of nonlithophysal rock on the basis of site-specific data and analyses of the data using techniques that are well established in geotechnical engineering practice. In this regard, DOE's information provides sufficient data on site geology to permit evaluation of the PCSA and the GROA design. As part of the performance confirmation program, DOE should confirm that the mechanical properties of the nonlithophysal rock encountered in the ESF are representative of such properties in the repository block.

Mechanical Properties of Lithophysal Rocks

To characterize the mechanical properties of lithophysal rock, DOE (i) tested 29 large-diameter specimens from the lithophysal rock units and used the results to group the rock mass into five categories on the basis of the values of strength and elastic stiffness, as identified in SAR Section 2.3.4.4.2.3.3.4 and BSC Appendix E (2004al); (ii) used lithophysal porosity data from the ECRB cross drift to define ranges of lithophysal porosity for the 5 rock mass categories

(SAR Figure 2.3.4-29); and (iii) used numerical model calculations that simulate laboratory compression of lithophysal rock specimens with different porosities to examine relationships between strength and elastic stiffness of lithophysal rock, as outlined in SAR Figure 2.3.4-30 and BSC Section 6.4.4.4.2 (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.

DOE determined the strength and stiffness parameters in SAR Table 2.3.4-16 and SAR Figure 2.3.4-30 from laboratory testing of large-diameter lithophysal rock specimens. Six of the 29 tested specimens in BSC Table 6-69 (2007be) were from the Tptpll subunit, and the other 23 were from the Upper Lithophysal subunit. In addition, DOE performed numerically simulated testing of lithophysal rock specimens to augment the laboratory test data (SAR Figure 2.1.4-30).

NRC Staff Evaluation: The NRC staff reviewed DOE's information regarding the mechanical properties of lithophysal rocks using guidance in the YMRP. DOE 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 geotechnical engineering practice. The NRC staff notes that the 29 specimens DOE 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, p. 113 (1981aa). Six of the specimens had a length-to-diameter (L/D) ratio of 1.0–1.5, and 23 had an L/D ratio of 1.7–2.1. Therefore, the values of L/D ratio for the specimens are smaller than the recommended value of 2.5–3.0. The NRC staff reviewed a relationship suggested in Jaeger and Cook, p. 144 (1979aa) 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 DOE's numerically simulated testing indicated uncertainties in the strength and stiffness data are encompassed by the upper and lower bounds that DOE defined in SAR Figure 2.3.4-30.

Furthermore, NRC staff confirmatory calculations in Ofoegbu, et al., p. 3-5 (2007aa) indicate that the upper and lower bounds DOE defined agree well with bounds based on 95 percent confidence limits. DOE indicated an additional lower bound that limits the value of unconfined compressive strength to a minimum of 10 mPa [1.45 ksi] as shown in SAR Figure 2.3.4-30. In BSC Section 6.4.4.4.2.6 (2007be), DOE stated this is suggested by the behavior of existing tunnels at the subsurface facility site. DOE's laboratory test results also indicated a minimum unconfined compressive strength of 10 mPa [1.45 ksi] as shown in SAR Figure 2.3.4-30. On the basis of its review of this figure, the NRC staff notes that DOE data also indicate a minimum Young's modulus of 5 GPa [725 ksi]. Although DOE did not incorporate lower bound data in SAR Figure 2.3.4-30, DOE's information on the characterization of mechanical properties of the lithophysal rocks in the GROA provides sufficient data on site geology to permit evaluation of the PCSA and the GROA design because the region with lower bound material properties is located in a small area.

As part of the performance confirmation program, DOE should confirm that the mechanical properties of the lithophysal rock encountered in the ESF and ECRB are representative of such properties in the repository block.

Other Geotechnical Properties at the Subsurface GROA

DOE provided thermal properties (i.e., thermal conductivity, thermal expansion coefficient, and heat capacity) for lithophysal and nonlithophysal rock on the basis of laboratory and field testing (SAR Section 1.1.5.3.1.2.3); *in-situ* stress based on two hydraulic fracturing tests (SAR Section 1.1.5.3.1.2.4); seismic velocities using downhole and surface-based geophysical testing, including tests from the ground surface and tunnel floor (SAR Section 1.1.5.3.1.3.1); and dynamic properties, such as shear modulus and damping ratio, from laboratory testing (SAR Section 1.1.5.3.2.6).

NRC Staff Evaluation: DOE used standard techniques for the laboratory and field tests and provided information to define potential uncertainties in the test results. Therefore, DOE's information characterizing the *in-situ* stress and thermal and dynamic properties of lithophysal and nonlithophysal rock provides sufficient data on site geology to permit evaluation of the PCSA and the GROA design; specifically, the performance of subsurface facility SSCs.

2.1.1.1.3.5.5 Fault Displacement Hazard Assessment

In SAR Section 1.1.5.2.4.1, DOE described the potential for displacement (movement by slipping) on faults that might affect the surface and subsurface GROA, the probability of fault displacements exceeding certain displacements, and the expert elicitation process that led to DOE's assessment. A fault that intersects the surface GROA and displaces bedrock, sediment, or soil in any direction (up, down, sideways, obliquely) could damage the foundation of surface facilities by shearing or tilting them and disrupting surface drainage and erosion-protection structures.

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, and degrade other components of the engineered barrier system. These hazards might affect health, safety, and the environment during operations.

PFDHA—Methodology

DOE conducted its PFDHA at the same time as its PSHA using the same procedures as discussed in TER Section 2.1.1.1.3.5.2. DOE assembled fault experts to estimate the likelihood (probability of occurring in any year) of particular faults in and near the GROA to exceed specific amounts of displacement {centimeters to meters [inches to feet] of slip}. In addition to assessing FDHs, DOE used the PFDHA frequency and magnitude information and estimates as input to the seismic ground motion hazard assessments its seismic experts made.

DOE used an expert elicitation process and a logic tree approach to organize the results of the expert elicitation to capture the uncertainties associated with a seismic and FDH assessment. DOE used experts from different areas of expertise closely related to seismic hazard to represent differences in experience, model selection, and analytical approach. DOE used the logic tree approach to ensure consistent and quantifiable results were obtained.

DOE used the hazard curves derived from the expert elicitation process to incorporate the variability in earthquake processes.

The process DOE followed in the PFDHA included specific characteristics and uncertainties that DOE needed to assess (i) identification of sources of fault displacement; (ii) evaluation of the location, frequency, and size of displacements; (iii) evaluation of subsidiary displacements as a function of magnitude and distance; and (iv) integration of these data into a hazard curve and associated uncertainty distribution (SAR Chapter 2, p. 2.2-66).

DOE divided its FDH into two categories: principal and distributed. Principal fault displacement occurs along a single, well-defined (obvious to a field observer) surface, which is also regarded as the primary source of seismic energy during an earthquake. Distributed fault displacement occurs on a series of surfaces (i.e., discontinuous faults) as a result of a principal-fault rupture and is regarded as of smaller scale and discontinuous in nature.

The expert elicitation teams used two different methods to generate FDH 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 data actually measured in trenches, or data from records of earthquakes correlated with the known seismogenic faults. The earthquake approach relates the frequency and magnitude of the faults' slip events to the frequency and magnitude of earthquakes on the seismic sources defined in the seismic-source models developed for the corresponding seismic hazard analysis (CRWMS M&O, 1998aa).

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 uses earthquake recurrence models from the seismic hazard analysis. For this approach, the experts assessed three probabilities, which the NRC staff evaluated: (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 displacement on other faults, primary or secondary.

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 and 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.

Because the maximum surface displacement of a fault may not coincide with the location for which the hazard curve is being generated (i.e., the demonstration point, as described next), an additional variable that randomized the rupture along the fault length was introduced. The probability of distributed faulting was determined from Basin and Range historical rupture data in which distributed faulting was mapped after the earthquake (e.g., Pezzopane and Dawson,1996aa) or through slip tendency analysis (Morris, et al., 1996aa).

NRC Staff Evaluation: The NRC staff observed the process of communication from the Probabilistic Fault Displacement expert panel to the PHSA panel and follow-up discussions at most of the elicitation meetings. NRC staff also reviewed much of the written information and most meeting summaries that emanated from the meetings and notes that the information was relevant and that formal procedures (e.g., NRC, 1996aa) were followed. On the basis of NRC staff assessment of the relevant information discussed and handed out at the elicitations and staff assessment of the experts' understanding of that information upon witnessing the elicitations and reviewing the resulting hazard curves, the range of hazard curves reflects the uncertainty the experts assessed in the PSHA calculations, as discussed in

SAR Section 2.2.2.1.5. The NRC staff also observed the geological evidence for recurrence and slip rates of many of the faults DOE investigated and considered in developing the seismic and FDH and notes the reported results are reasonable (Stamatakos, et al., 2003aa). The FDH Assessment methodology used to evaluate FDH for the preclosure period at Yucca Mountain is reasonable for the following reasons: (i) implicit in DOE's methodology is the acknowledgement that Yucca Mountain lies within a tectonically active region and is therefore potentially subject to earthquakes and fault displacement and (ii) a number of faults with the potential to create displacement hazards were identified and characterized, and hazard curves were generated for these faults.

The NRC staff notes the catalog of regional faults and the hazard curves derived for them are reasonable for use in other SAR sections. Also, on the basis of staff's understanding of DOE's data and the limitations of those data as discussed in expert elicitation meetings and NRC staff's independent analysis of slip tendency (Morris, et al., 1996aa), DOE's information provides sufficient data on site geology to permit evaluation of the PCSA and the GROA design.

<u>Probabilistic Fault Displacement Hazard Assessment (PFDHA)—Input Data and Interpretations</u>

The PFDHA integrated two data types: (i) known and/or documented faulting activity consisting of measurements of regional and local earthquakes and measurements of fault displacements within the last ~1.8 million years (Quaternary) and (ii) inferred potential faulting activity, on the basis of analysis of mapped geological faults, overall tectonic setting, and regional estimates of ongoing crustal strain. DOE 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 probabilistic FDH curves were largely based on the same detailed paleoseismic and earthquake data used to characterize these faults as potential seismic sources. 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 of smaller faults and fractures in the repository (Youngs, et al., 2003aa; CRWMS M&O, 1998aa).

DOE chose nine sites around Yucca Mountain as demonstration sites of the application of the PFDHA, as shown in SAR Table 1.1-67: (i) Bow Ridge fault, (ii) Solitario Canyon fault, (iii) Drill Hole Wash fault, (iv) Ghost Dance fault, (v) Sundance fault, (vi) an unnamed fault west of Dune Wash, (vii) a location 100 m [328 ft] east of Solitario Canyon fault, (viii) a location between Solitario Canyon fault and Ghost Dance fault, and (ix) a location within Midway Valley. 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.

Results of the PFDHA (CRWMS M&O, 1998aa) show that, except for the Bow Ridge and Solitario Canyon faults, mean fault displacements are 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 nine demonstration sites (SAR Table 1.1-67).

Individual FDH curves were developed to characterize fault displacements at each of the nine demonstration sites. These FDH 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. Example fault displacement curves for several of the nine demonstration sites are provided in SAR Figure 2.2-13.

NRC Staff Evaluation: NRC staff evaluated DOE's information regarding input to the PFDHA in the SAR and supporting documents. NRC staff also conducted its independent analysis of slip tendency (Morris, et al., 1996aa) and faults within the Yucca Mountain region (e.g., Morris et al., 2004aa). The input data to the PFDHA and its interpretation provide appropriate data on site geology to permit evaluation of the PCSA and the GROA design. Specifically, (i) DOE used suitable data, (ii) the methods DOE used to interpret the data were rigorous and appropriate, and (iii) DOE's interpretations are consistent with the regional data and concordant with the data used.

2.1.1.1.3.6 Site Igneous Activity

DOE provided information in SAR Sections 1.1.6, 2.2.1, and 2.3.11 on the known intrusive and extrusive (volcanic) igneous activity in the Yucca Mountain region as they pertain to PCSA and the GROA design. DOE also provided information on the probabilistic volcanic hazard analysis (PVHA) conducted. DOE stated that volcanic activity has occurred in the tectonically active Yucca Mountain region and could continue into the future. Thus, igneous activity may affect GROA design and preclosure repository performance.

In SAR Section 1.1.6, DOE 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. DOE also described the outcome of a PVHA undertaken in 1996 and 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 1×10^{-6} per year. Aspects of the hazard and risk that igneous activity in the postclosure period pose to repository performance are evaluated in TER Section 2.2.1.3.10.1.

In this TER chapter, NRC staff's evaluation of the information presented in SAR Section 1.1.6 is made consistent with the evaluation made in TER Chapters 2.5.4, 2.2.1.2.1, 2.2.1.2.2, and 2.1.1.7 because they also pertain to aspects of possible future igneous activity at the repository site. NRC staff review of the risk that igneous activity poses in the preclosure period also relies upon information given in General Information Section 5.2.1.5; in SAR Sections 1.2.2.1.6.5, 1.6.3.4.2, and 2.3.11; and on relevant DOE-provided reports. This review concentrates on volcanic (extrusive) surface activity as it is more likely to affect the repository surface facilities and workings in the active operation period than an intrusive event, even though DOE showed the probability of a future eruption within the preclosure period to be extremely low.

<u>Magma Types, Location, Style, and Timing of Igneous Activity in the Yucca Mountain Region</u>

<u>Rhyolitic Igneous Activity</u>

DOE assessed the types of igneous activity and location of past volcanism in the Yucca Mountain region in SAR Section 1.1.6.1. DOE assigned an age range of 13 to 10 million years ago (SAR Section 2.3.11.1) for the major volcanic flare-up that formed the rhyolitic ash-flow tuffs of Yucca Mountain, which are the host rocks for the repository. DOE determined that there has been a long time gap between the cessation of these large-scale, caldera-forming explosive eruptions and the present (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, DOE considered the chance of this type of volcanic activity recurring within that timeframe, or even within the postclosure performance period, to be exceedingly small (BSC, 2003ae) and concluded that such activity is not expected to recur in the area of Yucca Mountain within the next 1 million years.

NRC Staff Evaluation: NRC staff reviewed the information DOE provided using the guidance in the YMRP. The staff notes that DOE provided reasonable data on rhyolitic magmatic activity at the site to permit evaluation of the PCSA and the GROA design. Specifically, DOE's assessment provided a reasonable basis for its determination of the likelihood of future explosive volcanic activity of rhyolitic magma at the site, because DOE presented evidence 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 smaller scale basaltic igneous activity, DOE 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 80,000 years ago. This 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 magnetic surveys in 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, DOE 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).

DOE 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 future. Furthermore, several of the Quaternary 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. DOE 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 Evaluation: NRC staff reviewed the information provided in the SAR by conducting independent confirmatory studies to verify the style and frequency of past basaltic volcanism in the Yucca Mountain region (Hill and Connor, 2000aa; Connor, et al., 2000aa; Stamatakos, et al., 2007aa). On the basis of NRC staff's studies and consideration of the available information DOE presented, such as the type and number of basaltic volcanoes and their ages, DOE's approach to assessing the nature and timing of past, and possible future, basaltic igneous activity in the area around the repository provides reasonable data to permit evaluation of the

PCSA and the GROA design. Moreover, NRC staff notes, on the basis of its independent studies (Connor, et al., 2000aa; Hill and Connor, 2000aa) and on other peer-reviewed published information on the timing of eruptions of the Yucca Mountain basaltic volcanoes (Valentine and Perry, 2006aa, 2007aa; Valentine, et al., 2007aa), that DOE's assessment provides a reasonable basis to support its determination of the likelihood of magma supply and future basaltic magmatic activity in the area around the proposed repository, including intrusive and volcanic events, relevant to the preclosure period.

Relationship Between Seismic and Igneous Activity

In SAR Section 1.1.6.1.2, DOE stated that rising magma could cause seismic activity in the form of small earthquakes that in turn could trigger other larger earthquakes. DOE 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 trenches DOE excavated to investigate faults. DOE concluded that the cracks were caused by faulting. DOE showed that several such ash occurrences found in trenches dug across faults near Yucca Mountain were from the eruption of the youngest (80,000 years old) Lathrop Wells volcano.

NRC Staff Evaluation: NRC staff reviewed the information DOE provided using the guidance in the YMRP. Information concerning the possibility of a coseismic relationship between the ash occurrence and the ground cracks [i.e., the 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 coseismic relationship between the ash occurrence and the ground cracks has not been demonstrated. However, any potential relationship between seismic activity and igneous activity is taken into account by the individual probabilities for both future igneous and seismic activity. Whether or not these activities occur independently or dependently, the future occurrence of both has been considered. NRC staff notes 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 1×10^{-6} per year (see next subsection) and is beyond Category 2 event sequences.

Probabilistic Igneous Hazard Analysis

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

For the 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., 1×10^{-6} per year), as identified in SAR Table 1.6-1. On the basis of DOE's PVHA, DOE determined (i) the mean annual frequency of the likelihood of a basaltic dike intruding the underground repository as 1.7×10^{-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 to range from 4.8×10^{-9} to 1.3×10^{-8} (SAR Section 2.3.11). DOE performed other evaluations that supported these values, which NRC staff notes, as part of the postclosure review, are appropriate and consistent with information throughout the SAR (see TER Chapter 2.2.1.2.2).

These values indicated the general range of probabilities DOE determined for an igneous intrusion into, and a volcanic eruption within, the subsurface GROA. Actual probability values applicable to the preclosure period were not stated in the SAR, but were described in DOE's external events hazard screening analysis in BSC Section 6.3 (2008ai) as lower than 10^{-6} per year. A DOE-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, as DOE provided in its final report of the igneous consequence peer review panel (BSC, 2003ae), ranged between 1×10^{-8} and 1×10^{-7}. Therefore, the likelihood of future igneous activity directly impacting the subsurface repository site during the preclosure period is much lower than 1×10^{-6} per year. The potential effects on the repository site of a volcanic eruption from the nearby basaltic volcanic field, or from active volcanoes further away, are described and evaluated in the next subsection.

NRC Staff Evaluation: NRC staff reviewed DOE's information using the guidance in the YMRP and by conducting independent confirmatory studies of the style and frequency of past basaltic volcanism in the Yucca Mountain region (Hill and Connor, 2000aa; Connor, et al., 2000aa; Stamatakos, et al., 2007aa). DOE provided sufficient data to permit evaluation of the PCSA and the GROA design because (i) DOE provided reasonable consideration of, and an appropriate basis for understanding, the probability of future basaltic igneous activity in the Yucca Mountain region during the preclosure period; (ii) DOE provided sufficient identification, analysis, and data on the igneous activity to provide a technical basis for assessing the probability of recurrence; and (iii) NRC's independent studies also show that the probability of future igneous activity at the GROA is lower than 1×10^{-6} per year.

<u>Potential Hazard From Ash Fall from Distant Active Volcanoes and Volcanic Fields in the Region</u>

DOE described the potential effects of fallout of volcanic ash (tephra) on the GROA (SAR Section 1.1.6.3; BSC, 2008ai). DOE 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 in California, and also from the local fields of basaltic volcanic activity considered previously (i.e., from the Southwest Nevada Volcanic Field in general, or more specifically, from the adjacent Crater Flat part of the field). DOE 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 rhyolitic volcanoes in California (SAR Section 1.1.6.3; DOE, 2009ap).

DOE determined that these 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 50-year preclosure period for surface activities (DOE, 2009ap). Further, DOE recognized that this type of activity would likely deposit less ash fall at the repository site than closer located basaltic volcanoes (SAR Section 1.1.6.3), as discussed next.

DOE 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, p. 12 (1987aa). 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 as less than a 1 in 10,000 chance of recurring within the 50-year operational period of the buildings (DOE, 2009ap).

DOE stated that ash fall from nearby future basaltic eruptions in the Southwest Nevada Volcanic Field, similar to that at Lathrop Wells volcano, would deposit a range of ash thicknesses from 0.5 to 3 cm [0.2 to 1.2 in] on the repository site that encompasses the potential distal ash-fall thickness from small rhyolitic volcanoes. DOE found the average probability of recurrence of basaltic volcanism that could deposit a few centimeters [several inches] of ash on the repository site in the preclosure period was small and, on the basis of DOE-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 Section 6.3 (2008ai).

DOE calculated thicknesses of radionuclide-contaminated ash (tephra) accumulation at expected distances from a nearby volcanic eruptive vent with the ASHPLUME model for postclosure scenarios (SAR Section 2.3.11.4.1.1.2; BSC, 2004bk).

NRC Staff Evaluation: NRC staff reviewed the information DOE provided using the guidance in the YMRP and its own independent field observations and estimations of likely tephra-fall thicknesses, as well as expert knowledge derived from reviews of published information on these two distant caldera volcanoes. DOE reasonably 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 next 50 years. NRC staff notes that DOE's assessment of ash-fall hazard to the repository site provides a reasonable basis for understanding the probability of future basaltic ash falls on the GROA in the preclosure period because the likely sites and frequency of the volcanic activity are well understood. Therefore, DOE reasonably determined the thickness and probability of a deleterious ash fall and that these are below DOE's preclosure design criteria of 10 g/cm^2 [21 lb/ft^2] (BSC, 2004bk, 2008ai), equivalent to about a 10 to 20-cm [8 to 12-in] thickness of typical freshly fallen ash (as described next), and a probability of 1×10^{-6} per year.

Potential Hazard from Ash Fall Onto Site Facilities

The hazard at the GROA from distant volcanic activity in the preclosure period was accounted for by an ash-loading factor (mass per unit area), termed "deposition areal density." This was detailed in SAR Table 1.2.2.1 and SAR Section 1.6.3.4.3, and in DOE's report on ash fall hazard at the North Portal Operations Area Facilities in BSC (2004bk) and BSC Section 6.3 (2008ai). The hazard is pertinent to roof loading of buildings in the repository surface operations area by basaltic ash fall from a source in the Southwest Nevada Volcanic Field, which also encompasses ash fall from a distant rhyolitic source volcano. The factor is expressed by the probability of a future ash fall event (eruption) depositing a mass of ash per unit area that a building is not designed to withstand.

DOE developed the building roof design and the ash-loading limit to comply with internationally accepted standards (International Code Council, 2003aa). The probability was calculated by considering various ash densities for each potential volcanic source, the frequency of eruption for those volcanoes, and the probability that ash will fall in the North Portal area of the repository site.

DOE gave the frequency of occurrence of exceeding this mass limit, expressed as the roof live load of 10 g/cm^2 [21 lb/ft^2], as a mean of 6.4×10^{-8} per year. Further, also considering the uncertainty, DOE determined that the roof live load will not exceed an annual frequency of 6.8×10^{-7} (BSC, 2004bk, 2007av, 2008ai). The roof live-loading limit is equivalent to about a 10 to 20-cm [8 to 12-in] thickness of typical freshly fallen ash. This loading limit far exceeds the expected maximum ash fall of up to 3 cm [1.2 in] and more than accounts for possible differences in composition and wetness of the ash deposit, which may also affect its density by one to two times, as outlined in Sigurdsson, p. 565 (2000aa). Thus, DOE stated the threat ash fallout poses to building roofs in the preclosure period was smaller than the 1×10^{-6} per year design criterion (BSC, 2008ai).

DOE recognized the possibility of volcanic ash fall blocking ventilation and circulation pathways above waste-canister aging pads on the repository site. If ash fall did occur at thicknesses of up to 3 cm [1.2 in], DOE stated that there was sufficient space of approximately 40 cm [16 in] below ventilation system intakes such that clogging by ash would not occur. Even if an exceptional fall of 10 cm [4 in] occurred, it would have only a small effect based on the distance between roof and intakes {40 cm [16 in]}. DOE stated that the threat ash fallout poses to building ventilation systems in the preclosure period also fell outside the 1×10^{-6} per year design criterion and, furthermore, was mitigated by the design.

NRC Staff Evaluation: NRC staff reviewed DOE's information by performing independent calculations of likely ash-loading masses made on the basis of information DOE provided. DOE provided reasonable data to permit evaluation of the PCSA and the GROA design. Specifically, the values DOE used for ash densities and the calculated thicknesses at expected distances from a putative nearby future volcanic eruption are typical for this type of volcanic activity and the results of the estimates are consistent with NRC staff's results. Moreover, DOE's proposed design criteria mitigate the hazard posed by ash loading on building roofs and ash ingestion into ventilation systems in the preclosure period.

2.1.1.1.3.7 Site Geomorphology

In SAR Section 1.1.7, DOE assessed geologic landforms and geomorphic processes to identify geologic hazards that might affect structures or operations at the GROA during the preclosure period. These processes are agents of geologic change that may significantly alter surface topography and include erosional and depositional processes, such as running water, wind, rock weathering and soil development, and human earth-moving activities. DOE assessed the site's landscape response to climate change and erosional and depositional processes. These assessments are DOE's bases for evaluating whether a geomorphic hazard could affect site structures or operations. DOE presented geomorphic information and tectonic activity in SAR Section 1.1.7.1, and variability in Quaternary processes was assessed in SAR Section 1.1.7.2, as discussed next.

<u>Geomorphic Information and Tectonic Activity</u>

<u>Erosion, Erosion Rates, and Deposition</u>

DOE conducted geomorphic studies in the Yucca Mountain region to characterize the site, as described in BSC Section 3 (2004bi). On the basis of these studies, Yucca Mountain is described as a series of north-trending ridges and valleys controlled 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. DOE's mapping and trenching studies identified some faults that were active during the Quaternary (approximately last 2 million years) and were exposed at the surface. DOE observed boulder-controlled slopes (indicating stable or balanced transport processes on hillslopes), many angular ridges, narrow and V-shaped valleys, and some steep hillslopes and fault scarps not yet smoothed and eroded away. DOE concluded that these geomorphic observations support a slow rate of erosion for the region.

Additionally, DOE presented geomorphic information related to volcanism in the Yucca Mountain region that it used to determine erosion rates. DOE examined cinder cones (also known as scoria cones) and their associated basaltic lava flows in Crater Flat. The degree of cinder cone erosion was correlated with the length of time they had been exposed to erosion processes. Cones that formed 80,000 to 1 million years ago in Crater Flat are slightly eroded, but 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.

DOE stated that the site surface may be perturbed by geomorphic and tectonic processes during the 100-year preclosure period. Potential effects on landforms at the site include fault displacement of the land-surface causing scarps, land-surface tilting, ash fall from distant volcanic eruptions, debris slides, and disruption of surface drainage. DOE demonstrated that the potential effects are either negligible (due to low likelihood of occurrence, small magnitude/quantity, and long distance from the GROA) or are mitigable by design.

NRC Staff Evaluation: NRC staff reviewed DOE’s description of erosional and depositional processes and landforms in SAR Section 1.1.7 using guidance in the YMRP. NRC staff reviewed DOE’s description of the hazards posed by geomorphic processes at and near the repository site relevant to the preclosure period. The staff made field observations of faults and erosion of the 80,000-year-old Lathrop Wells cinder cone, as well as analog geologic sites, during independent structural geology and volcanology studies in the Yucca Mountain region (Connor, 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, 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 hillslope deposit sequences. The staff considers that hillslope 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 hillslopes around Midway Valley, the preservation of essentially unconsolidated sandy sediments on Yucca Mountain and Busted Butte hillslopes, and the exposure ages of hillslope boulder trains, among other indicators. NRC staff notes that DOE’s geomorphic investigations and descriptions are appropriate because DOE obtained rates of erosion on the basis of fault-scarp erosion of known ages, ages of boulders on hillslopes, erosion rates of cinder cones of known ages, and analyses of stream incisions and alluvial surfaces using standard and reasonable methods of analyses. DOE investigations and descriptions are generally consistent with NRC independent observations and analyses of geomorphic processes, landforms, and erosion rates. Therefore, DOE’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 during the preclosure period is reasonable for DOE to use in its PCSA and GROA design.

As part of independent analyses to confirm the reasonableness of DOE results, NRC staff identified potential neotectonic (Quaternary 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. NRC staff notes 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.

Variability of Quaternary Processes

Climate and Dust

In SAR Section 1.1.7.2, DOE described how climate variability during the Quaternary Period affected landforms and rates of erosional and depositional processes in the Yucca Mountain region. DOE concluded that its model of landscape response used for the Yucca Mountain region is area specific and builds upon a general semiarid landscape model BSC adopted, as described in BSC Section 3 (2004bi). Under present conditions, according to DOE, most runoff takes place during infrequent, intense, short-duration summer thunderstorms. This process activates unconsolidated slope material to produce debris flows. DOE stated in BSC Section 3 (2004bi) that such debris flows are infrequent events. As an example, DOE 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 is much longer than 500 years, as stated in BSC (2004bi).

DOE concluded that over the next 10,000 years, under climatic conditions similar to the present, the local rate of sediment accumulation around Yucca Mountain would remain approximately constant. 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, and occasional debris flows such as Jake Ridge, as described in BSC Section 3 (2004bi). Unless a future change in climatic regime occurs, the aggradational state dominating the valleys of the region will continue. If a climatic change toward wetter conditions occurs, DOE concluded in BSC Section 3 (2004bi) that substantially more than 10,000 years would be required for erosion to remove alluvium and start eroding bedrock in the valleys above the underground repository within Yucca Mountain.

DOE mentioned modern dust transport and deposition as examples of active surface processes in the vicinity of Yucca Mountain. Studies (e.g., Reheis and Kihl, 1995aa) DOE cited for southern Nevada and southern California measured dust deposition rates of silt and clay ranging from 4 to 16 g/m^2 [8.2×10^{-4} to 3.3×10^{-3} lb/ft^2] per year, as outlined in BSC Section 3 (2004bi). On the basis of these studies, eolian dust deposits have been accumulating below desert pavements as part of soil formation and on hillslopes in the vicinity of Yucca Mountain for at least the past 10,000 years.

NRC Staff Evaluation: NRC staff reviewed DOE's information using the guidance in the YMRP and its own field observations, as well as expert knowledge derived from reviews of general information on the geomorphological processes. NRC staff notes that DOE provided sufficient data to permit evaluation of the PCSA and the GROA design. Specifically, DOE's description of these surface features and processes is consistent with the climate setting of the region because they are generally consistent with NRC observations of landforms and rates of erosional and depositional processes in the Yucca Mountain region.

2.1.1.1.3.8 Site Geochemistry

DOE described Yucca Mountain site geochemistry in SAR Section 1.1.8 and references therein. In particular, DOE cited BSC Sections 3.3.5.1 and 5.2.2 (2004bi) for details of subsurface water chemistry and the geochemistry of rock units associated with the GROA. DOE cited SAR Sections 2.3.3 and 2.3.5 for additional information about 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, and the composition of airborne dust phases that may accumulate in the repository drifts.

In describing preclosure site geochemistry, DOE 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, DOE identified four factors associated with the preclosure period that would modify present-day geochemical conditions in the near-field environment: elevated temperatures, gamma radiation, underground construction activities, and underground ventilation processes.

DOE stated that although elevated temperatures (due to heat output from waste packages) and radiation fields (emitted by the waste forms) will persist into the postclosure period, heat output and gamma radiation will be at maximum values during the preclosure period (SAR Sections 1.1.8.1 and 1.1.8.4.2). Subsurface repository construction will introduce dust, residues from explosives, and other anthropogenic materials as potential chemical reactants in the repository drifts. In contrast to postclosure repository conditions, DOE’s preclosure facility design calls for continuous ventilation by fans in the subsurface to circulate air for workers during subsurface operations and to remove decay heat from the waste packages to meet the thermal limits of waste forms, waste packages, and host rock (SAR Section 1.3.5).

<u>Elevated Temperature and Ventilation Effects</u>

DOE assessed how elevated temperatures could modify dissolution, alteration, and precipitation reactions between rocks and the water in pore spaces and fractures. 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).

DOE included the modified composition of seepage into repository drifts (SAR Section 2.3.5.3) in postclosure performance assessment calculations because the water chemistry potentially affects the corrosion rates of engineered barrier materials. However, DOE stated that the continuous forced ventilation of hot, dry air in the repository drifts during the preclosure period would limit the availability of water in a region of dry rock called the dryout zone that would extend several meters into the surrounding rock from the drift walls (SAR Section 1.1.8.1). DOE also stated that the preclosure 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, DOE 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 Section 6.6 (2004bg) and SNL Section 7.5.2 (2008aj).

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review how DOE's description of site geochemistry addressed the potential for geochemical interactions in the near-field host rock under present-day conditions and subject to ventilation effects and elevated temperatures during the preclosure period. The NRC staff compared DOE's description of near-field geochemistry in the repository host rocks with the NRC staff's understanding of the geochemistry of the Yucca Mountain natural system, obtained from extensive prelicensing experience. The staff notes that DOE described the relevant geochemical information, including the 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. On the basis of NRC staff's understanding of coupled heat transfer processes in unsaturated tuffs and dryout zones in ventilated excavations, DOE reasonably described how elevated temperatures and forced ventilation during the preclosure period would limit geochemical interactions in the rocks around the drifts.

<u>Gamma Radiation and Ventilation Effects</u>

In describing the geochemistry of the Yucca Mountain site for preclosure conditions, DOE considered how gamma radiation from the emplaced waste packages might affect water–rock interactions in the repository near field. DOE conducted irradiation experiments to investigate radiation effects on repository host rock. DOE found that even at much higher doses than anticipated in the repository, gamma radiation damage in the rock samples was limited to small changes in mechanical properties of minerals due to the radiolysis of water in the samples. DOE 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. DOE 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 in the dryout zone would greatly reduce any geochemical interactions caused by radiolysis.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review how DOE related the potential effects of gamma radiation to the geochemistry of the repository near-field environment. DOE's experiments reasonably evaluated geochemical radiation effects because DOE used bounding radiation fields and site-specific rock samples that were representative of the near-field environment. DOE's experiments provided a reasonable basis that gamma radiation would not have an important effect on the near field rocks, because (i) DOE used doses that were much higher than expected for the preclosure period to assess geochemical effects of radiation and (ii) the observed effects were conservative because they excluded the attenuation of radiation by the additional shielding provided in a drift by the waste package components or by transportation, aging, and disposal containers. 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, DOE reasonably described how the geochemical effects of radiolysis of porewater in drift walls would be minimized by the presence of a dryout zone during the preclosure period.

Construction Activities and Ventilation Effects

DOE stated that subsurface construction activities, including excavation of the repository, will produce rock dust and limited amounts of anthropogenic materials in the drifts (e.g., explosives residue, diesel exhaust, lubricants, coolants, solvents) during the preclosure period. These materials could serve as potential geochemical reactants, particularly if particles settled on waste package surfaces and reacted to affect metal corrosion rates (SAR Sections 1.1.8.3 and 1.1.8.4.2). DOE also identified atmospheric dust, brought into the repository by the preclosure ventilation system, as a potential source of material for geochemical reactions in the drifts.

DOE found 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, 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 two factors, the presence of the dryout zone in the rock and 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 Evaluation: The NRC staff used the guidance in the YMRP to review the information DOE provided about construction activities and ventilation effects and notes that DOE provided reasonable 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. On the basis of the NRC staff's general understanding of coupled heat transfer processes in unsaturated tuffs and the formation of dryout zones in ventilated excavations, DOE has reasonably described the limited geochemical effects of dust during the preclosure period because the presence of elevated temperatures and the use of forced ventilation during the preclosure period would minimize the availability of water to react with potentially corrosive salts in the dust. Similarly, given the expected preclosure drift temperatures and the volatility range of ammonium salts, staff notes that DOE has reasonably described the limited effects of ammonium salts in contributing to the corrosivity of water in the near-field environment because the presence of elevated temperatures in the drifts and the use of forced ventilation would volatize and remove ammonium salts from the drifts during the preclosure period.

2.1.1.1.3.9 Land Use, Structures and Facilities, and Residual Radioactivity

DOE investigated the following to determine potential human-induced hazards at the site: (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. DOE presented this information in SAR Section 1.1.9.

Previous Land Use

DOE summarized previous land uses within SAR Sections 1.1.9.1 and 1.1.9.2 for the proposed land withdrawal area of 59,500 ha [147,000 acres]. 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 be among these agencies. DOE currently manages the land within the proposed withdrawal area (equivalent to the preclosure controlled area) through a series of memoranda of agreements, rights-of-way, and public land orders.

DOE identified existing mining claims located just outside and just within the southern boundary of the proposed preclosure controlled area. DOE also identified and described both patented and unpatented mining claims located about 15 km [9 mi] south of the proposed GROA (SAR Section 2.1.1.1.3.1). In addition to the mining claims, DOE identified a borrow pit located within the proposed withdrawal area.

NRC Staff Evaluation: NRC staff used the guidance in the YMRP to evaluate DOE's description of previous land uses. NRC staff notes that DOE provided sufficient information on land use to identify potential conflicts by comparing the information to publicly available maps and images.

Existing Structures and Facilities

DOE provided a summary, including location maps, of the existing structures and facilities in the proposed land withdrawal area in SAR Section 1.1.9.3. There are no civilian facilities within the GROA. Because the land has been under federal control for many years, the only nongovernment facilities located within the proposed land withdrawal area (preclosure controlled area) are water wells associated with the Nye County Early Warning Drilling Program. As noted previously, these facilities are 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 are associated with federal government activities, including surface facilities to support site characterization activities and environmental monitoring activities at Yucca Mountain. DOE 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 Evaluation: NRC staff used the guidance in the YMRP to review the information provided by DOE as discussed previously to evaluate DOE's description of previous and ongoing land uses, as well as existing structures and facilities, such as patented and unpatented mining claims, and roads. The NRC staff notes that DOE's information is reasonably descriptive and is consistent with publicly available information regarding man-made features, as discussed in TER Section 2.1.1.1.3.1.

Potential Exposure to Residual Radioactivity

DOE relied on several residual radiation 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, p. 35 (2000aa) and Tipton, p. 9 (1979aa)], included the proposed land withdrawal area along Fortymile Canyon, which includes Fortymile Wash. Other surveys DOE relied on included Area 25 of the NTS, as described in Hendricks and Riedhauser, p. 35 (2000aa) and Lyons and Hendricks Section 6.8 (2006aa). Area 25 is located east of the proposed Yucca Mountain site, and portions of Area 25 are within the withdrawal area. None of these surveys detected man-made radioactivity within the proposed land withdrawal area.

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

A 2006 radiological aerial survey DOE conducted 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 (Lyons and Hendricks, 2006aa).

DOE identified several sources of emissions at the NTS 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 include 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 continues to contribute to radioactive emissions. DOE relied upon information from the NTS air sampling stations that are required to monitor for radioactive airborne particulate and tritium contamination. Six of the sampling locations are near the boundaries and at the center of the NTS, as outlined in Wills Section 3.1 (2006aa). DOE estimated total tritium emissions from all sources to be 6,290 GBq [170 Ci] in 2005. Emissions 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 Table 3-13 (2006aa).

Offsite releases of radioactive material from the NTS are monitored using a monitoring network operated by the Community Environmental Monitoring Program and coordinated by the Desert Research Institute. DOE found that no airborne radioactivity related to historic or current NTS 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 Wills, p. iii (2006aa). An air sampling station that measures radionuclide air concentrations from the NTS is located at the southern boundary of the proposed land withdrawal area. On the basis of these measurements, DOE 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.

As a result of the surveys described previously, DOE determined that there are no indications of residual radioactivity from previous land uses within the GROA, but that there are two locations of residual radioactivity within the proposed land withdrawal area. One is 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 DOE, the Army

Ballistics Research Laboratory Test Range site is posted and fenced off in accordance with 10 CFR Part 835, DOE's regulations for radiological protection.

The other is 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 Section 2.2.1.5 (2001aa). According to DOE, any residual radioactivity within the proposed land withdrawal area will make a negligible contribution to worker and public radiation exposure.

NRC Staff Evaluation: On the basis of the review of the approach presented in the SAR and other information submitted in support of the SAR regarding surveys and reports of previous uses of radioactivity in the area at and around the Yucca Mountain site, DOE's data identifying residual radioactivity at the Yucca Mountain site are reasonable 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. NRC staff notes that the emissions from the NTS are reasonably characterized because of the mandatory reporting requirements for the operator of the NTS site. Further, NRC staff verified this information by reviewing the 2006 NTS Environmental Report (Willis, 2006aa).

NRC staff notes that the offsite monitoring data fully characterize any offsite sources that could contaminate the Yucca Mountain site. DOE reasonably identified the locations and source strengths of residual radioactivity from previous land uses near, but not in, the land withdrawal area. DOE's radiation surveys included the entire land withdrawal area and would detect 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 data are sufficient to evaluate the contribution to worker and public radiation exposure from residual radioactivity and that the residual radioactivity would make a negligible contribution.

DOE provided reasonable information on known radioactive sources, described previously, to evaluate the contribution to worker or public dose.

2.1.1.1.4 NRC Staff Conclusions

The NRC staff notes that DOE's general description of work conducted to characterize the Yucca Mountain site is consistent with the guidance in the YMRP. The NRC staff also notes that DOE reasonably characterized natural and human-induced hazards for the PCSA and the GROA design, as discussed in this chapter.

DOE stated that it would (i) monitor the location of a Quaternary fault with potential for significant displacement (SAR Table 1.9-9, DCP 01-05) and observe rock conditions to specifically evaluate the observed faults, during repository construction, to ensure that conditions cannot credibly lead to a breach of a waste package (otherwise, a standoff distance from the fault would be established) (TER Section 2.1.1.1.3.5.1.1) and (ii) evaluate stability of the cut and fill slopes near the aging pads and on the transportation routes to and from the aging pads under applicable seismic loading conditions to account for uncertainties in shear strength of alluvium before excavation of aging facility foundation (TER Section 2.1.1.1.3.5.4). As part of the detailed design process, DOE should (i) confirm the shear strength properties of alluvium, including uncertainty (TER Section 2.1.1.1.3.5.4); (ii) confirm the allowable maximum bearing pressure for mat foundations design on the basis of settlement criterion and during a design basis seismic event (TER Section 2.1.1.1.3.5.4); and (iii) confirm the stability of slopes under applicable seismic loading conditions using an approach that accounts for uncertainties in the shear strength of alluvium (TER Section 2.1.1.1.3.5.4). As part of the performance confirmation program, DOE should confirm that (i) the mechanical properties of the nonlithophysal rock encountered in the ESF are representative of such properties in the repository block (TER Section 2.1.1.1.3.5.4) and (ii) the mechanical properties of the lithophysal rock encountered in the ESF and ECRB are representative of such properties in the repository block (TER Section 2.1.1.1.3.5.4).

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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

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of DOE's 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). The objective of the review is to verify that DOE's information reasonably describes and discusses design of SSCs, SCs, equipment, radioactive wastes to be disposed, and operations to support an NRC staff evaluation of the GROA facility design and preclosure safety analysis (PCSA). 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 DOE's responses to the NRC staff requests for additional information (RAIs) (DOE, 2009dl–dn,dp,dq,ds,dt–dv,dx,ea–ee).

The DOE description and discussion of design of SSCs, SCs, equipment, and operational processes include (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.

2.1.1.2.2 Evaluation Criteria

The regulatory requirements for the description and discussion of design of SSCs, SCs, equipment, and operational process activities are in 10 CFR 63.21(c)(2), 63.21(c)(3)(i), 63.21(c)(4), and 63.21(c)(5). The regulations require that the SAR describe and discuss (i) structures including general arrangement and dimensions; (ii) material properties and specifications; (iii) analytical and design methods, applicable codes, and standards; and (iv) kind, amount, and specifications of the radioactive material proposed to be received and possessed at the GROA. The information provided by DOE should satisfy the general description for the PCSA as required in 10 CFR 63.112(a).

The NRC staff evaluated the information in DOE's SAR using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.2 (NRC, 2003aa). The relevant acceptance criteria follow:

- The license application contains a description of the location of the surface facilities and their designated functions sufficient to permit evaluation of the PCSA and the GROA 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 PCSA and GROA design.

- The license application contains descriptions and design details for structures, systems, components, and equipment of the subsurface facility, sufficient to permit evaluation of the PCSA and GROA design.

- The license application describes the characteristics of the spent nuclear fuel and high-level radioactive waste, sufficient to permit evaluation of the PCSA and GROA design.

- The license application provides a general description of the engineered barrier system and its components, sufficient to support evaluation of the PCSA and GROA design.

- The description of the operational processes to be used at the Geologic Repository Operations Area is sufficient for review of the PCSA.

In addition, the NRC staff used additional applicable guidance, such as NRC standard review plans and regulatory guides, to support the NRC staff's review. These additional guidance documents are discussed in the relevant sections.

2.1.1.2.3 Technical Evaluation

The structure of this Technical Evaluation Report (TER) chapter follows the review guidance provided in YMRP Section 2.1.1.2. TER Section 2.1.1.2.3.1 discusses the location and functions of surface facilities. TER 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). TER Section 2.1.1.2.3.3 details the SSCs, equipment, and utility systems for the subsurface facilities. TER Section 2.1.1.2.3.4 describes high-level radioactive waste (HLW) characteristics. TER 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. TER Section 2.1.1.2.3.6 covers the operational processes associated with the GROA and reviews the communication, instrumentation, and control systems for both surface and subsurface facilities. TER Section 2.1.1.2.3.7 presents a review of the description of design and operations of the subsurface facility and a focused review of the emplacement drift, which is a key component of the GROA subsurface operations. DOE provided the information on description and design of SSCs, SCs, and equipment in SAR Sections 1.2 through 1.14.

This chapter describes the NRC staff's review of DOE's (i) description of the SSCs, SCs, and equipment; (ii) operational activities; (iii) drawings and figures showing basic geometry and dimensions; and (iv) information on the materials. The NRC staff reviewed the functions of the SSCs and equipment in the context of operations and any interaction with other SSCs. The NRC staff also reviewed whether the codes and standards proposed for the SSC design are appropriate to perform its intended functions. The NRC staff evaluated the design description and functions of the SSCs, SCs, and equipment in the context of operations and used the results in the PCSA presented in subsequent chapters.

2.1.1.2.3.1 Description of Location of Surface Facilities and Their Functions

DOE provided an overview of the surface facilities and their associated operations in SAR Section 1.2.1. Information provided in the Yucca Mountain Repository General Information

Volume, Chapter 1, Section 1.1.2 presented a general description of the proposed geologic repository at Yucca Mountain, location of the GROA, and information on 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 surface facilities and their relative locations with respect to the site boundary. The surface facilities include waste handling facilities, surface transportation network, balance-of-plant facilities, flood control features, and support systems. The waste handling facilities include the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Waste Handling Facility (WHF), Receipt Facility (RF), and Aging Facility (AF). The IHF, CRCF, WHF, and RF are ITS structures. The aging pads of the AF are considered to be ITS. Other surface structures are not ITS, including the Central Control Center Facility (CCCF), Emergency Diesel Generator Facility (EDGF), Cask Receipt Security Station, and Low-Level Waste Facility (LLWF). DOE proposed a system of dikes (levees) and ditches to prevent inundation of surface facilities from a potential probable maximum flood (PMF). SAR Sections 1.2.3 to 1.2.7 described the design and functions of the surface waste handling facilities. Descriptions of 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 showed 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 of the waste handling facilities, as described in SAR Sections 1.2.3 through 1.2.7, is discussed next.

The IHF receives transportation casks containing naval SNF or HLW canisters and prepares the casks for unloading. The operations in the IHF 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 to load waste packages are the three CRCFs.

Each CRCF receives and unloads 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 are transferred to waste packages, and the waste packages are placed in the TEV for transporting to the subsurface. In the CRCF, the TAD canisters can also be moved from the transportation cask into an aging overpack for transportation to an AF.

The WHF receives uncanistered commercial spent nuclear fuel (CSNF) assemblies in a transportation cask. The CSNF assemblies are transferred, under water in the pool, into TAD canisters. The TAD canisters are 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 are received in transportation casks or aging overpacks. The DPCs are then transferred to a shielded transfer cask where the DPC is opened and the CSNF assemblies are transferred under water into the TAD canisters in the pool.

The RF receives transportation casks containing TAD canisters or DPCs and transfers 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 is designed to provide support to the aging overpacks containing HLW in the TAD canisters and DPCs. The main waste handling functions of the AF are to provide aging capability for the repository waste handling operations and to 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 DOE classified as non-ITS. The function of each non-ITS facility is briefly described next.

The EDGF houses 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 is 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 consists of space for local network equipment and functions, while the emergency operations center provides space for emergency management services and functions.

The CCCF provides functional space, structures, and internal systems to support the central control center, which is the technical support center for conducting emergency management activities. This provides centralized control and communication for plantwide monitoring and control. The CCCF has 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 stores dry and liquid low-level radioactive waste (LLW). The LLWF receives LLW from the IHF, CRCFs, WHF, and RF. Unloaded DPCs are delivered in a shielded transfer cask or other acceptable container and are stored in the LLWF for eventual disposal.

The Warehouse and Non-Nuclear Receipt Facility stores TAD canisters; empty, new waste packages; lids; pallets; spread rings; and shield plugs. No radioactive material is received or stored in this facility. The Aging Overpack Staging Facility serves as an outdoor area for storing empty aging overpacks and unloaded and noncontaminated aging overpacks.

Surface runoff flooding from a probable maximum precipitation event would inundate the surface facilities (SAR Section 1.6.3.4.5). The flood control features proposed in SAR Figure 1.2.2.-7 consist of ditches and dikes (levees) to collect and divert the surface runoff flow (potential flood) and prevent inundation of surface facilities. DOE classified the flood control features as ITS because they are intended to prevent inundation 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 Evaluation: The NRC staff conducted its review using the guidance in the YMRP and compared the information on the surface facility layout, contained in various SAR sections identified previously, with the proposed operations of handling HLW at the site and ultimate disposal in subsurface emplacement drifts. The general descriptive information in the SAR about the facilities on the nature of operations, location and distance from the boundary, design, and functions of the surface facilities at the GROA site is reasonable because the information is sufficient to permit an evaluation of DOE's PCSA and surface facilities systems design. In addition, the NRC staff notes that the information in the SAR on functions of the surface facilities at GROA is consistent with the overall HLW handling and disposal operations at the site.

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 DOE's information in SAR Sections 1.2.1 through 1.2.7 on description and discussion of design of the surface facility 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; HVAC systems; shielding and criticality control systems; fire suppression systems; piping and instrumentation (P&I) diagrams (P&IDs); and decontamination, emergency, and radiological safety systems.

2.1.1.2.3.2.1 Surface Structures

DOE described the structural design of the building facilities in SAR Sections 1.2.3 through 1.2.7. DOE used this information in PCSA and in the design and performance evaluation of the building facilities. On the basis of the PCSA, DOE designated the CRCF, IHF, WHF, RF, AF, and part of its flood control features as 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 GROA also contains a number of non-ITS facilities. Two of these facilities (LLWF and EDGF) will be covered in this TER section.

<u>ITS Structures</u>

SAR Section 1.2.4 provided the general description of the CRCF, and SAR Section 1.2.2.1 described the structural design of the CRCF. DOE indicated that the GROA would have three identical CRCFs constructed in phases. The CRCF building dimensions are 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 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 indicated that areas of the facility that fall outside the footprint of the CRCF and non-ITS areas are constructed using lighter concrete and steel framing. The mat foundations associated with ancillary areas (non-ITS structures) are reinforced concrete mats designed as necessary to adequately support the superstructures and are structurally independent of the ITS structures.

The IHF is composed of two seismically independent structures isolated by a seismic joint (SAR Section 1.2.3). The main structure consists 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 is 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 consists 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 is 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 is a 1.8 m [6 ft]-thick mat. DOE indicated 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 are composed of slabs on grade using lighter concrete construction and/or insulated metal panels on steel framing. These areas are supported by reinforced concrete mat foundations independent of the ITS structures.

The WHF is 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 is 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 is 30 m [100 ft] above grade, with the majority of the building approximately 24 m [80 ft] above grade. The below-grade pool substructure is 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 are 23 m [74 ft] wide and 19 m [61 ft] long. The bottom of the pool is 16 m [52 ft] below the at-grade concrete mat. The mat foundation at grade is 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 are 1.2 m [4 ft] thick. The main WHF superstructure is 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 are constructed of 1.2-m [4-ft]-thick concrete walls and roof slabs. Other elevated floor diaphragm slabs are generally 0.6 m [2 ft] thick. The below-grade portion of the pool consists of 1.8-m [8-ft]-thick exterior earth retaining walls. Interior rooms are 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 are categorized as non-ITS are supported by structurally independent foundations.

SAR Section 1.2.6 discussed the RF, stating it is 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 are approximately 96 m [315 ft] wide by 97 m [318 ft] long. The part of the structure that is considered ITS has dimensions of 61 m [200 ft] wide by 73 m [240 ft] long. The maximum height of the building is 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 is 1.2 m [4 ft]. The RF foundation mat is 2 m [7 ft] thick, and elevated floor diaphragm slabs are generally 0.5 m [1.5 ft] thick. Areas of the facility that are non-ITS are constructed on separate slabs on grade using lighter concrete and/or steel framing, which has insulated metal panels for the walls. These ancillary areas/rooms are next to the main RF structure, are seismically separated, and will not compromise the integrity of the main ITS structure in a design basis ground motion DBGM-2 event.

The 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 are to provide up to 2.1×10^7 kg [21,000 metric tons of heavy metal (MTHM)] of aging capability for the repository in 2,500 aging spaces and to protect TAD canisters and DPCs from external hazards. The AF consists of the following ITS components: (i) the aging pad, (ii) aging overpack, and (iii) overpack transfer systems. This section only describes the aging pads presented in SAR Figure 1.2.7-2. The detailed layout of aging pad area 17P was shown in SAR Figure 1.2.7-3 and consists of seven pads for about 1,250 vertical aging overpacks. The aging pad area 17R (SAR Figure 1.2.7-4) has eight pads with space for about 1,150 vertical aging overpacks, and two pads with space for 100 horizontal DPCs in horizontal aging modules, with 50 modules on each pad. Vertical aging overpacks are 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 is 1.8 m [6 ft] to enable access of the site transporter and to permit air circulation for cooling. Horizontal aging modules are arranged side by side.

The aging pads 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). DOE indicated (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 DOE, the concrete aging pads were designed to support the aging overpacks during credible design events and to withstand loads and load combinations imposed by natural phenomena. DOE considered flood drainage channels to carry away water from a PMF surrounding the aging pads. The distance from the aging pads to upslope hillsides and the location of the drainage channel precludes soil from the slope sliding onto the concrete aging pads and contacting the aging overpacks (see TER Chapter 2.1.1.7 for staff evaluation of stability of slopes near the AF). Aging pads provide for water runoff and are designed to consider concrete heating 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, DOE described ITS flood control features credited with preventing inundation of the surface facilities from a PMF at the site. DOE's proposed conceptual design includes 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 Evaluation: The NRC staff reviewed the information in SAR Sections 1.2.1 through 1.2.7 related to the description of the structural design of the ITS facilities using the guidance in the YMRP. The NRC staff reviewed the list of codes and standards, drawings, materials, and loads associated with each facility. These design codes and standards are reasonable because they are in conformance with standard engineering practices for nuclear material handling facilities. In addition, the NRC staff notes that the proposed materials for the construction of the ITS surface structures are appropriate for their intended use because they are in conformance with standard engineering practices. The description of design parameters and design methodologies is appropriate for the design of the surface ITS facilities. Furthermore, the NRC staff notes that the information DOE provided on the reinforced concrete structural components, steel structural components, and general arrangement drawings of the surface facilities is reasonable for use in the PCSA and in review of the design of the surface facilities (DOE, 2009dm).

For flood control features, the NRC staff notes that the information in the SAR and DOE's response to the NRC staff RAIs provide a reasonable description of the layout, function, and design bases and design criteria of the flood control features and can be used in other sections of this TER (see TER Section 2.1.1.7.3.1.3 for details).

Non-ITS Structures

SAR Section 1.2.8 provided DOE'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 Low-Level Waste Handling Facility (LLWF) handles radioactive material, and the Emergency Diesel Generator Facility (EDGF) houses ITS Diesel Generator and its components. These two facilities perform more important functions than other non-ITS facilities. Therefore, these two facilities are reviewed. SAR Section 1.2.8.1.3 listed the codes and standards that DOE proposed to use for the design of the non-ITS facilities. Additional information related to the structural design of the non-ITS facilities is in BSC Section 4.2.11.5 (2007av). 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 DOE's structural description of the EDGF. The EDGF has an overall footprint approximately 53 m [174 ft] wide by 30 m [98 ft] long. DOE 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 foundation of the EDGF structure is described 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 DOE, consists 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 is a 0.9-m [3-ft]-thick reinforced concrete slab. There are 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 LLWF has an overall footprint of approximately 80 m [263 ft] wide by 50 m [163 ft] long. 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 are shown in SAR Figures 1.2.8-12 through 1.2.8-14. The LLWF is a multistory building designed as a steel structure with concrete floor, concrete mat foundation, concrete shield walls, steel roof truss system, and interior and exterior structural steel bracing. The facility has four bays composed of half-height shielded walls for storage of LLW. A 45,359-kg [50-T] bridge crane is used to move large waste containers through the facility.

NRC Staff Evaluation: The NRC staff reviewed SAR Section 1.2.8 to evaluate the structural design description of the EDGF and LLWF buildings using the guidance in the YMRP. The facility descriptions explaining the functions and design of these facilities are reasonable because the information is sufficient to permit an evaluation of the PCSA and design of the facilities. In addition, the NRC staff reviewed DOE's information related to codes and standards to be used in the design and notes that they are reasonable because they are consistent with common industry practices.

2.1.1.2.3.2.2 Layout of Mechanical Handling Systems

DOE described mechanical waste handling systems, functions, ITS components, and design information, and layout drawings of mechanical handling systems in the surface facilities in various sections of the SAR. Major mechanical systems are used for waste handling operations at the IHF, CRCF, WHF, and RF, where similar systems are often used in multiple facilities.

Because of the system replication throughout the facilities, the NRC staff reviewed the information provided on equipment at the system level while noting any significant layout and interface distinctions between facilities. Detailed NRC staff evaluation of waste handling operations using these mechanical systems is in TER Section 2.1.1.2.3.6.1.

Cask Handling System

Similar cask handling systems are used in the CRCF (SAR Section 1.2.4.2.1), IHF (SAR Section 1.2.3.2.1), WHF (SAR Section 1.2.5.2.1), and RF (SAR Section 1.2.6.2.1) and consist of both cask and waste package preparation systems. The cask handling systems receive and export loaded and unloaded casks, canisters, and waste packages into and out of the facility. In addition, the cask handling systems prepare loaded or unloaded casks, canisters, and waste packages for canister transfer operations, reuse, or transportation out of the facility.

The ITS SSCs at the surface facilities were listed in SAR Table 1.9.1. Also, for the IHF, the ITS SSCs functions and components were described in SAR Sections 1.2.3.2.1.1.3.1 and 1.2.4.2.1.1.3.1. SAR Figures 1.2.3-2 and 1.2.3-3 showed the location of the ITS SSCs. The ITS SSC waste package handling crane is described for the IHF in SAR Section 1.2.3.2.4 and for the CRCF in SAR Section 1.2.4.2.4.

ITS SSCs functions and components of the cask preparation subsystem at the CRCF were described in SAR Section 1.2.4.2.1.1.3.1, and the locations were shown in SAR Figures 1.2.4-2 and 1.2.4-3. For the WHF and RF, the ITS SSCs of the cask handling subsystem functions and components were described in SAR Sections 1.2.5.2.1.1.3 and 1.2.6.2.1.1.3, respectively, with appropriate references to other sections of the SAR for ITS SSCs that are used at other facilities. The locations of the ITS SSCs were shown in SAR Figures 1.2.5-2, 1.2.5-3, and 1.2.6-2.

The applicable codes and standards for the cask handling equipment at all the surface facilities were identified in SAR Table 1.2.2-12.

NRC Staff Evaluation: The NRC staff evaluated DOE's description of the cask handling system using the guidance in the YMRP. The NRC staff reviewed the consistency among the system, equipment layout, cask handling operations, and process flow. In addition, the NRC staff assessed the appropriateness of the codes and standards proposed for the system design to perform the intended functions. DOE's descriptions of the cask handling system equipment layouts and functions in each facility are reasonable because they allow NRC staff to evaluate their intended activities and operations. The equipment layout information provided in the SAR to evaluate basic operations, as well as relationships and interdependencies with other subsystems within each facility, is reasonable. DOE used applicable codes and standards for SSC design and construction. Therefore, the NRC staff notes that DOE provided information about the description, design, function, and potential interaction among support SSCs for the cask handling system that is reasonable for use in the PCSA and design, as needed.

Canister Transfer System

The canister transfer subsystem was described in SAR Sections 1.2.3.2.2 (IHF), 1.2.4.2.2 (CRCF), 1.2.5.2.5 (WHF), and 1.2.6.2.2 (RF). The canister transfer system transfers canisters from transportation casks to waste packages or aging overpacks, transfers TAD canisters from aging overpacks to waste packages, and moves waste packages to the waste package positioning room after loading. In the WHF, the canister transfer system is also used to transfer loaded DPCs from transportation casks to shielded transfer casks. The canister transfer system was described in SAR Section 1.2.3.2.2.

The ITS SSCs of the canister transfer system at the surface facilities were summarized in SAR Table 1.9-1. Their functions and components were also described in SAR Sections 1.2.3.2.2.1.3 and 1.2.4.2.2.1.3 for the IHF, SAR Section 1.2.4.2.2.1.3 for the CRCF, SAR Sections 1.2.5.2.5.1.3 and 1.2.4.2.2.1.3 for the WHF, and SAR Sections 1.2.6.2.2.1.3 and 1.2.4.2.2.1.3 for the RF.

The applicable codes and standards for the canister transfer system were identified in SAR Table 1.2.2-12.

NRC Staff Evaluation: The NRC staff evaluated DOE's description of the canister transfer system using the guidance in the YMRP. The NRC staff reviewed the consistency among the system, equipment layout, canister transfer operations, and process flow. In addition, the NRC staff evaluated the appropriateness of the codes and standards proposed for the system design to perform the intended functions. DOE's descriptions of the canister transfer system equipment layout and functions in each facility are reasonable because they allow NRC staff to evaluate the activities and operations. The equipment layout information provided in the SAR to evaluate basic operations, as well as relationships and interdependencies with other subsystems within each facility, is reasonable. DOE used applicable codes and standards for SSC design and construction. Therefore, the NRC staff notes that the DOE-provided information about the description, design, function, and potential interaction among support SSCs for the canister transfer system is reasonable for use in the PCSA and design, as needed.

Waste Package Closure System

The waste package closure system consists of welding, stress mitigation, inerting, control and data management, and closure room material handling subsystems (SAR Section 1.2.4.2.3.1.3).

This waste package closure system is classified as non-ITS. However, this system is a critical component of the operation. The bridge of the remote handling subsystem is ITS. The waste package closure system is protected by preventing structural collapse of the bridge due to a spectrum of seismic events. This bridge is designed in accordance with American Society of Mechanical Engineers (ASME, 2005aa) NOG-1–2004.

The waste package closure system performs a seal weld between the spread ring and the inner lid, the spread ring and the inner vessel, and the spread ring ends; performs a seal weld between the purge port cap and the inner lid; performs a narrow groove weld between the outer lid and the outer corrosion barrier; performs nondestructive examination of the welds to verify the integrity of the welds and repair any minor weld defects; purges and fills the waste inner vessels with helium gas to inert the environment; performs a leak detection test of the inner lid seals to ensure the integrity of the helium environment in the inner vessel; and performs stress mitigation of the outer lid groove closure weld to induce compressive residual stress.

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 in accordance with the applicable sections of NUREG–1536 (NRC, 1997ae). The waste package closure system SSCs are 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 Evaluation: The NRC staff evaluated DOE's description of the waste package closure system in the CRCF and IHF facilities using the guidance in the YMRP. The NRC staff reviewed the consistency among the system, equipment layout, and process flow. In addition, the NRC staff assessed whether the codes and standards proposed for subsystem design are reasonable for the SSCs to perform the intended functions. DOE's descriptions of the waste package closure system equipment layout and functions in each facility are reasonable because they permit an evaluation of the activities and operations. The equipment layout information provided in the SAR to evaluate basic operations, as well as relationships and interdependencies with other systems within each facility, is reasonable. DOE used applicable codes and standards for the SSC design and construction.

Therefore, the NRC staff notes that the DOE-provided information about the description, design, function, and potential interaction among support SSCs for the waste package closure system is reasonable for use in the PCSA and design, as needed.

<u>Waste Package Load-Out System</u>

SAR Section 1.2.4.2.4 detailed the waste package load-out system. This system is located in the IHF and CRCF. The waste package load-out system receives sealed waste packages after closure operations and prepares them for transfer to the TEV.

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 door, the waste package transfer trolley (WPTT), the waste package handling crane, and the waste package shield ring.

The applicable codes and standards for the waste package load-out system were identified in SAR Table 1.2.2-12.

NRC Staff Evaluation: The NRC staff evaluated DOE's description of the waste package load-out systems using the guidance in the YMRP. The NRC staff reviewed the consistency among the systems' equipment layout and process flow. In addition, the NRC staff assessed whether the codes and standards proposed for subsystem design are appropriate for the SSCs to perform the intended functions. DOE's descriptions of the waste package load-out system equipment layout and functions in each facility are reasonable because they permit an evaluation of the activities and operations. The equipment layout information provided in the SAR to evaluate basic operations, as well as relationships and interdependencies with other subsystems within each facility, is reasonable. DOE used applicable codes and standards for SSC design and construction. Therefore, the NRC staff notes that DOE provided information about the description, design, function, and potential interaction among support SSCs for the waste package load-out system that is reasonable for use in the PCSA and design, as needed.

<u>SNF Assembly Transfer System</u>

The SNF assembly transfer system is in the Cask Preparation Area of the WHF and its functions and components were described in SAR Section 1.2.5.2.2. This system receives SNF assemblies from a DPC or transportation cask and places SNF assemblies using a spent fuel transfer machine (SFTM) in SNF staging racks or transfers SNF assemblies into a TAD canister. SNF assembly transfer occurs in the pool. Components of the SNF assembly transfer system are located in and above the pool.

The auxiliary pool crane and SFTM are ITS SSCs located above the pool. The boiling water reactor (BWR) lifting grapple, pool lid-lifting grapple, long-reach grapple adapters, pressurized water reactor (PWR) lifting grapple, SNF staging rack, truck cask lid-lifting grapples, truck cask handling frame, and pool cask handling yoke are 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.

The applicable codes and standards for the canister transfer system were identified in SAR Table 1.2.2-12.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of the SNF assembly transfer system using the guidance in the YMRP. The NRC staff reviewed the consistency among the subsystems, equipment layout, SNF assembly transfer operations, and process flow. In addition, the NRC staff evaluated the appropriateness of the codes and standards proposed for subsystem design to perform the intended functions. DOE's descriptions of the SNF assembly transfer system equipment layout and functions in the WHF are reasonable because they permit an evaluation of the activities and operations. The equipment layout information provided in the SAR to evaluate basic operations, as well as relationships and interdependencies with other subsystems within the WHF, is reasonable. DOE proposed applicable codes and standards for SCC design and construction. Therefore, the NRC staff notes that the DOE-provided information about the description, design, function, and potential interaction among support SSCs for the SNF assembly transfer system is reasonable for use in the PCSA and design, as needed.

<u>Dual-Purpose Canister Cutting System</u>

The DPC cutting system is located in the Cask Preparation Area of the WHF, as described in SAR Section 1.2.5.2.3. This system receives and opens various types of DPCs to access the SNF assemblies. DPC cutting is done outside the pool at the DPC cutting station. The DPC cutting jib crane, the DPC cutting station, and the lid-lifting grapple are ITS SSCs of the DPC cutting system. The functions and components of these ITS SSCs and the DPC operational process were described in SAR Section 1.2.5.2.3.

The principal codes and standards applicable to the DPC cutting system design were identified in SAR Table 1.2.2-12.

NRC Staff Evaluation: The NRC staff evaluated DOE's description of the DPC cutting system using the guidance in the YMRP. The NRC staff reviewed the consistency among the subsystem equipment layout, DPC cutting system operations, and process flow. In addition, the NRC staff evaluated the appropriateness of the codes and standards proposed for subsystem design to perform the intended functions. DOE's descriptions of the DPC cutting system equipment layout and functions in the WHF are reasonable because they permit an evaluation of the activities and operations. The equipment layout information provided in the SAR to evaluate the operations, as well as relationships and interdependencies with other subsystems within the WHF, is reasonable. Additionally, DOE used applicable codes and standards for the

SCC design and construction. Therefore, the NRC staff notes that the DOE-provided information about the description, design, function, and potential interaction among support SSCs for the DPC cutting subsystem is reasonable for use in the PCSA and design, as needed.

TAD Canister Closure System

The TAD canister closure system is in the cask preparation area of WHF, as described in SAR Section 1.2.5.2.4. TAD canister closure is the process that closes the loaded TAD canister by welding the shield plug and fully draining and 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.

The TAD canister closure system and the TAD canister welding machine are classified as non-ITS. The TAD canister closure jib crane, the lid-lifting grapple, and the shielded TAD canister closure station are ITS SSCs of the TAD canister closure system. 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 principal codes and standards applicable to the TAD canister closure system design were identified in SAR Table 1.2.2-12.

NRC Staff Evaluation: The NRC staff evaluated DOE’s description of the TAD canister closure system using the guidance in the YMRP. The NRC staff reviewed the consistency among the subsystems, equipment layout, TAD canister closure system operations, and process flow. In addition, the NRC staff evaluated the appropriateness of the codes and standards proposed for subsystem design to perform the intended functions. DOE’s descriptions of the TAD canister closure system equipment layout and functions in the WHF are reasonable because they permit an evaluation of the activities and operations. The equipment layout information provided in the SAR to evaluate the basic operations, as well as relationships and interdependencies with other subsystems within the WHF, is reasonable. Additionally, DOE used applicable codes and standards for SCC design and construction. Therefore, the NRC staff notes that the DOE-provided information about the description, design, function, and potential interaction among support SSCs for the TAD canister closure subsystem is reasonable for use in the PCSA and design, as needed.

2.1.1.2.3.2.3 Geologic Repository Operations Area Electric Power Systems

DOE described and discussed the electrical power system for the GROA in multiple sections of the SAR. Information related to the ITS electrical power system, normal (non-ITS) electrical power system, ITS and normal electrical power system direct current (DC) electrical power, ITS and normal electrical power system alternating current uninterruptible power supplies (UPS), ITS and normal diesel generators, and associated mechanical support equipment in the GROA facilities was provided in SAR Sections 1.4.1, 1.2.8, 1.9.1, 1.2.4, 1.2.5, 1.13, 5.5, and 5.6. Additional information relevant to the subsurface normal electrical power system distribution system was provided in SAR Sections 1.3.1–1.3.5, and information relevant to subsurface electrical power distribution concept of operations and functional design was provided in BSC (2008ca). Applicable codes and industry standards DOE cited and high-level, single-line electrical drawings for representative power subsystems were also included in these SAR sections.

Virtually all facilities in the proposed GROA utilize normal electric power that the electrical power system provides. The electrical power system also provides ITS electric power to designated SSCs in the surface facilities. Most ITS SSCs are powered by the normal electrical power system, as they are designed to fail in a safe condition if power is interrupted. DOE did not identify any ITS SSCs in the subsurface facilities. DOE provides separate subsurface normal electrical power system feeds that facilitate simultaneous operations and construction and expansion activities. During operations, some areas of the subsurface facility will be inaccessible to workers.

The electrical power system includes normal electrical power system and ITS electrical power system, and 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 switchgear, which can supply backup power to selected non-ITS loads during a loss of offsite power (LOSP). The EDGF houses two redundant and independent ITS diesel generators with ITS mechanical support systems and ITS 13.8-kV switchgear. Separate normal and ITS DC and UPS SSCs are located within various facilities to maintain uninterrupted power to designated controls and loads. DOE described physical and electrical separation and isolation between normal and ITS electrical power system SSCs, including descriptions of cable raceways and cabling. The SAR 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 is 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.

The switchyard, described in SAR Section 1.4.1.1.1, connects 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 switchgear 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 an LOSP, the feed breakers providing commercial power to the two 13.8-kV switchgears in the Standby Generator Facility are opened and the standby diesel generators are automatically started and connected to each switchgear. These operations 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 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 provided a single line diagram for the normal DC power system, including a third “swing” battery charger that the standby diesel generators can power. Normal electrical power system UPS are located in major operations facilities (SAR Section 1.4.1.1.5) 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. Single line electrical diagrams were provided for normal electrical power system 480/277-V and 208/120-V UPS in SAR Figures 1.4.1-8 and 1.4.1-9, respectively.

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

SAR Sections 1.4.1.2.3 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 will be 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 affects 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. The two electrical power system power feeds (13.8 kV) are converted to 480/277 V and 208/120V within alcoves located inside each subsurface access main. Normal 13.8-kV power is also 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 an LOSP, power to the exhaust fans can be 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 are equipped with connections for mobile diesel backup generators.

Various types of mechanical handling equipment such as the TEV and drip shield emplacement gantry (DSEG) will 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 will be located in the accessible areas.

The electrified third rail must extend into nonaccessible areas to provide power for the TEV, DSEG, and other remotely operated vehicles (ROVs) (SAR Section 1.3.3.5.1.1). DOE 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 are contingent on the subsurface transportation equipment design (SAR Section 1.3.2.4.6.4). DOE stated that commercially available materials will be used for the electrified third rail (SAR Section 1.3.4.5.7) and that these materials will be analyzed to determine whether the impact of the proposed materials on postclosure performance is acceptable (SAR Section 1.3.2.4.6.4). BSC (2008bz) provided a high-level conceptual description of the design of redundant, electrified third rail SSCs to accommodate transmission of three-phase power to a vehicle (requiring at least three power contact rails for each). DOE's description indicated that SSCs providing power to vehicles may be permanently installed within emplacement drifts, turnouts, and other nonaccessible areas. Inspection and potential maintenance operations for these areas will be performed using one or more types of ROVs. DOE 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 Evaluation: The NRC staff evaluated the design descriptions of the proposed non-ITS normal electrical power system using the guidance in the YMRP. To evaluate the description of the normal electrical power system, the NRC staff reviewed information provided in the SAR and supplemental materials to support an evaluation of the electrical power system design and architecture concepts and related codes and industry standards as cited by DOE. The design concept description of the TEV sliding collectors that interface with the electrified third rail indicated that the vehicle design could accommodate gaps in the third rail system. BSC (2008bz) did not present any details about potential compromise of power connection redundancy when third rail gaps are present, nor did SAR Sections 1.3.3 and 1.4.1 contain information regarding switching or other means for the electrified third rail to follow changing TEV, DSEG, or other vehicle track routes without mechanical interference while the vehicles move between access mains and any one of the multiple emplacement drifts. The SAR contained no description of power provisions to nonaccessible Enhanced Characterization of the Repository Block (ECRB) cross drift and exhaust mains and shafts where maintenance activities will be conducted, according to BSC (2008ca). For the design of the normal electrical power system providing power to nonaccessible subsurface areas, the NRC staff evaluated the following aspects: (i) design and operational information, (ii) design codes and standards, and (iii) provisions for adequate preventive and corrective maintenance operations. The evaluation focused on whether the design could reasonably perform the stated functions as defined by DOE and whether the design would interfere with deployment of alternative designs, if needed, to support operations during the preclosure period. In addition, the NRC staff evaluated the design described in the SAR and supplemental materials to support potential operations and/or waste retrieval during the preclosure period. The NRC staff notes that applicable codes and standards were listed and that the design descriptions DOE provided are reasonable to evaluate the design and operation concepts of the normal electrical power system.

The NRC staff notes that the conceptual descriptions of the electrified third rail and power provisions for ROVs in nonaccessible subsurface areas are reasonable. The SAR and supplemental materials provide a high-level conceptual description of the electrified third rail intended to provide power to TEV, DSEG, and other planned ROVs operating in emplacement drifts and turnouts, and describe multiple, additional planned ROVs and related power provisions that are intended to enable operations in nonacccessible areas. DOE described the ROVs as “yet-to-be-designed or specified” (BSC, 2008ca), representing DOE’s statement to further develop ROV designs and related power provision alternatives for vehicles operating in all nonaccessible areas, during the detailed design phase.

The NRC staff considers that the ability of the TEV, DSEG, and multiple planned ROVs to perform assigned tasks in nonaccessible subsurface openings is a fundamental component of preclosure GROA operations and planned observation and inspection activities as described in the SAR and supplemental materials. The NRC staff notes that the capability to reliably perform observations and inspections and potential maintenance of nonaccessible underground openings and SSCs within them is evaluated in TER Section 2.1.1.2.3.7.3 in the context of DOE maintaining access to TEV, DSEG, and ROVs into emplacement drifts for operations throughout the preclosure period (BSC, 2008ca; DOE, 2009bb,ea,ef,gk). The NRC staff considers that any SSCs located within nonaccessible areas that are intended to provide power or support power connections to vehicles operating in nonaccessible areas are subject to monitoring and maintenance plans, addressing such matters as timely and safe repair, to be consistent with and support NRC staff’s evaluations in TER Section 2.1.1.2.3.7.3.

The DOE information in the final design should include a complete design description of electrical power SSCs located in nonaccessible areas, including design, construction, connections, and configurable switching (i.e., at turnouts) and plans for inspection, observation, and maintenance and repair that are needed to support operations of all ROVs in nonaccessible areas and require external power provisions (BSC, 2008ca). DOE’s descriptive information regarding the non-ITS electrified third rail related to nonaccessible areas, along with its statement to develop detailed design descriptions in the final design, and monitoring and maintenance plans evaluated in TER Section 2.1.1.3.7.3, permits a reasonable understanding of the use of the electrified third rail and other means of providing power to ROVs operating in nonaccessible areas.

Therefore, DOE’s description of the normal electrical power system, including the electrified third rail and power provisions for other ROVs, provides confidence that the normal electrical power system design will be designed in accordance with applicable codes and standards and accepted industry practices and is reasonable for use in the PCSA, as needed.

ITS Electric Power System

DOE 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 UPS SSCs. DOE 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. DOE 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 DOE intends to apply to the final design of the ITS electrical power system (DOE, 2009do). DOE 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. DOE further described how it intends to 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 GROA. The main loads for the ITS electrical power system are those loads for which a backup power source is needed to perform required safety functions. 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 HVAC SSCs, were also described as redundant, independent operating SSC trains, each powered independently by one of the two ITS electrical power system trains. Typically, one of the redundant ITS electrical power system trains and the related ITS operating SSC train are operational while the alternate ITS electrical power system train and ITS respective operating SSC train are in standby. 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 are automatically engaged so the associated safety function can continue to perform. Upon recognition of an LOSP, the ITS electrical power system automatically disconnects from the offsite power grid and begins 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 rely on the ITS electrical power system. During an LOSP, both ITS diesel generators are 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 prevents 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 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, are sized to have sufficient capacity to support required loads for 8 hours.

NRC Staff Evaluation: The NRC staff evaluated the design descriptions for the proposed ITS electrical power system using the guidance in the YMRP. The NRC staff focused on the information provided in the SAR and supplemental materials to support an evaluation of the electrical power system design and architecture concepts and related codes and industry standards as cited by DOE.

The NRC staff notes that the design descriptions DOE provided are reasonable to evaluate the design and operation concepts of the ITS electrical power system. A safety evaluation of the design of the ITS electrical power systems is included in TER Sections 2.1.1.6.3.2.8.4 and 2.1.1.7.3.6. DOE's description of the ITS electrical power system reasonably shows that the ITS electrical power system design will be designed in accordance with applicable codes and standards and accepted industry practices. The information is reasonable for use in the PCSA, as needed, with regard to functions of the ITS electrical power system.

2.1.1.2.3.2.4 Heating, Ventilation, and Air Conditioning and Filtration Systems

DOE described the 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, DOE described the HVAC systems for the balance-of-plant facilities in SAR Sections 1.2.8.3.1 and 1.2.8.3.2. DOE 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, DOE stated that the HVAC system shall ensure reliable confinement and filtration of radiological releases from event sequences that involve breach of a waste container or damaged SNF assembly. DOE classified as ITS the HVAC system components required to mitigate the consequences of a radioactive release following an event sequence and provide cooling to ITS equipment.

DOE described the HVAC systems for the surface facilities in terms of ITS or non-ITS subsystems serving confinement zones or nonconfinement zones. Note that design review and evaluation of the ITS portion of the HVAC systems are presented in TER Section 2.1.1.7.3.1.2.2. Secondary confinement (i.e., an area with a potential for airborne contamination during normal operations) was identified for the pool room in the WHF only. Both tertiary confinement (i.e., areas where airborne contamination is not 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 system consists 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 are equipped with a recirculation supply subsystem and an exhaust subsystem. From the description DOE provided in SAR Section 1.2.2.3.2, each facility is 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, HVAC system components 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 has the necessary instrumentation and control (I&C) listed in SAR Table 1.2.2-14.

DOE 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 DOE, the location and arrangement of the HVAC systems within the surface facilities ensure no interference with the safety functions of adjacent equipment and/or other systems. In addition, DOE described the operational processes for the HVAC system and potential interaction between the HVAC system and other SSCs or support systems (e.g., electrical power, fire protection, radiation monitoring, and alarm systems) in SAR Section 1.2.4.4.2.

DOE stated that, for structural design, ITS components of the HVAC system are designed for deadweight, live, constraint of free displacement, system operational transient, fluid momentum, and external loads, pressure differential, and seismic events. DOE stated that the load combinations used in the design analysis of ITS HVAC systems, with the exception of seismic loads, are in accordance with ASME AG-1–2003, including 2004 addenda (AG-1a–2004) Articles SA–4212, SA–4216 (Table SA–4216), BA–4131, AA–4212 (Table AA–4212), and HA–4212 (Table HA–4212) (American Society of Mechanical Engineers, 2004ac). In addition, DOE considered 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 TER Section 2.1.1.7.3.1.2.2. In addition, according to DOE, the non-ITS components of the HVAC systems are 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.

DOE 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 casings 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).

DOE 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 RAI, DOE 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 Evaluation: The NRC staff evaluated the surface HVAC system information using the guidance in the YMRP. The NRC staff notes that DOE provided reasonable information on HVAC system descriptions that included design and design analyses along with information on materials of construction, fabrication, and applicable codes and standards; discussions of potential interactions of the surface facilities HVAC systems with other SSCs, including electrical power, fire protection, radiation monitoring, and alarm systems; and a description of the location and arrangement of the HVAC systems within each surface facility. Additionally, DOE reasonably addressed the ability of the HVAC systems to withstand the effects of natural phenomena by considering load combinations due to natural phenomena in structural design analysis. DOE described ventilation confinement zoning for the surface facilities and the design of the HVAC systems to maintain flow from low to higher potential for radioactive contamination that the NRC staff notes is a standard design and is reasonable.

2.1.1.2.3.2.5 Mechanical Handling Equipment

DOE provided a description and discussion of the design of major ITS specialized and one-of-a-kind mechanical handling equipment to be used in the IHF, CRCF, WHF, and RF operations. This included the following five ITS mechanical systems the NRC staff considered to be representative in terms of specialized operation and functionality: (i) canister transfer machine (CTM), (ii) cask handling crane (CHC), (iii) SFTM, (iv) canister transfer trolley (CTT), and (v) WPTT. In addition, DOE 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 other ITS mechanical handling systems are more prevalent in the nuclear industry.

<u>Canister Transfer Machine</u>

The CTM is a special-purpose overhead bridge crane with two trolleys. The first is a canister hoist trolley with a grapple attachment and hoisting capability. The second is a shield bell trolley that supports a shield bell. The bottom end of the shield bell supports a motorized slide gate, which when closed provides bottom shielding of the canister once the canister is inside the shield bell. 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 are for the canister hoist trolley and the two outer rails are for the shield bell trolley. The design and operation of the CTM is the same for all facilities.

DOE described the CTM in SAR Section 1.2.4.2.2.1.3 and showed the plan and elevation view of the CTM with a defined clearance envelope in SAR Figure 1.2.4.50. DOE stated that the CTM is to be designed in accordance with ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa). For the overhead cranes, the CTM, and the SFTM, DOE 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 (IHF only), and collision.

DOE provided approximate dimensions of the CTM in BSC Section 6.2.2.12 (2008bg). 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, DOE relied on ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods.

NRC Staff Evaluation: The NRC staff evaluated the CTM system description using the guidance in the YMRP. The NRC staff notes that the description of the CTM DOE provided is reasonable because basic drawings, approximate dimensions, geometry, materials, codes, and standards have been reasonably described in the SAR, along with potential interactions of the CTM with other ITS SSCs. Therefore, the design information DOE provided is reasonable to evaluate the CTM system and functions for use in the PCSA and design, as needed.

<u>Cask Handling Crane</u>

The CHC is a large gantry crane with a rated payload capacity ranging from 181,437 to 272,155 kg [200 to 300 T]. The CHC is a top running, double-girder-type bridge crane with a top running trolley. The CHC is used in all four surface facilities (IHF, CRCF, RF, and WHF). DOE (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 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.

The CHC upends loaded canisters in the preparation area and place them on the CTT. DOE used ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa) as the design code and standard for the CHC. As stated in SAR Section 1.2.2.2.9.2.1, for overhead cranes such as the CHC, DOE 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 (IHF only), and collision. In the IHF, the CHC is rated at 272,155 kg [300 T]. For the CRCF, the main hoist is rated at 181,437 kg [200 T] with an auxiliary hoist rated at 18,144 kg [20 T].

NRC Staff Evaluation: The NRC staff evaluated the description of the CHC system using the guidance in the YMRP. The NRC staff notes that the description of the CHC system DOE provided is reasonable because basic design, mechanical drawings, geometry, materials, and codes and standards have been described in the SAR, along with potential interactions of the

CTT with other ITS SSCs. Therefore, the design information DOE provided is reasonable to evaluate the CHC system and its functions for use in the PCSA and design, as needed.

Spent Fuel Transfer Machine

The SFTM transfers SNF arriving in transportation casks and DPCs into spent fuel racks and into TAD canisters or alternatively to a staging rack in the pool. The SFTM is a bridge-type crane that spans the pool of the WHF and runs on rails on the edge of the pool. The trolley runs on a set of rails on the bridge. DOE described the SFTM in SAR Section 1.2.5.2.2.1.3. DOE 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.

DOE stated that the SFTM is to 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, DOE 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 (IHF only), and collision. DOE provided approximate dimensions for the SFTM in BSC Section 6.2.2.14 (2008bg). The minimum clearance between the top of the SFTM and the ceiling of the WHF is 0.6 m [2 ft]. A retractable camera is at the end of a 4.6 m [15-ft] pole that is attached to the SFTM. Finally, DOE relies on ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods.

NRC Staff Evaluation: The NRC staff evaluated the description of the SFTM system using the guidance in the YMRP. DOE's description of the SFTM is reasonable because basic design, mechanical drawings, geometry, materials, codes and standards, and operations have been described in the SAR, along with potential interactions of the SFTM with other ITS SSCs. Therefore, the design information DOE provided is reasonable to evaluate the SFTM system and functions and for use in the PCSA and design, as needed.

Cask Transfer Trolley

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 are used in the bottom of the CTT. The pedestal is loaded into the CTT using the CHC. DOE described the CTT in SAR Sections 1.2.4.2.1.1.3.1 (CRCF), provided mechanical drawings in SAR Figures 1.2.4-26, and provided a process and instrumentation diagram in SAR Figure 1.2.4-27. DOE 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. The CTT is to 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, DOE used specific design codes and standards that address the pneumatic valves, pressure relief valves, air cylinders, air bearings/casters, air motors, and piping of the CTT. These additional codes are 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); API 527 (American Petroleum Institute, 1991aa); and ASME B31.3–2004 (American Society of Mechanical Engineers, 2004ab).

DOE provided approximate dimensions for the CTT in BSC Section 6.2.2.7 (2008bg). DOE relied on ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods. The CTT is to 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 produce greater movements, energy-absorbing features are used to minimize the effect of impact forces on the cask and to prevent tipover. In SAR Section 1.2.4.2.1.9, DOE stated that the load combination analysis for the CTT is in accordance with ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa). The CTT used in the IHF is part of the cask preparation subsystem and is rated at 240,404 kg [265 T]. The CTT to be used in the CRCF, WHF, and RF is rated at 181,437 kg [200 T].

NRC Staff Evaluation: The NRC staff evaluated the description of the CTT system using the guidance in the YMRP. The NRC staff notes that the description of the CTT DOE provided is reasonable because basic design, mechanical drawings, geometry, materials, codes, and standards have been described in the SAR along with potential interactions of the CTT with other ITS SSCs. Therefore, the design information DOE provided is reasonable to evaluate the CTT system and its functions. The information is reasonable for use in the PCSA and design, as needed.

<u>Waste Package Transfer Trolley</u>

The 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]. DOE provided basic mechanical design details of the WPTT in SAR Figures 1.2.4-88, instrumentation and process diagrams in SAR Figure 1.2.4-89, and the WPTT logic diagram in SAR Figure 1.2.4-90.

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. DOE provided approximate dimensions for the WPTT in BSC Section 6.2.2.17 (2008bg). The WPTT is remotely controlled.

DOE 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 process and instrumentation diagrams provided in the SAR contain 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.

DOE also provided information on potential interactions of the WPTT with other ITS systems. The WPTT interacts primarily with two ITS SSCs: the CTM in the package positioning room and the TEV in the waste package load-out room. Interlocks between these systems ensure safe interaction of the WPTT with the aforementioned ITS SSCs. In SAR Section 1.2.4.2.4.2, DOE provided additional details on the interactions of the WPTT with other SSCs. In SAR Section 1.2.2.2.1, DOE also discussed the WPTT withstanding seismic ground motions.

NRC Staff Evaluation: The NRC staff evaluated the description of the WPTT system using the guidance in the YMRP. The NRC staff notes that the description of the WPTT DOE provided is reasonable because basic design, mechanical drawings, geometry, materials, codes and standards, and operations have been described in the SAR, along with potential interactions of the WPTT with other ITS SSCs. Therefore, the design information DOE provided is reasonable to evaluate the WPTT system and its functions and the information is reasonable for use in the PCSA and design, as needed.

<u>Other ITS Mechanical Handling Equipment</u>

DOE categorized other ITS mechanical handling equipment that is not 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, (vi) racks, and (vii) waste package closure subsystem.

The crane systems consist of standard cranes with varying load ratings and will be designed following ASME NOG-1–2004 requirements (American Society of Mechanical Engineers, 2005aa) for Type 1 cranes, 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.

Special lifting components such as cask yokes, canister grapples, and lifting beams are attached to the end of standard mechanical handling equipment to lift and transport casks, overpacks, or canisters containing waste. DOE stated that these components will be designed in accordance with ANSI N14.6–1993 (American National Standard Institute, 1993aa), as modified by NRC NUREG–0612 [NRC Section 5.1.1(4) (1980aa)]. DOE stated that the design of the special lifting components accounts for load drop prevention as well as load drop onto a cask or canister and incorporates design features to prevent unintentional load disengagement. DOE further stated that these special lifting components are commonly used in nuclear facilities.

Shield and confinement doors and sliding gates are intended to protect facility personnel from direct exposure. DOE stated that the shield and confinement doors will be designed in accordance with ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). DOE 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. DOE 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. In SAR Section 1.2.4.1.6, DOE stated that rails will be designed in accordance with ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa).

Platforms include multilevel steel structures that provide personnel and tool access to the top of aging overpacks or transportation cask. DOE will use the methods and practices provided in American Institute of Steel Construction (1997aa) to design the platforms and include seismic considerations. DOE described platform categories in SAR Sections 1.2.3.2.1.1.2 (for IHF), 1.2.4.2.1.1.3.1 (for CRCF), 1.2.5.2.1.1.3 (for WHF), and 1.2.6.2.1.1.3 (for RF). DOE provided relevant mechanical design drawings in SAR Figures 1.2.3-26 (for IHF), 1.2.4-41 (for CRCF), 1.2.5-33 (for WHF), and 1.2.6-17 (for RF).

Racks are intended to stage SNF assemblies and TAD canisters. DOE stated that racks will be designed in accordance with the applicable provisions of ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). DOE addressed the thermal safety of the SNF 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. DOE also included a thermal barrier that encloses the bottom and sides of the canisters so that the canister temperatures do not rise to unsafe levels in the event of a fire. DOE 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 waste package closure subsystem performs operations, such as closure, welding, nondestructive examination, inerting, and stress mitigation, and contamination surveys of the exposed portion of the waste package and decontamination if necessary. This subsystem is located in the IHF and CRCF. DOE described the waste package closure subsystem in SAR Section 1.2.4.2.3. DOE classified this subsystem as non-ITS, with the exception of the remote handling system bridge, which DOE classified as ITS. The waste package closure subsystem components are designed using the methods and practices of the following codes and standards: (i) ANSI/AWS A5.32/A5.32M–97 (American Welding Society, 1997aa), (ii) ASME B30.2–2003 (American Society of Mechanical Engineers, 2003aa), (iii) NFPA 801 (National Fire Protection Association, 2003aa), and (iv) ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa). The inerting of the waste package is performed in accordance with the applicable sections of NUREG–1536 (NRC, 1997ae).

NRC Staff Evaluation: The NRC staff evaluated the description of other ITS mechanical handling equipment using the guidance in the YMRP. The NRC staff notes that the description is reasonable because basic design, mechanical drawings, geometry, materials, codes, and standards have been described in the SAR. Therefore, the design information DOE provided is reasonable to evaluate the other ITS mechanical handling equipment and functions, and the information is reasonable for use in the PCSA and design, as needed.

2.1.1.2.3.2.6 Shielding and Criticality Control Systems

DOE described and discussed SSCs to be used for shielding and criticality control for the proposed GROA operations in SAR Sections 1.10.3 and 1.14.

Shielding

DOE described the shielding design of the surface facilities in SAR Section 1.10.3. The facility shielding is designed to reduce dose rates from radiation sources such that worker doses are within the standards of 10 CFR Part 20 and are 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 is specified to be in accordance with ACI–349–01/349R–01 (American Concrete Institute, 2001aa) and ANSI/ANS–6.4–2006 (American Nuclear Society, 2006aa).

DOE 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). DOE's shielding design objectives were taken from NRC Regulatory Guide 8.8 (NRC, 1978ab). The shielding design considerations DOE provided define 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 used for the shielding evaluation is Type 04 concrete based on ANSI/ANS–6.4–2006, Table 5.2 (American Nuclear Society, 2006aa). Other materials used in the shielding evaluation were described in SAR Sections 1.2.3 to 1.2.8.

DOE 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, DOE used flux-to-dose-rate conversion factors taken from ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa) to develop dose rates. DOE assessed the basic design regarding protection of workers and the public using commonly accepted industry codes and standards, such as MCNP and SCALE codes. DOE's shielding assessment was summarized for various areas and components in SAR Tables 1.10-35 through 1.10-46.

NRC Staff Evaluation: The NRC staff evaluated the information presented in the SAR using the guidance in the YMRP to determine the adequacy of the descriptions for the shielding system location, functional arrangement, and interactions with other SSCs within each facility. In addition, the NRC staff evaluated DOE's information by comparison with industry standards, such as American Nuclear Society (1991aa) and International Commission on Radiological Protection (1996aa), and using NRC guidance documents. The NRC staff considers DOE's description to be reasonable because the information shows DOE's shielding assessment used codes, standards, and methods that are consistent with NRC guidance. Therefore, the description is reasonable to evaluate DOE's proposed design and the PCSA (see also evaluations in TER Section 2.1.1.8.3.3).

Criticality Control

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

In SAR Section 1.14.2.3.2.4 DOE stated that the waste form characteristic of fissile isotope concentration was the only criticality control parameter important for HLW glass. The NRC staff's review notes that in SAR Table 1.14-2 DOE incorrectly described the waste form characteristics of HLW glass as characteristics that did not need to be controlled. The NRC staff considers this to be not significant, because the concentration of fissile isotopes in the HLW glass is less than or equal to that described in SAR Table 1.14-1. DOE does not rely on control of the waste form characteristics of other waste forms, but uses the parameters discussed next.

Moderation is the most important control parameter DOE used. With the exception of certain types of DOE SNF (the exceptions are discussed under Neutron Interaction), all canisters used in the GROA remain subcritical when unmoderated. DOE described and discussed the control of neutron moderation and stated that the guidance of ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac), which NRC endorsed in Regulatory Guide 3.71 (NRC, 2005ac), is followed.

Neutron interaction between canisters needs to be controlled in the CRCF, where neutron interaction among several 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 in other locations is prevented by physical barriers that make it impossible to have enough canisters in the same location that could result in criticality.

DOE 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 [2,500 ppm] of soluble boron enriched to 90 wt% B-10 is credited as the primary criticality control parameter. The soluble neutron absorber is in the form of orthoboric acid (H_3BO_3), which is to 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, DOE developed procedural safety control (PSC)-9 and plans to 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).

NRC Staff Evaluation: The NRC staff reviewed description and design information for the criticality control systems using guidance in the YMRP. The NRC staff also reviewed information in DOE's SAR and RAI responses (DOE, 2009dh) concerning the availability of sufficient enriched soluble boron. DOE would develop procedures for manual inspection of the boron concentration and enrichment, and manual operation of the boric acid makeup subsystem. The NRC staff notes that DOE's information in SAR Section 1.14.2.3.2.2.4 does not indicate any configurations that need more than 30 percent of the minimum required soluble boron concentration. This information suggests a large margin to protect against dilution and uncertainty in the amount of soluble boron in the pool. Therefore, the NRC staff notes that soluble boron is reasonable to control criticality.

Criticality Monitoring and Alarms

DOE did not use a criticality accident alarm system to control the consequences of a potential criticality event. Instead DOE relied on its screening of criticality as beyond Category 2 to justify that a criticality accident alarm system is not needed. DOE 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). This statement was supported by accounts of false alarms. The NRC staff reviewed these accounts (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.

NRC Staff Evaluation: The NRC staff reviewed DOE's information regarding criticality monitoring and alarms using guidance in the YMRP. The NRC staff notes that DOE does employ a radiation monitoring system. In TER Section 2.1.1.3.3.2.7.5, the NRC staff reviewed DOE's screening of initiating events that could lead to criticality. In addition, in TER Section 2.1.1.4, the NRC staff reviewed DOE's calculation of the probability of event sequences that may lead to an end state important to criticality. The NRC staff notes that DOE concluded, on the basis of the calculations, that there is a negligible probability of a criticality event. Therefore, it is reasonable that DOE did not propose to install a criticality accident alarm system.

2.1.1.2.3.2.7 Fire Safety Systems

DOE described and discussed the fire safety systems in SAR Sections 1.4.3.2 and 1.4.5.1.2. These systems included the site water supply and distribution systems and other active and passive fire protection systems. DOE stated in SAR Section 1.4.3.2.1.2 that the site has a dedicated fire protection water supply and two water distribution systems (Loops 1 and 2). The distribution systems are composed of site fire pumps designed and installed per NFPA 22, NFPA 20, 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 is configured as a redundant system (e.g., two tanks with associated pumps are provided in two separate locations to feed Loop 1), because Loop 1 supplies the more critical operational facilities. Loop 2 supplies GROA support facilities such as Administration, Security, and general warehousing areas.

DOE's fire hazards analysis documents indicated that the building fire suppression systems, including the site water supply tanks and stationary pumps, were designated as non-ITS.

DOE described the fire water effluent collection in SAR Section 1.4.5.1.2.2. The Liquid LLW Management system includes a series of local containment tanks at the individual buildings. The tanks were sized in accordance with NFPA 801 (National Fire Protection Association, 2008aa) to contain 30 minutes of overhead sprinkler effluent, along with sufficient freeboard. DOE also included containment volume for the largest anticipated vessel spill within each handling facility. Supplemental design data were provided in DOE's RAI response (DOE, 2009dm). DOE illustrated that the design volume of the effluent tanks includes 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 an approach consistent with NFPA 801.

As described in SAR Section 1.4.3.2.1, DOE stated that standard water-filled pipe systems will be provided over the majority of the facilities. These systems deliver water through a pressurized piping network and discharge the water through specific sprinkler heads (nozzles) in the vicinity of a fire. The system is driven by water pressure provided by the site fire pumps. The systems described in the SAR will be designed per nationally recognized standards such as NFPA 13 (National Fire Protection Association, 2007ab). The NFPA requirements govern 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. DOE indicated that an Ordinary Hazard Group 2 density will be used. This designation will establish the water density and head spacing requirements for the systems. Automatic suppression systems are not planned for the subsurface facilities.

DOE described the double-interlock preaction system in SAR Section 1.4.3.2.1. These sprinkler systems are a variation of a traditional wet pipe system and are used in areas where moderator control is required (e.g., IHF, WHF, RF, and CRCF). The piping network, pipe supports, and overall system components are identical to a wet pipe system and are all designed using the same national standard. The preaction system is charged with air, rather than water. A sequence of events that includes two independent forms of fire detection is necessary before water is discharged through the system. Once the interlocks are made, the system relies upon water pressure to deliver suppression water to the source. The fire detection component of the preaction system includes 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 is 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. The fire control and suppression aspects of these systems were not credited in the PCSA.

Standpipes and manual hose stations were described in SAR Section 1.4.3.2.1.2 and will be provided for local, manual fire suppression. DOE stated that these stations are designed and installed per NFPA 14 (National Fire Protection Association, 2007ac). The hose stations are constantly pressurized, and water is 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] hose and will be designed to flow 0.9 m^3/min [250 gpm] per outlet. The location and number of these outlets were not specified for the surface facilities, and there are no manual standpipes planned for the subsurface facilities.

Although design details such as specific location, hose travel distance, and interconnection of standpipes were not provided, the overall design and installation of standpipes in accordance with nationally recognized design standards will ensure the desired level of manual firefighting protection will be met. The standpipe system is a completely manual system and is 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 are manual systems, similar to the standpipe and manual hose systems described previously.

Although specific locations, travel distances, and individual unit sizing data were not provided in the SAR, DOE stated that units will be sized and installed per NFPA 10 (National Fire Protection Association, 2007aa). In addition, DOE designated the extinguishers as non-ITS.

As described in SAR Section 1.4.3.2.2.1, the TEV will be provided with a dedicated, pre-engineered suppression system to protect the unit from electrical fires. DOE referenced BSC (2007bf) in the SAR; however, specific design data were not in the reference. The NRC staff did not identify any event sequences that relied upon the on-board TEV suppression to mitigate consequences. The suppression system is non-ITS, because the consequence of such a fire is limited given the fail-safe design of the TEV.

As DOE described in SAR Sections 1.4.3.2.1.3 and 1.4.3.2.2.2, site fire alarm systems are primarily notification systems for building occupants and onsite/offsite fire and security personnel. The building fire alarm systems will also play a role in HVAC control and will provide key input to double-interlock preaction suppression systems. These systems are to be installed in accordance with 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) are designated as ITS. Non-ITS fire alarm systems were detailed in SAR Sections 1.2 and 1.3. These systems are traditional installations that are well described by the national standard. As non-ITS systems, no special design bases outside of the national standard are being applied. Additional discussion regarding the design data pertaining to the ITS fire alarm functions of the double-interlock preaction system is provided in TER Section 2.1.1.7.3.8.

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

Fire barrier subsystems are 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 is taken for these barriers in the subsurface areas.

Passive fire resistance-rated assemblies will follow recognized construction practices to achieve the intended fire resistance ratings. This requirement extends to fire barriers, openings in fire barriers (e.g., doors and windows), or penetrations through barriers (e.g., pipe and duct penetrations). 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 storage). Furthermore, DOE designated these passive fire resistance-rated assemblies as non-ITS, and they were not credited in the PCSA.

NRC Staff Evaluation: The NRC staff evaluated DOE's description of the fire safety systems and DOE's statement to update the SAR (DOE, 2009dm) using the guidance in the YMRP. The design information provided to describe the traditional wet sprinkler systems, double-interlock preaction systems, standpipes, fire alarm systems, and fire barrier system is reasonable because it is consistent with nationally recognized standards.

The NRC staff noted during its review that a supplemental fire alarm system role will be to control fire/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. Information DOE provided in response to an RAI on how the interlock between smoke detectors and fire/smoke dampers of air handling units impacts the reliability of the cooling function (DOE, 2009dm) indicated that the HVAC control function of the fire alarm system would conflict with the containment requirements of the HVAC system. In response to this RAI, DOE (DOE, 2009dm) stated that PCSA identified the interlock between the smoke detector and air handling units results in an unacceptable reduction in reliability of the ITS cooling function. Therefore, air handler automatic shutdown on duct smoke detection is not warranted for this facility and uninterrupted HVAC confinement would take priority in all cases. Duct smoke detection will remain and will provide notification functions only. Shutdown of the air handlers will require manual intervention and will only be performed if a survey of the confinement conditions deems it safe to do so. The SAR will be updated to reflect this hierarchy in HVAC control.

As stated in DOE (2009dm), DOE will update SAR Sections 1.2.4.4.2 and 1.2.8.3.1.2 and SAR Figures 1.2.4-105, 1.2.4-106, 1.2.4-109, 1.2.4-110, 1.2.5-88, 1.2.5-90, 1.2.8-28, and 1.2.8-33 to indicate that (i) automatic smoke damper interlocks in confinement HVAC systems will be omitted and smoke detectors in the EDGF will be designated as non-ITS; (ii) duct fire detection and smoke damper systems will remain in all code-mandated locations (however, the detectors will provide an alarm function only); and (iii) automatic interlocks for air-handler shutdown will be removed in favor of manual controls. The revision will provide operators greater control over the HVAC confinement features of the facilities while still providing the code-required alarm and manual control functions. Although the final design may not fully comply with NFPA 90A

(National Fire Protection Association, 2002aa), DOE states that this variance is necessary to ensure that the confinement takes precedence.

The NRC staff also notes that even though specific details of all systems were not provided, the reference to recognized standards ensures that a satisfactory level of design and engineering would be provided to these systems. The NRC staff notes that the description of the fire safety systems within the surface facilities is reasonable to evaluate the fire safety system and its functions and is reasonable for use in PCSA, as needed. Although there are limited fire protection features planned for the subsurface facility, the NRC staff notes that the description of the fire safety systems for the subsurface facility is reasonable to evaluate the system and its functions and the information is reasonable for use in the PCSA and design, as needed.

2.1.1.2.3.2.8 Piping and Instrumentation Diagrams

DOE described and discussed Piping and Instrumentation Diagrams (P&IDs) of surface facility process subsystems in SAR Sections 1.2.1 through 1.2.8. This included the description and the design details of 13 process subsystem P&IDs.

The NRC staff's review of the P&ID of a process subsystem focused on understanding the function, operation sequence, logic of layout of the subsystem components, safety significance of ITS components, and the subsystem's role in the system. The design description is evaluated by assessing the relevance and appropriateness of codes and standards DOE proposed in the subsystem design. A representative sample of P&ID component descriptions, design codes and standards, equipment layout and arrangement, process flow, piping connections, potential interactions among support systems, and pressure relief systems were reviewed. To the extent practical, the NRC staff reviewed subsystem design descriptions and subsystem requirements to ensure that the relationships between all the major components shown on P&IDs, schematics, and other source documents are in general agreement with the operations information for the subsystem.

<u>Cask Cavity Gas Sampling Subsystem</u>

The cask cavity gas sampling subsystem is in the IHF, CRCF, RF, and WHF, and the subsystem is similar in all facilities. The IHF cask cavity gas sampling subsystem is classified as non-ITS. The subsystem samples gaseous contents of a loaded transportation cask before it is opened. The gas sample is 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 consists of 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 collects gas samples for analysis of the gaseous fission products. DOE 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, DOE described the design methods and applicable codes and standards. DOE 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).

NRC Staff Evaluation: The NRC staff evaluated DOE's description and P&ID for the IHF cask cavity gas sampling using the guidance in the YMRP. The NRC staff compared the subsystem description, subsystem functions, location, and functional arrangement of major components with operational processes and for potential interaction with other subsystems. The description of the P&ID of the cask cavity gas sampling subsystem for the IHF is reasonable to evaluate the layout, function, operation, design, and potential interactions among support SSCs. Therefore, the information is reasonable for use in the PCSA, as needed. In addition, the NRC staff notes that the provisions of ANSI/ANS-57.7–1988 (American Nuclear Society, 1988aa) are relevant and applicable to this low risk non-ITS subsystem, although the American National Standards Institute/American Nuclear Society has withdrawn the standard. Therefore, the NRC staff notes that the standard is reasonable for use as DOE proposed.

Liquid LLW Sampling and Sump Collection Subsystem

The liquid LLW sampling and sump collection subsystem (SAR Section 1.2.3) is similar in IHF, 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. DOE classified this subsystem as non-ITS. The subsystem contains primary piping that transfers waste water effluents to the liquid LLW sampling tank. The effluents are pumped from the LLW sampling tank through a system of piping, valves, and pumps to trucks that transfer the effluents to the LLW collection tank. Sample lines on secondary piping contain an access port to grab sample for analysis. This subsystem is also designed to collect water from the fire suppression system. DOE provided the design basis requirements for this subsystem in SAR Section 1.2.4.3.2. In addition, DOE proposed to comply with Regulatory Guide 1.143, Table 1, excluding footnotes (NRC, 2001ab).

NRC Staff Evaluation: The NRC staff evaluated DOE's description and P&ID for the Liquid LLW subsystem using guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes for potential interactions with other subsystems. The description and design information and P&ID for the liquid LLW sampling and sump collection subsystem are reasonable to understand its design, layout, function, and operation and the information may be used in the PCSA, as needed. The NRC staff evaluated the excluded footnotes and notes the exclusions are reasonable, because the excluded provisions normally apply to HLW systems and are not applicable to this non-ITS Liquid LLW subsystem.

Waste Package Inerting Subsystem of Waste Package Closure System

The waste package inerting subsystem is part of waste package closure system, and the system is similar in the CRCF and IHF buildings. 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 dries the waste package and then pressurizes 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) and other P&I related to the waste package inerting process. DOE classified this subsystem as non-ITS. DOE proposed the following codes and standards for the design of 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); (ii) ANSI/AWS A5.32/A5.32M–97(American Welding Society, 1997aa); (iii) ASME B30.20–2003 (American Society of Mechanical Engineers, 2003aa); (iv) NFPA 801

(National Fire Protection Association, 2008aa); (v) ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa); and (vi) the inerting of the waste package in accordance with the applicable sections of NUREG–1536 (NRC, 1997ae).

DOE stated that the waste package and its closure system were selected for prototyping programs prior to their use in the repository. DOE also stated that its prototype testing program demonstrated the overall feasibility of 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 stated that it encountered problems during the demonstration of the inerting subsystem with seating purge port plug. Further, the temporary heating system located inside the inner vessel of the waste package mockup did not function properly, resulting in insufficient simulation of the expected temperature range. DOE recorded recommendations and lessons learned, and planned a retest. DOE stated that potential design modifications may be required. In addition, to clarify the performance capability of the waste package closure system, DOE will perform capability demonstrations (full system qualification testing) to ensure conformance with waste package safety criteria (DOE, 2009dr).

NRC Staff Evaluation: The NRC staff reviewed the information provided in the SAR and response to the NRC staff RAI (DOE, 2009dr) for the waste package inerting subsystem description and P&ID using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The description and design information for this subsystem are reasonable because the functions, locations and arrangement of components, codes, and standards for the subsystem were described in the SAR. Therefore, the NRC staff notes that the design information and P&ID DOE provided are reasonable to evaluate the waste package inerting subsystem functions and to use in the PCSA, as needed.

WHF Pool Water Treatment and Cooling System

DOE 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 are classified as non-ITS. In SAR Section 1.2.5.3.2, DOE provided information on functions, location, and components for the pool water treatment and cooling system that consists 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.

The pool water treatment subsystem removes crud and particulates using filters, radionuclides, and other ionic species; maintains optical clarity of pool water to allow identification of 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 includes cameras and sumps to monitor any leak.

DOE 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 Evaluation: The NRC staff evaluated the description and P&ID information for the pool water treatment and cooling subsystem using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The description and design information for the pool water treatment and cooling subsystem are reasonable because the functions, operations, design, and proposed codes and standards for the subsystem were described in the SAR. The NRC staff notes that the pool water treatment and cooling subsystem is commonly used in the nuclear industry and the standards DOE proposed are reasonable for this subsystem. Therefore, the design information DOE provided is reasonable to evaluate the pool water treatment and cooling subsystem functions and to use in the PCSA, as needed.

Cask Decontamination Subsystem

DOE 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 is classified as non-ITS, but the decontamination pit and seismic restraints are classified as ITS. The decontamination pit includes seismic restraints to ensure that the transportation cask or shielded transfer cask inside the decontamination pit is restrained to prevent tipover. ITS SSCs in the cask decontamination subsystem are designed in accordance with ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). The design of the decontamination pit and seismic restraints is in accordance with ANSI/AISC N690–1994 Sections Q1.2 (design methodologies) and Section Q1.4 (selection of appropriate material) and Table Q1.5.7.1 (meeting load combinations) (American Institute of Steel Construction, 1994aa).

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the cask decontamination subsystem using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The description and design information for the cask decontamination subsystem are reasonable because the functions, operations, design, and proposed codes and standards for the subsystem were described in the SAR. Therefore, the NRC staff notes that the design information DOE provided is reasonable to evaluate the cask decontamination subsystem functions and to use in the PCSA, as needed.

Cask Cooling and Filling Subsystem

DOE 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 DPCs and casks, and to fill the casks and 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 is located in the 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 is classified as a non-ITS system but has both ITS and non-ITS components. The pressure relief valves, which are used to implement the overpressure protection function, are classified as ITS components. DOE stated it will design the SSCs in the cask cooling and filling system using 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, 2004aa,ab).

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the cask cooling and filling subsystem using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. Though the cask cooling and filling subsystem P&ID (SAR Figure 1.2.5-69) identified an ITS function for cask overpressure protection, the P&ID figure title did not indicate the subsystem as ITS. This information is inconsistent with that discussed in BSC (2008bx). DOE recognized this inconsistency in SAR Section 1.2.5.3.4, which stated a subsystem is designated as non-ITS while portions of the structure or components in the subsystem are ITS. DOE stated in response to an RAI (DOE, 2009du) that it will address this 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 on 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 further identified similar inconsistencies (DOE, 2009ec) in SAR Chapter 1 and provided revised ITS designation for components and subsystems (see table in response for detailed list). DOE 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 notes that DOE's statement in response to a supplemental RAI (DOE, 2009ec) and the proposed revision to the relevant sections of the SAR are reasonable to show that DOE will apply the appropriate ITS classification process to make the SAR text, tables, and figures consistent.

The description and design information for the cask cooling and filling subsystem are reasonable because the functions, as well as relationships and interdependencies with other subsystems, were described. Therefore, the P&ID and design information DOE provided are reasonable to evaluate the cask cooling and filling subsystem functions and to use the information in the PCSA, as needed.

<u>WHF TAD Canister and Shielded Transfer Cask Drying and TAD Canister Inerting Subsystem</u>

In SAR Section 1.2.5.3.5, DOE described the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem. DOE also presented P&ID for this drying and inerting subsystem in SAR Figure 1.2.5-73. The system consists of a forced helium dehydrator or vacuum drying to drain and dry the TAD or shielded transfer cask when it is taken out of the WHF pool, at which time it is filled with pool water. The forced helium dehydrator is also used to inert the TAD canister filled with spent fuel in the WHF pool. If a vacuum system is used, the system consists of a vacuum pump, filter, and condenser, which dries the TAD/shielded transfer cask while it is removed from the pool and filled with pool water. In SAR Section 1.2.5.3.5.2, DOE described the operational process. In SAR Figure 1.2.5-73, DOE presented a simplified P&ID that shows components and their arrangement in this non-ITS subsystem.

The codes, standards, and regulatory guidance that DOE 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). These codes, standards and guidance are commonly used in the industry for the design of similar systems.

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The NRC staff notes that the description and design information for the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem are reasonable because the functions, layout of components, operations, and relationships and interdependencies with other subsystems were described. Therefore, the P&ID and design information DOE provided are reasonable to evaluate the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem functions and to use the information in the PCSA, as needed.

WHF Water Collection Subsystems

In SAR Section 1.2.5.3.6, DOE presented the description, function, and design information of the water collection systems in WHF. These systems consist of floor drains, collection tanks, and pumps to collect small amounts of water that are discharged or leaked from process SSCs, decontamination and wash water, and fire suppression water. There are two water collecting systems— C2, which collects normally noncontaminated water, and C3, which collects water that has the potential to be contaminated. The contaminated water will be transferred to LLWF for treatment. SAR Figures 1.2.5.-74 and 1.2.5-75 showed P&IDs for the C2 and C3 systems, respectively. These subsystems are classifies as non-ITS. DOE provided the codes and standards generally applicable for these systems in SAR Sections 1.2.3.3 and 1.2.4.

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information of the WHF water collection systems using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The NRC staff notes that the description and design information for the WHF water collection system are reasonable because the functions, as well as relationships and interdependencies with other subsystems, were described. Therefore, the P&ID and design information DOE provided are reasonable to evaluate the WHF water collection system functions and to use the information in the PCSA, as needed.

Emergency Diesel Generator Facility

The 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, DOE provided P&ID information for the following ITS subsystems of each train (Trains A and B are similar trains; only Train A is evaluated): (i) ITS diesel generator fuel oil system, (ii) ITS diesel generator air start system, (iii) ITS diesel generator jacket water cooling system, (iv) ITS diesel generator lubrication oil system, and (v) ITS diesel generator air intake and exhaust system.

ITS Diesel Generator Fuel Oil System

In SAR Section 1.2.8.2.1 and Figure 1.2.8-18, DOE provided a description of the design and P&ID information for the ITS diesel generator fuel oil system. The ITS diesel generator fuel oil system consists of an underground diesel fuel oil storage tank, from which fuel is 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 draws 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 gear-driven, positive-displacement 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. SAR Figure 1.2.8-17 showed the interface between ITS diesel generator Train A and the mechanical system that supports 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 Evaluation: The NRC staff evaluated information in SAR Section 1.2.8.2.1 on ITS diesel generator fuel oil system's purpose, function, operation, and design descriptions using the guidance in the YMRP. The NRC staff reviewed the general design information of the ITS diesel generator fuel oil system through reviewing the P&ID schematics described in SAR Figures 1.2.8-17 and 1.2.8-18. The codes and standards identified in SAR Section 1.2.8.2.1.8 for this system are commonly used in the nuclear power plant emergency diesel generator system, and the NRC staff considers them appropriate. The NRC staff notes that the information in the SAR is reasonable to evaluate the ITS diesel fuel oil system, its function, operation, and design. Therefore, the information is reasonable for use in the PCSA, as needed.

ITS Diesel Generator Air Start System Train A

In SAR Section 1.2.8.2.2 and Figure 1.2.8-20, DOE described the diesel generator air start system. 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 downstream of the ITS isolation gate valve are classified as ITS, and components upstream of the ITS isolation gate valve (the compressor, after cooler, and air dryer) are classified as non-ITS (SAR Section 1.2.8.2.2). The air receiver is maintained at operating pressure. The system alarms 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 P&ID. SAR Figure 1.2.8-21 showed the logic diagram for the ITS diesel generator air compressor.

The ITS diesel generator air start system is 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 Evaluation: The NRC staff evaluated the description and P&ID information for the diesel generator air start system Train A using the guidance in the YMRP. The description and design information for the diesel generator air start system Train A are reasonable because the function, operation, and logic of the layout of components have been described and the proposed design codes and standards are commonly used in the design of diesel generators in nuclear power plants. Therefore, the design and P&ID information DOE provided are reasonable to evaluate the diesel generator air start system Train A functions and to use in the PCSA and design, as needed.

The NRC staff notes that the P&ID did not clearly show the system interface and boundary between the ITS and non-ITS portions. In responding to an NRC staff RAI on lack of information on system interface and boundary between ITS and non-ITS portions, DOE stated (DOE, 2009du) that "additional details related to system interface system boundaries will be determined during detailed design." The NRC staff considers DOE's RAI response that addresses the concerns on system interface and boundary between ITS and Non-ITS portions in the detailed design is reasonable.

Diesel Generator Water Jacket Cooling System

In SAR Section 1.2.8.2.3 and Figure 1.2.8-22, DOE described the ITS diesel generator jacket water cooling system. 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 Figure 1.2.8-17). The system is designed such that the cooling water chemistry criteria are 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 (2007aa).

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the diesel generator jacket water cooling system using the guidance in the YMRP. The description and design information in the P&ID reasonably describe the function, operation, and logic of the layout of components of the diesel generator water jacket cooling system. The NRC staff identified an inconsistency in not designating the system as ITS when a subsystem or component is ITS. In response to an NRC staff RAI (DOE, 2009dq), DOE clarified its procedure for designating an ITS title to the system when it contains an ITS component and stated it will be addressed in the detailed design. The proposed design methodologies and design codes and standards for the diesel generator jacket water cooling system are commonly used in the nuclear industry and are deemed reasonable by the NRC staff. The NRC staff notes that the description, design, and P&ID information DOE provided on diesel generator jacket water cooling system are reasonable to evaluate the system and its functions. The information is reasonable for use in the PCSA, as needed.

Diesel Generator Lubricating Oil System Train A

In SAR Section 1.2.8.2.4, DOE described the ITS diesel generator lubricating oil system and its P&ID for Train A, which was presented in SAR Figure 1.2.8-23. 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 Evaluation: The NRC staff evaluated the description and P&ID information for the diesel generator lubricating oil system using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The codes and standards committed to be used in the design are commonly used in the nuclear industry and are deemed applicable.

DOE did not consider the design features that can detect and control system leakage and its consequences or describe it as part of the overall design methodology. Additionally, features that allow the system to be isolated from other portions of the system in the event of excessive leakage were not described or included in the design. ANSI/ANS–59.52–1998, Section 4 (American Nuclear Society, 1998aa) states that a gravity drain system is an acceptable alternative; however, consideration shall be given to potential system leakage and its consequence. Additionally, the scope of ANSI/ANS–59.52–1998 (American Nuclear Society, 1998aa) excluded engine-mounted components except to define interface requirements. DOE stated that it will address these issues in its detailed design (DOE, 2009dt) and listed proposed codes and standards for engine-mounted component design (DOE, 2009dt,ec). The NRC staff considers DOE's statement to address the NRC staff concerns in the detailed design to be reasonable because the diesel generator lubricating oil system is commonly used in nuclear facilities, and staff expects the detailed design to satisfactorily address these concerns. The NRC staff notes that the description, P&ID, and design information DOE provided are reasonable to evaluate the diesel generator lubricating oil system functions. The information is reasonable for use in the PCSA, as needed.

<u>ITS Air Intake and Exhaust System Train A</u>

In SAR Section 1.2.8.2.5, DOE described the ITS air intake and exhaust system Train A, and its P&ID was 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 allow air to be routed through intake piping, an intake filter, an in-line silencer, and a turbocharger. The system is designed to reduce the potential exhaust gas from entering through the air intake. For this reason, the exhaust piping is 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, DOE listed codes and standards for the design of the ITS air intake and exhaust system.

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the air intake and exhaust system using the guidance in the YMRP. The NRC staff compared the subsystem description, functions, location, and functional arrangement of major components with operational processes and for potential interactions with other subsystems. The description and design information for the air intake and exhaust system in the SAR are reasonable because the functions, layout of components, and operation were described. Additionally, the

NRC staff notes that the cited codes for the ITS air intake and exhaust system are routinely used in the design of diesel generators in nuclear power plants, and therefore their use in design is reasonable. Therefore, the design and P&ID information DOE provided are reasonable to evaluate the air intake and exhaust system functions. The information is reasonable for use in the PCSA and design, as needed.

On the basis of the evaluation discussed previously, the description and P&IDs of various surface facilities subsystems are reasonable. The NRC staff notes that DOE's information regarding the description and P&IDs of various surface facilities subsystems is reasonable for use in the PCSA and design, as needed.

2.1.1.2.3.2.9 Decontamination, Emergency, and Radiological Safety Systems

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

In SAR Section 1.2.1.3, DOE provided an overview of the decontamination systems. DOE 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 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, DOE described the cask decontamination subsystem of the WHF. The only ITS components of the cask decontamination system are the decontamination pit and the seismic restraints discussed in SAR Section 1.2.5.3.3.1.3. In SAR Section 5.7, DOE described the emergency plan to mitigate the consequences of a potential radiological accident. The emergency plan describes the safety systems to be put in place; specifically, these 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 emergency planning procedures provided in SAR Section 5.7 are evaluated in detail in TER Section 2.5.7. DOE's emergency and radiological safety systems were described in SAR Section 5.11 as part of the operational radiation protection program description. DOE 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 RMS are designated as non-ITS. The systems are used to monitor the surface and subsurface areas and effluents from the GROA release points. Monitoring equipment is intended to alert operators through a central monitoring station of any radiological release and potential Category 1 or Category 2 event sequences or conditions.

DOE stated it will perform 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). DOE stated that it will also design and use of 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 Evaluation: The NRC staff evaluated the description of the decontamination, emergency, and radiological safety systems using the guidance in the YMRP. The descriptions of the WHF decontamination and the emergency and radiological safety systems are reasonable because the functions of the systems and the design codes and standards have been described in the SAR. Therefore, the NRC staff notes that the description and design information DOE provided are reasonable to evaluate the WHF decontamination and emergency and radiological safety systems functions and the information is reasonable for use in the PCSA and design, as needed.

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 section, the NRC staff evaluates DOE'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 understanding the geometrical and other physical characteristics of the SSCs, their functions, and the design features DOE used to accomplish the functions. Functions of the subsurface facility openings and structures identified in the NRC staff's review are summarized in TER Table 2-1 and show how functions are linked to design features of the openings.

Table 2-1. Functions of the Subsurface Facility Structures Based on NRC Staff Evaluation of DOE Description of the Subsurface Facility Design

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

Table 2-1. Functions of the Subsurface Facility Structures Based on NRC Staff Evaluation of DOE Description of the Subsurface Facility Design (continued)		
Structure	**Functions**	**Controlling Design Features**
Turnout	(1) Limits radiation dose rate in the access main (2) Controls access to emplacement drift (3) Regulates air flow into the emplacement drift (4) Provides smooth elevation transition from access main to emplacement drift (5) Supports crane rails for the TEV and DSEG (6) Supports a third rail for power supply	(1) Curvature and length (2) Emplacement access doors (3) Air flow regulator; stability of roof and walls (4) Invert slope and elevation (5) Stability of invert (6) Stability of invert
Exhaust Main	(1) Exhaust conduit for heated air from emplacement drifts (2) Provide remote access for inspection and maintenance	(1) Stability of walls and roof (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 (2) Drip shield installation (3) Fresh air conduit for waste package ventilation	(1) Stability of roof and walls (2) Stability of shape and dimension of drift opening (3) Stability of drift opening
Emplacement Drift Invert Structure	(1) Foundation of crane rail and power supply third rail (2) Drip shield alignment and interlocking	(1) Stability of invert structure; serviceability of crane rail (2) Serviceability of crane rail

2.1.1.2.3.3.1 Subsurface Facility Layout and Development Plan

Subsurface Facility Layout

In the context of the GROA subsurface facility, DOE described the layout of subsurface facility structures in SAR Sections 1.3.1.1 and 1.3.2.2.1. According to DOE, the subsurface facility consists of nonemplacement and emplacement areas. The nonemplacement area consists 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 consists of a horizontal array of emplacement drifts divided into four panels (SAR Figure 1.3.1-1). Panel 1, the smallest panel, consists of six emplacement drifts in the central area of the subsurface facility and will be developed first as DOE stated. Each emplacement drift is 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 is connected to a ventilation exhaust main, which, in turn, is connected to an exhaust shaft (SAR Figure 1.3.1-1).

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

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to evaluate DOE's description of the subsurface facility layout. The design specifications DOE generated are consistent with the geometrical conditions that DOE used for PCSA and postclosure performance assessment. The NRC staff considers the following as key assumptions in DOE's proposed subsurface layout: standoff parameters, repository emplacement capacity, standoff distance from faults, cross-sectional diameter, horizontal spacing, invert grade, axis orientation of emplacement drifts, and grading of portals and shaft collar.

Subsurface Facility Development Plan

DOE described the subsurface facility development plan in SAR Section 1.3.1. DOE 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. DOE 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 includes a ventilation system that uses 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 Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the subsurface facility development plan. The information on the sequence and time estimate for drift development and waste emplacement, and on the duration of ventilation is reasonable because it enables the staff to evaluate the development schedule for the subsurface facility structures and to use in the PCSA, as needed.

Thermal Load Design

DOE described its approach to thermal management in SAR Section 1.3.1.2.5. DOE stated 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. DOE specified thermal load control parameters, including (i) maximum waste package thermal power at emplacement of 18 kW for a CSNF waste package or 11.8 kW for a naval 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. DOE stated that the actual waste stream that will be emplaced in the drifts depends on a number of variable and unspecified factors. DOE 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 Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the approach to managing the repository thermal load. The thermal load description DOE provided includes reasonable information regarding the target design and the approach DOE will use to determine whether a waste package arrangement satisfies the target design. Therefore, the information is reasonable to evaluate DOE's thermal load design and to use in the PCSA and postclosure performance assessment, as needed.

2.1.1.2.3.3.2 Nonemplacement Areas of the Subsurface Facility

DOE described the nonemplacement areas of the subsurface facility in SAR Section 1.3.3. DOE 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 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 TER Table 2-1. The functions support operation of ITS TEV or operations and activities, such as thermal management, that control parameter values that DOE used for postclosure performance assessment.

<u>North Portal and North Ramp</u>

DOE described the North Portal and North Ramp in SAR Section 1.3.3.1.1. The North Portal connects 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 consists 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 consists of a reinforced concrete slab with embedded anchor bolts to support crane rails for TEV and a third rail for power supply (SAR Section 1.3.3.4.1).

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the North Portal and North Ramp designs. On the basis of DOE's description, the NRC staff determines that the North Portal, North Ramp, and invert need to be sufficiently stable during the preclosure period to support functions listed in TER 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's evaluation of the North Portal and North Ramp designs' capability to perform the functions through the preclosure period is presented in TER Section 2.1.1.2.3.7. The NRC staff notes that the description of the North Portal and North Ramp designs in the SAR provides the basic geometry and layout, identifies the construction materials, and defines the intended functions. Therefore, the description is reasonable to evaluate the North Portal and North Ramp designs, to use in the PCSA, and to consider ventilation aspects of thermal load in postclosure performance assessment.

Access Mains

DOE described the access mains in SAR Section 1.3.3.1.2. DOE stated that the subsurface facility includes three access mains that 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 are connected to the emplacement drifts via turnouts that accommodate the turning radius of the TEV. Ground support for the access mains consists of fully grouted rock bolts and wire mesh, except at the intersections with the turnouts, where the ground support includes fully grouted rock bolts with steel-fiber-reinforced shotcrete and occasional lattice girders (SAR Section 1.3.3.3.1). The access main invert consists of a reinforced concrete slab with embedded anchor bolts to support crane rails for the TEV and a third rail for power supply (SAR Section 1.3.3.4.1).

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the access mains. On the basis of DOE's description, the NRC staff notes that the access mains and invert need to be sufficiently stable during the preclosure to support the functions listed in TER Table 2-1. The NRC staff evaluation of the capability of the access mains' design to perform the functions through the preclosure period is presented in TER Section 2.1.1.2.3.7. The description of the access mains in the SAR provides the basic geometry and layout, identifies the construction materials, and defines the intended functions. Therefore, the description is reasonable to evaluate the access mains' design, to use in the PCSA, and to consider ventilation aspects of thermal load in postclosure performance assessment.

Turnouts

DOE described the turnouts design in SAR Section 1.3.3.1.4. DOE stated that the turnouts connect the emplacement drifts to the access mains and contain facilities and equipment to control access and ventilation to the emplacement drifts. The turnout cross sections vary in shape and dimensions, from a rectangular section at the intersection with the access main to a circular section with a 5.5-m [18-ft] diameter at the intersection with the emplacement drift (SAR Figure 1.3.3-13). The invert of the turnout slopes up toward the emplacement drift. As described in SAR Section 1.3.3.1.4, the invert slope increases from 1.35 percent at the access main intersection to a maximum of 1.75 percent and decreases thereafter to 0 percent at the intersection with the emplacement drift. DOE stated in SAR Section 1.3.3.1.4 that the curvature and length of the turnout are 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 varies 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 consists of fully grouted rock bolts with steel-fiber-reinforced shotcrete and occasional lattice girders. The rock bolts have a nominal length of 5 m [16.4 ft] and are spaced in a square-grid pattern at 1.25-m [4.1-ft] centers. The shotcrete is 0.1 m [0.3 ft] thick. Ground support for the other turnout segments consists of stainless steel friction-type rock bolts with stainless steel welded wire fabric. The rock bolts have a nominal length of 3 m [9.8 ft] and are spaced in a square-grid pattern at 1.25-m [4.1-ft] centers. The fabric is W4 × W4 with 75-mm [3-in] center-to-center wire spacing. The turnout invert consists of reinforced concrete in the segments closest to the access main and carbon steel invert structure with ballast in the segments closest to the emplacement drift entrance (SAR Section 1.3.3.4.1).

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the turnouts. On the basis of DOE's description, the NRC staff determines that the turnouts should satisfy the following geometrical configurations: (i) the turnout curvature and length should be sufficient to protect the access main from radiation and (ii) the turnout invert slope should provide a smooth transition from the invert of the access main to the invert of the connected emplacement drift. Also, the turnout invert and cross section should be sufficiently stable during the preclosure period to (i) support crane rails for transportation of waste packages or drip shields and a third rail for power supply, (ii) provide the operating envelope for the DSEG, (iii) provide access to the connected emplacement drift, and (iv) function as a fresh air conduit for ventilation of disposed waste packages. The NRC staff's evaluation of the turnouts design's capability to perform the functions through the preclosure period is presented in TER Section 2.1.1.2.3.7. The description of the turnouts in the SAR provides the basic geometry and layout, identifies the construction materials, and defines the intended functions. Therefore, the description is reasonable to evaluate the turnouts' design, to use in the PCSA, and to consider ventilation aspects of thermal load in postclosure performance assessment.

Exhaust Mains

DOE described the exhaust mains design in SAR Section 1.3.3.1.3. DOE stated that each exhaust main connects 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 connects to an access main via a turnout. According to DOE, the exhaust mains have the same diameter as the access mains {i.e., 7.62 m [25 ft]}, except the exhaust main for Panel 1 has a diameter of 5.5 m [18 ft]. DOE will use an isolation barrier where an exhaust main and access main 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, 3-west, and 1 have separate but closely spaced exhaust mains. DOE explained that separate exhaust mains are needed to allow concurrent development in Panel 4 and waste emplacement in Panel 3-west or development in Panel 4 adjacent to a waste-loaded Panel 1. According to DOE (DOE, 2009dm), the majority of the exhaust main lengths for Panels 4, 3, and 1 are parallel and have a centerline-to-centerline spacing of approximately 22.9 m [75 ft]. The primary function of an exhaust main is to remove hot ventilation air from the repository during the preclosure period (SAR Section 1.3.3.1.3). Thus, the exhaust main plays a key role in DOE's thermal management strategies to satisfy DOE's thermal performance requirements. According to DOE, 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 DOE stated. DOE explained in SAR Section 1.3.1.2.1.6 that the exhaust mains, like the exhaust shafts, will be nonaccessible because of high temperature and potential high radiation levels.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the exhaust mains. On the basis of DOE's description, the NRC staff determines that 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 understanding of the functions of the exhaust mains is summarized in TER Table 2-1. The NRC staff's evaluation of the exhaust mains' design capability to perform the functions through the preclosure period is presented in TER Section 2.1.1.2.3.7. The description of the exhaust mains' design in the SAR and subsequent information provided to respond to an NRC staff RAI (DOE, 2009dm) provide the basic geometry and layout, identify the construction materials, and define the intended functions. Therefore, the description is reasonable to evaluate the exhaust mains' design, and use in the PCSA, and to consider ventilation aspects of thermal load in postclosure performance assessment.

<u>Shafts and Ventilation Raises</u>

DOE described the design of shafts and ventilation raises in SAR Section 1.3.3.1.5. DOE stated that the subsurface facility includes three intake shafts and six exhaust shafts (SAR Figure 1.3.3-8). The shafts 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. DOE 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; DOE implied the concrete liner will be stress free. As DOE described in SAR Section 1.3.3.3.1, ground support for the shaft base where the shaft intersects 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. DOE stated in SAR Section 1.3.1.2.1.6 that the exhaust shafts will be nonaccessible 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 DOE, the subsurface facility also includes two short vertical openings, referred to as raises, as described in SAR Section 1.3.3.1.5. DOE stated that one raise will connect the exhaust main of emplacement drift Panel 1 to the 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 Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of shafts and ventilation raises. On the basis of DOE's description, the NRC staff determines that 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 TER Table 2-1. The NRC staff's evaluation of the shafts' and raises' design capability to perform the functions through the preclosure period is presented in TER Section 2.1.1.2.3.7. The NRC staff determines that the description of the shaft and raises in the SAR provides the basic geometry and layout, identifies the construction materials, and defines the intended functions. Therefore, the NRC staff determines that the description is reasonable to evaluate the shafts' and ventilation raises' designs and to use in the PCSA and postclosure performance assessment.

Subsurface Facility Openings Dedicated to Performance Confirmation

DOE described the design of underground openings dedicated to performance confirmation in SAR Section 1.3.3.1.6. DOE stated that the subsurface facility will include an observation drift and three alcoves dedicated to performance confirmation. The observation drift and one alcove are 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 extends under Panel 1 and ramps 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 extends southward under the emplacement drift as shown in SAR Figure 1.3.3-18. DOE stated that the observation drift and alcove will be used to install 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 DOE, 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. DOE 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 Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the underground openings dedicated to performance confirmation. On the basis of DOE's description, the NRC staff determines that the rock pillar overlying the observation drift and alcove under Panel 1 needs to be sufficiently stable during the preclosure period to provide space and a platform for instrumentation and equipment needed to monitor the rock pillar under the emplacement drift for performance confirmation. The NRC staff's understanding of the functions of the observation drift and alcove is summarized in TER Table 2-1. 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 TER Section 2.1.1.2.3.7. The description of the observation drift and alcoves in the SAR provides the basic geometry and layout, identifies the construction materials, and defines the intended functions. Therefore, the description is reasonable to evaluate the observation drift and alcoves designs, to use in the PCSA, and to consider ventilation aspects of thermal load in postclosure performance assessment.

On the basis of the NRC staff's evaluation discussed previously, the NRC staff notes that DOE's description of underground openings in the nonemplacement areas of the subsurface facility is reasonable for use to evaluate DOE's design, PCSA, and postclosure performance assessment.

2.1.1.2.3.3.3 Emplacement Areas of the Subsurface Facility

DOE described the emplacement areas of the subsurface facility in SAR Section 1.3.4. DOE stated that the emplacement areas of the subsurface facility 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 is connected to an access main via a turnout, and the opposite end is connected to an exhaust main. Each emplacement drift consists of the drift opening, ground support for stabilizing the immediately surrounding rock, and an invert that carries 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). DOE stated that this drift orientation favors drift stability considering the prevalent orientation of rock joints. According to DOE, the drifts will be laid out in a parallel pattern and spaced 81 m [266 ft] horizontally from centerline to centerline. DOE 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 has a circular cross section with a nominal excavated diameter of 5.5 m [18 ft] (SAR Figure 1.3.4-4). DOE 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 DOE-specified ventilation efficiency. Other features of the emplacement drift design described in SAR Section 1.3.4.2.3 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 DOE, geologic mapping of drifts will include documentation of fractures, fault zone characteristics, stratigraphic contacts, and lithophysal content.

DOE described the emplacement drift ground support in SAR Section 1.3.4.4.1. According to DOE, an initial ground support and a final ground support will be installed in each emplacement drift. The initial ground support consists 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 consists 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 are 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. DOE stated that the emplacement drift ground support is 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 consists of a steel invert structure and crushed tuff ballast fill (SAR Section 1.3.4.5). The steel invert structure consists 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 are spaced 1.5 m [5 ft] center to center and bolted to the longitudinal beams. The two outermost longitudinal beams are attached to stub columns that transfer loads to the drift floor. The stub columns are anchored to the underlying rock. In addition, the ends of the transverse beams are attached to plates that are 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 are mounted on the two outer longitudinal beams, also referred to in the SAR as rail runway beams. DOE stated in SAR Section 1.3.4.5.1 that the steel invert structure and crane rail will be fabricated from corrosion-resistant steel. DOE also mentioned a third rail that will be used for 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. DOE stated that the top of the ballast will coincide with the top of the steel structure.

The steel invert structure provides 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 also functions as the foundation for the crane rail system for operation of the TEV, DSEG, and remote-controlled inspection vehicle.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's description of the emplacement areas of the subsurface facility. On the basis of the description, 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 envelop 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 understanding of the functions of the emplacement drifts, invert structure, and ground support is summarized in TER Table 2-1. 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 TER Section 2.1.1.2.3.7. The description of the emplacement drift, invert structure, and ground support designs in the SAR provides the basic geometry and layout, identifies the construction materials, and defines the intended functions. Therefore, the description is reasonable to evaluate the design of underground openings in the emplacement areas of the subsurface facility and to use in the PCSA and postclosure performance assessment.

On the basis of the NRC staff's evaluation discussed previously, the NRC staff notes that DOE's description of the emplacement areas of the subsurface facility is reasonable for use to evaluate DOE's design, PCSA, and postclosure performance assessment.

2.1.1.2.3.3.4 Waste Package Transportation and Emplacement System

DOE described and discussed the TEV design in SAR Sections 1.2 and 1.3. DOE used this information in the PCSA and iterative design of the TEV (SAR Sections 1.6 through 1.9). DOE designated the TEV as ITS.

<u>General Description of TEV and Functions</u>

DOE 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. DOE 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 to the surface facility. DOE also proposed to use the TEV for retrieval operations, if needed, by reversing the emplacement operations. DOE 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, DOE provided the layout of the surface rails that illustrated the specific routes of the TEV at the surface (SAR Figures 1.2.1-1 to 1.2.1-3). DOE also provided general descriptions related to the role of the TEV in these surface facility areas, such as the CRCF and IHF. When appropriate, DOE 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, DOE 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. DOE discussed the subsurface crane rail, which is an integrated rail system that connects the IHF and CRCF buildings to the subsurface emplacement areas. DOE provided detailed location, length, direction, and magnitude of expected slopes that the TEV is designed to travel. It also included the specification for turning radii of 61 m [200 ft] or larger within the subsurface facilities to allow TEV travel without binding the wheels. DOE 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 Evaluation: The NRC staff reviewed the general description of the TEV using the guidance in the YMRP. The NRC staff also reviewed DOE'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 of 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 tipover initiating events, (v) compute throughput, and (vi) determine bounding values for TEV component reliability calculations. On the basis of this review, the NRC staff notes that DOE, through proper diagrams, delineated sufficient sequences of TEV activities in the surface facilities to reasonably understand TEV operations at the GROA. DOE provided reasonable information on the route lengths, grades, curvature, activities, and interactions with other SSCs necessary to support a TEV design review. Therefore, DOE's TEV description provides reasonable information to evaluate the TEV functions and is therefore reasonable for use in the PCSA and design, as needed.

<u>TEV Design Information</u>

DOE described the TEV conceptual design in 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) layout of drive motors, lifting motors, shielding enclosures, shield doors, ITS mechanical switch, extendable base plate, third rail power with shielded conductors, sensors (speed, range, and temperature), fire-suppression system, communication devices, PLCs, interlock switches, and video cameras.

DOE stated that the electric drive motors and the lifting jacks were selected with a type of gearing unit that prevents the load from back-driving the units under runaway or loss of power conditions. DOE 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. DOE 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 are more difficult to repair inside the radiation environment of the drifts.

In addition to design information in the SAR, DOE provided other supplemental references; in particular, TEV drawings, dimensions, weight, materials of construction, and subsystem specifications (BSC, 2008bz,cb, 2006aj). DOE also discussed appropriate industry codes and standards for the wheel and rail design. An example of a 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 Evaluation: The NRC staff reviewed the TEV design description information using the guidance in the YMRP. On the basis of the information provided in the SAR and the additional information included in the supplemental documents, the NRC staff notes that DOE provided reasonably sufficient details on TEV design, functions, codes, and standards because the details enable an understanding of the TEV design description needed for the TEV design evaluation presented in TER Section 2.1.1.7.

<u>TEV Structural and Thermal Analysis</u>

DOE provided calculations performed to size TEV components. DOE also discussed the methodology employed, the key TEV performance computations, and the specifications resulting from the analyses. DOE utilized 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 the calculations.

DOE included an analysis that evaluated the impact of a collision between the TEV and an emplaced waste package in BSC (2007cd). The ANSYS®, an industry-accepted simulation software, results were presented to demonstrate 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) 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. DOE’s analyses used methods that are common in the engineering community to define boundary conditions and to derive reasonably accurate simulation results that could be validated during the “live load” confirmation program.

DOE 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 the thermal parameters (i.e., density, conductivity, and specific heat) of the different constituents of the shielding enclosure materials. The thermal analyses, which were performed using ANSYS®, provided sufficient basis for

DOE's quantification of the temperature harshness within the emplacement drifts. DOE utilized 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 Evaluation: The NRC staff reviewed the description of structural and thermal analyses using the guidance in the YMRP. The NRC staff verified that DOE included computation of expected TEV chassis frame deflections under waste package and shielding weight. In addition, DOE provided 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 description of the analyses is reasonable because it (i) included the methodology applied using either industry-accepted standards or simulation packages, (ii) utilized reasonable boundary conditions, and (iii) resulted in bounding parameters and specifications needed for the proper TEV design. Therefore, the description is reasonable to evaluate the structural and thermal analysis of TEV.

2.1.1.2.3.3.5 Waste Package Emplacement Pallet System

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

DOE 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 have a single design (SAR Figure 1.3.4-13) that consists of two waste package supports containing V-shaped cradles to accommodate all waste packages diameters and is connected by square tubes. According to SAR Figures 1.3.4-11 and 1.3.4-12, the waste package emplacement pallet length is 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 is 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 is 1,970 kg [4,340 lb] as reported in SNL (2007ap).

In BSC (2007ca), DOE stated that the waste package emplacement pallet is considered a Class 2 vessel plate-type support and its design is 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, DOE considered 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 is Alloy 22, as outlined by the American Society of Mechanical Engineers SB–575, UNS N06002 (2001aa), which was chosen to provide only Alloy 22-to-Alloy 22 contact surfaces for the waste packages. According to DOE, 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 is Stainless Steel Type 316, as described by the American Society of Mechanical Engineers SA–240, UNS S31600 (2001aa). According to DOE, this was selected because the general corrosion rate of the stainless steel tubes in the repository-relevant environment is low.

DOE stated that the waste package emplacement pallet would 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; and 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 Evaluation: The NRC staff evaluated the description of the waste package emplacement pallet using the guidance in the YMRP. The description of the waste package emplacement pallet is reasonable because the dimensions, weights, materials, fabrication, functional features, design analyses, and applicable codes and standards were appropriately described. Therefore, the design information DOE provided is reasonable to evaluate the waste package emplacement pallet functions in the PCSA and design, as needed.

2.1.1.2.3.3.6 Drip Shield Emplacement System

DOE described the DSEG design in SAR Section 1.3.4.7.2. DOE designated the DSEG as non-ITS because it is not 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, DOE described the functions of the DSEG that included 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 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 subsequent slow crawl speed of 0.5 m/min [1.5 ft/min], and (vi) returning to the surface facility and to repeat the process.

DOE described the DSEG design as a self-propelled, rail-based crane, which is similar to the TEV based on nuclear and industrial crane technology. The main components of the DSEG included (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 PLC network that communicates with the Central Control Center 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) I&C containing articulated cameras, ultrasonic sensors, and high-intensity lights. DOE 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). DOE also provided the specific routes for the DSEG from the surface to the subsurface facility. Furthermore, DOE described the motion of the DSEG including stops, inspection, and calibration of its precise position, which closely resembles the TEV operational sequence toward the subsurface.

DOE 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 height of 102 cm [40 in] lift. DOE also specified that additional clearance is required at different locations of DSEG operations with appropriate justification.

DOE further described the DSEG in BSC Section B.4.2 (2008bk) in support of PCSA. BSC Table B4.3-1 (2008bk) listed dependencies and interactions with other SSCs. DOE's description provided design similarities between the DSEG and the TEV. The DSEG relies on a number of infrastructure components, such as the subsurface crane rail {85 kg/m [171 lb/yd]}, several rail switches, radiation monitoring equipment located near the access doors near the access main and toward the North Portal, redundant third rail power, a communication system with the CCCF, and subsurface ventilation. DOE used 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 Evaluation: The NRC staff reviewed the description of the DSEG and its installation operations as described in SAR Section 1.3.4.7.2 and the supporting documents (BSC, 2007ca,cg–ci) using the guidance in the YMRP. On the basis of this information, the NRC staff notes that DOE reasonably identified the dimensions to compare (i) envelopes among the DSEG, the drip shield, and the drift openings and (ii) the relationship between the DSEG motion and the interlocking requirements of the drip shields. DOE provided information to explain the DSEG operations, which 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) define different design speeds during distinct modes of operations.

Due to the similarities between the DSEG design and the TEV design, the NRC staff notes that the information provided in the SAR and supplemental documents was reasonable because it described the functions, design requirements, considerations for the thermal and radiological severity of the environment, redundancy of electrical drive power, measures to prevent derailment, monitoring and control systems, and consistency with accepted industry standards. Therefore, the DSEG description is reasonable to evaluate the DSEG design and operation to use in the PCSA and postclosure performance assessment.

2.1.1.2.3.4 Description of Waste Form Characteristics

DOE provided information on the kind, amount, and specification of the radioactive material proposed to be received and possessed at the GROA. The objective of the NRC staff review is to evaluate whether sufficient information is presented in the SAR for use in the PCSA. LLW produced as a result of GROA operations will be temporarily stored onsite, although it will not be disposed at the GROA.

2.1.1.2.3.4.1 High-Level Radioactive Waste Characteristics

DOE described 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). DOE discussed how it uses the CSNF characteristics in the PCSA (SAR Section 1.8.1.3.1) and creates representative CSNF characteristics for DOE's ALARA analysis (SAR Section 1.10.3.4.1).

DOE described 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 SNF consists of SNF from numerous test and research reactors {2.3×10^6 kg [2,265 MTHM]} and naval SNF {65,000 kg [65 MTHM]}. DOE 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.

<u>Physical Characteristics of Radioactive Waste</u>

The CSNF inventory of the repository is 63×10^6 kg [63,000 MTHM]. SAR Tables 1.5.1-2 and 1.5.1-3 summarized physical characteristics of PWR and 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 are retained in the UO_2 matrix, but some of the more mobile fission products and activation products accumulate in gap regions and grain boundaries. In SAR Section 2.3.7.7.3.1, DOE discussed these isotopes, which are available for instantaneous release when the cladding is breached. DOE used the rod breakage fraction of CSNF in evaluating radionuclide isotopes used in normal and accident conditions.

HLW glass is highly radioactive waste that has been mixed with silica and/or other glass-forming chemicals that are melted and poured into canisters where they solidify 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). DOE used 500 kg [0.5 MTHM] per canister equivalence for DOE HLW to determine how many canisters can be disposed of at the repository. DOE expects to receive approximately 9,300 DOE HLW canisters containing a total of 4.7×10^6 kg [4,667 MTHM]. A 2,300 kg [2.3 MTHM] per canister equivalence is used for the 275 commercial HLW glass canisters from West Valley with a total of approximately 640,000 kg [640 MTHM] of HLW.

DOE SNF waste form comes from a range of backgrounds with a variety of fuel types, moderators, enrichments, shapes, and chemistries. The approximately 2.3×10^6 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, DOE developed 34 groups of DOE SNF, including naval SNF, that are based on the characteristics DOE believes have the greatest impact on release and criticality. SAR Table 1.5.1-23 listed these 34 fuel groups and described how they are analyzed. SAR Table 1.5.1-24 described the ranges of the properties of the 34 groups.

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 CSNF. DOE stated in SAR Section 1.5.1.4.1.1, “In a few cases …. Naval Spent Fuel has nonintact cladding……”. DOE modeled Naval SNF as CSNF, which does 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 Evaluation: The NRC staff reviewed DOE’s description of physical characteristics of HLW using the guidance in the YMRP.

Records of spent fuel characteristics, maintained by NRC-licensed power reactors, are reviewed by the NRC staff. DOE reasonably described the waste form composition and amount and storage unit of material because (i) the amount of waste DOE intends to dispose of at the repository is accounted for in the same MTHM unit as that in the Nuclear Waste Policy Act and (ii) the amount allocated for disposal meets the limit of 7×10^7 kg [70,000 MTHM].

The description of the CSNF physical characteristics was based on data NRC licensees supplied. The description of the amount of CSNF is reasonable to the NRC staff because the amount allocated for disposal, when combined with other types of waste, is less than the Nuclear Waste Policy Act-mandated limit of 7×10^7 kg [70,000 MTHM]. DOE’s use of rod breakage fractions is consistent with Interim Staff Guidance (ISG) 5 p. 7 (NRC, 2000af). Although DOE 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. Therefore, the NRC staff considers not taking credit for burnable poison absorbers to be conservative. DOE bounded the cooling time with PSC-20, which requires at least 5 years’ cooling time as per 10 CFR Part 961. Therefore, the NRC staff considers this specified cooling time reasonable.

The NRC staff reviewed DOE’s description of grouping DOE SNF waste forms in the SAR and supporting documents (DOE–Idaho, 2000aa) and notes that DOE’s description of the DOE SNF groupings is reasonable because these groupings were based on the fuel properties that were most important to the design and safety analyses. DOE’s description of the range of waste form characteristics is reasonable because the range of the waste form characteristics includes parameters ITS and repository performance. The NRC staff also notes that creating groups of similar types of DOE SNF and using representatives of the groups for analysis purposes is reasonable.

The cladding of naval SNF is not relied upon for safety. Because DOE models naval SNF as CSNF, which does not credit cladding, the NRC staff notes that the small amount of nonintact cladding in naval SNF is not risk significant. Therefore, DOE's description of damaged fuel cladding is reasonable.

<u>Thermal Characteristics</u>

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. DOE chose to analyze limiting values so that the uncertainties are bounded by the maximum cases.

DOE calculated the heat generation rate from the HLW radionuclide inventory and displayed the results in SAR Table 1.5.1-19. DOE imposed a limit that the maximum allowable canister temperature and maximum allowable heat generation rate are 400 °C [752 °F] and 1.5 kW/canister, respectively. HLW canisters that do not meet this limit will not be disposed of (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. DOE imposed 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 (SAR Table 5.10-3).

According to DOE, 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 canister surface in the emplacement drift should be less 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 canister as a function of time after emplacement.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of thermal characteristics and heat generation rate using the guidance in the YMRP and notes that DOE's description is reasonable because the DOE-provided description enabled the NRC staff to evaluate the CSNF heat generation rate. In addition, the thermal characteristics include conservatism to bound uncertainties to permit an evaluation of thermal calculations. Because all canisters that can be used for disposal have thermal characteristics less than the aforementioned DOE-imposed limits, the actual case will be bounded by DOE's thermal calculation results.

<u>Nuclear Characteristics</u>

DOE used the SCALE computer code 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 DOE used for its HLW glass consequence analysis.

DOE used the ORIGEN code to develop a template of radionuclide inventories, at 10-decay intervals between 5 and 100 years, for typical SNF, which were 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, DOE 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. DOE 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 Evaluation: The NRC staff reviewed DOE's description of the radionuclide inventories using the guidance in the YMRP, with a focus on the methods DOE used to generate the radionuclide inventories and conservatisms in the models and calculations. The NRC staff considers the use of SCALE to be reasonable because it is an industry-accepted code. DOE's description of the radionuclide inventories to be reasonable because proper use of SCALE would provide a useable model of the radionuclide inventory.

Source Term Characteristics

DOE 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, DOE 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 is 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 DOE claims will bound at least 95 percent of the fuel inventory, and it is used in shielding calculations for some transfer shield designs to limit shield weight. For normal operation airborne releases, DOE 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.

DOE did not identify Category 1 event sequences for HLW glass. Potential doses to the public were discussed in SAR Section 1.8.3.2 and doses to workers in SAR Section 1.8.4. DOE stated that the maximum per canister inventories were provided in SAR Table 1.5.1-21 and used in the PCSA. DOE also stated that the Savannah River Site HLW canister represents 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 ensures that the dose limits in SAR Tables 1.8-30 and 31 are met.

DOE discussed its SNF shielding sources term characteristics in SAR Section 1.10.3. Neutron and gamma energy spectra and source intensity and fuel composition DOE used in shielding calculations for a homogenized TRIGA-FLIP fuel was 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, DOE discussed the gamma and neutron source terms for the naval SNF canisters, which DOE considered in the IHF design. The source term assumes a cooling time of 5 years, and it is 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. DOE did not develop a source term to analyze doses from a DOE SNF or naval canister breach, because it determined that breaches of these canisters were beyond Category 2 event sequences.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of the radiation source term using the guidance in the YMRP. The NRC staff notes that the description of the source term includes conservatisms and sufficient information to evaluate the shielding and dose calculations. The NRC staff's evaluation of the source terms is documented in TER 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 considers that dividing the DOE SNF into groups with only a representative of the group being analyzed is reasonable given that the other members of the DOE SNF groups will still be subjected to DOE's waste form and waste package qualification program, as specified in SAR Table 5.10-3.

DOE made numerous assumptions about the characteristics of the waste forms, which are provided in various parts of the SAR. The NRC staff's evaluation relies on these key assumptions on limiting values and/or bounding values of waste form characteristics. In addition, the waste form characteristics evaluated in this section formed the bases for DOE's PCSA. To ensure the validity of the PCSA results for the preclosure period, DOE made the following key assumptions on the characteristics of emplaced waste: (i) at least 5 years of cooling time are needed for CSNF; (ii) the heat generation rate is provided in SAR Table 1.5.1-11 and limits on cladding and canister temperature are listed in the SAR; and (iii) source term characteristics are provided in SAR Tables 1.14-1, 1.8-5, 1.10-23, 1.10-21, 1.10-22, 1.8-6, and 1.8-7.

2.1.1.2.3.4.2 Description of Low-Level Radioactive Waste

DOE described how it intends to handle and process LLW that will be produced at the GROA in SAR Section 1.4.5.1. More specifically, DOE 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. DOE performed PCSA for the LLW assuming 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. DOE assessed the consequences of these event sequences assuming unfiltered radionuclide release and no significant worker exposures were identified (SAR Section 1.7.5).

NRC Staff Evaluation: The NRC staff reviewed the LLW description using the guidance of TER Section 2.1.1.2.2, although LLW is not discussed in particular. DOE provided reasonable information to enable an evaluation of any effects of LLW in PCSA because it provided the information on LLW characteristics, LLW handling operations, and event sequences to conduct PCSA. Additionally, the NRC staff notes that DOE's calculation results, which show no significant worker exposure is likely from unfiltered radionuclide release of LLW, are reasonable.

2.1.1.2.3.5 Waste Package, Canisters, Casks, and Engineered Barrier System Components

This section provides NRC staff's review and evaluation of DOE's overview description of canisters, casks, and the EBS. The EBS is composed of the waste package, waste package emplacement pallet, drip shield, and the invert structure. The following four sections detail the NRC staff's evaluation of the description of the (i) waste package, (ii) waste canisters, (iii) aging overpack and shielded transfer cask, and (iv) drip shield.

2.1.1.2.3.5.1 Waste Packages

DOE 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). DOE proposed to use the waste package as an engineered barrier for disposal of SNF and HLW. The waste packages would be loaded with TAD, HLW, DOE, and naval SNF canisters at the surface facilities. DOE classified waste packages as ITS because they are relied upon to protect against the release of radioactive gases or particulates during normal operations and Category 1 and Category 2 event sequences. Moreover, DOE classified waste packages as ITWI because, after repository closure, they are relied upon for postclosure performance.

DOE 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 is the most commonly used configuration, which accounts 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 canisters account for approximately 23 percent of the waste packages.

All waste package configurations have a single design that consists 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 includes an inner cylinder, bottom inner lid, and top closure inner lid. The outer corrosion barrier includes an outer cylinder, bottom outer lid, and top closure outer lid. In addition, a purge port is added to the top closure inner lid and the inner vessel is helium filled (SAR Figures 1.5.2-3 through 1.5.2-8).

DOE used codes and standards typically used in the industry for the waste package design. As DOE specified, the inner vessel is 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). DOE stated the inner vessel will be stamped with an N symbol and, therefore, is identified as a pressure vessel. Furthermore, the outer corrosion barrier is 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 DOE, the outer corrosion barrier will not be stamped with an N symbol and, therefore, is not identified as a pressure vessel.

Although all waste packages have a single design, different waste package configurations 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 ranges from 369.7 to 585.0 cm [145.57 to 230.32 in], the nominal diameter ranges from 183.1 to 212.60 cm [72.07 to 83.70 in], and the maximum loaded weight ranges from 40,800 to 73,500 kg [90,000 to 162,000 lb].

DOE stated that the materials used for the waste package 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 is 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 is 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 is carbon steel SA 516 (UNS K02700).

According to DOE, fabrication materials and processes 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) comply with NC-4000 Sections IX and Section III, Division 1; (ii) the welding filler materials comply with NC-2400 Section III, Division 1; (iii) the heat treatment procedure complies 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 Evaluation: The NRC staff reviewed the general description of the waste package and its components using the guidance in the YMRP. DOE provided information regarding the principal characteristics of the waste package and its components that included dimensions, weights, materials, fabrication, and welding. DOE also provided a reasonable characterization of the functional features of the waste package and its components (i.e., confinement for the preclosure period and restricting radionuclide transport to the environment for the postclosure period) and presented a general discussion on applicable codes and standards used for the waste package and its components. Therefore, the waste package description is reasonable to evaluate the waste package design and operation to use in the PCSA and postclosure performance assessment.

2.1.1.2.3.5.2 Waste Canisters

DOE described the design of waste forms and waste packages in SAR Section 1.5.1. DOE 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 DPCs. On the basis of the PCSA, DOE designated these waste canisters as ITS.

TAD Canisters

DOE provided the performance specifications in SAR Section 1.5.1.1.1.2.1.3 for the TAD canisters. These specifications are generally based upon nuclear safety design bases developed from the PCSA and/or transportation and storage requirements. In SAR Figure 1.5.1-5, DOE also provided a conceptual drawing of the TAD canister. The information DOE provided included key dimensions, weights, fabrication specifications and materials of construction, sealing (welding) specifications, and drying processes. In SAR Section 1.5.1.1.1.2.1.4, DOE 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 that has 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, is specified not to exceed 49,215 kg [54.25 T]. To facilitate underwater handling, DOE specified that the TAD lid should be designed for underwater handling, and the canister and lid can be centered while submerged. The TAD canister is specified to have lifting features to allow overhead lifting when the canister is open, empty, and vertical, while a closed, loaded TAD should be capable of being lifted by its lid.

DOE 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 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. DOE chose this material because of its resistance to degradation. 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, DOE 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 are specified to have a minimum thickness of 11 mm [0.4375 in] while the nominal thickness is to be based on structural requirements to maintain the stored geometry of the SNF inside the canister. The length of the neutron absorber plates is specified to 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, DOE described the TAD canister containment characteristics pertaining to welding of the TAD lid. DOE stated that the TAD design will meet either of two requirements: (i) welding specifications will be in accordance with NRC 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).

DOE specified 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, which makes this an ITWI. SAR Section 1.2.5.3.5 described the TAD canister drying and inerting systems, which consist of a generic, forced helium dehydrator system package and a traditional vacuum drying system.

NRC Staff Evaluation: The NRC staff reviewed DOE's description and design information for the TAD canisters using the guidance in the YMRP. The NRC staff compared the physical dimensions (internal and external) of the TAD canister with the waste package dimensions. The NRC staff also verified the consistency between the dimensions of the proposed SNF packages to be placed inside the canister and the internal dimensions of the canisters. In addition, the NRC staff reviewed the information on material, specifications, and codes proposed for TAD canister design for consistency.

The NRC staff notes that the general description and design details of the TAD canisters, including the materials of construction, details of shell and internal component fabrication, and codes and standards are reasonable. The drying and backfilling information is reasonable because drying and backfilling with helium are consistent with standard industry practice and DOE indicated that it will follow the guidance given in NUREG–1536 (NRC, 1997ae) for draining and drying the TAD canisters. Therefore, the descriptive information pertaining to drying and inerting of the canister is reasonable because DOE describes the drying and inerting processes as well as the regulatory guidance (NUREG–1536) that will be followed.

The NRC staff notes that leak testing performed in conformance with ANSI N14.5–97 (American National Standards Institute, 1998aa) is reasonable. The description of the closure design criteria is reasonable because DOE refers to the guidance in Interim Staff Guidance 18 (NRC, 2003af) for welding and the use of ANSI N14.5–97 for demonstrating leak-tightness.

<u>DOE Standardized Canister</u>

SAR Section 1.5.1.3.1.2.1.1 presents the design description of the DOE standardized canister. There are four different DOE standardized canisters, but DOE stated that the functions and requirements are the same. DOE specified that the standardized canister has two different diameters with differing wall thicknesses: (i) a large-diameter standardized canister has 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 has an outer diameter of 457 mm [18 in] and wall thickness of 9.525 mm [0.375 in]. DOE 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 is 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. The standardized canisters are 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 are to be fabricated from Stainless Steel Type 316L SA-479 (bar).

In SAR Section 1.5.1.3.1.2.2, DOE described the operational processes used for drying, sealing, inerting, and leak testing the canisters. DOE specified that the inerting process will utilize an inert gas such as helium. In terms of sealing, the standardized canister boundary components are joined with full-penetration welds. DOE stated that these welds shall 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). DOE stated that the type of weld inspection will be a volumetric inspection using ultrasonic testing. The final closure weld is performed using an ASME-acceptable welding procedure. Prior to transportation to the repository, any required threaded plugs are installed and seal welded in place to establish an ASME-acceptable containment boundary. DOE would 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, DOE presented information on the structural design methodology used for the standardized canister. DOE stated that the standardized canisters have been designed to the 1998 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998aa). DOE also described the finite element analyses performed on the canisters by which the design can be evaluated.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of DOE standard canisters using the guidance in the YMRP. On the basis of the evaluation of the information presented, DOE's description of the standardized canister is reasonable because the standardized canister components, basic drawings, geometry, materials, and codes and standards used have been described in the SAR. The NRC staff notes that DOE provided sufficient detail of the numerical (finite element) analyses performed to support the NRC staff design evaluation in TER Chapter 2.1.1.7.

<u>HLW Canister</u>

SAR Section 1.5.1.2.1.2 described four 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, DOE showed the physical characteristics of each HLW canister including length, outside diameter, wall thickness, and the material type.

SAR Section 1.5.1.2.1.7 provided the design codes and standards for the HLW canisters. DOE 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 an 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). For the West Valley Demonstration Project canister, a dye penetration of all fabrication welds will be performed in accordance with 2001 ASME Boiler and Pressure Vessel Code, Section V (American Society of Mechanical Engineers, 2001aa). Further, the canister will be required to pass pressure and helium leak tests.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of HLW canisters using the guidance in the YMRP. On the basis of the evaluation of the information presented, DOE's description of the HLW canisters is reasonable because the geometry, materials, and codes and standards used have been described in the SAR. The codes and standards for the fabrication welding and nondestructive evaluation of the welds are provided, as well as those for the leak testing of the canister. The NRC staff notes that these codes and standards are applicable because they are commonly used in the industry for the intended purpose, and proper application of these is expected to result in canisters meeting their intended leak-tight performance. In SAR Section 1.7.2.3.1, DOE discussed a full-scale experimental testing of HLW canisters. The NRC staff considers this information reasonable for a design evaluation presented in TER Chapter 2.1.1.7. Therefore, DOE's information is reasonable for use in the review of the PCSA and design, as needed.

<u>Dual-Purpose Canister</u>

DOE briefly discussed the DPCs in SAR Section 1.5.1.1.1.2.1.2. A DPC is used to store CSNF at the utility site, licensed under 10 CFR Part 72, and could also be used to ship the SNF, licensed under 10 CFR Part 71. Currently, DOE plans to accept DPCs at the repository. In terms of storage use, the DPCs would be placed in an appropriate aging overpack for aging. The NRC staff notes, however, that DOE stated that current DPC designs are not appropriate for disposal. For disposal, the SNF in the DPCs will be repackaged into a TAD canister and this operation will be performed in WHF.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of DPC using the guidance in the YMRP. Because the DPC systems are licensed for storage at utility sites under 10 CFR Part 72 and for transportation under 10 CFR Part 71, the NRC staff determines that sufficient information in the form of Final Safety Analysis Reports for various DPC system vendors is available. The NRC staff notes that the general description and discussion of design details of the DPC canisters, including the materials, details of fabrication of the containment shell and canister internal components, and codes and standards, have been described reasonably for review of the PCSA and design. The evaluation of design and analysis of the DPC is in TER Chapter 2.1.1.7.

<u>Naval Canister</u>

In SAR Section 1.5.1.4.1.2.1, DOE described the naval short or naval long SNF canisters to accommodate different naval fuel assembly designs. SAR Figure 1.5.1-29 showed a typical naval SNF canister. DOE specified that the naval SNF canister is fabricated from a 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 has six, 7.6-cm [3-in]-diameter threaded holes for lifting purposes. The shield plug is 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 has a 471-cm [185.5-in] nominal length {maximum length is 475 cm [187 in]}, and the naval long SNF canister has a 535-cm [210.5-in] nominal length {maximum length is 538 cm [212 in]}. The maximum outer diameter of the naval SNF canister is 168.9 cm [66.5 in]. The canisters are sized to fit within a waste package. The maximum design weight of the loaded long or short naval SNF canister is 44,452 kg [98,000 lb]. However, DOE noted that for establishing a margin in crane capability, the canister has been assigned a maximum weight of 49,215 kg [108,500 lb].

Design codes and standards for the naval SNF canister are 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 Evaluation: The NRC staff reviewed DOE's description of the naval SNF canisters using the guidance in the YMRP. On the basis of the evaluation of the information presented, DOE has provided reasonable descriptive information defining the canister's principal design characteristics. The descriptive information of the naval SNF canisters is reasonable because the geometry, materials, codes and standards, and design analysis approaches used have been described in the SAR.

2.1.1.2.3.5.3 Aging Overpack and Shielded Transfer Casks

DOE provided in SAR Sections 1.2.7 and 1.2.5.4 information on aging overpack and transfer casks. The objective of the NRC staff review is to evaluate whether sufficient descriptive information is presented in the SAR pertaining to aging overpacks and the shielded transfer cask.

CSNF will be aged at the repository in the TAD canisters and in DPCs. The TAD can be previously loaded at a utility site or loaded at the repository in the WHF, while the commercial DPCs are loaded at the utility site. 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 AF design criteria. The TAD canister will utilize a vertical overpack, while the commercial DPCs will use concrete vertical aging overpacks and concrete horizontal aging modules for aging.

There are 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.

<u>Aging Overpacks</u>

Aging overpacks were detailed in SAR Section 1.2.7, which discussed the proposed AF. DOE described the vertical aging overpack in SAR Section 1.2.7.1.3.2.1 and the horizontal aging modules in SAR Section 1.2.7.1.3.2.2. DOE stated that 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 the occurrence of 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, DOE classified the aging overpacks 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, DOE provided a general description of vertical aging overpacks. DOE described the vertical aging overpack as a cylinder with a metal liner surrounded by steel-reinforced concrete that is surrounded by an outer steel shell. A vertical aging overpack has 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 can support the inserted canister during the aging process. A vertical aging overpack is fitted with a bolted lid, which also provides shielding and protection, and is designed to protect the internal canister against impact/collision and drop loads. The overpack is also designed to provide passive cooling through convective movement of the air surrounding the canisters. Bottom inlets and top outlets allow ventilation air to be passively drawn into the annular area between the TAD or vertical DPC canister and the metal liner. The inlet and outlet designs are designed to prevent radiation streaming. A conceptual drawing of the vertical aging overpack in SAR Figure 1.2.7-6 showed most of the design features described here.

DOE 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 is used behind each horizontal aging module and at each end of the row 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 is equipped with temperature sensors to measure outlet air temperature.

SAR Section 1.2.7.8 listed the design codes and standards for the vertical aging overpack and horizontal aging modules. DOE 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, Appendix A (1997aa).

NRC Staff Evaluation: The NRC staff reviewed the aging overpack information using the guidance in the YMRP and notes that the information is reasonable because DOE described basic drawings, geometry, materials, and applicable codes and standards, and the proposed codes and standards are consistent with industry practice. Therefore, DOE's description is reasonable to evaluate the aging overpack design and for use in the PCSA, as needed.

<u>Shielded Transfer Cask</u>

SAR Section 1.2.5.4 detailed the shielded transfer cask. The shielded transfer cask is used for processing TAD canisters and DPCs in the WHF. DOE indicated that the shielded transfer cask is also used for moving horizontal DPCs from the AF to the WHF. The shielded transfer casks are designed to provide integral shielding, structural strength, and passive cooling functions. DOE classified shielded transfer casks as ITS, and the nuclear safety design bases and design criteria were given in SAR Table 1.2.5-3.

A shielded transfer cask is required to maintain its structural integrity, retain the canister, and continue to provide shielding when subjected to drops, tipover, 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 is to be used in the WHF, it must also be compatible with the pool water. DOE specified that the materials of construction for the shielded transfer cask design are in accordance with 2004 ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (American Society of Mechanical Engineers, 2004aa).

The shielded transfer casks perform different functions when handling TAD canisters and DPCs; however, some common design features are utilized to standardize operations and maintenance. SAR Figures 1.2.5-76 to 1.2.5-78 showed the general design features of the shielded transfer casks. A vertical DPC shielded transfer cask is designed to contain a DPC, which is 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 is designed to be lifted by trunnions using an overhead crane and is 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 is designed to contain a TAD canister (in a vertical orientation) and can be moved and lifted in a manner similar to the vertical DPC shielded transfer cask. The TAD shielded transfer cask is also 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 is designed for a single horizontal DPC. After loading, the cask is rotated to a vertical position and lifted by its trunnions using an overhead crane. This cask is also 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 Evaluation: The NRC staff reviewed the information on the shielded transfer cask presented in SAR Section 1.2.5.4.2 using the guidance in the YMRP and notes that SAR Table 1.2.2-12 provides a general reference to codes applicable to the design of surface structures and mechanical handling equipment. DOE uses design methodology in accordance with the 2004 ASME Boiler and Pressure Vessel Code, Section III, Subsection NC (American Society of Mechanical Engineers, 2004aa) for the design of shielded transfer casks. The NRC staff notes that this is reasonable because this methodology is an accepted industry practice. In addition, DOE’s description of the shielded transfer cask is reasonable because the NRC staff is able to understand the system’s basic functions and its general design features. Furthermore, the NRC staff notes that the materials and the codes and standards proposed for the design are reasonable.

2.1.1.2.3.5.4 Drip Shield

DOE described and discussed the drip shield design in SAR Section 1.3.4.7. DOE has proposed to use the drip shield as an EBS during the postclosure period to divert the liquid moisture around the waste package and down to the drift invert and protect the waste packages from rockfall. DOE classified the drip shield as non-ITS because it is not relied on to prevent or mitigate Category 1 and Category 2 event sequences. DOE classified the drip shield as ITWI because it is relied upon to prevent or substantially reduce the rate of movement of water and radionuclides.

DOE stated that the drip shield has a single design and is uniformly sized to enclose all waste package configurations and is designed for both corrosion resistance and structural strength. The drip shield incorporates 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 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 varies between 2,821 and 2,886 mm [111 and 113.6 in], the width varies between 2,526 and 2,535 mm [99 and 99.8 in], and the length is 5,805 mm [228.5 in]. The drip shield weight is 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 RAI, DOE provided drip shield main assembly, subassemblies, and components drawings (DOE, 2009dr).

According to DOE, the drip shields are designed to form a continuous barrier throughout the entire length of the emplacement drift by interlocking the drip shield segments. DOE stated that the drip shield is designed to accommodate an interlocking feature to prevent the separation between contiguous drip shield segments and a minimum lift height of 1,016 mm [40 in] is required to interlock the drip shield segments. Furthermore, the drip shield interlocking feature includes water diversion rings and connector plates that 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, DOE 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).

DOE stated that, except for the attachment to the Alloy 22 base, drip shield components are intended to be connected to each other by welding. According to DOE, the Alloy 22 base plates are intended to be mechanically attached to the titanium components by Alloy 22 pins because titanium and Alloy 22 cannot be reliably welded together. DOE 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, DOE extracted codes and standards as applicable for materials, welding, postweld heat treatment, and postweld nondestructive examination of the drip shield from BSC (2007bu). DOE 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, DOE 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 DOE, 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 response to an NRC staff RAI (DOE, 2009dr) for justification that welding Titanium Grade 7 to Grade 29 using Grade 28 as filler metal is appropriate, DOE stated that this dissimilar welding joint mitigates hydrogen embrittlement. In addition, DOE 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 Evaluation: The NRC staff reviewed the drip shield information using the guidance in the YMRP and notes that the information is reasonable to evaluate the design and functions of the drip shield and drip shield interlocking feature. The NRC staff also notes that the codes and standards pertaining to drip shield design and fabrication in the prototype program are appropriate. With respect to welding Titanium Grade 7 to Grade 29 using Grade 28 as filler metal, the two examples DOE cited are not comparable to the dissimilar welding joint in 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). This example demonstrates that welding between ruthenium-containing alloy (Titanium Grade 26) using palladium-containing alloy (Titanium Grade 7) has been used in the industry; however, this example is not similar to the welding joint between Titanium Grade 7 and Titanium Grade 29 using Titanium Grade 28 as filler material in the drip shield, because both Titanium Grades 26 and 7 are single α-phase material and they have similar mechanical properties. 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 industry practice recommends testing such welding joints to ensure weld strength. On the basis of the NRC evaluation of DOE's response (DOE, 2009dr), the NRC staff notes that DOE did not provide sufficient technical basis to justify the adequacy of dissimilar weldment configuration used in the drip shield. However, the prototype program DOE proposed in BSC (2007bu) is intended to test welding, identify any deficiencies in the design, and ultimately demonstrate the suitability of the design. Therefore, the information DOE provided regarding the drip shield design and prototype program is reasonable for use to evaluate DOE's assessment of drip shield performance.

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 DOE's description of operational processes in the surface and subsurface facilities of the GROA and onsite transportation provided in the SAR. The operational process of a facility considers (i) the operational sequences and material flow, (ii) the major waste processing functions performed, and (iii) the waste form inventory present within the facility.

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 will evaluate (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) confirm whether the proposed operation plan will permit permanent disposal of the mandated quantity of HLW within the period DOE stipulates. These topics are covered in TER Section 2.1.1.2.3.6.1.

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. Surface and subsurface facility control systems are covered in TER Section 2.1.1.2.3.6.2.

2.1.1.2.3.6.1 Operational Processes

DOE described GROA process activities in SAR Section 1.2 to enable identification of hazards and event sequences in the PCSA. DOE's description includes operations in IHF, CRCF, WHF, RF, AF, subsurface facility, and 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 TER Sections 2.1.1.2.3.2.2 and 2.1.1.2.3.2.5, respectively.

Surface Facility Operations

DOE 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 facility 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 facility, 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 in SAR Sections 1.2.6.2.1.2 and 1.2.6.2.2.2, respectively.

DOE proposes to construct three identical CRCFs. The main operations in the CRCF involve handling of canisters containing different waste forms, transportation casks, aging overpacks, and waste packages. The waste form canisters handled in the CRCF are TAD canisters, HLW canisters, DPCs, and DOE SNF canisters. The overall operations in the CRCF are performed using four mechanical handling subsystems: cask handling, canister transfer, waste package closure, and waste package load out. The process subsystems include cask cavity gas sampling and water collection subsystems. The facility is 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 include the HVAC equipment room, electrical rooms, maintenance areas, and waste package closure support rooms. The major mechanical equipment used in the facility is overhead bridge cranes, CTTs, CTMs, WPTTs, and associated lifting fixtures and devices.

The transportation casks on rail- or truck-based trailers are received in the cask handling area. The casks are moved onto the cask transfer trolley after the impact limiters are removed from them. In the cask preparation area, the cask cavity is sampled and depressurized and lid bolts are removed. The aging overpacks are received on a site transporter, and the lid bolts are removed. In the canister transfer subsystem, the canisters are transferred from the transportation casks into a waste package, aging overpack, or placed in the staging area. A staging area is provided for TAD, HLW, and DOE SNF canisters. However, in response to the NRC staff RAI (DOE, 2009dx), DOE indicated that TAD canister staging is not part of normal operations. The CTM is operated remotely to remove cask lids, transfer canisters to waste packages on the WPTT, and place inner lids on waste packages. Waste package closure subsystems are 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 include transfer of sealed waste packages to the load-out area in the WPTT and loading of waste packages onto the TEV.

The main operations in the IHF involve handling of naval SNF containers or HLW containers, transportation casks, and waste packages. Similar to the CRCF, the overall operations in the IHF are performed using four mechanical handling subsystems: cask handling, canister transfer, waste package closure, and waste package load out. The process subsystems include cask cavity gas sampling and water collection. The major operational areas 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 is overhead bridge cranes, cask transfer trolleys, CTMs, WPTTs, and associated lifting fixtures and devices.

The main operations in the RF involve handling of TAD canisters or DPCs, transportation casks, and aging overpacks. The overall operations in the RF are 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 is 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 is overhead bridge cranes, cask transfer trolleys, CTMs, and associated lifting fixtures and devices.

The CRCF, IHF, and RF 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 in BSC Section 6, Figure 15, and Attachments A and B of BSC (2008ab,ao,bd). Human interactions during operations are discussed in BSC Section E6 (2008ac,as,be) as a part of analysis of human failures.

The overview of the operational processes for WHF was described in SAR Sections 1.2.5.1.2 and 1.2.5.2.1.2. At the WHF, the main operations include handling of transportation casks and aging overpacks, SNF assemblies, and DPC and TAD canisters. Overall operations are performed using five mechanical handling subsystems: cask handling, SNF assembly transfer, DPC cutting, TAD canister closure, and canister transfer. The facility is 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 pool, DPC cutting area, TAD closure area, and canister transfer area. The process subsystems include cask cavity gas sampling, pool water treatment, and cooling subsystems. The major rooms to support waste handling operations include the HVAC equipment room, electrical rooms, maintenance areas, and waste package closure support rooms. The major mechanical equipment used in the facility is overhead bridge cranes, cask transfer trolleys, CTMs, SFTM, and associated lifting fixtures and devices. The facility includes 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 1.2.3-17, 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 Section 6, Figures 16–17, and Attachments A and B (2008bq). Human interactions during operations are discussed in BSC Section E6, Appendix E (2008bq) as a part of the analysis of human failures.

The overview of the operational processes for the WHF was discussed in SAR Section 1.2.5.1.2, and the detailed description of the operations for the cask handling, SNF assembly transfer, DPC cutting, TAD canister closure, and canister transfer was discussed in 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, respectively. The cask handling system receives transportation casks containing uncanistered CSNF, rail casks with DPCs, shielded transfer casks with DPCs, and aging overpacks with DPC. For transportation casks containing uncanistered SNF, CHCs are used for removing impact limiters, upending, and moving to the preparation station. After sampling, venting, and cooling, the cask interior is filled with borated water and the cask lid unbolted and moved to the pool. For rail casks containing DPCs, impact limiters are removed, upended, and placed on a cask transfer trolley. DPCs in shielded transfer casks are upended and placed on the cask transfer trolley. DPCs in an aging overpack are transferred to a shielded transfer cask at the WHF using the CTM. All casks containing DPCs are transferred to the DPC cutting station. The cask handling system also transfers TAD canisters from shielded transfer casks to aging overpacks using the CTM and sends 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 moves the SNF assemblies from transportation casks or DPCs to TAD canisters or a staging rack and from the staging rack to TAD canisters. After loading the TAD canisters, a TAD canister shield lid and shielded transfer cask lids are replaced prior to moving the TAD canisters out of the pool.

NRC Staff Evaluation: The NRC staff reviewed the information on surface facility operations using the guidance in the YMRP. The information on surface facility operations is reasonable because the description of (i) waste handling operations including the process flow diagram; (ii) subsystem/equipment interactions and interfaces between the subsystems; and (iii) human interactions for CRCF, WHF, IHF, and RF provided 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 TER Sections 2.1.1.3.3.2.1 and 2.1.1.3.3.2.3.4.

The NRC staff notes that DOE's operational description does not include the maximum lift heights of the casks, canisters, and waste packages. DOE, however, discussed the probability of failure from drop for various containers from normal operating and two-block lift heights for inclusion in the event sequence analysis and analyzed the passive reliability of containers for different drop heights and impacts [e.g., BSC Table 6.3-7 (2008ac)] (see detailed NRC staff's evaluation in TER Section 2.1.1.4.3.3.1.1). The staff notes that operations lift heights and collisions of various waste containers are used in calculating their passive reliability. DOE's information enables the NRC staff to understand the operations of the surface facility and is reasonable to evaluate the surface facility operations and to use in the PCSA.

Intrasite Surface Operational Processes

DOE 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 AF, LLWF, EDGF, and the intrasite transportation system. For example, site transportation activities discussed included movement of the transportation casks, security/radiological inspections, and transfer of aging overpacks and horizontal casks from one surface facility to another. AF activities considered in the key intrasite activity, as described in BSC Section 4.3.4 (2008at), included 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 included onsite loading, onsite transfer to the LLWF, unloading at the LLWF, storage at the LLWF, and the offsite disposal process. BSC Attachments B and C (2008at) provided additional details on the operations. The balance-of-plant activities included 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 will provide emergency power in the event of an LOSP.

DOE provided an intrasite operational process flow diagram in BSC Figure 14 (2008at). BSC Attachment C (2008at) included onsite transportation routes and the relative location of the aging pads, LLWF, and buffer areas within the GROA. DOE 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.

DOE 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 prime mover (SAR Section 1.2.8.4.3). All three are ITS and are further described and evaluated in TER Section 2.1.1.7. In BSC Appendices B1.4, B2.3, and B3.3 (2008au), DOE 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, DOE listed human interactions and described human-induced failures during intrasite operations. For the site transporter, in particular, DOE provided additional detail on human interfaces. This included a list of 15 remote control activation devices (pushbuttons and selector switches) that the intrasite SSCs' human operators manipulate or activate.

NRC Staff Evaluation: The NRC staff reviewed the intrasite operations using the guidance in the YMRP and notes that DOE provided reasonable operational information for the intrasite operational processes because the descriptions of these processes are sufficient for NRC staff review of identification of hazards and initiating events. Further, the information on operational activities and procedures of the intrasite operations involving site transport and emplacement of the aging cask in an AF is reasonable because DOE provided waste form handling operations including a process flow diagram, subsystem/equipment and interactions and interfaces among the subsystems, and human interactions.

<u>Subsurface Operational Processes</u>

DOE described the subsurface operations in SAR Section 1.3.1 and summarized these operations in BSC Appendix B (2008bj). The subsurface operations included activities such as WP load out, WP emplacement, drip shield loadout, drip shield transport, and drip shield emplacement. DOE provided a process flow diagram in supplemental document (BSC, 2008bj) detailing 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. DOE 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 is the TEV.
The non-ITS SSCs mentioned were the invert system, crane rail switches, ventilation system, access door, the DSEG, fire protection system, the electric power system of the third rail, and the communication and control system of the control center.

The subsurface operations have two distinct normal operations sequences: waste package emplacement and drip shield emplacement. The normal TEV emplacement operation consists of approximately 30 steps, such as opening its front shield doors, driving forward, lifting rear shield doors, extending the base plate, lowering the shielded enclosure, and lifting the waste package. These operations were described in SAR Section 1.3.3.5.2.1 and BSC (2008bz).

DOE also briefly described the construction operations that will occur at the same time as emplacement operations during a portion of the preclosure period, as outlined in

BSC Appendix B (2008bj). These operations include excavation using common drill and blast techniques, as well as mechanical excavators, which could potentially affect waste handling operations.

DOE indicated that the primary human interactions with the subsurface operations are related to the communication and control of the TEV by the operators in the control center. DOE described the use of high-intensity lights and a camera onboard the TEV to provide feedback to operators in the control center. DOE also described the use of PLCs that accept initiating commands from the operators to execute predefined, preprogrammed instructions and maneuvers. DOE discussed the subsurface operations as it relates to human interactions in BSC Section 6.4 (2008bk).

NRC Staff Evaluation: The NRC staff reviewed the information on subsurface operation processes using the guidance in the YMRP and notes descriptions of the subsurface operational processes and procedures involving waste package transport and emplacement using TEV are reasonable because DOE provided waste form handling operations including a process flow diagram, subsystem/equipment and interactions and interfaces among the subsystems, and human interactions. The NRC staff also notes that the description of emplacement drift construction operations provided reasonable support for NRC to evaluate DOE's event sequence development analysis. Additionally, the information on human interactions is reasonable for identification of hazards and initiating events in the PCSA.

Waste Form Throughput

According to DOE, the repository is designed for 7×10^7 kg [70,000 MTHM] of radioactive waste, as shown in SAR Table 1.5.1-1 and BSC Section 6 (2007bh). More specifically, the waste to be disposed of in the repository includes 6.3×10^7 kg [63,000 MTHM] of CSNF and HLW of commercial origin, 4.7×10^6 kg [4,667 MTHM] of defense HLW, 2.3×10^6 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. DOE estimated an annual rate of waste handling including 3×10^6 kg [3,000 MTHM] of CSNF, 763 defense HLW canisters, 179 DOE standardized canisters, and 24 naval canisters.

Most waste shipped to the repository site will be canistered. DOE 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. HLW glass will be in HLW canisters, DOE defense SNF will be either in DOE standardized canisters or MCOs, and naval SNF will be loaded in U.S. Navy-designed canisters.

According to DOE (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 once 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 before being 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 was 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 DOE's PCSA are based on these throughput numbers, which have been used as a point estimate in the event tree analysis for quantification of event sequences. These throughput numbers are 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.

NRC Staff Evaluation: The NRC staff reviewed DOE's waste throughput information using the guidance in the YMRP and notes that DOE's throughput numbers listed in SAR Tables 1.2.1-1 and 1.7-5 are reasonable 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). Therefore, DOE provided sufficient throughput information to support its PCSA for event sequence development and categorization. DOE indicated that the quantification of event sequences involving MCOs were not supported in the SAR (DOE, 2009bl) and the event sequences involving MCOs were not reviewed in TER Section 2.1.1.4. Because the waste throughput is one of the baseline assumptions for DOE's PCSA, it is important for DOE to make sure the waste throughput numbers in SAR Tables 1.2.1-1 and 1.7-5 are not exceeded throughout the preclosure period so that event sequence categorization and the PCSA results are valid.

Operational Period

DOE indicated in the SAR and General Information Section 2.2 (DOE, 2009av) that the total preclosure period is 100 years, while the receipt and emplacement operations period is projected to span 50 years. In addition, as stated in General Information Section 1.1.2.1 (DOE, 2009av), the surface facilities have a design operating life of 50 years. DOE also used screening criteria of 2×10^{-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 TER Section 2.1.1.3.

NRC Staff Evaluation: The NRC staff reviewed the information on the operational period DOE proposed using the guidance in the YMRP and notes that DOE used the 50-year operational period consistently in initiating event screening in the PCSA. Because the 50-year operation period is one of the baseline assumptions for DOE's PCSA, it is important that this operational period not be exceeded in order that the event sequence categorization and PCSA results are valid.

2.1.1.2.3.6.2 Instrumentation and Control Systems

DOE 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 includes 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 DOE reasonably described (i) control philosophy, conceptual process diagrams, and digital control logic diagrams for ITS and non-ITS controls; (ii) design codes, standards, and accepted 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 nonaccessible areas, the NRC staff also evaluated the reasonableness of the I&C and related communication systems design descriptions to support potential operations and/or waste retrieval.

Surface and Subsurface Facility Instrumentation and Control

Most normal facility operations are based on repetitive cask unloading, transfer, repackaging, and reloading steps; hence, the normal facility production/throughput functions implement automation where practical. Such automation typically uses non-ITS PLCs or other non-ITS digital devices to control machines that have been 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 are 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, 1.3.5, and 1.2.8 for relevant surface operating facilities. ITS HVAC and ITS electrical power system SSCs, including ITS diesel generators, also include

ITS I&C SSCs, as generally described in SAR Sections 1.2.2.3 and 1.4.1.2, respectively. DOE described no ITS I&C SSCs for subsurface facilities.

In general, ITS controls are made up of individual instruments, sensors, or devices that are hardwired to control devices to perform safety-related control functions and interlock and other protective functions. DOE stated that all ITS controls and interlocks that implement safety functions needed for preventing event sequences and mitigating consequences are 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), DOE 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. DOE 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 DOE intends to apply to the final design of the ITS controls (DOE, 2009do). DOE proposed to use IEEE 308, 379, 384, and 603 (Institute of Electrical and Electronics Engineers, 2001aa,ab and 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. DOE further described how it intends to interpret the applicability of the 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. DOE 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, 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). DOE 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. DOE further stated that potential future upgrade of the ITS interlocks will be based on 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. DOE 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 Evaluation: The NRC staff reviewed DOE's description of I&C SSCs using the guidance in YMRP Section 2.1.1.2.2. DOE identified the standards and codes for the design and environmental qualification of ITS control systems, and stated that it will conduct periodic testing and maintenance of ITS I&C. The NRC staff notes that DOE's description of ITS I&C allows a reasonably clear understanding of how the ITS I&C in surface facilities will be designed and operated and may be used in the PCSA, as appropriate. A safety evaluation of the design of ITS I&C SSCs is included in TER Sections 2.1.1.6.3.2.8.2.1 and 2.1.1.7.3.5.

The NRC staff reviewed DOE's description of non-ITS I&C SSCs in the GROA and notes that DOE provided discussions, applicable codes and standards, and descriptions of controls and monitoring for the GROA and a high-level description of distributed control philosophy in the SAR. DOE generally provided more specific information regarding the function and use of particular non-ITS I&C SSCs within descriptions of other GROA non-ITS facilities and operating systems in the SAR. Therefore, DOE's description of non-ITS facility I&C allows a reasonably clear understanding of how non-ITS I&C used in the surface and subsurface facilities will be designed and applied, and is reasonable to use in the PCSA, as needed.

<u>Special Vehicle Instrumentation and Control</u>

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 include the mechanical location switch and shield door motors (DOE, 2009dp). DOE 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).

PLCs are used for remote-controlled, non-ITS operations of the TEV. In response to an NRC staff RAI (DOE, 2009dm), DOE 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. DOE stated that the onboard PLC is non-ITS because it does not perform ITS safety functions and is not 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 are functionally independent of each other. Once the ITS location switch is deactivated, a remote operator cannot inadvertently open the TEV shielded enclosure doors using the non-ITS PLC on the

TEV. The TEV has an air conditioning unit and a fire protection system for onboard temperature-sensitive electronic components.

The non-ITS DSEG is 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 is controlled by an onboard PLC network, which controls cameras, high-intensity lights, and thermal and radiological sensing instruments. Like the TEV, the DSEG has an air conditioning unit for temperature-sensitive electronic components and a fire protection system. Because the DSEG is a custom-designed vehicle, there is no special industry code or standard that identifies its appropriate control, communications, or monitoring design criteria. DOE indicated in SAR Table 1.3.2-4 that DSEG design generally conforms to ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa).

A number of planned non-ITS ROVs have been proposed for performing visual inspections, material sampling, and potential maintenance and repair tasks within emplacement drifts after the emplacement of waste packages, and in other nonaccessible subsurface areas, according to SAR Section 1.3.1.2.1.6 and BSC (2008ca). Remote observations will be conducted using onboard video cameras to monitor the drift condition, drift ground support system, and emplaced waste packages. Inspections 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. A potential need for additional, unplanned, special purpose ROVs designed specifically for performance of off-normal or unplanned inspection, observation, maintenance, or repair activities in nonaccessible areas was described in the SAR. Because the specifications and designs of these unplanned ROVs would be reactive to specific unforeseen problems or needs, no design or operations concepts were provided.

In SAR Section 4.2.1.8, DOE 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 is expected to move among the drifts and deploy I&C and supporting communications. The emplacement drift ROV will have enhanced monitoring capabilities. Because the ROVs are custom-designed vehicles, there are no special industry codes or standards that identify appropriate design criteria for control, communications, and monitoring systems.

NRC Staff Evaluation: The NRC staff reviewed DOE’s description of special vehicle I&C using the guidance in the YMRP. DOE provided appropriate codes and standards for the ITS controls. The NRC staff notes, on the basis of its review, that the information provided is reasonable to evaluate the TEV ITS I&C design and to use in the PCSA, as needed. A safety evaluation of the design of TEV ITS I&C SSCs is included in TER Sections 2.1.1.6 and 2.1.1.7.3.5.

Regarding DOE’s descriptions of non-ITS I&C for the TEV and DSEG, the NRC staff notes that, despite the high-level conceptual description and associated diagrams, the information is sufficient to provide a reasonably clear understanding of the design and intended operations of non-ITS I&C SSCs for this special equipment. The NRC staff notes that DOE provided appropriate codes and standards for design of the onboard PLC for the TEV and DSEG.

The NRC staff reviewed descriptions, outline drawings, and other diagrams for the planned ROVs provided in the SAR and BSC (2008ca) and notes that the information is reasonable to evaluate the intended concepts for design and operations. The ability of the ROVs to perform assigned tasks in nonaccessible subsurface openings is a fundamental component of planned observation and inspection activities as described in the SAR and supplemental materials. The capability to reliably perform observations and inspections, and potential maintenance, of nonaccessible underground openings and SSCs within them throughout the preclosure period is discussed in TER Section 2.1.1.2.3.7.

DOE described these ROVs as "yet-to-be-designed or specified (BSC, 2008ca)," representing a DOE statement to further develop specifications and detailed designs of the various planned ROVs and related special vehicle instrumentation and controls during the detailed design phase. A detailed design should provide a complete design description for each ROV, including tasking, configuration, major components, method of locomotion, power requirements, and method for provision of communications between each vehicle and remote operators/inspectors for all ROVs that are planned to operate in nonaccessible areas (BSC, 2008ca). DOE's information along with this statement to provide the details in the detailed design for planned ROVs is reasonable to evaluate the use of ROVs and to use in the PCSA, as needed.

<u>Subsurface Ventilation Instrumentation and Control</u>

Drift temperature, pressure, and relative humidity are very important parameters indicating the effectiveness of the ventilation system and are closely monitored. DOE 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), because the primary purpose of the subsurface facility sensors will be to monitor temperature, barometric pressure, relative humidity, and dose rate in places of the subsurface facility which are accessible to repository personnel to ensure that ventilation to the emplacement drifts is maintained at design values.

The sensors are non-ITS; therefore, commercial-grade sensors, which are environmentally qualified for the expected environment, will be used. However, DOE 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, 2005aa). In accordance with Regulatory Guide 8.8 (NRC, 1978ab), the monitoring sensors will be located and/or shielded to maintain occupational radiation exposures ALARA.

NRC Staff Evaluation: The NRC staff reviewed the description of controls for the subsurface ventilation system provided in the SAR using the guidance in the YMRP. The SAR information is reasonable because DOE provided discussions, applicable codes and standards, and descriptions of controls and instrumentation for monitoring operations of the subsurface ventilation system. The subsurface ventilation I&C system is non-ITS; however, selection of sensors suitable for the anticipated environment and locations/shielding to provide ALARA occupational exposures were described. The NRC staff notes that the SAR information on the I&C for subsurface ventilation system provides a reasonable description of the characteristics and operation of subsurface ventilation monitoring and is reasonable for use in the PCSA, as needed.

Digital Control Management Information Systems

The Digital Control Management Information Systems (DCMIS) is part of the proposed non-ITS I&C system and provides control and monitoring for the GROA facilities during the preclosure period (SAR Section 1.4.2.1). The DCMIS relies on the proposed communications system (described next) to “connect” operators in the CCCF with non-ITS monitoring and controlling SSCs distributed throughout the GROA and nearby support facilities. In other words, the DCMIS provides the CCCF operators with the capability to control and monitor the operations of the TEV, DSEG, and ROVs. The major components of the DCMIS are controllers, human–machine interface consoles, input and output modules, engineering workstations, data historians, networks and network interface devices, and foreign-device interfaces.

DOE provided applicable codes and standards for the DCMIS in SAR Section 1.4.2.1.3. In its response to an NRC staff RAI (DOE, 2009du), DOE identified additional applicable industry codes and standards for network interface design to protect the DCMIS from undesired interactions and intrusions.

The DCMIS architecture utilizes 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 make data available to a redundant supervisory network, which facility operators and managers may access to monitor the status of operations. The DCMIS is also 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). DOE stated that it would incorporate criteria contained within NIST 800–53 (National Institute of Standards and Technology, 2007aa) and other standards, which provide for the incorporation of security controls to help guard against such intrusion. DOE also stated that the repository cyber security program is risk based and provides 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 process remote operator’s 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 is no information in the SAR regarding the required design characteristics for the local controllers, it can be inferred from other SAR sections that local controllers should be used to regulate GROA operations so that the combination of hardware and software is configured to wait for permissive signals from either a local or remote operator before the automation functions can proceed. DOE provided the design characteristics for the local controllers to include separation of power supply feeds and digital input/output modules, diagnostics, shielding, etc. (DOE, 2009dn). The high-level plans for periodic calibration and surveillance requirements for analog signals are also defined.

Controllers and input/output modules (non-ITS, non-Class 1E equipment) are distributed throughout the GROA and are close to the signal source. To ensure that the DCMIS can perform monitoring functions during a loss of normal power, portions of the DCMIS will be powered by ITS UPS (SAR Sections 1.4.2.1 and 1.4.1.1.5).

NRC Staff Evaluation: The NRC staff reviewed DOE’s description of DCMIS using the guidance in the YMRP and notes that the standards and codes identified for the DCMIS design are appropriate for use in this facility because these codes and standards are general industry standard protocol design standards for this type of system. DOE reasonably described and discussed design information for the security aspects of the DCMIS and for local controllers. DOE's information is reasonable to permit an NRC staff evaluation of the use of DCMIS to conduct and support GROA operations and may be used in the PCSA, as needed.

Communications System

The communications system, which is non-ITS, facilitates interchange of video, voice, and data communications for surface and subsurface facilities during the preclosure period. Communications are provided for the 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 is supported by a dual-ring network topology that provides the physical transport media within a Synchronous Optical NETwork (SONET) communications backbone. The proposed SONET architecture consists of a redundant fiber optic ring connecting all network nodes. One ring is typically the active ring and is referred to as the working facility, while the other ring is the standby ring, referred to as the protection facility (Black and Walters, 2001aa). DOE identified codes and standards applicable to the communications systems in SAR Section 1.4.2.4.3.

Radio-frequency wireless transmission communications systems are provided to interconnect the DCMIS and the TEV, DSEG, and ROVs. 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, DOE (DOE, 2009eg) stated that it would 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 an LOSP, the facility communications system has been designed to be powered by UPS, although the design time period for such sustained operations was not provided.

DOE described the dual-ring network for the subsurface facilities as physically separate and independent from the dual-ring SONET network that serves the surface facilities; however, the surface and subsurface networks are interconnected by firewall SSCs as shown in SAR Figure 1.4.2-8. SAR Figure 1.4.2-8 illustrated how SONET nodes are installed in several electrical equipment alcoves positioned along the access mains and how they 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 subsurface communication system makes extensive use of both wired and wireless technology. The wired component consists of both fiber optic and copper cabling. A radio frequency radiating coaxial cable antenna wirelessly connects 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 carries a battery backup power system that maintains onboard communications with the CCCF for an undefined period of time if normal power becomes unavailable. DOE confirmed that the radiating cables, antennas, and transceivers are installed in the access mains and alcoves of the subsurface facility and, in some cases, they are located in the air intake shafts.

DOE proposed to use antennas located in the access main and just inside drift entrances (BSC, 2008bz) to provide reliable wireless data communications between DCMIS and the TEV when located in nonaccessible turnouts and emplacement drifts. DOE also suggested use of a slotted microwave guide system (DOE, 2009ee) as an alternate communications approach. The system, according to DOE, would provide payload data rates of up to 54 Mbps and should provide adequate capacity for communications with the TEV. DOE 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). DOE stated it would further develop these and other potential in-drift communications alternatives during the detailed design phase (DOE, 2009ee).

The SAR and supplemental materials did not further describe the ability of the antenna, slotted microwave guide, power line carrier, or other alternative concepts to accommodate reliable communications between a vehicle in nonaccessible areas and the balance of the DCMIS, nor did it describe switching or other means for slotted microwave guide or power line carrier concepts to follow changing TEV, DSEG, or other vehicle track routes without mechanical interference while the vehicles move between access mains and any one of the multiple emplacement drifts. DOE's description indicated that components providing or supporting communications between vehicles and remote operators may be permanently installed within emplacement drifts and in turnouts and other nonaccessible areas; however, no provisions or plans for inspection or preventive or corrective maintenance for the described concepts were presented.

According to BSC (2008ca), the SAR contained no description of communications provisions to nonaccessible ECRB cross drift and exhaust mains and shafts where maintenance activities will be conducted. Inspection and potential maintenance operations for these areas will be performed using one or more types of ROVs.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of the non-ITS communications systems in the GROA facilities using the guidance in the YMRP. DOE clearly described the intended fundamental design and operations plans for the system. The standards and codes listed pertaining to the protocols and interfaces used for communications systems are industry standards that NRC staff considers appropriate. The NRC staff reviewed DOE's response to an NRC staff RAI (DOE, 2009ee) and notes that the descriptions of in-drift communications to the TEV through antennas, slotted microwave guide, electrified third power rail, or other alternative means are potentially feasible concepts and that the information provided is reasonable to evaluate the intended concepts for design and operations.

The ability of the TEV, DSEG, and multiple types of ROVs to perform assigned tasks in nonaccessible subsurface openings is a fundamental component of descriptions of preclosure GROA operations and planned observation and inspection activities as described in the SAR and supplemental materials. The non-ITS communications system provides for remote control of the vehicles and transmission of acquired images, video, and data to remote operators and is required to facilitate the described operations of these vehicles. The capability to reliably perform observations, inspections, and potential maintenance of nonaccessible underground openings and the SSCs within them throughout the preclosure period is important to the evaluations in TER Section 2.1.1.2.3.7.

DOE stated it will further develop potential in-drift communications alternatives during the detailed design phase and included information that indicates that this design will also provide communications within turn-out areas (DOE, 2009ee). Information in the SAR and supplemental materials vaguely described multiple additional ROVs and required communications provisions that will enable them to operate in other nonacccessible areas. DOE described the ROVs as "yet-to-be-designed or specified," representing a DOE statement to further develop ROV designs and related communications alternatives providing communications between vehicles operating in all nonaccessible areas and remote operators during the detailed design phase (BSC, 2008ca). The NRC staff notes that a detailed design should provide a complete design description on communications SSCs located in nonaccessible areas including design, construction, connections, and configurable switching (i.e., at turnouts) and plans for inspection, observation, and maintenance and repair, if needed, during the preclosure period for all ROVs that are planned to operate in nonaccessible areas and must communicate with remote operators/inspectors (DOE, 2009ee). The NRC staff also notes that any SSCs located within nonaccessible areas that are intended to provide or support communications between vehicles operating in nonaccessible areas and remote operators are subject to provisions of monitoring and maintenance plans, addressing such matters as timely and safe repair, consistent with the evaluation in TER Section 2.1.1.2.3.7.

Because an NRC staff evaluation of the communications designs and applicable monitoring and maintenance plans related to the operations of the TEV, DSEG, and other ROVs in nonaccessible areas can await review of final design, a detailed design is not required at this time. DOE's conceptual design description regarding communications related to ROVs operating in nonaccessible areas, along with DOE statement that it will further develop in-drift communications alternatives during the detailed design phase, provides general information and a reasonable understanding of the intended operation of the communications system, and confidence that detailed information will be available with the detailed design (BSC, 2008ca).

Therefore, DOE's description of the communications system, including its statement on further development of design alternatives and detailed design on communications with ROVs when operating in nonaccessible areas, is reasonable and provides confidence that the communications system design will be in accordance with applicable codes and standards and accepted industry practices.

Radiation/Radiological Monitoring System

The function of the RMS is radiological monitoring of both surface and subsurface facilities during the preclosure period. DOE categorized the RMS as non-ITS. SAR Figure 1.4.2-3 provided the RMS functional block diagram for the GROA. The major components are area radiation monitors, continuous air monitors, and airborne radioactivity monitors. The facility RMS is powered by a set of UPS. Although the monitoring equipment can alert the operators to the occurrence of Categories 1 or 2 event sequences, or potential off-normal radiological releases, the RMS does not alert operators to take manual action to mitigate an analyzed event.

Area radiation monitors are not required for the subsurface facilities, because administrative controls will be used to prevent personnel from entering areas that potentially contain high levels of radiation. However, continuous air monitors are located at strategic locations 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 standards and codes applicable to the RMS were identified in SAR Section 1.4.2.2.2, which contains recommendations for obtaining valid samples of airborne radioactive material in effluents and the guidelines for sampling from ducts and stacks. DOE 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). DOE 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 has been heated by decay heat from the emplaced waste. The exhaust shaft sensors will monitor for airborne radioactivity.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of RMS using the guidance in the YMRP and notes that the standards and codes identified in the SAR are applicable for the RMS. The codes for area radiation and continuous air monitoring are applicable because these codes are generally accepted industry standards. Additionally, ANSI N42.18–2004 (American National Standards Institute, 2004aa) states that, regardless of the temperature of the effluent air stream, the instrument system should be capable of operating with less than a 5 percent change in calibration or response over a temperature range of 0–60 °C [32–140 °F], and where greater extremes are expected or greater accuracy is required, protection from the environment is to be provided. Therefore, DOE's information is reasonable to evaluate the RMS for use in the PCSA, as needed.

Environmental/Meteorological Monitoring System

The environmental/meteorological monitoring system, which DOE classified as a non-ITS system, monitors seismic and meteorological parameters for the GROA through the preclosure period and transmits the collected data through the DCMIS so that the data are available in the CCCF. The system performs only monitoring functions, and there are no control functions associated with it. Remotely located environmental/meteorological equipment is powered by solar panels with battery backup, and by UPS for other equipment. DOE confirmed (DOE, 2009dq) that the sensors/monitors used are not Class 1E equipment and there are no plans to qualify these environmental sensors/monitors (SAR Section 1.13) for harsh environments.

Meteorological instruments 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 are 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)] with the exception that the elevation locations for key sensors are commensurate with the operations requirements for the GROA.

The seismic monitoring subsystem consists of triaxial accelerometers, which are 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 Evaluation: The NRC staff reviewed the description of the environmental/meteorological monitoring system provided in the SAR using the guidance in the YMRP and notes that the standards and codes DOE provided for the environmental and meteorological monitoring system are applicable. The NRC staff notes that DOE's approach to design the sensors and monitors consistent with their respective location is reasonable because DOE described the normal acceptable ranges of environmental conditions at those locations. Therefore, the description of design information is reasonable to evaluate the design and operation of the environmental/meteorological monitoring system and use in the PCSA, as needed.

2.1.1.2.3.7 Design of Subsurface Facility Structures, Systems, and Components

This TER section evaluates the design of subsurface structures and systems to determine their capability to perform functions that DOE credited to the structures and systems. The NRC staff's evaluation of the design of SSCs ITS is presented in TER Chapter 2.1.1.7.
The evaluation in this section focuses on the design of structures and systems that are not evaluated in TER Chapter 2.1.1.7 but which DOE relied on to perform functions important to subsurface facility operations

DOE described the subsurface facility operations in SAR Section 1.3.1.2. DOE stated that the operations 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. DOE explained in SAR Sections 1.3 and 1.4 and in its responses to the NRC staff RAIs (DOE, 2009bb,ed) that the subsurface facility structures and systems are designed to provide several functions to support the subsurface facility operations. These functions 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.

To make a case that the subsurface facility structures and systems will perform these functions through the preclosure period, DOE indicated that underground openings and the inverts will remain stable and retain their as-designed alignments and grades through the preclosure period. DOE (DOE, 2009ed) stated that it established appropriate design criteria and basis to ensure stability of the structures and systems and will implement a monitoring, inspection, and maintenance program to ensure any deterioration of the structures and systems will be detected and corrected timely. As explained in SAR Section 1.11 and reviewed by the NRC staff in TER Section 2.1.2, DOE relied on the availability of functions of the subsurface facility structures and systems to make a case that waste packages will be accessible through the preclosure period, such that any necessary retrieval could be performed by reversing the operational procedures used for waste emplacement. In addition, DOE relied on the functions to set the environment for the PCSA of subsurface operations and, in particular, to exclude structural failures that could potentially initiate event sequences (DOE, 2009ed). Furthermore, DOE relied on the functions to support capability to install drip shields to satisfy the geometry and interlocking that DOE specified for postclosure performance assessment.

Therefore, the NRC staff's evaluation in this TER section focuses on determining whether (i) the subsurface facility structures and systems are reasonably designed to provide the functions and (ii) DOE's monitoring and maintenance programs are reasonable to ensure availability of the functions through the preclosure period.

2.1.1.2.3.7.1 Thermal Load and Ventilation Design

DOE 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. DOE categorized the subsurface ventilation system as non-ITS, because it does not prevent or mitigate an event sequence, and as non-ITWI, because the subsurface ventilation system does not function as a barrier to potential release during the postclosure performance period.

Thermal Management Analysis

DOE performed a three-step thermal management analysis to ensure compliance with 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 determines 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, DOE and utilities (BSC, 2007cb). DOE's analysis also assumed that TAD canisters having a heat load as high as 22.0 kW will be aged at the repository aging pads until the emplacement thermal load limit (18.0 kW) is met. For the second step of analysis, DOE 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 DOE's analysis involved evaluating the thermal-hydrologic, geomechanical, and geochemical response to the loading arrangement determined in the previous step (SNL, 2008ai). DOE 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 RAI (DOE, 2009ea), DOE 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, and (iii) the maximum average line load will be 2.0 kW/m [0.61 kW/ft]. According to DOE, 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 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 TSPA {11.45 kW maximum waste package and 1.45kW/m [0.44 kW/ft] linear heat loads}. The thermal load reference case DOE used to assess postclosure performance assumes 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 RAI (DOE, 2009eb), DOE 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. DOE 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 Evaluation: The NRC staff reviewed the information on thermal loading strategy and thermal management using the guidance in the YMRP. The NRC staff evaluated the

DOE-provided information on design assumptions, constraints, design technical basis, uncertainty, and analytical or modeling techniques employed.

The NRC staff reviewed DOE's analytical thermal loading calculations and notes that DOE information on thermal characteristics of the waste in the waste package emplacement plan is reasonable because the emplacement plan is consistent with the expected waste receipt and operations at the surface facilities. The NRC staff also evaluated DOE's thermal-hydrologic, geomechanical, and geochemical studies of the repository for a given waste package emplacement sequence that satisfies the preclosure and postclosure temperature limits. The analytical methods used to assess the repository performance are standard engineering techniques and methods commonly used, and are reasonable for their intended use. DOE reasonably described this thermal analysis technique using a process flow diagram, which was highlighted in SAR Figure 1.3.1-9. On the basis of these evaluations, DOE provided a reasonable description and technical basis information for the thermal loading strategy.

Although DOE 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), the SAR did not provide reasonable supporting documentation and calculations. However, DOE stated that 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. As discussed in TER Section 2.2.1.3.6.3.3, the NRC staff recognizes that peak local temperatures may vary depending on a number of factors including emplacement sequence, waste characteristics, modeling parameters, and the influence of the natural system. The NRC staff notes that DOE stated that it will [SAR Section 1.3.1.2.5; DOE Enclosure 1 (2009ct); DOE (2009eb)] develop a comprehensive emplacement plan prior to actual waste emplacement with specific information on waste characteristics, waste package emplacement location, ventilation duration, and use this emplacement strategy to achieve the preclosure and postclosure performance temperature limits (SAR Table 1.3.1-2). As mentioned previously, DOE stated it would to provide comprehensive emplacement plans annually after emplacement starts. Considering these evaluations and DOE statements, the NRC staff notes that the information in the SAR and the previously cited DOE's statements on proposed thermal loading are reasonable for use in the postclosure performance assessment.

Subsurface Facility Ventilation System

DOE designed 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 include fans, isolation barriers, airflow regulators, access doors, and instrumentation for controlling and monitoring the system. An interconnected system of subsurface openings that consists of intake ramps, access and exhaust mains, access turnouts, emplacement drifts, intake and exhaust shafts, and shaft access drifts is 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, DOE described the operation of the ventilation system during simultaneous emplacement and development in which isolation barriers are 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 isolate (i) inlet airflow from exhaust airflow and (ii) the emplacement and development area was provided in SAR

Section 1.3.5.1.3.2. DOE 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, can vary the drift airflow rate between 0 and 47 m^3/s [0 and 100,000 cfm]. DOE stated that the total power required for ventilation fans at the exhaust shaft will be approximately 1,343 kW [1,800 hp].

DOE provided information on the operability of ventilation system components under normal and off-normal conditions. According to DOE, 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 if the other fails. DOE also contends that small-diameter exhaust shafts will normally operate with only one fan that can deliver 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, are designed according to International Building Code Seismic Use Group I and II (International Code Council, 2003aa). DOE stated that it will use NFPA 801 and 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. These components are not 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) is not necessary.

NRC Staff Evaluation: The NRC staff reviewed the description of the subsurface facility ventilation system provided in SAR Section 1.3.5 using the guidance in the YMRP. On the basis of this review, the NRC staff notes that DOE reasonably described the subsurface ventilation system design as it provides the design basis, component descriptions, system functioning information, interfaces, general operating procedure, design codes and standards, and off-normal performance analysis. The codes and standards DOE proposed are accepted by the industry and used in industrial installations. On this basis, the NRC staff notes that DOE has used appropriate industry codes and methods for designing subsurface ventilation systems.

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 specifically provide details on the operation and function of these components in the event of power failure. However, DOE 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 an immobile emplacement access door does not cause a safety hazard and 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. The

NRC staff notes that, in the event of a power failure, the components of the subsurface ventilation system will continue to operate normally on the basis of DOE's fan installation design that has multiple sources of backup power, and temporarily nonfunctioning equipment does not pose any safety hazard.

Subsurface Ventilation System Maintenance

DOE 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 DOE, emplacement access doors will require regular periodic inspection with the bulkhead and frame requiring minimal maintenance. DOE stated that emplacement door components will have a modular design that facilitates easy replacement. DOE does not plan any routine maintenance activities for door actuators, which will be remotely monitored and replaced, if necessary. DOE also anticipates that the 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. DOE 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, DOE showed that waste package components will not reach the 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 shows that the waste package temperature will be below values mentioned in SAR Table 1.3.1-2. DOE also stated that the probability of an emplacement sequence within the drift, where a naval SNF waste package (12.9 kW) is placed beside a CSNF with the limiting thermal load (18.0 kW), is extremely small. In the third analysis, DOE showed that the ventilation system is capable of maintaining normal airflow with 94 percent localized blockage of a single emplacement drift. DOE also stated that any potential rock fall 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 Evaluation: The NRC staff reviewed the subsurface ventilation system maintenance considerations using the guidance in the YMRP. On the basis of the anticipated low frequency of use and a definite replacement plan, DOE has a reasonable inspection, testing, and maintenance program for emplacement access doors. NRC staff notes that DOE stated it will have a scheduled maintenance plan for airflow regulators, and DOE's statement that locating the regulator on the access main side will protect maintenance personnel from radiation is reasonable. DOE's intention to use commercially available instrumentation for monitoring and following manufacturer's recommendations for maintenance, calibration, and testing is reasonable. DOE cited standard techniques and tools used to perform the thermal analyses and documented important results in SAR Section 1.3.5.3.2.

Therefore, DOE's information regarding the subsurface ventilation system is reasonable for use in the PCSA and postclosure performance assessment.

On the basis of the NRC staff's evaluation of DOE's thermal management analysis, discussed previously, DOE has justified the waste package emplacement sequence through an analysis that accounts for site-specific thermal properties, uncertainties, and engineering input parameters such as ventilation efficiency. The NRC staff also notes that DOE information on ventilation is reasonable because it (i) provided design information and a design basis for subsurface ventilation system components; (ii) provided plans for inspection, maintenance, and replacement for critical components; and (iii) identified accepted codes and standards for ventilation system design. On the basis of review of the information provided in the SAR and supporting documents, the NRC staff notes that DOE's description and discussion of the subsurface ventilation system and thermal design of subsurface facilities provided in SAR Sections 1.3.5 and 1.3.1.2.5 are reasonable.

2.1.1.2.3.7.2 Underground Openings in Accessible Areas

DOE provided the design of underground openings in accessible areas of the subsurface facility in SAR Section 1.3.3. DOE 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 DOE's approach to assure adequate functionality of the openings and their SSCs during the preclosure period, DOE classified the openings as accessible or nonaccessible on the basis of personnel accessibility because of thermal and radiation conditions DOE determined (DOE, 2009bb). According to DOE (DOE, 2009bb), the accessible openings 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 DOE to assure adequate functionality of the openings. DOE 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. DOE 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 Evaluation: The NRC staff used the guidance in the YMRP to review DOE's design of underground openings in the accessible areas of the subsurface facility. The NRC staff notes that the excavation methods DOE selected will minimize construction damage to the surrounding rock and thereby enhance stability of the openings. DOE used well-established empirical methods to select the ground support system (SAR Section 1.3.3.3). DOE selected materials for steel ground support, grout for fully grouted rockbolts, and shotcrete in conformance with established industry standards (SAR Section 1.3.3.3.3).

The NRC staff notes that DOE's design and monitoring and maintenance plan will reasonably ensure that the accessible openings in the subsurface facility will be stable enough during the preclosure period to support the functions DOE described (TER Table 2-1). Therefore, 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) were appropriately designed to satisfy DOE's assumptions in the PCSA or TSPA regarding the geometry and serviceability of the openings during the preclosure period.

2.1.1.2.3.7.3 Underground Openings in Nonaccessible Areas

DOE provided the design of underground openings in nonaccessible areas of the subsurface facility in SAR Sections 1.3.3 and 1.3.4. According to DOE (DOE, 2009bb), the nonaccessible openings consist of the emplacement drifts, turnouts, exhaust mains, exhaust shafts, and shaft access drifts. DOE 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 are high enough to make these openings inaccessible to personnel. DOE stated in SAR Section 1.3.2.4.4.3 that the nonaccessible underground openings have been designed to function without planned maintenance during the preclosure period. In addition, in its response to an NRC staff RAI (DOE, 2009bb,ed), DOE stated that it will monitor the nonaccessible openings remotely to detect progressive deterioration and promptly implement appropriate maintenance to prevent structural failures that could initiate event sequences or interfere with DOE's plan to keep waste packages accessible and ventilated through the preclosure period.

SAR Sections 1.3.3.3 and 1.3.4.4 described DOE's approach to the subsurface facility opening design. DOE selected the ground support system using empirical methods, as described in BSC Section 6.3 (2007an) and BSC Section 6.4 (2007ao), and site-specific rock mechanical properties. DOE then assessed the stability of the resulting design using numerical modeling, as outlined in BSC Section 6.7 (2007an) and BSC Section 6.5 (2007ao). In the numerical model analyses, DOE 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. DOE 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 Section 6.4.3 (2007an) and BSC Section 7.2 (2007ao). DOE also concluded that the repository thermal loading and potential seismic ground motion will not have a significant effect on the damaged zone. According to DOE in BSC Section 7 (2007an), 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 by more than approximately 50 °C [122 °F] during the preclosure period, as described in BSC Section 7 (2007an). DOE also assessed the effect of ground support on stability of the openings. DOE's analysis indicated that rockbolts 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 rockbolts 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). Therefore, DOE concluded that the nonaccessible openings will be stable without planned maintenance through the preclosure period.

According to BSC Section 7.3 (2007ao) and DOE (2009bb), DOE will monitor the mechanical performance of the nonaccessible openings using remotely operated equipment to assess maintenance needs. According to DOE, 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, DOE 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).

DOE will inspect the entire length of the openings annually for the first few years and progressively less frequently if DOE determines that the inspection frequency could be reduced (DOE, 2009bb). DOE stated that maintenance of the openings will be performed only as a contingency measure in cases of significant failure or deterioration (DOE, 2009bb). DOE may inspect areas of failed ground support more frequently to determine when to initiate repair or maintenance. DOE 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).

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's design of underground openings in the nonaccessible areas of the subsurface facility. NRC staff focused on determining whether the design of the nonaccessible openings will satisfy functional requirements DOE established and whether the monitoring and maintenance programs will assure satisfactory structural performance of the openings through the preclosure period.

The NRC staff notes that satisfactory performance of underground openings in nonaccessible areas of the subsurface facility can be assured through monitoring and maintenance and DOE's design was based on well-established empirical rules for ground support system designs and site-specific rock mass mechanical properties. DOE's analysis of the design was based on well-established procedures and numerical analysis computer codes. However, the NRC staff notes that DOE's statement that the underground openings will be stable even without ground support appears inconsistent with DOE's information (DOE, 2009ed) that rock spalling at the drift circumference is expected but will be mitigated by the stainless steel liner. The expectation that the rock will spall or ravel at the drift circumference is consistent with DOE'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 Section 7.3.2 (2004al). The spalling or raveling can be mitigated using the types of surface protection (perforated stainless steel liner, wire mesh, or shotcrete) included in DOE's ground support design, if the surface protection covers the rock surface and is anchored in areas of the rock that will not be affected by spalling or raveling. The spacing and penetration length of rockbolts included in DOE's ground support design are likely to sufficiently anchor the stainless steel liner, shotcrete, or wire mesh. Therefore, DOE's design of the underground openings relies on the surface protection (perforated stainless steel liner, wire mesh, or shotcrete) to assure stability of the openings and on the rockbolts to assure stability of the surface protection. Hence, DOE's statement that the openings will be stable without ground support is inconsistent with the expected behavior of the rock DOE described.

The effectiveness of rockbolts to anchor surface-protection ground support elements could be undermined if the rockbolts corrode during the preclosure period. DOE expects stainless steel rockbolts to perform better than carbon steel rockbolts because the stainless steel material is less susceptible to general corrosion than carbon steel. However, the NRC staff notes that DOE did not assess the potential for other corrosion mechanisms, such as stress corrosion cracking, or provide data to support DOE's expectation that stainless steel rockbolts will perform satisfactorily during the preclosure period. Therefore, DOE did not sufficiently support the assertion that the ground support for emplacement drifts and other nonaccessible openings will function without planned maintenance through the preclosure period (SAR Section 1.3.2.4.4.3; DOE 2009bb,ed,ef).

However, the NRC staff notes that the ground support system failure during the preclosure period can be mitigated using an effective monitoring program to assess the mechanical performance of the openings and to perform timely maintenance. Such a program would be consistent with standard practice in underground space engineering and is needed to assure satisfactory performance of the openings because DOE (i) did not provide data to justify relying on the mechanical ground support components to perform satisfactorily for 100 years;

(ii) modeled the mechanical behavior of Category 1 lithophysal rock using a Young's modulus of 1.9 GPa [2.8 × 10^5 psi] and unconfined compressive strength of 10 MPa [1,450 psi], as detailed in BSC Table 6-4 (2007an) and BSC Table 6-1 (2007ao), which would underestimate potential thermal stress, overestimate the available rock strength and, hence, underestimate potential thermally induced rock damage, considering DOE's data in SAR Figure 2.3.4-30 (see TER Section 2.1.1.1.3.5.4 for details); and (iii) used the thermal loading for the drift wall temperature instead of the exhaust air temperature, as shown in BSC Table 6-8 (2007ao) to model the thermal-mechanical performance of the exhaust airways (exhaust mains and shafts and shaft access drifts), which would underestimate potential thermal stresses and, hence, thermally induced rock damage. DOE's assumptions in items (i) through (iii) above could result in underestimating potential damage of the openings due to rock failure, and effects of rock damage on the capability of the underground openings to support functions, where DOE stated that the underground openings provide (i) base support for crane rails and the operating envelope for the TEV, DSEG, and remote-controlled equipment for the inspection of loaded emplacement drifts and (ii) fresh air and exhaust air conduits for ventilation of disposed waste packages.

The ability to perform effective monitoring in these areas is contingent on availability of power and communications provisions enabling remote inspection and observation. Power, communications, and vehicle SSCs required for remote monitoring, inspection, and observation in nonaccessible areas are evaluated in TER Sections 2.1.1.2.3.2.3 and 2.1.1.2.3.6.2.

For underground openings on the exhaust-air side of the emplacement drifts, the NRC staff requested DOE to clarify its approach to ensure there is sufficient airflow through the exhaust airways if an exhaust main or shaft, shaft access drift, or ventilation raise were blocked by rock collapse. In response, DOE (DOE, 2009ea) stated that its analysis shows that blockage of an exhaust airway due to rockfall is unlikely and, if such blockage were to occur, the configuration of the subsurface facility will ensure sufficient airflow to maintain the design basis ventilation flow rates. DOE also stated that it will monitor (i) any rubble accumulation in the openings to ensure the accumulation does not become a detrimental blockage to ventilation or an obstacle to an inspection vehicle and (ii) any damaged area to ensure the damage does not progress (DOE, 2009ea). For the emplacement drifts and turnouts, the NRC staff requested that DOE clarify its plans to monitor convergence of the openings through measurements at preselected locations and clarify how the plan will adequately preserve operating envelopes and detect potential instability timely. DOE (DOE, 2009ef) stated that 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. In response to an NRC staff request that DOE clarify its approach for using monitoring data to plan for timely maintenance, DOE (DOE, 2009gk) stated that there is no specific time limit for corrective maintenance but it will perform maintenance to correct any encountered problems in a timeframe that meets operational needs. DOE stated that as part of the design process it will provide monitoring and maintenance plans for underground openings in nonaccessible areas of the subsurface facility and the invert and rail structures in the emplacement drifts, and include sufficient details to demonstrate that the plans will assure (i) preservation of an adequate operating envelope for the TEV and DSEG, (ii) timely and safe repair of damaged emplacement drifts or exhaust airway opening, and (iii) sufficient airflow for waste package ventilation. As part of maintaining the functionality of the emplacement drifts throughout the preclosure period to enable remote operations of TEV, DSEG, and ROVs, DOE maintenance plan should include sufficient details to ensure timely and safe repair of invert, rails, and power/communications equipment inside the emplacement drifts (DOE, 2009bb,ea,ef,gk).

On the basis of the evaluation of DOE's responses (DOE, 2009bb,ea,ef,gk), the NRC staff notes that DOE proposes to use (i) monitoring and maintenance as necessary to preserve the operating envelope and operational needs for the TEV and DSEG and (ii) remote-controlled equipment for the inspection of loaded emplacement drifts and to ensure sufficient fresh air and exhaust air flow for ventilation of disposed waste packages.

2.1.1.2.3.7.4 Invert Structure and Rails

DOE described the invert structure in SAR Section 1.3.4.5. The steel invert structure provides a platform that supports the emplacement pallets, waste packages, and drip shields (SAR Section 1.3.4.5.1). The invert also provides a platform that supports the crane rail system for operation of the TEV for emplacement, recovery, and potential retrieval of waste packages, and for operations of the DSEG and the remotely operated inspection vehicles. According to DOE (SAR Section 1.3.4.5.3), the invert is a non-ITS system because it is not relied on to prevent or mitigate a Category 1 or Category 2 event sequence (SAR Table 1.9-1), and the invert is classified as non-ITWI because no credit is taken for the diffusivity of the invert ballast (SAR Table 1.9-8).

DOE used conventional structural methods to design the invert (SAR Section 1.3.4.5.6) and indicated that the design minimizes the need for maintenance during the preclosure period (SAR Section 1.3.4.5.2). The invert was designed to withstand gravitational, thermal, and seismic loading (SAR Section 1.3.4.5.5) and its performance should not be affected by corrosion during the preclosure period (SAR Section 1.3.4.5.1). The steel invert includes transverse beams bolted to four longitudinal beams (SAR Figure 1.3.4-8). The two outermost longitudinal beams at either end of the invert section are 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 are mounted on the two outer longitudinal beams or rail runway beams. After installation of the invert steel structure, the ballast is placed in lifts and compacted to specifications. The ballast material is crushed tuff and fills the voids between the drift rock and the invert steel frame. Completion of the invert structure assembly is followed by installation and alignment of the crane rails.

DOE stated that subsurface facility structures and systems in nonaccessible areas (e.g., turnouts and emplacement drifts) will be inspected remotely to ensure that the onset of a condition that may lead to a structural failure is detected in a timely manner and repaired as needed (DOE, 2009ed). DOE acknowledged that, in the case of nonaccessible areas, high temperature and radiation influence the ability to maintain the structures and systems, making "maintenance inside the nonaccessible areas challenging but not impossible."

Design Criteria and Design Bases

As part of the design criteria (SAR Section 1.3.4.5.5), DOE indicated that the invert was designed for the appropriate worst-case combinations of construction loads, waste package and pallet loads, drip shield loads, thermal loads, and seismic loads. DOE also stated that the invert structures were 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 was designed to accommodate the relatively small structural displacement expected to occur in the emplacement drifts (SAR Section 1.3.4.5.1). Slotted holes are 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 DOE (DOE 2009ed), these design features 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 is to 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 Evaluation: The NRC staff reviewed the design criteria and design bases DOE proposed for the design of the invert structure and rails, using the guidance in the YMRP. The NRC staff notes that the design criteria used for the design of the invert structure and rails are reasonable and consistent with standard engineering practice for the design of similar risk NRC-licensed nuclear facilities.

Design Codes and Standards

DOE specified codes and standards used in the design of the invert structure in SAR Section 1.3.4.5.8. For instance, the structural steel shapes and plates conform to ASTM A 588/A 588M–05 (ASTM International, 2005aa), the crane rail is in accordance with ASTM A 759–00 (ASTM International, 2001aa), the structural steel bolts conform to ASTM A 325–06 (ASTM International, 2006ae), and welding is in accordance with AWS D1.1/D1.1M (American Welding Society, 2006aa).

NRC Staff Evaluation: The NRC staff reviewed the design codes and standards DOE proposed for the design of the invert structure and rails, using the guidance in the YMRP. The codes and standards are in conformance with standard engineering practices and are applicable for their intended use.

Design Loads and Load Combinations

For the design of the invert, the load combinations include the following loads (SAR Section 1.3.4.5.9.1):

Gravitational Loads

Dead loads include the weight of framing and permanent equipment, and attachments. Live loads 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 were designed to withstand DBGM–2 seismic events {associated to a mean annual probability of exceedance (MAPE) of 5×10^{-4}}. The TEV rail and rail runway beams were designed with DBGM–1 seismic loads (MAPE of 1×10^{-3}), as described in BSC Section 3.2.4 (2007cj). DOE 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 were computed on the basis of the equivalent static load method in accordance with NRC NUREG–0800, as outlined in NRC Section 3.7.2 (1987aa). The SAR did not mention whether seismic fault displacements were included in the invert and rails design.

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 are designed in the longitudinal members of the steel invert and the rails in emplacement drifts (BSC, 2007cj) for temperatures up to 200 °C [392 °F].

NRC Staff Evaluation: The NRC staff reviewed the loads and load combinations DOE proposed for the design of the invert structure and rails, using the guidance in the YMRP. The NRC staff notes that the loads and load combinations used for the design of the invert structure and rails are consistent with standard engineering practice for the design of similar risk NRC-licensed nuclear facilities. The NRC staff notes, however, that DOE did not include the potential for fault displacement. The NRC staff notes that seismic faults may lead to displacements of several centimeters [inches] for fault events with an MAPE of 1×10^{-6} (SAR Table 2.3.4-55), which is the frequency threshold for subsurface facilities evaluation. Furthermore, DOE 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 NRC staff notes that seismic fault displacement of the invert and rail will be monitored as part of the DOE maintenance program. The maintenance program is evaluated in TER Section 2.1.1.2.3.7.3.

Design and Monitoring of the Invert Structure and Rails

DOE intends to monitor the invert and rails during the preclosure period, but relies on their design to exclude event sequences that involve potential structural failures. In addition, DOE stated (SAR Section 1.3.4.5.4) that the steel invert structure and rails are not expected to be subjected to any administrative procedure or PSC to prevent event sequences. Nevertheless, DOE 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 regardless of their design bases, subsurface facility structures will be inspected remotely to ensure 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. Furthermore, in response to a clarification question on the previously mentioned NRC staff RAI, DOE stated that a robust design combined with inspection, monitoring, and maintenance will ensure the rails will be functional throughout preclosure period.

NRC Staff Evaluation: The NRC staff reviewed DOE's information regarding the design and monitoring of the invert structure and rails using the guidance in the YMRP. DOE did not provide design calculations for the invert structure and rails. In response to an NRC staff request that DOE clarify its plan to ensure adequate functionality of the invert structure and rails through the preclosure period, DOE (DOE, 2009gl) stated that it will develop an inspection plan for the invert structure and rail that includes a method to measure the rail alignment and grade using photogrammetry, laser scanning, or a method based on using fixed laser targets. DOE will evaluate damage to the invert structure or rail detected through the inspection to determine potential impact to repository operations and need for maintenance. DOE (2009gl) stated that it will develop and implement a remediation method for each case.

On the basis of NRC staff's evaluation and DOE's statement in responses to RAIs discussed previously (DOE, 2009ed,gl), the NRC staff notes that DOE's design of the invert structure and rails along with DOE's statement to monitor and maintain the invert structure and rails, as necessary, is reasonable. The information provides an understanding of the structural capabilities of the invert structure to withstand the effects of operational activities and natural phenomena.

2.1.1.2.4 NRC Staff Conclusions

The NRC staff notes that DOE's description of structures, systems, components, equipment, and operational process activities for the GROA is consistent with the guidance in the YMRP. The NRC staff also notes that DOE reasonably described and discussed the design of surface and subsurface structures, equipment, instruments and controls, and operations of the GROA facility as discussed in this chapter, and the information may be used in the PCSA, as needed.

DOE stated that as part of the design process, it will provide more details on (i) electrical power SSCs for subsurface facility nonaccessible areas (TER Section 2.1.1.2.3.2.3), (ii) surface facility fire protection (TER Section 2.1.1.2.3.2.7), (iii) Remotely Operated Vehicles (ROVs) operating in nonaccessible areas in the subsurface (TER Section 2.1.1.2.3.6.2), (iv) communication SSCs located in nonaccessible areas in the subsurface (TER Section 2.1.1.2.3.6.2), and (v) plans for monitoring and maintenance of underground openings and rails in the emplacements drifts (TER Section 2.1.1.2.3.7.3).

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CHAPTER 3

2.1.1.3 Identification of Hazards and Initiating Events

2.1.1.3.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of DOE's identification of hazards and initiating events in both surface and subsurface facilities of the geologic repository operations area (GROA) during the preclosure period. The objective of the review is to assess the information DOE presented in identifying hazards and initiating events that might affect the GROA design. In this chapter, natural, human-induced, and operational hazards are evaluated. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.6 (DOE, 2008ab); supplemental documents referenced in the SAR; and DOE's response to the NRC staff's requests for additional information (RAIs) (DOE, 2008ah; 2009ap,bg,dn,dx,dy,ed,ej,ey,fa,fe,fh,fi,fm,fn). This information addressed how DOE screened each potential hazard to assess its potential to initiate an event sequence, including specific hazard identification methodology for each type of hazard, screening criteria, data used, and specific analyses conducted.

2.1.1.3.2 Evaluation Criteria

The regulatory requirements for identification of hazards and initiating events as they pertain to the preclosure period are set forth in 10 CFR 63.112(b) and (d). 10 CFR 63.111(c) requires the SAR to include a preclosure safety analysis (PCSA) of the GROA. The 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. An event sequence, pursuant to 10 CFR 63.2, includes one or more initiating events and associated combinations of repository system or component failures. 10 CFR 63.2 also states that Category 1 event sequences are expected to occur one or more times before permanent closure of the GROA and other event sequences having at least 1 chance in 10,000 of occurring before permanent closure are Category 2 event sequences.

10 CFR 63.112(b) requires the PCSA to include an identification and systematic analysis of naturally occurring and human-induced hazards at the GROA, including a comprehensive identification of potential event sequences. Pursuant to 10 CFR 63.112(d), the PSCA must also include the technical basis for either including or excluding specific hazards in the safety analysis. For the purposes of this analysis, 10 CFR 63.21(c)(5) provides that it is assumed that GROA operations will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application.

The NRC staff reviewed DOE's identification of hazards and initiating events information using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), Section 2.1.1.3. The relevant acceptance criteria are (i) technical bases and assumptions for methods used to identify hazards and initiating events are adequate, (ii) site data and system information are appropriately used to identify hazards and initiating events, (iii) determination of frequency or probability of occurrence of hazards and initiating events is acceptable, (iv) adequate technical bases for including and excluding hazards and initiating events are provided, and (v) the list of hazards and initiating events that may result in radiological releases is acceptable.

The NRC staff used additional guidance, such as NRC standard review plans, interim staff guidance (ISG), and regulatory guides, where applicable. These additional guidance documents are discussed in the relevant sections that follow.

2.1.1.3.3 Technical Evaluation

The NRC staff's review of DOE's identification of hazards and initiating events is integrated with its review of site-related information in Technical Evaluation Report (TER) Section 2.1.1.1 and facility-related information in TER Section 2.1.1.2. The NRC staff used the information from these chapters to assess whether all external and internal (operational) hazards at the GROA are identified. For each credible hazard and initiating event, the NRC staff reviewed the identification methodology and data used, the estimated frequency of occurrence and associated uncertainty, and the screening of these initiating events. The NRC staff conducted an audit review of DOE's assessment of hazards and initiating events at the GROA facilities. The NRC staff selected several hazards and initiating events on the basis of their risk potential (e.g., hazards and initiating events with potential to pose significant consequences to the public and worker safety) for review. In addition, some hazards and initiating events were selected for review because they could have high annual frequency of occurrences or annual frequency of occurrences close to the boundaries between Category 1 and Category 2 or Category 2 and beyond Category 2. Additionally, the NRC staff used experience gained in licensing facilities that handle spent nuclear fuel in selecting some of the hazards and initiating events for this risk-informed review.

2.1.1.3.3.1 Naturally Occurring and Human-Induced External Hazards

DOE 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 screening for their potential to initiate event sequences at the Yucca Mountain repository. The NRC staff reviewed and evaluated the naturally occurring and human-induced external hazards by examining DOE's (i) identification of hazards, (ii) screening criteria, and (iii) implementation of the screening criteria.

2.1.1.3.3.1.1 Identification of Hazards

In SAR Section 1.6.3.2, DOE identified potential naturally occurring and human-induced external hazards at the repository that could initiate event sequences (SAR Table 1.6-8). DOE compiled these potential external hazards using the NRC guidance [i.e., NUREG/CR–2300 (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa)] and industry standard [i.e., American Institute of Chemical Engineers (1989aa)] and considered potential hazards specific to the underground facilities (Ma, et al., 1992aa) of the Yucca Mountain repository.

NRC Staff Evaluation: The NRC staff evaluated the information provided in SAR Section 1.6.3.2 regarding DOE's identification of potential naturally occurring and human-induced external hazards using the guidance in the YMRP to determine whether the specific hazards provided in SAR Table 1.6-8 constitute a comprehensive list of external initiating events for screening. DOE used NRC guidance documents developed for the identification of external hazards at nuclear power plants. The use of these NRC guidance documents is reasonable because the types of external hazards identified in the NRC guidance for nuclear power plants 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).

DOE followed NRC guidance and standard industry practice to identify external hazards. The NRC staff also notes that the methods selected are reasonable for the available data on the site and GROA because DOE includes hazards that are specific to the geologic repository at Yucca Mountain as identified in Ma, et al. (1992aa). The NRC staff also reviewed the International Atomic Energy Agency Standard NS–G–1.5 (International Atomic Energy Agency, 2003ab) list of external hazards considered at nuclear power plants in other countries. On the basis of its review, DOE's list of hazards is consistent with the hazards identified in the International Atomic Energy Agency analyses. On the basis of this information, the initial list for screening of naturally occurring and human-induced potential hazards provided in SAR Section 1.6.3.2 is comprehensive and can be used in other sections of the SAR to conduct the PCSA.

2.1.1.3.3.1.2 Screening Criteria

DOE discussed the criteria used to screen external hazards in SAR Table 1.6-1. DOE stated that these criteria were based on NUREG/CR–5042 and NUREG–1407 (Kimura and Budnitz, 1987aa; NRC, 1991aa). The criteria were used to identify those hazards that either are not applicable to the Yucca Mountain repository or have negligible potential to initiate an event sequence over the 100-year preclosure period. The screening criteria used were

1. Can the external event occur at the repository?

2. Can the external event occur at the repository with a frequency greater than 10^{-6}/yr, (i.e., have a 1 in 10,000 chance of occurring in the 100-year preclosure period)?

3. Can the external event, which would be severe enough to affect the repository and its operations, occur at the repository with a frequency greater than 10^{-6}/yr?

4. Can a release, which results from the external event, be severe enough to affect the repository and its operations and occur with a frequency greater than 10^{-6}/yr?

Additionally, DOE used the Requirement EXT–B1 (screening criterion iii) of the American Nuclear Standard Institute/American Nuclear Society (ANSI/ANS)–58.21–2007 (American Nuclear Society, 2007ab) to screen out external hazards that are too slow to develop during the preclosure period to affect the repository significantly. This requirement states that an event can be screened out if the event is slow to develop and it can be shown that sufficient time would be available to eliminate the source of the threat or provide a reasonable response (American Nuclear Society, 2007ab).

NRC Staff Evaluation: The NRC staff examined the discussion provided in SAR Section 1.6.3.4 regarding the screening criteria of external hazards using the guidance in the YMRP to assess whether DOE's screening criteria are consistent with NRC guidance or standard industry practice. The NRC staff notes that DOE's screening criteria are based on NRC guidance and industry standards (e.g., NRC, 1987aa, 1991aa; American Nuclear Society, 2007ab). The NRC staff also notes that DOE's screening criteria are consistent with the guidance in NUREG/CR–2300 (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa). Therefore, DOE's screening criteria are reasonable to screen out external hazards because they are consistent with NRC guidance and standard industry practice.

2.1.1.3.3.1.3 Screening Implementation

As described next, the NRC staff reviewed DOE's screening process to assess whether DOE reasonably screened each of the 89 potential hazards identified in SAR Table 1.6-8. This includes review of DOE's estimation of the annual frequency of occurrences of the hazards. The NRC staff's review of the DOE-identified potential hazards is organized into five areas on the basis of technical subject matter. TER Table 3-1 groups these hazards for the NRC staff's review.

DOE addressed explosion-related hazards outside the GROA under industrial and military activities and inside the GROA under intrasite operations. The NRC staff reviewed the screening of all explosion-related hazards in TER Section 2.1.1.3.3.1.3.5.

Table 3-1. Grouped External Hazards Used in the NRC Staff Review

Review Area	Specific Hazards
Geologic and Geotechnical Hazards	• Seismic Hazards: earthquake, surface and subsurface fault displacement, liquefaction and lateral spread • Volcanic Hazards: lahar, intrusive and extrusive volcanic activities, ash fall • Hill Slope Processes: avalanche, landslide, mass wasting • Slow Geologic Processes: epeirogenic and orogenic diastrophism; tectonic activity (large-scale folding, faulting, uplift, and depression of the Earth's crust); sedimentation; erosion including denudation, coastal erosion, and stream erosion; glaciation and glacial erosion • Processes Affecting Soil Stability: settlement, soil shrink–swell consolidation, static fracturing, subsidence • Subsurface Drift Degradation Processes: drift degradation, fracturing–fractures, rock deformation and rockburst • Undetected Geologic Processes
Weather-Related Hazards	• High Wind: barometric pressure, extreme wind, extreme weather and climate fluctuations, hurricanes • Tornado/Tornado-Generated Missiles • Lightning
Aircraft Crash Hazards	• Aircraft Impact
Industrial and Military Activities	• Induced Air Overpressure • Induced Seismic Motion • Release of Radiological or Toxic Materials • Waste Management • Mining • Commercial Rocket Launch • Release of Onsite Hazardous Materials • Turbine-Generated Missiles
Other Hazards	• External Flooding • Loss of Power • Loss of Cooling • External Fire • Explosions • Extraterrestrial • Waste and Rock Interactions • Perturbation of Groundwater • Improper Design or Operation • Undetected Past Human Intrusions • Sabotage, Terrorist Attack, and War

The NRC staff's review of external hazards focused on the technical basis for excluding potential hazards. Using the guidance in the YMRP, the NRC staff examined whether the technical bases for excluding external hazards and initiating events are reasonable and consistent with site information. To do this, the NRC staff determined whether (i) DOE reasonably used site-specific data to identify external hazards and (ii) DOE's determination of the frequency or probability of occurrence of external hazards is reasonable.

2.1.1.3.3.1.3.1 Geological/Geotechnical Hazards

DOE provided information on geological/geotechnical hazards that could affect the repository facilities during the 100-year preclosure period. SAR Table 1.6–8 identified 5 seismicity-related and 29 nonseismic geological/geotechnical hazards. The seismic hazards include (i) lateral spread, (ii) liquefaction, (iii) earthquake, (iv) surface fault displacement, and (v) subsurface fault displacement.

The nonseismic geological/geotechnical hazards include

- Avalanche
- Coastal erosion
- Denudation
- Dissolution
- Drift degradation
- Epeirogenic diastrophism
- Erosion
- Fracturing–fractures
- Glacial erosion
- Glaciation
- Lahar
- Landslide
- Mass wasting
- Orogenic diastrophism
- Rockburst
- Rock deformation
- Sedimentation
- Settlement
- Soil shrink–swell consolidation
- Static fracturing
- Stream erosion
- Subsidence
- Tectonic activity–uplift and depression
- Undetected geologic features
- Undetected geologic processes
- Volcanic activity
- Volcanism (including intrusive igneous activity, extrusive igneous activity, and ash fall)

The NRC staff assesses these hazards according to the following groups: (i) seismic related, (ii) volcano-related processes, (iii) slow geologic processes, (iv) hill slope processes, (v) processes affecting soil stability, (vi) subsurface-drift-degradation-related processes, and

(vii) undetected geologic processes and features. The NRC staff reviewed dissolution-related hazards in TER Section 2.1.1.3.3.1.3.5.

<u>Seismic Hazards</u>

DOE presented seismic hazard information in SAR Section 1.6 and in associated documents. Seismic ground motion from an earthquake may damage structures, systems, or components (SSCs). If an earthquake triggers fault displacement within the GROA, any structure located above or nearby the fault plane could experience a shearing motion. Also, a cohesionless soil mass could transform into liquid during an earthquake as a result of increased pore pressure resulting in reduced effective stress. Any lateral movement of the soil mass because of this liquefaction may cause lateral spread. Liquefaction of soil below the important to safety (ITS) structures would cause stability problems and may induce damage to the structures.

DOE described the basis for screening of seismic hazards in BSC Section 6.1 (2008ai). DOE screened in all seismic hazards except liquefaction and lateral spread (BSC, 2007bq). DOE (BSC, 2008ai) indicated that liquefaction and lateral spread would not take place at the Yucca Mountain site, because the site soils are dense and not saturated.

NRC Staff Evaluation: The NRC staff reviewed DOE's seismic hazard screening information using the guidance in the YMRP to determine whether DOE used appropriate site-specific information and analysis to include or exclude seismic hazards as initiators of event sequences.

The NRC staff's evaluation of DOE's earthquake information, including the probabilistic seismic hazard analysis, is provided in TER Sections 2.1.1.1.3.5 and 2.1.1.1.3.5.2, where the NRC staff notes that the earthquake-hazard curves derived from the catalogue of regional faults are reasonable for the PCSA. In addition, the NRC staff evaluated the seismically initiated event sequences at the repository facilities in TER Section 2.1.1.4.3.4.2. On the basis of its review discussed in TER Section 2.1.1.4.3.4.2, the NRC staff notes that DOE appropriately assessed the seismically initiated event sequences. The NRC staff evaluated DOE's information on seismically induced rockfall in TER Section 2.1.1.2.3.7.3. On the basis of its review discussed in TER Section 2.1.1.2.3.7.3, the design and information presented on underground openings in inaccessible areas of the subsurface facility is reasonable to preclude effects of seismically induced rockfall.

Because DOE screened in all hazards except liquefaction and lateral spread, the focus of the NRC staff's review is to assess the basis for excluding liquefaction and lateral spread. The NRC staff reviewed the information provided on liquefaction and lateral spread at the Yucca Mountain site (BSC, 2007bq) and notes that the water table at the surface facilities area is approximately 390 m [1,270 ft] below the ground surface, as described in BSC Section 6.1.4.4 (2007bq). The unsaturated soil mass above the water table will not pose a liquefaction hazard or lateral spread hazard, because a saturated soil is necessary to develop excess pore pressure during an earthquake for liquefaction to take place (e.g., Terzaghi, et al., 1996aa). Therefore, DOE's exclusion of liquefaction and lateral spread as initiators of event sequences is reasonable.

<u>Volcano-Related Hazards</u>

DOE presented five volcano-related hazards in SAR Section 1.6.3.4.3 and BSC Section 6.3 (2008ai). DOE provided additional information in response to an NRC staff RAI (DOE, 2009ap).

A nearby volcanic eruption may produce a lava flow at the surface, a mud flow of volcaniclastic materials (known as a lahar), or airborne volcanic ash. In the subsurface, magma from the volcano may flow through fractures and affect the emplacement drifts. Airborne volcanic ash can clog the heating, ventilation, and air conditioning (HVAC) filters and the natural ventilation ports of the aging casks. Settled ash provides an additional load on the roof of a waste handling facility. DOE provided information on five volcano-related hazards—lahars, volcanic activity, volcanism-intrusive igneous activity, volcanism-extrusive igneous activity, and volcanic ash fall—and screened out all five as not having the potential to initiate event sequences in the preclosure period.

Four volcano-related hazards (lahars, volcanic activity, volcanism-intrusive igneous activity, volcanism-extrusive igneous activity) are associated with a direct intersection of the surface or subsurface repository facilities by a volcanic event. DOE estimated the mean frequency of intersection of the subsurface repository footprint by a volcanic event to be 1.7×10^{-8}/yr and the mean frequency for the development of an eruptive conduit to the surface to be 4.7×10^{-9}/yr. These result in a probability of direct volcanic impacts to repository facilities well below 10^{-4} during the preclosure period, making direct volcanic impacts beyond Category 2 events.

Ash fall hazards could originate from volcanic eruptions away from the immediate repository location. Ash fall hazards could result in these initiating events: blockage of overpack vents on the aging pads, accumulation of additional loads on surface facility roofs, clogging HVAC filters, and loss of cooling associated with HVAC failure. DOE estimated a mean annual frequency of 6.4×10^{-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); the roofs of the surface facilities are designed with a live load of 10.25 g/cm^2 [21 psf]. In addition, using the bulk density of ash from Mount St. Helens in 1980, DOE 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, DOE indicated that it would take maintenance and remedial actions after an ash fall event to remove ash and unclog the vents of the HVAC system and aging overpacks (SAR Section 1.6.3.4.3). In addition, temporary ventilation systems would be used, if necessary. Consequently, DOE excluded ash fall from further consideration.

NRC Staff Evaluation: The NRC staff reviewed DOE's information about volcano-related hazards using the guidance in the YMRP to determine whether DOE reasonably estimated the frequency or probability of occurrence of volcanic hazards to include or exclude volcanic-related hazards as initiators of event sequences.

For the four volcanic hazards associated with direct impacts to the repository facilities, the technical basis for DOE's exclusion relied on the low probability of igneous intersection of the repository. The estimated frequency of the hazards is reasonable because DOE relied on the same frequency and probability analyses of volcanic events for the postclosure period, and the NRC staff notes in TER Sections 2.2.1.2.1, 2.2.1.3.10, and 2.2.1.3.13 that those analyses are reasonable. The NRC staff notes that even though the area DOE used to estimate the probability was based on the subsurface facility footprint, applying uncertainties of two orders of magnitude on DOE's estimated probabilities of 4.7×10^{-9}/yr for the development of an eruptive conduit to the surface would screen out direct volcanic impacts to repository facilities. Consequently, the estimated probability of occurrence of those events would remain below 10^{-6}/yr. The NRC staff therefore notes that DOE reasonably estimated the frequency or probability of occurrence of direct volcanic impacts to the repository and that DOE provided a reasonable technical basis to exclude direct volcano-related hazards from initiating event sequences.

For ash fall hazards, DOE determined that the probability of aerial ash fall density greater than 10 g/cm^2 [20 psf] is well below 10^{-6}/yr. The NRC staff notes that DOE's assessment of the ash-loading hazard to surface facilities, presented in SAR Section 1.6.3.4.3, and the response to an NRC staff RAI (DOE, 2009ap) provide a reasonable basis for DOE's calculation for probability of future basaltic ash falls on the repository GROA in the preclosure period. The thicknesses at expected distances from a nearby future volcanic eruption estimated from the ASHPLUME model were consistent with those estimated in other parts of the SAR for possible postclosure deposition of radionuclide-contaminated ash (tephra) in the event of a future eruption (SAR Section 2.3.11.4.1.1.2; BSC, 2004bk)—a scenario that is reviewed in detail in TER Section 2.2.1.3.13.1. DOE showed in BSC Section 6.8 (2008ai) that the wet handling facility (WHF) pool, the only place that may have a radiological consequence because of loss of cooling, would be able to maintain shielding for at least 30 days. Because DOE reasonably estimated the annual frequency of ash fall, DOE provided a reasonable technical basis to exclude ash fall as a potential initiating event and, even if ash fall occurs, remedial action can be taken to remove ash from HVAC and aging overpacks to prevent clogging.

Slow Geologic Processes

DOE provided information on potential hazards from slow geologic processes in SAR Section 1.6.3.4.2 and BSC Section 6.2 (2008ai). Slow geologic processes include 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. These generally slow processes may render the waste emplacement areas unsuitable for placing or disposing waste forms during the preclosure period, necessitating a long-term solution. DOE screened out these processes as not having potential to initiate an event sequence during the preclosure period, largely because these hazards would progress slowly over time, which would allow for remedial actions.

DOE stated that continental-scale vertical movements of the Earth's surface and its compressional deformation could occur relatively uniformly over the repository site. As these effects would progress slowly, there would be sufficient time to take any remedial actions to prevent event sequences from developing. Similarly, the rate of the tectonic activities, resulting surface uplift, and depression are slow enough not to significantly affect the repository in the preclosure period. DOE indicated that even if this activity manifests itself in the preclosure period, there would be sufficient time to take necessary remedial actions (BSC, 2008ai). Consequently, DOE excluded diastrophism and tectonic activity from further consideration as external hazards (SAR Section 1.6.3.4.2; BSC, 2008ai).

DOE described sedimentation as the transport and deposition of particles by wind and water. It is a slow process occurring unevenly at the site area, with topography playing a major role in the location and amount of sedimentation. DOE excluded this external hazard from further consideration on the basis of the overall slow rate of the sedimentation process and its dependence on topographic features. The slow rate of progression will provide ample time to consider waste relocation if sedimentation effects pose a hazard during the preclosure period (BSC, 2008ai).

DOE stated that the progression rate of both denudation and erosion is slow and would allow sufficient time for remedial actions to be taken to prevent event sequences from developing

(SAR Section 1.6.3.4.2). Consequently, DOE 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, DOE also excluded coastal erosion as a potential hazard. Currently, there are no intermittent or continuous flowing streams at the site. Consequently, DOE excluded stream erosion as a potential hazard in the preclosure period.

DOE stated that the current climatic conditions at the repository site would not allow glacier formation. Therefore, DOE 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 Evaluation: The NRC staff reviewed DOE's information on exclusion of tectonic activity using the guidance in the YMRP to examine whether DOE's technical basis for excluding tectonic activity, sedimentation, erosion, and glaciation as potential initiating events is reasonable and consistent with site information. To do this, the NRC staff determined whether DOE showed that these hazards are not credible for the repository facilities, because they would progress slowly over time and allow remedial actions.

DOE's assessment that the potential effects of tectonic movement and sedimentation would be inconsequential for the repository safety during the preclosure period is reasonable because tectonic movement and sedimentation would progress slowly over the preclosure period. The NRC staff therefore notes that tectonic movement and sedimentation are not credible hazards for the repository facilities and that the probability of such hazards is low.

DOE's screening of denudation and erosion from hazard consideration is reasonable because denudation and erosion rates are slow. The site is significantly inland from nearby coastlines, rendering coastal erosion noncredible. Currently there is no stream inside the GROA boundary near the planned surface facilities. Although openings for ventilation shafts for the subsurface facilities could be located near existing channels on the slopes of Yucca Mountain, information on the location and design of ventilation shafts obtained during the construction phase can be reviewed to ensure that ventilation shafts will not be adversely affected by erosion in existing channels on the Yucca Mountain slopes. Therefore, erosion would not initiate event sequences during the preclosure period.

There are no glaciers at or near the repository site, and the climatic conditions are not favorable for formation of a glacier. Consequently, DOE's assessment that glacial erosion and glaciation are not initiating events at the repository during the preclosure period is reasonable.

Hill Slope Processes

DOE provided information in SAR Section 1.6.3.4.2 and BSC Section 6.2 (2008ai) on hill slope processes that included avalanche, landslide, and mass wasting events. DOE provided additional information in responses to an NRC staff RAI (DOE, 2009fe).

Triggered by avalanche, landslide, and mass wasting events, loose soil, rock, or ice/snow may slide down from nearby hill slopes under the force of gravity and bury or otherwise impact parts of the surface facilities. As listed in SAR Table 1.6-8, these hazards include avalanche, landslide, and mass wasting events. DOE determined that further evaluation was not warranted, as it is unlikely that sufficient snow, ice, or loose rock would develop into an avalanche at the repository site. Additionally, the surface construction site would be leveled and compacted. DOE screened out avalanches as not having the potential to initiate an event sequence during the preclosure period, as described in DOE Section 1.6.3.4.2 (2008ab). DOE 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).

NRC Staff Evaluation: The NRC staff reviewed the hill-process information using the guidance in the YMRP to determine whether DOE used appropriate site-specific information and analysis to screen out avalanche, landslide, and mass wasting hazards from initiating event sequences.

The NRC staff notes that DOE provided reasonable site-specific data on maximum monthly snowfall and extreme temperature range and terrain slopes in BSC (2008ai) to assess whether avalanches can be a potential hazard. On the basis of these site-specific data, the anticipated snowfall near the surface facilities is small. Additionally, 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 notes that DOE's assessment that a snow avalanche would not be a potential initiator for event sequences to affect the surface facilities is reasonable.

DOE did not reasonably use site data to assess whether landslide and mass wasting can be a potential hazard. DOE stated in its response to an NRC staff RAI (DOE, 2009fe) that the flat topography of the surface GROA was not conducive to generating a mass wasting event. However, on the basis of DOE's layout of the surface facilities, the NRC staff notes that certain facilities would be built at the base of Exile Hill's eastern slope and that mass wasting events, although rare, do occur and cannot be excluded by simply pointing to the absence of suitable topography and geology. The 1984 Jake Ridge event is an example of such a mass wasting event (SAR Sections 1.1.7.2.1 and 1.1.7.2.2; DOE, 2009fe). The topographic and geologic characteristics of Jake Ridge and Exile Hill are comparable. Their tops are composed of competent, not easily erodible tuff rock, and their steepest topographic slopes are in a 20 to 30 percent range and are partly covered with loose rock debris and colluvium. Therefore, a mass wasting event analogous to the Jake Ridge event could occur on the eastern slopes of Exile Hill.

Because DOE did not justify why a mass wasting event analogous to the Jake Ridge event could not be triggered on the eastern slopes of Exile Hill, the NRC staff notes that DOE did not provide reasonable technical bases to exclude potential mass wasting and landslide hazards. However, the NRC staff notes that in addition to excluding a mass wasting event on Exile Hill eastern slopes based on the absence of suitable topography and geology in SAR Section 1.1.4.1.2.2, DOE also excluded this event based on two storm water drainage diversion channels that would be constructed to protect the surface GROA and the North Portal from storm water runoff and debris flow emanating from Exile Hill. The NRC staff reviewed the design details of the diversion channels that DOE stated it would use to mitigate mass wasting or landslide events in addition to probable maximum flood (PMF) in TER Section 2.1.1.7.3.1.3, and the NRC staff notes in that TER section that the diversion channels would protect the surface GROA and the North Portal from storm water runoff and debris flow emanating from Exile Hill. Therefore, the potential landslide and mass wasting hazard would be prevented because of the diversion channels.

<u>Processes Affecting Soil Stability</u>

DOE provided information about the potential impact of processes affecting soil stability in SAR Section 1.6.3.4.2 and in BSC Section 6.2 (2008ai). DOE provided additional information in its response 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. DOE screened out 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 remedial actions to be taken to prevent event sequence development.

DOE dismissed settlement of surface facility structures from being a potential hazard at the repository site as the effects would develop slowly over time, allowing remedial actions (such as alternative locations of placing waste forms) until a longer term solution is implemented (BSC, 2008ai). Soil consolidation and soil shrink–swell due to drying and wetting can result in fissures and cracks in the ground (DOE, 2009ey). 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, DOE stated that repository site subsidence would be localized (BSC, 2008ai). DOE excluded subsidence on the basis of the overall slow progress that would allow necessary remedial actions to be taken to prevent event sequence development.

NRC Staff Evaluation: The NRC staff reviewed the information on the processes affecting soil stability using the guidance in the YMRP to determine whether the technical bases for excluding external hazards and initiating events are reasonable.

The NRC staff reviewed DOE’s assessment of potential hazards from settlement of surface facility structures in TER Section 2.1.1.1.3.5.4. On the basis of its review discussed in TER Section 2.1.1.1.3.5.4 regarding site-specific geotechnical conditions and stability of subsurface materials, the NRC staff notes that DOE’s information and analyses are reasonable to assess the engineering design and performance of the structural foundation related to potential settlement hazard.

Because the climate at the repository is arid, the soil is not expected to undergo repeated wetting and drying cycles. Therefore, the NRC staff does not expect fissures in the soil or consolidation of the soil mass due to shrink–swell cycles to be a hazard at the site. Additionally, the NRC staff notes that the subsurface facilities are in relatively competent rock mass at a depth of 300 m [984 ft] or more (SAR Section 1.1.5.3.1.1). DOE stated that 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, quite small. Because the excavations will be supported and the extraction ratio is small, NRC staff does not expect that the subsurface facilities will experience massive collapse of the thick {more than 300 m [984 ft]} overlying strata. Therefore, the potential subsidence resulting from massive collapse of rock strata above the emplacement drifts is not likely.

<u>Subsurface Drift Degradation Processes</u>

DOE provided information about drift degradation hazards in SAR Section 1.6.3.4.2 and in BSC Section 6.2 (2008ai). DOE provided additional information in its response to an NRC staff RAI (DOE, 2009ey).

DOE indicated that drift degradation processes include drift degradation, fracturing–fractures, rock deformation, and rockburst. Stress-induced fractures (i.e., cracks, joints, faults) in rock are expected around the emplacement drifts; however, their effects would be localized around the drifts. It is unlikely that these types of fractures would form at the emplacement drift walls and roofs 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 under a very high stress field. All these hazards can contribute to drift degradation. DOE assessed the potential degradation of the emplacement drifts during the preclosure period and concluded that drifts will be stable without ground support (BSC, 2007ai). DOE 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). DOE concluded that these hazards would not cause any adverse effects on the GROA facilities during the preclosure period (BSC, 2008ai).

NRC Staff Evaluation: The NRC staff reviewed DOE's information on potential drift degradation using the guidance in the YMRP to examine whether DOE's technical bases for excluding external hazards and initiating events are reasonable. To do this, the NRC staff determined whether (i) site data were reasonably used in identifying external hazards and (ii) the hazards could occur within the preclosure period. DOE used site-specific data for subsurface drift degradation process hazard identification. The NRC staff notes that the stress-induced deformation and associated fracture formation in the rock mass would contribute to degrading the stability of subsurface excavations in the preclosure period; however, rockburst potential would be negligible due to the relatively low stress field in conjunction with relatively softer rock mass, as given in BSC Tables 3-4 and 3-5 (2007an).

Undetected Geologic Processes and Features

DOE discussed the impact of undetected geologic processes and features in SAR Section 1.6.3.4.2 and BSC Section 6.2 (2008ai). DOE screened out undetected geologic processes and features as not having the potential to initiate event sequences during the preclosure period, because these processes and features would either (i) occur too slowly over the preclosure period to have any consequences or allow enough time to implement mitigation measures or (ii) not occur at the repository site.

NRC Staff Evaluation: The NRC staff reviewed the information on undetected geologic processes and features using the guidance in the YMRP. The NRC staff notes that although undetected geologic features and processes (e.g., new fault and/or fracture zones, new evidence of past or current seepage in drifts) might be discovered during the repository construction phase, time would be available during preclosure operations to address potential hazards from the newly discovered geologic features and processes. Because time would be available during preclosure operations to address the potential hazards from newly discovered geologic features and processes, DOE's screening of these features and processes as potential event initiators is reasonable.

2.1.1.3.3.1.3.2 Weather-Related Hazards

DOE provided weather-related hazard information in SAR Sections 1.6.3.4.4 and 1.6.3.4.6. SAR Table 1.6-8 identified seven weather-related hazards: (i) barometric pressure, (ii) extreme wind, (iii) extreme weather and climate fluctuations, (iv) hurricanes, (v) tornadoes, (vi) missile impact, and (vii) lightning. Unless the repository facilities are appropriately designed against these hazards (i.e., the design bases and design criteria of ITS SSCs account for values appropriate for the site and/or facility), these SSCs may sustain damages leading to radiological consequences. The NRC staff's evaluation of weather-related hazards evaluates how DOE quantified the weather-related hazards that could affect the repository facilities' operations during the 100-year preclosure period. The NRC staff's assessment is provided in three hazard groups: high winds, tornado and tornado-induced missiles, and lightning. Note that hazards from missile impact deal only with tornado-generated missiles. Any hazard from missile testing at the Nevada Test and Training Range (NTTR) is reviewed in TER Section 2.1.1.3.3.1.3.3.1.

<u>High Winds</u>

DOE presented information on high-wind hazards in SAR Section 1.6.3.4.4 and BSC Section 6.4 (2008ai). High-wind-speed-related hazards include barometric pressure, extreme wind, extreme weather and climate fluctuations, and hurricanes. Rapidly falling barometric pressure indicates an onset of stormy weather that could result in high wind speeds. Extreme weather and climate fluctuations include larger weather or climate fluctuations from normal—besides extreme heat or cold, flooding, and drought—that could result in high winds. Hurricanes can also generate high winds. A structure can sustain damages from high wind speeds if it is not designed appropriately. The design basis wind speed is related to both straight and tornado winds.

American Society of Civil Engineers (2006aa) classifies the Yucca Mountain area as a special wind region requiring site-specific data. Consequently, DOE collected wind data from Site 1, approximately 1.0 km [0.6 mi] south of the North Portal. DOE estimated the maximum 3-second gust straight wind for the Yucca Mountain site to be 193 km/h [120 mph] at an annual frequency of occurrence 1.0×10^{-6} (BSC, 2007dc). DOE estimated the wind speed using the Fisher-Tippett Type I extreme value distribution suggested by Simiu and Scanlan (1996aa) and ASCE/SEI 7–05 (American Society of Civil Engineers, 2006aa). Additionally, DOE indicated that wind speed capable of causing significant damage to ITS structures is not expected from hurricanes at the repository site as the nearest sea or ocean is Santa Monica Bay near Los Angeles, California, approximately 360 km [225 mi] away (BSC, 2008ai).

As it travels over land, including the mountainous region to reach Yucca Mountain, a substantial amount of hurricane energy would dissipate. In addition, no rivers and estuaries in the intervening areas could serve as pathways to transmit hurricane storm surge to the Yucca Mountain site. SAR Section 1.6.3.4.4 specified that the maximum design tornado wind speed at the repository site is 304 km/h [189 mph] for ITS structures, which is larger than the estimated straight-wind speed. Consequently, DOE determined that straight wind is not an external hazard severe enough to affect the repository.

NRC Staff Evaluation: The NRC staff reviewed the information and the methodologies DOE used to assess the straight-wind-related hazards using the guidance in the YMRP. The NRC staff's review focused on whether DOE reasonably (i) characterized the straight wind speed and (ii) used site-specific data information in wind speed characterization. The NRC staff referenced Simiu and Scanlan (1996aa), ASCE/SEI 7–05 (American Society of Civil Engineers, 2006aa), and ANSI/ANS–2.8 (American Nuclear Society, 1992ab) and DOE's response to an NRC staff RAI (DOE, 2009fe). Specifically, the NRC staff evaluated how DOE screened out initiating events arising from high winds that could affect an ITS structure.

On the basis of its review of information presented in the SAR Section 1.6.3.4.4 and BSC (2008ai, 2007dc) regarding straight wind hazards, the NRC staff notes that DOE used reasonable probabilistic methodology to estimate the straight wind speed, as suggested by ASCE/SEI 7–05 (American Society of Civil Engineers, 2006aa) and Simiu and Scanlan (1996aa). In addition, DOE's characterization of the hazards associated with straight wind is reasonable because DOE used site-specific wind information.

The NRC staff notes that DOE's assessment that hurricanes generated in the Pacific Ocean would not enhance the straight wind speed significantly is reasonable because the coastline is at least 360 km [225 mi] away with mountains in the intervening region. For comparison, ANSI/ANS–2.8 (American Nuclear Society, 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.

Tornado/Tornado-Generated Missiles

DOE presented information on tornado-related hazards in SAR Section 1.6.3.4.4 and BSC Section 6.4 and Attachment A (2008ai). An ITS structure can sustain damages from either tornado wind or impact of a tornado-generated missile if it is not appropriately designed.

DOE estimated the site-specific tornado characteristics at the Yucca Mountain site using information from NUREG/CR–4461 (Ramsdell and Andrews, 1986aa). A 4° latitude-longitude box was used as the region has relatively low tornado activities. However, DOE developed a modified 4° box keeping Yucca Mountain near the center to better estimate the tornado strike probability using information from sixteen 1° boxes [given in Ramsdell and Andrews (1986aa)] surrounding the site, as detailed in BSC Attachment A (2008ai). The tornado strike probability includes both point strike and lifeline. Lifeline probability is based on frequencies that account for the finite size of the surface facilities, whereas point strike probability does not include consideration of structure dimensions.

DOE used a 100-year preclosure period with an exposure factor of 0.5 to account for the 50-year life of these structures. This represents the fraction of time the facility or a system is at risk of a tornado or a tornado-generated missile strike. Site transporters and the transport and emplacement vehicle (TEV) are exposed to a potential tornado only when they are outside the waste handling buildings or the subsurface facility. On the basis of the fraction of the transit time to which these equipment would be vulnerable to a tornado strike, DOE estimated that site transporters and the TEV would be susceptible to a tornado or a tornado-missile strike for 3.9 and 4.1 years, respectively. The 3.9-year exposure time for the site transporter was based on the time estimated for moving aging overpacks between the surface facilities and aging pads. Similarly, exposure time for the TEV was based on the time estimated for moving waste packages to the underground facility. The exposure time for each operation was assumed to take 2 hours for the site transporter and 3 hours for the TEV. Using this information and considering the dimensions of the facilities, DOE estimated that the probability of a tornado strike during the preclosure period to the surface facilities including the Railcar and Truck Buffer area and aging pads would exceed 1.0×10^{-4} in the preclosure period and, therefore, tornado strike is a credible hazard to these facilities. DOE estimated the probabilities of a tornado striking a site transporter or a TEV would be less than 1.0×10^{-4} and, therefore, concluded that a tornado strike does not pose a credible hazard during preclosure operations.

DOE estimated the potential damage from tornado wind and tornado-generated missiles given a tornado strike with an annual frequency of occurrence of 10^{-6}. DOE used data and methodology given in Ramsdell and Andrews (1986aa) to estimate this conditional probability. This conditional probability of structural damage was estimated using the overhead doors at the entry vestibules as a surrogate for the Canister Receipt and Closure Facility (CRCF), Receipt Facility (RF), and WHF and using a sheet metal exterior wall for the steel structure of the Initial Handling Facility (IHF). DOE considered a structure damaged if the surrogate for that structure sustained damage at a given wind speed. DOE estimated the damage probability of the overhead door of these facilities using information relating door damage with the wind speed, as recommended by Texas Tech University (2006aa). The CRCF was used in the estimation as it has higher tornado strike probability due to its larger footprint. The structural failure probability of the surrogate overhead door of the CRCF was estimated to be less than 1 in 10,000 during the preclosure period. Therefore, DOE concluded that waste handling facilities would not sustain any structural damage from tornadoes to initiate event sequences during the preclosure period.

As no realistic surrogate was identified for the Railcar and Truck Buffer area, aging pads, transportation casks, and aging overpacks, DOE estimated the pressure associated with the tornado and correlated it with the failure probability of a material with a specified thickness, as given in the International Atomic Energy Agency safety guide (International Atomic Energy Agency, 2003ab). On the basis of analysis results, DOE concluded in BSC Section A3.2 (2008ai) that the probabilities of adverse tornado wind effects on the Railcar and Truck Buffer area, aging pads, transportation casks, and aging overpacks would be lower than the screening threshold of 1.0×10^{-4} for the preclosure period.

In BSC Section A3.3 (2008ai), DOE used the classification of tornado-generated missiles of Coats and Murray (1985aa) to assess the effects of these missiles on ITS structures. On the basis of tornado wind speed estimated at an annual frequency of occurrence of 10^{-6}, DOE concluded that heavy missiles, such as utility poles or automobiles, would not be expected at the repository during the preclosure period. For the common sources of tornado-generated missiles when construction and operation of different waste handling facilities are occurring simultaneously, DOE estimated a 10.5-year duration for timber planks and 7.6-cm [3-in]-diameter pipes. The analysis showed that the probability of such missiles impacting the waste handling facilities would be lower than 10^{-4} for the preclosure period (BSC, 2008ai). After the construction is complete, DOE indicated that (i) the tornado-generated missiles are expected to be small debris onsite and (ii) imbedded pipes will not become a tornado-generated missile as the expected tornado wind speed would be lower than required to uproot them from Earth. DOE estimated the penetration depth of a 5 × 10 cm [2 × 4 in], 2.3-kg [5-lb] piece of lumber generated by a 304-km/h [189-mph] wind. Results showed that the estimated penetration depth was significantly smaller than the wall thickness of aging overpack, waste handling buildings, transportation casks, and TEV. Consequently, DOE eliminated tornado-generated missiles as initiators of event sequences during the preclosure period.

NRC Staff Evaluation: The NRC staff reviewed the information DOE used to assess the tornado and tornado-generated missiles hazards, as given in SAR Section 1.6.3.4.4 and BSC Section 6.4 and Attachment A (2008ai) using the guidance in the YMRP. In addition, the NRC staff reviewed DOE's response to an NRC staff RAI (DOE, 2009ey). The NRC staff referenced NUREG/CR–4461 (Ramsdell and Andrews, 1986aa) and data provided by Texas Tech University (2006aa) and the International Atomic Energy Agency (2003ab).

DOE used the data and methodology proposed in NUREG/CR–4461 (Ramsdell and Andrews, 1986aa) to assess the tornado hazard at the repository facilities. Because this document is used as NRC guidance and in the industry, the data and methodology used are reasonable to assess the tornado hazard at the repository. Additionally, DOE used the tornado strike data near the repository site from NUREG/CR–4461 (Ramsdell and Andrews, 1986aa) to estimate the annual strike frequency. Therefore, DOE used appropriate site-specific data in its assessment of tornado hazard. The NRC staff also notes that use of a 4° box would be appropriate for this assessment because the number of observed tornadoes in the Yucca Mountain region is low.

DOE's use of the 50-year exposure time to estimate the tornado strike frequencies for the GROA facilities is reasonable because this exposure time is consistent with the expected duration of operations at the waste handling facilities, as indicated in SAR Chapter 1. Because DOE estimated the 3.9-year exposure time for the site transporter using the overpack numbers and the 4.1-year exposure time for the TEV using the waste package numbers consistent with the throughput numbers listed in SAR Table 1.7.5, these exposure times are reasonable. On the basis of its review of the DOE analysis, the NRC staff notes that tornado strike is a potential hazard to the repository facilities except for site transporters and the TEV, because the estimated annual frequency exceeds the Category 2 limit.

DOE's use of the data Texas Tech University (2006aa) provided to the National Weather Service to assess the potential tornado-strike damage to the WHF, Railcar and Truck Buffer area, and aging pads is reasonable because these data are for commercial structures; nuclear facilities will experience less damage than commercial structures when subject to tornado strike because there are more stringent design specifications for nuclear grade structures. The NRC staff notes that using data from the International Atomic Energy Agency is reasonable because these data are applicable to nuclear power plant designs and nuclear waste handling facilities. The approach of using surrogates to estimate the damage potential is reasonable because the surrogates used are weaker than the waste handling facilities. Consequently, this approach will lead to a conservative estimation. DOE converted the static pressures at failure to equivalent wind speeds using the Bernoulli's equation and subsequently combined the probability of structural failure at the converted wind speed with the tornado strike probability when estimating failure probabilities of transportation casks and aging overpacks; the NRC staff notes this approach is reasonable because it results in a bounding probability.

DOE's analyses of tornado-generated missile strikes on ITS structures and systems using site-specific tornado wind speeds, construction information, and available missile materials at the site are reasonable. DOE assumed rigid missiles that are widely used in assessing missile penetration depth in nuclear power plants and other structures. In addition, the NRC staff notes that DOE's use of the missile classification scheme McDonald (1999aa) proposed is reasonable for this assessment because this method has been previously used in analyses related to licensing activities for NRC-regulated fuel cycle facilities.

DOE assessed the potential for initiating an event sequence during the preclosure period by a tornado-missile strike using site-specific information. The NRC staff notes that DOE's determination that heavy tornado-generated missiles, such as utility poles or automobiles, would not be credible at the repository site is reasonable because the expected wind speed at this site is significantly less than 400 km/h [250 mph] for these items to be credible missiles in a tornado strike (Coates and Murray, 1985aa).

The NRC staff also notes that DOE's assessment is reasonable that all light and medium tornado-generated missiles would be construction related for the first 10.5 years when simultaneous operations and construction would continue. The approach DOE used to determine that light and medium construction-related missiles would not be a credible threat based on the tornado strike probability of the WHF coupled with the 10.5-year period is reasonable because a conservative period of 10.5 years was used in the estimation. The penetration depths DOE estimated resulting from a 5 × 10 cm [2 × 4 in], 2.3-kg [5-lb] piece of lumber generated by a 304-km/h [189-mph] wind are conservative because (i) the formula used for missile penetration is for rigid bodies, unlike these missiles from onsite debris, and (ii) the wind speed used in the calculation exceeds that expected at the site.

As discussed, DOE used the Bernoulli's equation to estimate the probability of failure of transportation casks and aging overpacks. The estimated annual frequency of damage of transportation casks and aging overpacks on aging pads is well below 10^{-6}. Consequently, DOE's exclusion of the tornado wind hazards for the Railcar and Truck Buffer Area and aging pads in its PCSA is reasonable.

Lightning

DOE presented the lightning hazard information in SAR Section 1.6.3.4.6 and BSC Section 6.6 (2008ai). An ITS structure can sustain damages from a lightning strike if not designed appropriately. 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 strike density ranges from 0.06 to 0.4 strikes/km^2/yr [0.16 to 1.04 strikes/mi^2/yr]. A National Oceanic and Atmospheric Administration report stated that the strike density for the Yucca Mountain area is 0.2 flashes/km^2/yr [0.52 flashes/mi^2/yr] (BSC, 2008ai). Assuming the protected area of the GROA is 2.7 km^2 [1.04 mi^2], DOE estimated that the annual average is 0.54 lightning strikes (SAR Section 1.6.3.4.6; BSC, 2008ai). As this annual strike rate exceeds 1 in 10,000 over the 100-year preclosure period, DOE proposed design features for the ITS structures and systems to withstand a lightning strike (SAR Section 1.6.3.4.6; BSC, 2008ai).

DOE proposed to install a lightning protection system for buildings and outdoor elevated structures (BSC, 2007av) including all ITS facilities (BSC, 2008ai) in accordance with (i) National Fire Protection Association (NFPA) 780–2004 (National Fire Protection Association, 2004aa), (ii) Underwriters Laboratories 96A (Underwriters Laboratories, 2005aa), and (iii) Regulatory Guide 1.204 (NRC, 2005ad). Measurements (Schnetzer, et al., 1995aa) have shown that a major portion of the current associated with a lightning bolt is carried by the rebars in a reinforced concrete structure, termed as "Faraday cage" effects, which places any waste forms within the reinforced concrete structures at a lower risk. DOE also proposed using Faraday cage effects for the Railcar and Truck Buffer area and aging facility (BSC, 2008ai). These facilities would have air terminals bussed together. The air terminals would be connected by at least two down conductors to the grounding system at the site. DOE recognized that casks and canisters may be vulnerable to a lightning strike during transportation between different facilities. Therefore, it analyzed the effects of direct lightning strike on a representative transportation cask, aging overpack, and TEV (BSC, 2008ai). Results in BSC (2008ab) 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. As the walls of an aging overpack, transportation cask and canister, and TEV would be thicker than 12 mm [0.47 in], DOE (BSC, 2008ab) concluded that there would not be a breach of containment resulting in radioactive release.

NRC Staff Evaluation: The NRC staff reviewed the lightning information DOE provided using the guidance in the YMRP. Specifically, the NRC staff evaluated how DOE screened out initiating events arising from lightning strikes that could affect an ITS SSC in addition to design features incorporated to withstand a lightning strike. Additionally, the NRC staff reviewed NFPA 780–2004 (National Fire Protection Association, 2004aa), Underwriters Laboratories 96A (Underwriters Laboratories, 2005aa), and Regulatory Guide 1.204 (NRC, 2005ad). The NRC staff's review focused on whether DOE (i) reasonably estimated lightning strike frequency and (ii) provided design features to ITS SSCs to reduce lightning strike potential and withstand a lightning strike without any radiological consequences.

DOE reasonably estimated the annual strike frequency because the lightning strike data collected for the Yucca Mountain region from the National Oceanic and Atmospheric Administration, a reliable source, are reasonable.

The NRC staff notes that the special design features DOE proposed to install on ITS SSCs to reduce the strike potential and the estimated pit are reasonable because the lightning protection system is designed following industry standard codes and NRC guidance. In addition, the transportation cask, aging overpack, and the TEV would be able to continue performing their safety functions after a lightning strike because the transportation cask, aging overpack, and the TEV DOE proposed have sufficient thickness to withstand the damage caused by a direct lightning strike.

Summary of NRC Staff's Evaluation

On the basis of the evaluation discussed previously, the NRC staff notes that DOE used reasonable methodology and site-specific data to assess weather-related hazards at the repository site. DOE showed that straight wind would not be a potential initiator for event sequences, because the estimated 3-second gust wind speed at 10^{-6}/yr frequency of occurrence is bounded by the design basis tornado wind speed at the site. DOE's use of a surrogate to show that hazards associated with the expected tornado wind speed would not initiate a credible event sequence is reasonable because the selected surrogates are weaker than the reinforced concrete or steel structures of the waste handling facilities. Consequently, using surrogates would result in conservative estimates.

DOE also implemented widely accepted tornado-missile classification schemes and site-specific data to show that no heavy missiles could be generated due to the low wind speed expected. In addition, DOE implemented widely used methodology to assess the damage imparted to a concrete and a steel structure by a credible missile during the operation period. Although the methodology conservatively assumes all tornado-generated missiles are rigid, the results showed that even in these conservative cases, ITS structures and systems would continue their safety functions. The proposed lightning protection systems are reasonable because they followed industry standard guidance and would protect ITS SSCs.

On the basis of these evaluations, the weather-related hazard information is reasonable.

2.1.1.3.3.1.3.3 Aircraft Crash Hazards

DOE provided activities involving aircraft in SAR Section 1.6.3.4.1 and BSC (2007ak,ap). DOE provided additional information in its response to the NRC staff's RAIs (DOE, 2008ah, 2009fh,fi). The NRC staff reviewed DOE's aircraft crash hazard analysis including the assumptions, data, and methodology DOE used to assess the hazards from aircraft crashes that could affect ITS structures at the GROA. An ITS structure can sustain significant damage from a crashing aircraft impact including fire damage resulting from the ignition of the aviation fuel carried onboard the aircraft. DOE screened out hazards associated with aircraft crashes using a screening criterion of 1 in 10,000 assuming an operational period of 50 years for the surface facilities and therefore no event sequence assessment was conducted.

Potential Hazardous Flight Activities

DOE 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 include flights at nearby civilian, U.S. Department of Energy (DOE)-controlled, and military airports; airways; military training routes and areas; air refueling routes; restricted airspace; and military operating areas of the NTTR and the restricted airspace over the Nevada Test Site (NTS). The U.S. Air Force controls airspace over the NTTR. The restricted airspace above the NTS includes R-4808N and R-4808S. DOE controls airspace R-4808N, which is subdivided into R-4808A, R-4808B, R-4808C, R-4808D, and R-4808E (BSC, 2007ap). The repository surface facilities are located beneath airspace R-4808E, as shown in BSC Figure 6-1 (2007ap). Airspace R-4808S is jointly controlled by the NTS, Nellis Air Traffic Control Facility, and Federal Aviation Administration (FAA) Los Angeles Air Route Traffic Control Center (BSC, 2007ap).

<u>Civilian, DOE-Controlled, and Military Airports, and Helipads</u>

DOE provided information on the civilian, DOE, and military airports in BSC Tables 7–1 and 7–2 (2007ap). These airports are more than 40 km [25 mi] away from the repository facilities. DOE screened out any hazards associated with the aircraft landing at and taking off from these airports because, according to DOE, these airports are a sufficient distance away from the GROA. Consequently, DOE concluded that the activities associated with these airports would not pose significant hazards to the GROA facilities.

Two helipads, Area 29 and the Field Operations Office, are within 32 km [20 mi] of the North Portal. DOE indicated that helicopters flying to these helipads may come close to the repository facilities. DOE proposed 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 can contain radioactive materials, as outlined in BSC Section 3.3.3 (2007ak). Additionally, any helipad would be located 0.8 km [0.5 mi] away from the relevant surface facilities/areas, as shown in BSC Table 1 (2007ak). On the basis of this assumed flight restriction, DOE eliminated any hazards to the repository facilities from helicopters.

NRC Staff Evaluation: The NRC staff reviewed the information related to the civilian, DOE-controlled, and military airports, and helipads and associated hazards using the guidance in the YMRP. DOE reasonably screened out any hazards associated with aircraft landing at and taking off from the civilian, DOE, and military airports because these airports are more than 40 km [25 mi] away from the repository facilities. On the basis of NUREG–0800 (NRC, 1987aa), which is referenced in the YMRP and has been used by the NRC staff in licensing activities related to fuel cycle facilities, and DOE–STD–3014–96 (DOE, 1996aa), landing and taking off at these airports would not pose significant hazards to the repository facilities at this distance. Additionally, DOE–STD–3014–96 (DOE, 1996aa) was developed for generic safety analysis of DOE safety-related facilities. The development was overseen by a committee comprising members from DOE, FAA, and the U.S. Environmental Protection Agency. The NRC staff notes that the use of these criteria in NUREG–0800 (NRC, 1987aa) and DOE–STD–3014–96 (DOE, 1996aa) is reasonable because the information is from credible sources.

The NRC staff also notes that DOE reasonably identified Area 29 and the Field Operations Office helipads for assessing potential hazards to the repository facilities; however, DOE eliminated any potential hazards from the helicopter flights using procedural and operational controls. On the basis of DOE–STD–3014–96 (DOE, 1996aa), DOE indicated that a separation distance of 0.4 km [0.25 mi] would be sufficient to eliminate any helicopter crash hazard potential. On the basis of DOE's separation-distance requirement, helicopter flights would not pose a credible hazard to the repository facilities.

Jet Routes and Federal Airways

DOE identified the jet routes and federal airways within 160 km [100 mi] of the GROA in BSC Table 6.5 (2007ap). DOE indicated that it conservatively screened out the jet routes and federal airways that are more than 48 km [30 mi] from the North Portal. Among the jet routes and federal airways, jet routes J-86 and J-92 and federal airways V-105 and V-135 are the closest to the North Portal. The nearest edge of jet routes J-86 and J-92 is approximately 10 km [6 mi] away from the North Portal. Federal airways V-105 and V-135 are approximately 18 km [11 mi] away (BSC, 2007ap). The nearest edge of the other jet routes and federal airways is more than 97 km [60 mi] away with jet route J-110 located 58 km [36 mi] away.

NRC Staff Evaluation: The NRC staff reviewed the information on jet routes and federal airways using the guidance in the YMRP. DOE's assessment of potential crash hazards of the federal airways and jet routes within 48 km [30 mi] of the North Portal is reasonable because aircraft flying these routes experiencing problems leading to a crash are not expected to fly more than 48 km [30 mi]. The aircraft either would crash before traveling that distance or would be able to find an airport suitable for landing. There are several airports nearby, as listed in BSC Table 7–1 (2007ap). Additionally, the distance criterion is consistent with NUREG–0800 (NRC, 1987aa). DOE's assessment of hazards associated with jet routes J-86 and J-92 and federal airways V-105 and V-135 is reasonable because these are located less than 48 km [30 mi] from the North Portal.

Beatty Corridor

DOE defined the Beatty Corridor as the 42-km [26-mi]-wide broad corridor running parallel to U.S. Highway 95 and the Nevada–California border (BSC, 2007ak). General aviation, commercial, and military aircraft use this corridor. DOE identified the flights through the Beatty Corridor that can potentially pose a crash hazard to the repository facilities.

NRC Staff Evaluation: The NRC staff reviewed the information on flights through the Beatty Corridor using the guidance in the YMRP. The NRC staff notes that DOE reasonably identified flights through the Beatty Corridor as potential hazards because aircraft can fly within several kilometers [miles] of the repository facilities while transiting this corridor.

Military Operations Areas

DOE identified military operations areas (MOAs) within 160 km [100 mi] of the GROA, as shown in BSC Figure 6-1 (2007ap). Two NTTR MOAs, Desert and Reveille, are close to the North Portal, as shown in BSC Figure 6-1 (2007ap). The Desert MOA is approximately 86 km [55 mi] away from the North Portal, and the Reveille MOA is approximately 114 km [71 mi] away, as detailed in BSC Table 6-2 (2007ap). DOE screened out activities of these two MOAs as they are farther than 48 km [30 mi] away from the North Portal.

NRC Staff Evaluation: The NRC staff reviewed the information on the nearby MOAs using the guidance in the YMRP and notes that activities in the Desert and Reveille MOAs would not pose any significant hazard to the repository facilities, because DOE provided information that shows both MOAs are more than 80 km [50 mi] from the repository facilities.

Restricted Airspace Over Nevada Test Site

The airspace above the NTS includes R-4808N and R-4808S. DOE controls R-4808N, which is subdivided into R-4808A, R-4808B, R-4808C, R-4808D, and R-4808E (BSC, 2007ap). The surface facilities are located beneath the airspace R-4808E, as shown in BSC Figure 6-1 (2007ap). DOE (BSC, 2007ap) indicated that flight activities at R-4808N can potentially pose a crash hazard to the repository facilities. Airspace R-4808S is jointly controlled by the NTS, Nellis Air Traffic Control Facility, and FAA Los Angeles Air Route Traffic Control Center (BSC, 2007ap). At the nearest point, airspace R-4808S is approximately 10 km [6 mi] from the North Portal. DOE included flight activities in R-4808S as flights through jet routes J-86 and J-92 and federal airways V-105 and V-135, as detailed in BSC Table 8-1 (2007ap).

NRC Staff Evaluation: The NRC staff reviewed the information on the restricted airspace using the guidance in the YMRP. DOE’s assessment that flight activities at R-4808N can potentially pose a crash hazard to the repository facilities is reasonable because the surface facilities are located beneath this restricted airspace. The NRC staff notes that DOE’s approach of supplementing flight activities in jet routes J-86 and J-92 and federal airways V-105 and V-135 with flights in airspace R-4808S is reasonable because these routes are close to R-4808S. This is conservative because an aircraft needs clearance to enter airspace R-4808S. Additionally, civilian aircraft are not permitted below 6,096 m [20,000 ft] above mean sea level in R-4808S, as described in BSC Appendix F (2007ap).

Military Training Routes

BSC Table C–1 (2007ap) listed the military training routes that access the NTTR. Only three routes—IR-286, VR-222, and VR-1214—are closer than 32 km [20 mi] to the repository facilities. The IR-286 route is approximately 24 km [15 mi] from the North Portal. Approximately 21 flights use this route annually. VR-222, closest to the repository facilities, is approximately 18 km [11 mi] from the North Portal. DOE estimated that 550 sorties use this route annually (BSC, 2007ap). In this route, flights are limited to 457 m [1,500 ft] above ground level and are typically flown at 152 to 305 m [500 to 1,000 ft] above ground level (BSC, 2007ap). Route VR-1214 is approximately 29 km [18 mi] away, and approximately 300 sorties use it annually. DOE has screened out flight-related activities in these military routes because, according to DOE, they pose a negligible hazard to the repository facilities.

NRC Staff Evaluation: The NRC staff reviewed the information on military training routes using the guidance in the YMRP. On the basis of the distance of the military training routes to the repository facilities, DOE’s assessment that these military training routes, except IR-286, VR-222, and VR-1214, are too far away to pose any credible hazard to the repository facilities is reasonable. Because of the flight-height limitation in Route VR-222, given the elevation of the ridge line above the North Portal, these flights will typically be below the ridge line while passing close to the GROA and the mountain ridge would act as a natural barrier for the aircraft. Therefore, the flights in Route VR-222 would not pose a hazard to the repository facilities. IR-286 route is approximately 24 km [15 mi] from the North Portal. Approximately 21 flights use this route annually. Consequently, on the basis of the distance and flight frequency, the NRC staff notes that the flights using IR-286 would pose negligible hazards to the repository facilities. Route VR-1214 is approximately 29 km [18 mi] away, and approximately 300 sorties use it annually. Again, considering the distance, flights on this route would not pose a significant hazard to the repository facilities.

Ordnance, Dropped Objects, and Ground-to-Ground Missile Testing at Nevada Test Site

DOE estimated that approximately 5 percent of the military aircraft in NTTR carry ordnance; however, ordnance is not armed until the aircraft is over the land of the R-4807 and R-4806 bombing ranges, as outlined in BSC Section 6.1.1 (2007ak). DOE indicated that inert and live ordnance are used in training at the R-4806 and R-4807 ranges (BSC, 2007ap). The closest point of R-4806 is approximately 43 km [27 mi] from the North Portal. Similarly, the closest point of R-4807 is approximately 45 km [28 mi] from the North Portal. NTTR does not allow actual launch of air-to-air missiles due to safety concerns (BSC, 2007ap). Additionally, on the basis of information from the U.S. Department of the Air Force, as detailed in BSC Section 7.2.1 (2007ap), no ordnance used from 1993 to 2003 strayed outside the designated hazard areas. Therefore, DOE excluded any hazards from stray deployed ordnance, as described in BSC Section 7.2.1 (2007ap).

An aircraft transiting R-4808N can carry ordnance onboard; however, overflights with live or hung ordnance are prohibited except in emergencies. Although ordnance may be carried onboard, use of ordnance is prohibited within the Electronic Combat (EC) South range. On the basis of BSC Assumption 3.3.2 (2007ak), an aircraft is not allowed to carry any onboard ordnance while transiting the flight-restricted area. Consequently, DOE did not consider any effects from accidently dropped or intentionally jettisoned ordnance in its hazard assessment, as outlined in BSC Section 6.7 (2007ak).

The most recent ground-to-ground missile tests were conducted in R-4808A Area 26 in June 2000. There is no plan for future ground-to-ground missile testing. Therefore, BSC (2007ap) did not consider ground-to-ground missile testing a potential hazard to the repository facilities.

NRC Staff Evaluation: The NRC staff reviewed the information on ordnance, dropped objects, and ground-to-ground missile testing at NTS using the guidance in the YMRP. On the basis of the distance and historical evidence that no ordnance used from 1993 to 2003 strayed outside the designated hazard areas, DOE reasonably screened out hazards from stray ordnance.

Although flights with ordnance are possible in the EC South Range and R-4808N, ordnance discharge is not allowed by the U.S. Department of the Air Force. Additionally, transit protocol for the flight-restricted airspace prohibits carrying any ordnance. Therefore, DOE reasonably screened out the potential hazards from dropped ordnance on the GROA facilities.

In addition, the NRC staff notes that DOE controls the airspace of R-4808N, where ground-to-ground missile tests took place in the past. Any future ground-to-ground missile testing in that airspace would have to be coordinated with DOE, which would evaluate the impact with respect to NTS activities including the repository (DOE, 2008ah). Consequently, it is reasonable that any future ground-to-ground missile test would not initiate an event sequence at the repository facilities.

Radar and Communication Jamming Activities

DOE indicated that it included the flights at the EC South Range of R-4807 for assessing potential hazards to the repository facilities. In addition, radar and communication jamming activities are conducted in this range. DOE indicated that frequency management is used in these activities to ensure there is no interference with other federal or civil transmitters or receivers (BSC, 2007ap). Additionally, DOE indicated that any radio frequency emission near the NTS is coordinated with DOE.

NRC Staff Evaluation: The NRC staff reviewed the information related to the radar and communication jamming activities conducted in the EC South Range using the guidance in the YMRP. DOE's inclusion of the flights at the EC South Range for assessing potential hazards is reasonable because this range is the closest restricted area of the NTTR to the North Portal. DOE's assessment that radio frequency emission used in training activities at the EC South Range would not pose any additional hazard to the public is reasonable because appropriate frequency management is used to avoid interference with other federal or civil transmitters or receivers. This is also supported by the U.S. Department of the Air Force Section 3.3.1 (1999aa).

Low Altitude Training and Navigation

The Low Altitude Training and Navigation (LATN) West area is adjacent to R-4808E and is approximately 1.6 km [1 mi] from the North Portal, as described in BSC Section 7.2.2 (2007ap). LATN West is used by A-10s and helicopters for low-altitude training. This airspace is activated when no airspace suitable for this type of training is available within the NTTR complex. Aircraft using LATN West are included with the aircraft in the NTTR that pose a crash hazard to the repository facilities, as detailed in BSC Section 7.2.2 (2007ap).

NRC Staff Evaluation: The NRC staff reviewed the information related to the activities conducted in LATN West using the guidance in the YMRP. The NRC staff notes that DOE reasonably identified hazards associated with activities in LATN West. Although the Yucca Mountain ridgeline provides a visual separation, as outlined in BSC Appendix F (2007ap), DOE included the aircraft involved in the activities in LATN West with the aircraft in the NTTR that pose a crash hazard to the repository facilities.

Aircraft Hazard Frequency Analysis

DOE used the list of potential aircraft-flight-related hazards to the repository facilities during the 100-year preclosure period, as shown in BSC Table 8.1 (2007ap), to assess the cumulative annual frequency of the initiating events in BSC (2007ak).

Flight-Restricted Airspace

In BSC Section 3.3.1 (2007ak), DOE 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. Only 1,000 overflights by military aircraft would be allowed annually, as described in BSC Section 3.3.2 (2007ak). These flights would be straight and level. No tactical maneuvering, carrying ordnance onboard, and electronic jamming activities would be allowed, as detailed in BSC Section 3.3.2 (2007ak). Additionally, accelerating to join the formation and use of piddle pack would not be allowed while overflying this airspace, as outlined in DOE (2008ah) and BSC Section 3.2.17 (2007ak).

DOE currently allows military aircrafts to transit R-4808N but requires that pilots observe certain avoidance areas. On the basis of interactions with DOE, the U.S. Department of the Air Force Warfare Center at Nellis Air Force Base, Nevada, has revised Air Force Instruction 13–212, Volume 1, Addendum A, to include flight restriction with a future implementation date before receiving and possessing nuclear waste. DOE would implement these flight restrictions at a future date through the National Nuclear Security Administration, Nevada Site Office.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on the flight-restricted airspace using the guidance in the YMRP. DOE has control of R-4808N and has the authority to implement additional flight restrictions in that air space. The NRC staff also notes that it is currently possible to monitor flights near the repository area [e.g., BSC Table II-1 (2007ak)]. The actual controls for restricting flights over the repository area, as stated in SAR Section 5.8.3, are not fully developed yet (DOE, 2008ah); however, DOE stated that it will develop these controls that would be in place before receiving and possessing nuclear waste to restrict maneuvering and other activities, as assumed in BSC (2007ak), to estimate the annual aircraft crash frequency on the repository facilities while transiting the flight-restricted air space.

Effective Area

Using the equations from DOE–STD–3014–96 (DOE, 1996aa), DOE calculated the effective areas of structures and equipment in the GROA, which may contain radioactive waste, including various handling facilities, rail and truck staging areas, and the aging pads. The effective area is the horizontal area such that, if an aircraft crashed into any point of that area, the crash would be considered to impact the structure identified as a target of concern. The total area for the GROA is the sum of the areas of all potential targets of concern. Results of these total effective area calculations were given in BSC Table 19 (2007ak) for various aircraft and vary 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.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on the effective area using the guidance in the YMRP. DOE estimated effective aircraft crash target areas for all applicable repository facilities using well-known equations for calculating the direct fly-in, shadow, and skid effective target areas, as described in DOE–STD–3014–96 Appendix B (DOE, 1996aa). The NRC staff notes that using these equations is consistent with the guidance in NUREG–0800 Section 3.1.5.6 (NRC, 1987aa) and is conservative for estimating aircraft crash frequencies at the Yucca Mountain repository because these equations are based on the fundamental target collision theory and are consistent with these authoritative references.

Aircraft Crash Rates

To estimate the annual frequency of aircraft crashes onto the repository facilities, DOE used three types of crash rates for (i) aircrafts including general, commercial, and military aviation in the Beatty Corridor; (ii) small military aircrafts overflying the flight-restricted area; and (iii) small military aircrafts with mishaps initiating outside the flight-restricted area. Crash rates of general aviation aircraft flying through the Beatty Corridor per flight mile were taken from Kimura, et al. (1996aa) and listed in BSC Table 15 (2007ak). DOE updated the crash rate of air taxi aircraft using information from the National Transportation Safety Board. DOE used the military aircraft crashes from Kimura, et al. (1996aa) for both normal and special flight modes. Kimura, et al. (1996aa) developed the crash rate of military aircraft using mishap information from 1975 through 1993. As the mishap rate (number of mishaps per flying hour) is decreasing, DOE updated these crash rates with recent mishap information from the U.S. Department of the Air Force, as described in BSC Attachment IV (2007ak).

For the crash frequency of small military aircraft overflying the flight-restricted area around the GROA, DOE relied on data from U.S. Department of the Air Force (2007aa) from 1990 through 2006. DOE stated that using F-16 mishap rates for all small military aircraft is conservative, because crash rates for other types of aircraft currently flying within the NTTR, such as F-15s and A-10s, are lower than for F-16s. The resultant value of 1.7 × 10^{-8} crashes/km [2.74 × 10^{-8} crashes/mi] for F-16s in the normal flight mode was used in calculating crash frequencies for small military aircraft overflying the Yucca Mountain flight-restricted area. Aircraft conducting these overflights are required to be in a normal transit mode, not conducting maneuvers or other activities.

DOE used a crash rate per NTTR unit area to estimate annual crash frequency onto the repository ITS facilities resulting from small military aircraft operating in the NTTR outside the flight-restricted area. In BSC Section 3.2.14 (2007ak), DOE estimated the crash rate using the actual crash statistics (18 crashes) on the NTTR in 16.5 years over an area of 38,850 km^2 [15,000 mi^2], giving an areal crash density of 2.8 × 10^{-5} crashes/yr-km^2 [7.3 × 10^{-5} crashes/yr-mi^2], which DOE rounded up to 2.9 × 10^{-5} crashes/yr-km^2 [7.5 × 10^{-5} crashes/yr-mi^2] in the actual calculations.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on aircraft crash rates using the guidance in the YMRP and notes that it is appropriate for DOE to use the crash rate information for general aviation and air carrier aircraft from Kimura, et al. (1996aa), an authoritative document for aircraft crash rates. Additionally, DOE used the National Transportation Safety Board information to estimate the crash rate of air taxi aircraft because the National Transportation Safety Board is the authoritative source for civil aviation mishaps. Therefore, the estimated crash rate is reasonable.

The NRC staff notes that the method DOE used to update the military aircraft crash rate with recent mishap information is reasonable because this method has previously been used in analyses related to licensing activities for NRC-regulated facilities. The NRC staff also notes that it is reasonable for DOE to use the F-16 crash rate to represent all small military aircraft of concern—namely, F-15, F-16, and A-10, as detailed in BSC Section 3.2.13 (2007ak)—because the F-16 (a single engine aircraft) has the highest crash rate. Therefore, using the F-16 crash rate is conservative.

The use of actual crash information for the aircraft flying in the NTTR but outside the flight-restricted airspace above the repository facilities is reasonable because the data reflect the type of operations and aircraft expected in the NTTR.

<u>Crashes From Flights Through the Beatty Corridor</u>

DOE calculated the annual probability of crashes onto the repository facilities from aircraft transiting the Beatty Corridor by multiplying the number of flights through this corridor, the crash rates per mile for air carriers and taxis and small military aircraft, and the exponential functions giving the decline in the areal crash density with distance from the edge of the air corridor, and the Yucca Mountain effective area. These exponential functions differ for each type of aircraft and come from a model by Solomon (1975aa, 1988aa) that has been used previously in analyses related to licensing activities for NRC-regulated facilities. The result was 2.9 × 10^{-8} crashes/yr onto the repository facilities from aircraft transiting this corridor.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on crashes from flights through the Beatty Corridor using the guidance in the YMRP. The NRC staff notes that DOE's use of the method Solomon (1975aa, 1988aa) proposed to estimate the annual crash frequency of aircraft transiting through the Beatty Corridor is reasonable as it is an alternative method to that proposed in NUREG–0800 (NRC, 1987aa) and has been used in other licensing activities (e.g., Palo Verde nuclear power plant). Similar to the NUREG–0800 (NRC, 1987aa) model, the Solomon model assumes that the flights in an air corridor follow a straight-line path. However, the probability that an aircraft would crash onto a facility diminishes with the orthonormal distance from the designated flight path in the Solomon model. Solomon (1975aa, 1988aa) modeled the decay of this probability by a double exponential distribution symmetrical against the flight path. The decay factor in exponential distribution Solomon (1975aa, 1988aa) proposed is 1.0 for military aircraft.

In addition, the NRC staff notes that DOE's assessment of annual crash frequency into the Yucca Mountain effective area from flights by different types of aircraft using the Beatty Corridor is reasonable. This is because DOE's estimated annual flight count in the Beatty Corridor is a 400 percent increase of that estimated using the FAA information to account for the uncertainties and future flight growth; therefore, this is a conservative count.

<u>Crashes From Military Flights Over Flight-Restricted Airspace</u>

In its response to the NRC staff's RAIs (DOE, 2008ah, 2009fi), DOE used the NUREG–0800 (NRC, 1987aa) formula to calculate the annual crash frequency due to the maximum 1,000 annual flights permitted over the flight-restricted airspace. The estimated annual crash frequency was 8.1×10^{-7} crashes/yr onto the repository facilities from these flights over the flight-restricted airspace.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on crashes of military flights over flight-restricted airspace using the guidance in the YMRP. The NRC staff notes that DOE used the formula from NUREG–0800 (NRC, 1987aa) for flights through a corridor to estimate crash frequency from small military aircraft overflying the flight-restricted area; this model is reasonable for overflights where the corridor width is just the diameter of the restricted area. Many types of crashes were eliminated because of the restrictions imposed on aircraft maneuverability during overflight of the flight-restricted airspace; therefore, most crashes would be from engine failures. In most such cases, the engine failure would have occurred at a considerable distance from the GROA because the aircraft would be flying 4,267 m [14,000 ft] above mean sea level. After engine failure, an aircraft may not always travel a straight line, especially in crashes with large glide, as implicitly assumed in the NUREG–0800 (NRC, 1987aa) model. Consequently, the crash hazard at the repository surface facilities from overflights of the flight-restricted area is expected to be less than that estimated using the NUREG–0800 formula (NRC, 1987aa).

<u>Crashes From Military Flights Outside Flight-Restricted Airspace</u>

DOE provided information on locations of the 18 crashes and tracks of actual aircraft flights during 1 day from the U.S. Air Force (DOE, 2009fi). None of the 18 crashes, which occurred within the 38,850 km^2 [15,000 mi^2] NTTR, were within 48 km [30 mi] of the Yucca Mountain North Portal. Knowing the crash locations and the fact that no crashes occurred within 48 km [30 mi] of the North Portal, DOE used a Bayesian analysis to update the crash rate given in BSC Section 3.2.14 (2007ak). This updated crash rate was used to estimate the annual frequency of small military aircraft crashes onto the repository facilities by flights outside the flight-restricted airspace. DOE stated the estimated crash frequency was approximately 1×10^{-6} crashes/yr (DOE, 2009fi).

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on crashes of military flights outside flight-restricted airspace using the guidance in the YMRP. The NRC staff notes that (i) none of these 18 crashes occurred within 48 km [30 mi] of the North Portal and (ii) the NTTR operating area immediately adjacent to the repository facilities (EC South Range) is used for electronic countermeasures operations rather than any combat maneuvering. The use of a Bayesian method to update the crash frequency on the basis of evidence of zero crashes within 48 km [30 mi] of the North Portal in 16.5 years is reasonable because using this method to update information is a standard risk assessment practice (Atwood, et al., 2003aa). In addition, use of a beta distribution prior with a mean of 0.1 for the fractional reduction in the crash density is reasonable on the basis of crash distributions DOE provided in DOE Figures 3 and 4 (2009fi).

DOE's estimation of annual crash frequency onto the repository facilities is conservative because no credit was taken to account for the distance many crashing aircraft traveled potentially being shorter than the radius of the restricted area {9 km [5.6 mi]}.

To confirm whether this estimate is reasonable, the NRC staff developed its own estimate for crash frequencies from small military aircraft as discussed next. Information in BSC Attachment III (2007ak) on 282 small military aircraft crashes shows that only 72 out of 282 (0.255) are engine-failure-related crashes. Virtually all other crashes resulting from other causes, such as flight directly into the ground, collisions, and loss of control, would have a distance from mishap initiation to crash location much shorter than the 9-km [4.9-nautical mi or 5.6-statute mi] restriction radius. Hence, the crash density would be reduced by a factor of 0.255, as only engine failure events can reach the repository facilities from outside the flight-restricted area.

To address uncertainties, the NRC staff considered that the relative standard error (sample standard deviation divided by mean) in an event frequency estimate is equal to the reciprocal of the square root of the number of events [$1/\sqrt{(18)} = 0.2357$]. This means the mean plus one standard deviation point on the uncertainty distribution would be at a crash frequency value 1.2357 times the value used. This is modest compared to the conservatism of not crediting the effect of 9-km [4.9-nautical mi or 5.6-statute mi] radius restrictions, which would have reduced the crash frequency by about a factor of 0.255. In fact, DOE indicated that, in case of engine failures, which constitute this remaining fraction of 0.255, military pilots follow an explicit procedure to try to recover the aircraft in the case of emergency. This procedure includes zooming to gain altitude, gliding, pointing the aircraft toward the nearest airfield, and attempting engine restart. As a result of this procedure and considering where the aircraft operates, pilots experiencing engine failures will generally not be pointing their aircraft toward the repository facilities.

The NRC staff also performed an independent estimate of the crash frequency from small military aircraft operating in the NTTR due to flights outside the flight-restricted airspace. One would expect 3.39 crashes over the 16.5-year time period if the areal crash density was truly uniform over the area within 48 km [30 mi] of the GROA. However, no crashes actually occurred within 48 km [30 mi] of the North Portal. This indicates that the crash rate density near the GROA is smaller than DOE estimated. In fact, a conservative Bayesian assumption would be a crash rate corresponding to 0.5 crashes in 16.5 years. The crash rate density, using this conservative estimate, would be lower by a factor of 0.5/3.39 = 0.1475. Hence, the NRC staff's conservatively estimates the crash frequency for non-overflights to be 9×10^{-7}—about the same value as DOE obtained (DOE, 2008ah)—and both values are conservative. Therefore, the NRC staff notes that DOE's estimate of crash frequency at the repository for flights outside the flight-restricted airspace is conservative.

Total Annual Crash Frequency

Combining the crash frequencies from flights through the Beatty Corridor and military flights over and outside flight-restricted airspace, DOE estimated the annual frequency of aircraft crashing onto the repository facilities to be 1.78×10^{-6} crashes/yr. This combined crash frequency is slightly smaller than the screening criterion of 2×10^{-6}/yr BSC used (2007ak). This screening criterion of 2×10^{-6}/yr 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). DOE concluded that aircraft crash hazards could be screened out from further consideration because of various conservatisms in these calculations, including no credit taken for the effect of the flight-restricted area.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided on the total annual crash frequency using the guidance in the YMRP. The NRC staff notes that DOE's estimation of the total crash frequency (1.78×10^{-6} crashes/yr) from the three sources (Beatty Corridor, military overflights, and military flights outside the flight-restricted area) into the effective area of the aboveground GROA is based on conservative calculations. Although the estimated annual frequency of aircraft crashes based on a 100-year preclosure period does not satisfy DOE's screening criterion of 1.0×10^{-6}/ yr given in SAR Table 1.6-1 and BSC Section 6.9 (2008ai), DOE stated in SAR Section 1.6.3.4.1 and BSC (2007ak) that all surface operations of radioactive waste would be completed in 50 years. Therefore, it used a 50-year operational period for surface facilities to convert frequency into probability. This probability of an aircraft crash onto the surface facilities is 0.9×10^{-5} over the preclosure period (100 years), which is less than 1×10^{-4}. Therefore, DOE's estimate of aircraft crash hazard onto the repository facilities and its screening of aircraft crash hazards from further consideration are reasonable.

2.1.1.3.3.1.3.4 Industrial and Military Activity-Related Hazards

DOE identified industrial and military facilities and associated activities in SAR Sections 1.6.3.4.8 and 1.1.1.3. Additional information and analysis were presented in BSC (2008an) and DOE (2009fe). On the basis of guidance provided in NUREG–0800 Sections 2.2.1 and 2.2.2 (NRC, 1987aa), DOE described all facilities and activities within 8 km [5 mi] of the repository. Additionally, as suggested in NUREG–0800 Sections 2.2.1 and 2.2.2 (NRC, 1987aa), facilities and activities at distances exceeding 8 km [5 mi] from the repository that can affect the safety-related features at the repository facilities were described in SAR Section 1.6.3.4.8 and BSC (2008an). BSC Figure 1 (2008an) provided the location of these facilities. NTS land use information in this figure was from the final environment impact statement for the test site and offsite locations (DOE, 1996ab). Locations of the active mines were from Driesner and Coyner (2006aa). DOE 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.

Although DOE included fog and shipwreck as contributors to nearby industrial and military facilities accidents (SAR Table 1.6-8; BSC, 2008an), these potential hazards were not dealt with explicitly. As the repository site is far from a seashore, a shipwreck affecting preclosure operations is not credible. Additionally, fog is dealt with indirectly in the assessment of hazards from nearby facilities.

Induced Air Overpressure

DOE provided information on air overpressure hazards resulting from explosive and flammable materials within the GROA and at facilities or activities at the adjoining NTS in SAR Sections 1.6.3.4.8 and 1.1.1.3, BSC (2008an), and DOE (2009fe). 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 inflammable 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, DOE (BSC, 2008an) estimated that an explosion of the entire diesel fuel in the tank would produce an air overpressure of 6.9 kPa [1 psi] at a distance between 52 and 166 m [0.03 and 0.10 mi]. As trains with loaded transportation casks will not travel closer than 52 m [0.03 mi] to this tank, no damage to the transportation casks is expected from such an explosion.

On the basis of information on activities conducted at different facilities in the NTS, DOE (BSC, 2008an) identified that the following 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. All these facilities are at least 32 km [20 mi] from the repository. On the basis of methodology given in Regulatory Guide 1.91 (NRC, 1978ac), DOE estimated that 5,900 kt [1.3×10^{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 facilities. DOE (BSC, 2008an) stated that 92 kt [2×10^{8} lb] would most likely exceed any TNT inventories of the NTS facilities. Therefore, DOE concluded that an explosion at the NTS facilities would produce an air overpressure much smaller than 6.9 kPa [1 psi] at the repository facilities and, consequently, ITS SSCs would not sustain any damage.

Additionally, Lathrop Wells Road, approximately 11 km [7 mi] away, is closest to the repository facilities. Some hazardous materials are transported over this road to support the Work for Others Program. In addition, U.S. Highway 95 is used to haul significant quantities of munitions, propellants, explosives, and radioactive materials. At the 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. DOE will construct a new rail line connecting the repository operations area with the commercial line. DOE (BSC, 2008an) 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.

NRC Staff Evaluation: The NRC staff reviewed DOE’s induced air overpressure information using the guidance in the YMRP. 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 induced air overpressure.

DOE’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 NTS is consistent with Regulatory Guide 1.91 (NRC, 1978ac). DOE used the peak positive incident air overpressure criterion of Regulatory Guide 1.91 (NRC, 1978ac) 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 DOE 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, Air Force, 1990aa). In addition, the NRC staff notes that DOE assumed that the entire tank would be filled with diesel vapor at the upper flammable limit. The NRC staff notes that this assumption is bounding and consistent with Regulatory Guide 1.91 (NRC, 1978ac) and is therefore reasonable. As the repository will be located farther away than the estimated safe distance {between 52 and 166 m [0.03 and 0.10 mi]}, DOE's determination that the diesel tank at the rail maintenance yard would not initiate an event sequence at the repository is reasonable.

The NRC staff independently verified the distances of the facilities within the NTS by examining BSC Figure 1 (2008an) and other NTS maps. The NRC staff also reviewed the fact sheet for the Device Assembly Facility (DOE, 2010ao) to assess any hazards an accident could pose at that facility. On the basis of this fact sheet, each of the five cells in the Device Assembly Facility can handle a maximum of 250 kg [550 lb] of TNT (DOE, 2004ae). Additionally, the NRC staff estimated the quantity of TNT-equivalent explosives using the approach in Regulatory Guide 1.91 (NRC, 1978ac). On the basis of its review, the NRC staff notes that DOE's assessment is reasonable that the estimated amount of explosives at a distance of 32 km [20 mi] {5,900 kt [1.3×10^{10} lb]} would possibly exceed the TNT inventories at the NTS facilities. Additionally, on the basis of Regulatory Guide 1.91, Figure 1 (NRC, 1978ac), the NRC staff notes that detonation of 1,860 kg [4,100 lb] of TNT-equivalent explosive at the Area 11 explosive ordnance disposal facility would not damage ITS SSCs at the repository from the generated air overpressure. Therefore, DOE's determination that an accidental explosive detonation at any of these facilities would not initiate an event sequence at the repository facilities is reasonable.

According to Regulatory Guide 1.91 (NRC, 1978ac), 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 60,000 kg [132,000 lb] of explosives. On the basis of the quantity of TNT needed at a distance of 21 and 32 km [13 and 20 mi] from the repository, as calculated in BSC Table 3 (2008an) and Regulatory Guide 1.91 (NRC, 1987ac), the NRC staff notes that an accidental explosion on the nearby highway or rail route would not generate a strong enough induced air overpressure to damage any ITS SSC at the repository facilities.

Induced Seismic Motion

DOE provided information on hazards from induced seismic motion from activities at the adjoining NTS in SAR Sections 1.6.3.4.8 and 1.1.1.3, BSC (2008an), and DOE (2009fe). DOE identified activities at several facilities at the NTS that can generate ground motion from underground explosions that may be potentially hazardous to the repository facilities. These activities include stockpile stewardship, damaged nuclear weapons program, conventional demilitarization activities, and blasting at nearby mines.

Stockpile management includes operations to store and maintain nuclear weapons stockpile. No activities would generate seismic motion affecting the repository (BSC, 2008an). Experiments and testing of nuclear devices were previously conducted in NTS Areas 1 through 10 for continued stewardship of the nuclear weapons' stockpile, but are currently not authorized. If limited underground nuclear testing commences, Yucca Flat Area (Area 6) and Pahute Mesa Area (Areas 19 and 20) would probably be the selected locations

(BSC, 2008an). DOE concluded, on the basis of the 14-year test data (Walck, 1996aa), that the ground motions at Yucca Mountain from nuclear tests would be bounded by moderate to large earthquakes in the region. DOE indicated that the response spectra measured at rock and soil sites near the repository and at the NTS from a moderate to large earthquake were always larger than the underground nuclear explosions. Additionally, secondary seismic effects, associated with coseismic release of strain, aftershocks, and collapse of cavity, are not significant at distances beyond 5 to 10 km [3 to 6 mi] even with the largest underground nuclear explosions (BSC, 2008an). On the basis of the same rationale, DOE concluded 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 earthquakes, as discussed before. Similarly, according to DOE, 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 Evaluation: The NRC staff reviewed the induced seismic motion information using the guidance in the YMRP. In addition, the NRC staff reviewed 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 induced ground motion. Induced seismic motion at the repository may have damaging effects similar to earthquake-induced seismic motion.

On the basis of the analysis given in Walck (1996aa), the NRC staff notes that ground motions at Yucca Mountain from nuclear blasts at the NTS would be bounded by moderate to large earthquakes in the region. Ground motions generated by underground nuclear blasts from 1977 through 1990 at the NTS, measured at stations of rock and soil near the repository site, were more than two times smaller than those from an equivalent earthquake.

As observed in underground nuclear tests, secondary seismic effects are not significant at distances exceeding 5 to 10 km [3 to 6 mi]. The repository facilities will be more than 24 km [15 mi] away (the closest area being Area 6). Therefore, DOE's determination that secondary seismic effects from potential future underground nuclear tests would not be credible hazards to the repository is reasonable. Similarly, the NRC staff notes that activities associated with damaged nuclear weapons at G-tunnel, approximately 40 km [25 mi] away, would not pose a hazard to the repository facilities, because of the large distance. The ground motion 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 is less than an underground nuclear blast. Similarly, a ground motion from a nearby mine blast would be bounded by earthquake ground motion. Therefore, DOE's assessment that induced seismic motions from underground explosions of nuclear and conventional explosives will not initiate an event sequence at the repository is reasonable.

Release of Radiological Materials and Toxic Chemicals

DOE provided information on hazards from released radiological materials and toxic chemicals from the NTS facilities in SAR Sections 1.6.3.4.8 and 1.1.1.3, BSC (2008an), and DOE (2009fe). DOE identified several facilities at the NTS that use or will use radiological materials or toxic chemicals on the basis of the description and activities conducted therein. These facilities include 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). DOE evaluated whether the activities could result in a radiological or chemical release that could impact the GROA mainly by the distance to the repository.

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), DOE 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 at 32 km [20 mi] away.

DOE identified that nuclear criticality activities currently performed at the Los Alamos National Laboratory in New Mexico would be relocated to the western section of the Device Assembly Facility in Area 6, renamed the Criticality Experiments Facility (BSC, 2008an). On the basis of the final environmental impact statement for this relocation, noninvolved workers would receive a minimal radiation dose from an accident. Consequently, DOE concluded that an accident at this facility would not be a hazard to the repository facilities due to large distance (BSC, 2008an).

The Radiological/Nuclear Countermeasures Test and Evaluation Complex is being constructed to conduct activities related to combating terrorism. When completed, this facility would be classified as a Hazard Category 2 nuclear facility (potential for onsite consequences) and is anticipated to 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. DOE does not expect that the activities at this complex would 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, DOE (BSC, 2008an) concluded that there would not be any impact to the repository operations.

NRC Staff Evaluation: The NRC staff reviewed the information regarding potential radiological material and toxic chemical releases from the NTS facilities using the guidance in the YMRP. The NRC staff also consulted some of the environmental impact statements or environmental assessments that BSC (2008an) referred to in DOE's assessment. Specifically, the NRC staff evaluated the type, quantity, and distance of the repository to the facilities that handle these materials.

In the JASPER Facility, radionuclides are used as the target materials in the experiment. The NRC staff notes that a 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 local contamination. Therefore, DOE's assessment that the dose to a repository worker would be negligible from an accident at the JASPER Facility as the repository would be 32 km [20 mi] away is reasonable.

The NRC staff consulted the final environmental impact statement for relocating the activities in Technical Area 18 of the Los Alamos National Laboratory to Area 6 in the NTS (DOE, 2002ac). The NRC staff notes that the highest risk of latent cancer fatality of a noninvolved worker at a distance of 100 m [330 ft] would be on the order of 10^{-9}/yr. Consequently, DOE's assessment that an accident at this facility (when operating) would not initiate an event sequence at the repository because of large distance is reasonable.

The NRC staff consulted DOE (2004ac) for the Radiological/Nuclear Countermeasures Test and Evaluation Complex and notes 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 hazardous radioactive and chemical materials would be limited to those onsite (DOE, 1992aa). This facility is 32 km [20 mi] from the repository. Consequently, DOE's assessment that a potential accident at this facility would not pose a hazard to the repository facilities and would not initiate an event sequence because of large distance is reasonable.

The NRC staff consulted the final environmental assessment for release of biological simulants and chemicals at the NTS (DOE, 2004ad). The biological simulants mimic some identifiable characteristics (except the higher risk associated with biological agents used in biological weapons) and are not toxic to healthy humans. The NRC staff notes that release of low concentrations of chemicals and biological simulants is permitted in Area 5 and other areas of the NTS (DOE, 2004ad; BSC, 2008an). Additionally, the NRC staff notes that the released materials are not detectable beyond the NTS boundaries and do not affect the involved and noninvolved workers or members of the public. On the basis of the previous discussion, the released biological simulants and chemicals would not affect the repository workers because of large distance and atmospheric dispersion and therefore would not initiate an event sequence.

Waste Management Programs

DOE provided information on hazards from waste management programs at the NTS facilities in SAR Sections 1.6.3.4.8 and 1.1.1.3 and BSC (2008an). The primary mission of the waste management programs are to dispose low-level radioactive waste (LLW) generated at the NTS and from other DOE-approved waste generators (BSC, 2008an). An accident at the waste management facility in Areas 3, 5, or 6 may result in a radioactive release.

NTS 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. DOE did not identify any hazards at these facilities that may affect the repository.

NRC Staff Evaluation: The NRC staff reviewed the information regarding waste management programs at the NTS facilities using the guidance in the YMRP. The NRC staff notes that the LLW is disposed of in subsidence craters in Area 3 and shallow pits and trenches in Area 5. Because the repository is a sufficient distance away from both sites {more than 40 km [25 mi]}, 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). Being at least 32 km [20 mi] away, any leak from this facility would be dissipated before it reaches the repository facilities and, therefore, would not pose a hazard to the repository. Therefore, liquid waste treatment would not initiate an event sequence at the repository because of large distance.

Mining

DOE 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). There are no mining claims in the repository, and Public Land Orders preclude any mining claims in the controlled area. Although there are unpatented mining claims at the southern edge of the proposed land withdrawal area, they are 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 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. 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 will not be permitted. Therefore, DOE concluded that the nearby mining operations would not have any impact to the repository and its operation, because of the large distance from the repository facilities. Additionally, no significant sources of oil or gas have been found in southern Nevada or adjacent areas of California and Arizona. 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. In the unlikely event that a commercially viable deposit is discovered near Yucca Mountain, DOE's control over the land withdrawal area will preclude such operations near the repository (BSC, 2008an).

NRC Staff Evaluation: The NRC staff reviewed the information regarding the hazards from mining-related activities near the repository facilities using the guidance in the YMRP. Specifically, the NRC staff evaluated the locations of the nearby mines and description of their activities to assess the potential hazards.

DOE obtained the locations of the nearby active mines from Driesner and Coyner (2006aa), which provides mine information in the State of Nevada. The NRC staff notes that use of mine information from Driesner and Coyner (2006aa) is reasonable. All mining activities are outside the 8-km [5-mi] zone surrounding the repository. Most of the mining, quarrying, milling, and exploration operations in southern Nevada are much farther away. DOE's control over the land withdrawal area as stated in BSC (2008an) will preclude new, future activities within the 8-km [5-mi] zone surrounding the repository.

Commercial Rocket Launch and Retrieval

DOE assessed hazards to the repository facilities associated with potential launch and recovery of reusable rockets by Rocketplane Kistler in NTS Areas 18 and 19 in SAR Section 1.1.1.3.6.1 and BSC Section 6.3.3.2 (2008an). National Aeronautics and Space Administration selected Rocketplane Kistler to provide delivery services to the International Space Station using the K-1 reusable vehicle. Kistler Aerospace Corporation proposed a potential launch site in Area 18 and a recovery area in Area 19; however, currently no operational facilities exist. The FAA issued an environmental evaluation with a finding of no significant impact for this activity. A detailed flight hazard analysis will be made before the FAA grants a license to the launching activities. Therefore, DOE concluded that currently there is no hazard associated with the commercial rocket launching activities.

NRC Staff Evaluation: The NRC staff reviewed the information regarding the hazards of potential launch and retrieval activities at the NTS using the guidance in the YMRP. The NRC staff evaluated whether the hazard currently exists and what would be done if these activities start in the future. Additionally, the NRC staff reviewed Federal Aviation Administration (2007aa) on the development of U.S. commercial space transportation.

In October 2007, National Aeronautics and Space Administration notified Rocketplane Kistler regarding termination of the contract to provide delivery services to the International Space Station (Whitesides, 2007aa). Additionally, even if there are future rocket launching and retrieval activities in Areas 18 and 19, the NRC staff expects that a detailed flight hazard analysis would be conducted before the FAA would grant a license. The NRC staff notes that no hazards currently exist from NTS Areas 18 and 19 because no contract is in place for Rocketplane Kistler to launch and recover reusable rockets in NTS Areas 18 and 19.
In addition, the potential future FAA licensing action would provide time for DOE to assess this hazard and take any necessary actions to mitigate potential effects.

<u>Release of Onsite Hazardous Materials</u>

DOE assessed hazards to the repository facilities associated with potential onsite release of hazardous materials in SAR Section 1.6.3.4.9 and BSC Section 6.11 (2008an). DOE conducted the screening analysis following Regulatory Guide 1.78 (NRC, 2001af) on any hazards from release of hazardous materials at nearby facilities.

Chlorine and helium are the two chemicals, listed in Regulatory Guide 1.78 Table 1 (NRC, 2001af), that will be stored onsite (SAR Section 1.6.3.4.9; BSC, 2008an). Chlorine tablets will be used for the water treatment system, and helium will be used for inerting the waste containers. Additionally, argon, a potential asphyxiant, will also be stored onsite (SAR Section 1.6.3.4.9). 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, 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). DOE 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, DOE 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 Evaluation: The NRC staff reviewed the information on potential release of onsite hazardous materials using the guidance in the YMRP and Regulatory Guide 1.78 (NRC, 2001af). DOE's use of Regulatory Guide 1.78 (NRC, 2001af) to identify the hazardous materials that will be stored onsite is reasonable because Regulatory Guide 1.78 provides guidance to identify the onsite chemicals that may potentially affect the habitability of the operation rooms and force abandonment. The NRC staff notes that the solid chlorine (chlorine tablets) to be used for water treatment cannot affect personnel at other locations by becoming airborne. Additionally, helium and argon gases would be supplied to the surface facilities from sources outside the buildings. Therefore, any released gases would be dispersed into the atmosphere. Similarly, any diesel gases from a spill would be localized and would not affect operations at other locations. DOE's statement that remote monitoring equipment in the operation rooms can continue monitoring even if abandonment is necessary is reasonable.

Turbine-Generated Missiles

DOE assessed hazards to the repository facilities associated with potential offsite 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. DOE (BSC, 2008ai) stated that the hazard from turbine missiles is generally associated with large turbines in nuclear power plants. As there are no nuclear power plants near the repository facilities, DOE excluded this hazard as a potential initiator of event sequences (SAR Table 1.6-8; BSC, 2008an).

NRC Staff Evaluation: The NRC staff reviewed the information on hazards related to turbine-generated missiles using the guidance in the YMRP. Failure of the massive rotor of a turbine with high rotational speed can generate high energy missiles that can affect ITS SSCs. The NRC staff confirmed that there are no nuclear power plants near the repository facilities. Additionally, there are no other plants within 8 km [5 mi] of the repository that use large turbines based on its review of BSC Figure 1 (2008an).

Fog

DOE did not assess the effects of fog on the repository facilities during the preclosure period directly; instead, it included fog effects indirectly in assessing the effects of nearby industrial and military-related facilities.

NRC Staff Evaluation: The NRC staff reviewed the assessment of hazards from nearby industrial and military facilities using the guidance provided in the YMRP. Fog decreases visibility; however, fog would not affect atmospheric dispersion of hazardous materials. Fog may sometimes indicate a temperature inversion condition of the atmosphere, which may enhance the air overpressure at certain locations, depending on the topography. As discussed, the distance between the repository facilities and the explosion source is large, making any air overpressure enhancement resulting from fog insignificant. Therefore, fog is not a hazard to the repository operations.

2.1.1.3.3.1.3.5 Other Hazards

DOE provided information for the remaining 14 hazards (SAR Table 1.6-8). DOE categorized these hazards into the following groups:

- External flooding
- Loss of power
- Loss of cooling capability
- External fire
- Explosions
- Extraterrestrial activity
- Waste and rock interaction, geochemical alterations, and dissolution
- Perturbation of groundwater system
- Improper design and operation
- Undetected past human intrusions
- Security-related hazards (namely, sabotage, terrorist attack, and war)

External Flooding

DOE provided information on external flooding at the repository facilities in SAR Section 1.6.3.4.5, SAR Table 1.6–3, and BSC Section 6.5 (2008ai). DOE 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, DOE concluded that dam failure, river diversion, flooding due to ice cover, and high river stage cannot occur at the GROA. The repository is approximately 402 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. 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, DOE concluded that external flooding due to high lake level or dam failure cannot occur at the GROA.

DOE further evaluated external flooding resulting from rainstorms because of the potential for severe rainstorms to occur at Yucca Mountain. Average annual precipitation at the NTS is less than 81 cm [10 in]. Projected maximum daily precipitation within 50 km [31 mi] of Yucca Mountain is less than 13 cm [5 in]. The 6-hour probable maximum precipitation for the GROA is estimated at approximately 30 cm [12 in]. DOE determined that potential flooding due to melted snow and ice is less severe and less frequent than from rainstorms. Thus, DOE evaluated rainstorms as a bounding case for potential flooding resulting from storm precipitation.

DOE's flood hazard analysis estimated the million-year flood (i.e., with 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 are designed to convey up to 1,557 m^3/s [55,000 ft^3/s] of flood flow. To ensure service at the design capacity, DOE will implement a standard maintenance practice on the flood protection features. DOE estimated PMF frequency as 1.1×10^{-9}/yr (BSC, 2008ai). In the final disposition, DOE screened out external floods from further consideration because the PMF frequency is less than 10^{-6}/yr and the building roof drainage system is designed to accommodate rainfall criteria.

NRC Staff Evaluation: The NRC staff evaluated information and analysis DOE provided on external flood hazards using the guidance in the YMRP. The NRC staff also reviewed the document describing the flood hazard curve (BSC, 2008cd).

DOE provided applicable site-specific data on rainfall in BSC (2008ai,cd). Because intense precipitation can occur at Yucca Mountain, DOE evaluated external flooding at the GROA due to rainstorms using a probabilistic flood hazard curve to estimate annual flood frequencies. In the process of developing the flood hazard curve, DOE estimated a probable maximum precipitation based on annual precipitation values the National Oceanic and Atmospheric Administration reported for the semiarid Southwest. The NRC staff notes that developing a probabilistic flood hazard curve is a reasonable method to estimate annual flood frequencies because this method is commonly used in the industry. DOE's estimates on probable maximum precipitation are reasonable because DOE used quality data from the National Oceanic and Atmospheric Administration, a reliable source. In addition, DOE estimated a PMF for the GROA based on flood peak simulations using the HEC–1 hydrologic model the U.S. Army Corps of

Engineers, Hydrologic Engineering Center developed. The NRC staff notes that the hydrologic model is reasonable as it is commonly used in the industry for flood hazard assessments. The PMF frequency does not exceed the limit of 10^{-6}/yr. Additionally, the diversion channel has the capacity to accommodate a million-year flood (annual probability of exceedance 10^{-6}) and is reviewed in TER Section 2.1.1.7.3.1.3. Therefore, the external floods will not initiate an event sequence because the flood control features have the capacity to accommodate a million-year flood.

Loss of Power

DOE provided information on loss of power to the repository facilities in SAR Section 1.6.3.4, SAR Table 1.6-8, and BSC Section 6.7 (2008ai). DOE identified loss of electrical power to be an initiating event in the repository facilities as it is a normal occurrence in nuclear facilities. Several natural hazards were identified to cause loss of offsite and/or onsite power: extreme weather and climate fluctuations, frost, hail, sandstorm–dust storm, and grid failure.

DOE (BSC, 2008ac,as,be,bk,bq) estimated the frequency of a loss of electrical power event occurring at the Yucca Mountain facilities to be 3.6×10^{-2}/yr on the basis of estimated mean frequency for the entire United States from 1986 through 2004, as provided in NUREG/CR–6890 (Eide, et al., 2005aa). This estimated frequency includes plant, switchyard, grid, and weather-related information. NUREG/CR–6890 (Eide, et al., 2005aa) also estimates the likelihood of a loss of power event lasting more than 24 hours (composite estimate) to be 1.79×10^{-2}/yr. Assuming that the waste handling operations would continue for the first 50 years of the preclosure period, DOE estimated the annual frequency of the initiating event due to loss of electrical power lasting more than 24 hours to be 3.2×10^{-2} (BSC, 2008ac,as,be,bk,bq). DOE concluded that a loss of external power event would occur during the facility operation as a normal occurrence, because the annual probability is more than 10^{-6}.

NRC Staff Evaluation: The NRC staff reviewed the information on loss of power using the guidance in the YMRP. Specifically, the NRC staff evaluated the annual frequency of occurrence of a loss of offsite power event at the repository to determine whether DOE used appropriate analysis and information to develop this annual frequency.

DOE's use of NUREG/CR–6890 (Eide, et al., 2005aa) to develop the estimate of the frequency of loss of electric power at the Yucca Mountain repository and the likelihood of a loss of power event lasting for more than 24 hours is reasonable because NUREG/CR–6890 (Eide, et al., 2005aa) has synthesized loss of electric power events to U.S. nuclear power plants collected over a period of 19 years (1986 through 2004). Although the number of events that occurred and the number of reactors considered in the study are not uniform across the country, NUREG/CR–6890 (Eide, et al., 2005aa) indicates that there are significant geographical differences in grid-related outage events among 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 4.18×10^{-2}/yr that is more than double the mean value of 1.86×10^{-2}/yr for the entire country. Therefore, DOE's use of the western region outage performance frequency value to estimate the outage frequency due to grid events in its PCSA is reasonable.

Loss of Cooling Capability

DOE provided information on loss of cooling capability to the repository waste handling facilities in SAR Section 1.6.3.4.7 and BSC Section 6.8 (2008ai). Water supply at the repository facilities may be disrupted due to dam failure; extreme weather and climate fluctuations including drought, high summer temperature, and low winter temperature; presence of fungus, bacteria, and algae; ice cover; low lake level; low river level; river diversion; and sandstorm (BSC, 2008ai). Three underground wells supply water to the repository through a 3,217,600-L [850,000-gal] storage tank. The storage tank feeds the delivery systems for deionized water at the fuel handling pool, fire water, potable water, and water for the cooling tower. As the water supply for the Yucca Mountain repository is obtained from groundwater sources, DOE 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). In addition, DOE concluded that any blockage from sandstorm or dust storm would not be a concern as the entire water supply infrastructure would comprise underground pipes and covered tanks. DOE further concluded that hazards that can 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 the flow of cooling water (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, waste forms do not exceed their temperature limits for 30 days of room heat up resulting from a loss of water supply to the relevant HVAC subsystems (BSC, 2007dd). Therefore, DOE concluded that if water supply is disrupted due to pipe freeze and rupture, there would be sufficient time for operations personnel to arrange for alternative sources of cooling water. Consequently, DOE (SAR Section 1.6.3.4.7; BSC, 2008ai) screened out the loss of cooling capability as a potential initiator of event sequences.

NRC Staff Evaluation: The NRC staff reviewed the information on loss of cooling capability to the repository waste handling facilities using the guidance in the YMRP. The NRC staff notes that because water will be supplied from three underground wells, hazards that can affect the source(s) of water, such as dam failure, ice cover, low lake or river level, and river diversion, would not be credible concerns. As change in groundwater supply is gradual and takes place over time, there will be sufficient time available to seek out alternate sources for water. DOE estimated how long the WHF pool and HVAC system can survive without new water before compromising safety functions on the basis of appropriate assumptions, input data, and analytical methods. These estimates showed that sufficient time is available to take necessary remedial actions to restore the water supply. Therefore, DOE's screening out loss of cooling capability hazard is reasonable.

<u>External Fire</u>

DOE provided information on external fire at the repository facilities in SAR Section 1.6.3.4.10. BSC Section 6.12 (2008ai) provided additional information on the estimated annual frequency of ignition. In its response to an NRC staff RAI (DOE, 2009fa), DOE summarized an analysis to establish the required firebreak width to keep the ITS SSCs safe from potential wildfires.

DOE 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 5.5×10^{-3} fires/yr in the GROA with a surface area of 2.7 km^2 [1.0 mi^2] (BSC, 2008ai). DOE [SAR Section 1.6.3.4.10; BSC Section 6.12 (2008ai)] proposed a separation distance of 10 m

[33 ft] that would be maintained vegetation free between fuel sources (brushes and vegetation) and the structures, as recommended in NFPA 1144 (National Fire Protection Association, 2008ab). DOE indicated that it would use administrative control 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. DOE calculated the heat released from the postulated fire moving toward an aging overpack at the corner of an aging pad (DOE, 2009fa) using the handbook of the Society of Fire Protection Engineers (1995aa). The estimated radioactive heat flux was 0.89 kW/m^2 [0.078 Btu/ft^2-sec]. As 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], DOE concluded that the noncombustible aging overpacks and waste handling facilities would not sustain any damage from the postulated fire separated by a 10-m [33-ft] distance from the structures. DOE 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 aging pads) would not sustain significant damage and would be capable of maintaining their intended safety functions from these vegetation fires. On the basis of these analyses, DOE screened out 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 providing protection from approaching wildfires and the ITS SSCs have sufficient capacity to resist the effects of these fires.

NRC Staff Evaluation: The NRC staff reviewed the external fire information DOE presented using the guidance in the YMRP. Specifically, the NRC staff evaluated the description of potential vegetation characteristics near the repository site and the analysis DOE provided to screen out initiating events arising from fires originating outside the GROA boundary (vegetation or wildfire).

The NRC staff notes that it is reasonable for DOE to use the U.S. Forest Service data on wildfires classified by the ecoregion to estimate the annual frequency of wildfire in the Yucca Mountain region because U.S. Forest Service is a reliable source. The methodology DOE used is reasonable because this approach was used by a government agency for wildfire hazard management, climate studies, and research purposes. Additionally, the wildfire statistics DOE used are reasonable because they were collected by U.S. Forest Service.

The NRC staff independently estimated the annual frequency of wildfires on the basis of the Bailey tropical/subtropical desert division, which translates to 1.61×10^{-4} fires/yr 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 translates to 2.7 km^2 [1.0 mi^2] of GROA land}. Although this estimate exceeds the estimate DOE presented, both estimates put the annual frequency of wildfire occurrence in a Category 2 initiating event. DOE estimated the heat flux that an aging overpack located at the corner of the aging pad would experience. The NRC staff reviewed DOE's estimate and notes that it is conservative for the 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 due to the presence of a buffer zone. DOE's determination that the vegetation at the repository site is light fire load is reasonable because the biomass of living and dead vegetation around the aging pad area is 0.2 kg/m^2 [2.8×10^{-4} lb/ft^2], lower than the light fuel load of 0 to 34 kg/m^2 [0 to 4.8×10^{-2} lb/ft^2], defined in NFPA 80A Section 4.3.5.2 (National Fire Protection Association, 2007ag). The calculation DOE provided made several conservative assumptions for the fire characteristics. The industry-standard Society of Fire Protection Engineers (1995aa) methodology was used in DOE's analysis and is therefore reasonable. The analysis assumed that 40 percent of the total heat released by a vegetation fire would be in the form of radiative heat and would be transferred to the aging cask. Typical radiative fractions are on the order of 20 to 40 percent, with 40 percent being a conservative assumption, as per the

Fire Protection Handbook (National Fire Protection Association, 2003ac). The estimated radiative heat flux to an aging overpack located in the most vulnerable position, the corner of an aging pad, is 0.89 kW/m^2 [0.078 Btu/ft^2-sec] if a 10-m [33-ft]-wide vegetation-free barrier is maintained. The minimum heat flux necessary to ignite certain types of paper and wood products is 10 kW/m^2 [0.88 Btu/ft2-sec], based on Society of Fire Protection Engineers (1995aa). Therefore, DOE reasonably analyzed the potential for ignition from external sources within the surface facilities area.

The NRC staff also reviewed the cask fire fragility analysis presented in BSC Attachment D, Section D2 (2008ac) and compared the expected exposures from wildfires. The NRC staff notes that DOE's determination is reasonable that a higher intensity with substantially longer duration fire exposures would be needed to challenge the shielding of an aging overpack and cause spalling of the concrete overpack. More than 34.5 cm [13.6 in] of concrete spalling would be needed before radiation exposure to firefighters or other personnel would be an issue, as described in BSC Section D.2.2.3.1 (2008ac). Therefore, DOE's screening out of the external fire hazard is reasonable.

Explosions

DOE provided information on explosion hazards to the repository facilities in SAR Section 1.7.1.2.2 and BSC Section 6.0.5 (2008au). DOE indicated that Area 70A will have a diesel oil storage tank of capacity 454,250 L [120,000 gal]. This tank will be supplied by a 37,850-L [10,000-gal] tanker truck. BSC Section 6.0.5 (2008au) 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 (NRC, 1978ac) states that an air overpressure of 6.9 kPa [1 psi] would not have any adverse effect on ITS SSCs. DOE evaluated whether the waste handling facilities would be subjected to an air overpressure larger than 6.9 kPa [1 psi] as a result of an explosion.

The postulated event in both the storage tank and tanker truck is vapor-cloud explosion. BSC (2008au) analyzed the effects of vapor-cloud explosions assuming the entire volume of diesel fuel available participated in the explosion. This was accounted for by converting the diesel fuel to an equivalent TNT mass. For the Area 70A storage tank, an overpressure of 6.9 kPa [1 psi] would develop 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]. Because the distance between the Area 70A storage tank and the waste handling facilities would exceed 164 m [539 ft], DOE (BSC, 2008au) concluded that a diesel storage tank explosion in Area 70A would not cause any adverse effect. The route the tanker truck would use to reach the Area 70A storage tank would be more than 46 m [150 ft] from a transportation cask. As this distance exceeds the standoff distance where an overpressure of 140 kPa [20 psi] is expected to develop from an accidental explosion of the tanker truck, DOE (BSC, 2008au) concluded that the transportation cask would not suffer any adverse effects. Therefore, DOE screened out explosion hazards from being a potential initiator of event sequences in the repository facilities and for the transportation casks because the ITS SSCs will be too far away to incur any explosion-related damage.

Additionally, BSC Table 6.0-2 (2008au) excluded any adverse effects from fuel tank explosions of the site transporter, cask tractor, cask transfer trailer, or site prime mover citing a design requirement of explosion-proof fuel tanks. DOE stated that the potential mechanism to be implemented in the explosion-proof fuel tanks of these ITS vehicles is yet to be selected in this preliminary design. The feature will be selected in the detailed design phase (DOE, 2009fa).

NRC Staff Evaluation: The NRC staff reviewed DOE's assessment of potential explosion hazards using the guidance in the YMRP. Specifically, the NRC staff evaluated the description of explosion sources and the analysis DOE provided to screen out initiating events arising from explosions. An explosion generates an air overpressure. If an ITS SSC is not designed to withstand the load imposed by the air overpressure, it may fail and lead to a release of radioactive materials.

The NRC staff notes that DOE used reasonable methodology to estimate the safe standoff distance from an accidental explosion at the storage tank in Area 70A and the tanker truck. The separation distance between the storage tank and different waste handling facilities is more than the estimated safe standoff distance. Similarly, the proposed route the tanker truck would take to reach the storage tank would also be more than the estimated safe standoff distance. Therefore, the NRC staff notes DOE's determination that any event sequence at the waste handling facilities from a vapor cloud explosion at the storage tank site would be beyond Category 2 and need not be analyzed is reasonable. The NRC staff also notes that any accidental explosion of the fuel tanks of the site transporter, cask tractor, cask transfer trailer, or site prime mover would not develop an initiating event because of the explosion-proof design requirement.

Extraterrestrial Activity

DOE provided information on extraterrestrial activity at the repository site in SAR Section 1.6.3.4.11 and SAR Table 1.6-8. DOE also discussed the potential impact on the repository facilities by the extraterrestrial objects in BSC Section 6.13 (2008ai). An asteroid is an extraterrestrial object with a size greater than 50 m [164 ft] (BSC, 2008ai) and can cause significant damage; however, DOE indicated that the frequency of asteroid impacts is relatively small. The return periods for smaller asteroids are hundreds to thousands of years (BSC, 2008ai) and, therefore, asteroids will not be credible hazards for 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 (BSC, 2008ai) and unconsolidated composition. 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) assumed that meteorites are less than 50 m [164 ft] in diameter. DOE (BSC, 2008ai) categorized meteorites into three classes on the basis of their composition to assess the hazards they posed: (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.

The Earth's atmosphere acts as a shield against meteorites. Because of frictional heating, most meteorites disintegrate while descending through the Earth's atmosphere. Both iron and hard stone meteorites smaller than approximately 10 kg [22 lb] tend to burn up in the atmosphere and do 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] or even larger tend to fragment in the atmosphere. Soft rock and ice meteorites 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. DOE 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.

Using information on the number of meteorites impacting the Earth's surface from Bland and Artemleva Table 2 (2006aa), DOE (BSC, 2008ai) 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 1.8×10^{-7} to 5.8×10^{-10}/yr. Hard stone meteorites will impact the GROA at a frequency varying between 6.4×10^{-7} and 1.2×10^{-9}/yr for the same mass range. Meteorites with higher mass would have a lower annual probability of striking the GROA. On the basis of the estimated annual frequency, DOE (SAR Section 1.6.3.4.11; BSC, 2008ai) concluded that meteorite strike would not initiate event sequences at the repository during the preclosure period.

Additionally, approximately 17,000 tracked space objects (man-made objects) reentered the Earth's atmosphere from 1957 through 1999 (BSC, 2008ai). Most of these objects burnt up completely before reaching the Earth's surface; however, a small portion of them may reach the Earth's surface and cause damage. DOE (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. DOE 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 reenter the Earth's atmosphere. Assuming that the space debris impacts the surface facilities at a 90° angle, the total area of affected surface facilities has been estimated to be 0.31 km^2 [0.12 mi^2]. The probability that space debris would strike a surface facility is 1.3×10^{-5} impacts over the operational period of 50 years, which makes it a beyond Category 2 event and, hence, excluded from further consideration (BSC, 2008ai).

NRC Staff Evaluation: Using the guidance in the YMRP, the NRC staff reviewed the information and analysis on extraterrestrial activity presented in SAR Section 1.6.3.4.11 and supporting documents to evaluate DOE's screening of initiating events arising from extraterrestrial object impacts on a safety-related structure.

DOE 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 invariant over the same period. These assumptions are reasonable for the type of analysis DOE 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 also notes that the sources of meteorite information (e.g., Bland and Artemleva, 2006aa; Ceplecha, 1994aa) are reasonable for the analysis as they are from established literature. In addition, the size range used in the analysis is reasonable because large-sized meteorites would disintegrate into the atmosphere resulting in small-sized objects. 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 analysis presented in BSC Section 6.13 (2008ai) is reasonable. Because the estimated annual frequency of a meteorite striking the GROA is less than 10^{-6}, DOE's determination that a meteorite strike during the preclosure period would not be a potential initiator of the event sequences in the repository is reasonable.

The NRC staff also notes that DOE'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 reasonable because any nontracked objects are too small to cause any significant damage to any hardened structure that would be used in the repository facilities. The NRC staff independently reviewed more recent 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. The U.S. Space Command closely tracks the larger sized objects, which may survive the reentry. The NRC staff notes that DOE's analysis is conservative with respect to the recent data (Klinkrad, et al., 2001aa).

Waste and Rock Interaction, Thermal Loading, Geochemical Alterations, and Dissolution

DOE provided information on waste and rock interaction, geochemical alterations, thermal load, and dissolution at the repository site in SAR Table 1.6-8, BSC Section 4.4 (2008ai), and DOE (2009ey). DOE screened out waste and rock interaction as an external hazard in SAR Table 1.6-8 because the release of waste from a waste package was identified as strictly postclosure related. With regard to thermal loading, DOE stated in SAR Section 1.3.5 that forced ventilation during the preclosure period would moderate any temperature rise. DOE 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 Evaluation: The NRC staff reviewed the information and technical basis DOE provided on waste and rock interaction, thermal loading, geochemical alterations, and dissolution hazards using the guidance in the YMRP. Specifically, the NRC staff evaluated the information and rationale DOE provided to screen out initiating events arising from waste and rock interaction, thermal loading, geochemical alterations, and dissolution 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 actual observations in the Exploratory Studies Facility, as described in BSC Section 6.6 (2004bg) and SNL Section 7.5.2 (2008aj). Additionally, the NRC staff reviewed the information DOE 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.

DOE used reasonable site data on mineral dissolution and actual observations including model prediction to determine that the rates of progression of these hazards are too slow to be a hazard during the preclosure period. Consequently, DOE's exclusion of waste and rock interaction, thermal loading, geochemical alterations, and dissolution from further consideration as potential initiating events is reasonable.

Perturbation of Groundwater

DOE provided information on perturbation of groundwater at the repository site in SAR Table 1.6-8, BSC Section 4.4 (2008ai), and DOE (2009ey). DOE 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 Evaluation: The NRC staff reviewed the information and rationale DOE provided on the hazard associated with perturbation of groundwater using the guidance in the YMRP.

The NRC staff 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 Sections 8.3 and 8.10 (1979aa). On the basis of the gradual changes the models indicated, DOE's description and technical basis for excluding the perturbation of the Yucca Mountain groundwater system are reasonable.

<u>Improper Design and Operation</u>

DOE provided information on improper design and operation of the GROA facilities in SAR Table 1.6-8 and BSC Section 4.4 (2008ai). DOE identified improper design and operation as one of the hazards unique to the Yucca Mountain repository. However, DOE considered this hazard to be an internal hazard and, consequently, excluded it from the list of credible external hazards, as outlined in BSC Section 4.4 (2008ai).

NRC Staff Evaluation: The NRC staff reviewed the information and rationale DOE provided on hazards associated with improper design and operation of the GROA facilities using the guidance in the YMRP. Improper design of ITS SSCs and their operation during the preclosure period may initiate an event sequence. The NRC staff separately reviewed DOE's assessment of operational hazards in the repository facilities from improper design and operation in TER Section 2.1.1.3.3.2. Therefore, improper design and operation is not applicable with respect to external hazards and DOE reasonably excluded such hazards from further consideration.

<u>Undetected Past Human Intrusions</u>

DOE provided information on undetected past human intrusions at the repository site in SAR Table 1.6–8 and BSC Section 4.4 (2008ai). DOE 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. DOE classified undetected past human intrusion as a nonapplicable external hazard in BSC Section 4.4 (2008ai) because either signs of past human intrusion would be detected during repository construction or erosion of the condition would proceed too slowly to affect the repository facilities during the preclosure period. DOE 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 Table G-1 (2008ab).

NRC Staff Evaluation: The NRC staff reviewed the information and technical basis DOE provided on hazards associated with undetected past human intrusions using the guidance in the YMRP. DOE's information and technical basis to screen out this hazard as not having the potential to initiate an event sequence in the repository facilities during the preclosure period is reasonable because (i) either 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.

Security-Related Hazards

DOE provided information on security-related hazards at the repository site in SAR Table 1.6-8 and BSC Section 4.4 (2008ai). DOE identified sabotage, terrorist attack, and war as security-related events and screened out these hazards as not having the potential to initiate an event sequence in the repository facilities during the preclosure period by stating that security threats are not within the scope of its hazard evaluation for PSCA. In addition, DOE stated that it considered security-related hazards by including safeguards and security systems consistent with the physical security criteria in 10 CFR Part 73 (BSC, 2008bu). Therefore, DOE did not evaluate these events in its PCSA.

NRC Staff Evaluation: The NRC staff reviewed the information on security-related hazards and DOE's classification of these events using the guidance in the YMRP. DOE reasonably assessed security-related hazards by establishing safeguards and security systems to protect repository operations from security-related hazards. Because the establishment of these safeguards and security systems would address the security-related hazards, it is reasonable for DOE not to include security-related hazards in the PCSA.

2.1.1.3.3.2 Operational (Internal) Hazards and Initiating Events

DOE identified internal hazards and initiating events at the GROA in SAR Section 1.6.3. These hazards and associated initiating events are internal to the process or operations and are generally associated with failure of equipment, either system or component, and human-initiated events. The NRC staff's review focused on those initiating events that involved handling operations in the surface facilities with a potential for direct radiation exposure or radionuclide release and that resulted in event sequences with high frequency of occurrence or were close to a categorization boundary. The NRC staff reviewed DOE's methodology and its implementation for identifying initiating events, screening and grouping of initiating events, and quantifying initiating events. The NRC staff further separated its review of DOE's quantification and screening into equipment and human-induced failures at surface and subsurface facilities, fire hazards, internal flood hazards, and criticality hazards.

2.1.1.3.3.2.1 Identification of Internal Initiating Events

DOE described how it identified initiating events in SAR Section 1.6.3.1. DOE provided additional details regarding the process in the surface facility event sequence development analysis documents in BSC Section 4.3.1 (2008ab,bo,ao,bd) and documented the analysis results in BSC Section 6.1.3 (2008ab,bo,ao,bd). DOE described a process for developing initiating events using the master logic diagram approach described in the American Nuclear Society/Institute of Electrical and Electronics Engineers (1983aa), Stamatakos (2002aa), and the Electric Power Research Institute (2004aa). A master logic diagram systematically relates loss of top-level safety functions to lower level failure events via a hierarchical, top-down decomposition of safety systems. DOE provided master logic diagrams to identify potential hazards at each process step in BSC Attachment D (2008ab,bo,ao,bd). DOE 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. DOE provided tables of HAZOP deviations in BSC Attachment E (2008ab,bo,ao,bd). DOE compiled the list of internal initiating events in SAR Table 1.6-3 in each of the event sequence development analysis documents in BSC Table 10

(2008ab,bo,ao,bd). As documented in BSC Section 6.2 and in Attachment F (2008ab,bo,ao,bd), DOE then grouped individual initiating events that are associated with similar operations and with the same system response for further analysis.

NRC Staff Evaluation: The NRC staff reviewed DOE's identification of initiating events using the guidance in the YMRP to determine whether the methodologies used for identifying initiating events were appropriate for identifying initiating events that could lead to risk-significant event sequences. The NRC staff reviewed the discussions of the methodology in SAR Section 1.6.3.1 and in BSC Section 4.3.1 (2008ab,bo,ao,bd), and then examined BSC Chapter 6 and Attachments D and E (2008ab,bo,ao,bd) to understand how DOE applied these methodologies.

The NRC staff notes that the use of master logic diagrams and hazard and operability studies are standard approaches that industry and NRC use to identify initiating events. For example, NRC references the HAZOP methodology described in American Institute of Chemical Engineers (1989aa) in the YMRP. The NRC staff notes that additional events, beyond those identified in SAR Table 1.6-3, are included in the fault trees that DOE used to quantify groups of initiating events. These are either identified directly in the fault tree or were identified through the evaluation of human reliability. The NRC staff notes that using fault trees and using human reliability analysis to develop potential initiating events are standard approaches for modeling contributors to system failure.

The NRC staff reviewed DOE's identification of initiating events to determine whether site data and system information were reasonably used in the identification of internal initiating events (i.e., would not result in an underestimate of risk). To perform its review, the NRC staff conducted an audit of selected event sequences to cover a range of initiating event types and facilities. The NRC staff also conducted a more detailed audit of the initiating event identification for CRCF (BSC, 2008ab) and for WHF (BSC, 2008bo). The NRC staff mapped selected events from the MLD and HAZOP tables into the fault trees and then examined how the events were included in the fault trees.

DOE's process flow diagrams are developed to a level of detail so that challenges at different operational steps can be identified because the operations described in BSC Section 6.1.2 and Attachment B (2008ab,bo,ao,bd) describe, for each process step, which pieces of equipment are being used in the step and how the equipment is used. For example, DOE 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 to account for how many crane lifts or slide gate operations would be required to carry out a particular step. The NRC staff notes that DOE reasonably considered other operating modes, such as maintenance activities, in identifying initiating events. For example, DOE stated that maintenance will not be performed on equipment that is in operation, and several fault trees include the possibility of the failure to reset systems following maintenance. Similarly, for mechanical systems such as cranes that DOE quantified using empirical data, maintenance-related failures are implicitly included in such estimates of failure rates. DOE briefly discussed testing and maintenance for those systems for which a fault tree was developed.

The NRC staff notes that DOE's determinations regarding maintenance, testing, and ancillary operations are consistent with the information reasonably available.

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

DOE discussed grouping of initiating events identified in SAR Table 1.6-3 and in Section 4.3.4.4 of the surface facility event sequence development analysis documents in BSC Section 4.3.4.4 (2008ab,bo,ao,bd). DOE stated that events from the master logic diagram that involve the same SSCs, operations response, and the same pivotal event system response are grouped together. DOE documented these groupings in BSC Section 6.2 and in Attachment F (2008ab,bo,ao,bd). For categorization purposes, these initiating event groups are further combined with events that pertain to the same operational area/activity and that lead to the same end state.

DOE discussed screening of initiating events in SAR Section 1.7.1.2.1 and in the surface facility event sequence reliability and categorization analysis documents in BSC Section 6.0 (2008ac,as,be,bq). DOE identified criteria for screening of initiating events that includes screening by design features and screening by subsuming less significant events into existing events. Internal events that DOE screened out were listed in SAR Table 1.7-1. The screened events corresponding to SAR Table 1.7-1 are provided in BSC Table 6.0-2 (2008ac,as,be,bq).

NRC Staff Evaluation: The NRC staff reviewed the SAR and supporting documents to determine whether reasonable technical bases for the inclusion and exclusion of internal initiating events were provided. To conduct its review, the NRC staff reviewed two types of selected audits. First, the NRC staff reviewed a subset of event sequences selected to cover a range of initiating event types and facilities and examined how the initiating events for these sequences were quantified. In addition, the NRC staff audited the initiating event identification for CRCF (BSC, 2008ab,ac) and for WHF (BSC, 2008bo,bq). The NRC staff selected these two facilities to provide a representative sample of handling both canistered wastes and canistered and uncanistered spent fuel. The NRC staff mapped selected events from the master logic diagram and HAZOP deviation tables into the fault trees. The NRC staff examined whether the events were reasonably included in the fault trees developed to quantify the initiating event group represented by the small bubbles in BSC Attachment F (2008ab,bo,ao,bd). Consistent with the approach outlined for grouping of the "small bubble" initiating event groups, the NRC staff further grouped these small bubbles according to the type of challenge that they posed, corresponding to the top event of the fault trees, as documented in BSC Attachment B (2008ac,as,be,bq), and the system response event trees to which the initiating event fault trees were assigned in Attachment A of these documents. For mechanical challenges to containment, the NRC staff used the challenges identified in BSC Table 6.3-7 (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.

DOE grouped the initiating events from the master logic diagram into groups on the basis of safety function and similarity of challenges to safety systems, as discussed in BSC Section 6.2 and Attachment F (2008ab,bo,ao,bd). The methodology of grouping initiating events by safety function and by similarity of the challenges posed to the system (i.e., system responses) is consistent with the guidance in NUREG/CR–2300 Chapter 3 (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa).

For the CRCF, the NRC staff notes that the initiating event groups shown in BSC Attachment F (2008ab) could be further grouped into the following bins corresponding to safety system response: (i) loss of shielding resulting from human or equipment failure and (ii) loss of containment or shielding due to thermal or mechanical challenges to waste containers. Thermal challenges to containment or shielding are evaluated in TER Section 2.1.1.3.3.2.4. Mechanical challenges comprise drops of heavy {equiv [>9.1 t]} loads onto waste containers, flat-bottomed waste container drops (from below, at, or above operational height), collisions and side impacts to waste containers, inadvertent lateral motions leading to shearing impacts to waste containers, and waste container tipovers. Initiating events for WHF could be grouped similarly, with the addition of events initiated by cask sampling errors, events leading to pool water spills, and events associated with damage to bare fuel assemblies during preparation operations for transportation or dual-purpose casks or spent fuel transfer machine operations. These groups correspond approximately to the main types of top events and failure scenarios identified in the fault trees, as well as to the challenges for which passive reliabilities are computed and shown in BSC Tables 6.3-7 and 6.3-8 (2008ac,bq). These groups encompass a wide range of structural, thermal, and human-induced challenges, and these challenges encompass major safety functions, such as shielding, containment, and criticality prevention. The NRC staff considers that any potential initiator not listed in SAR Table 1.6-3 would likely fall into one of these groups and therefore be similar to the existing initiating event groups identified in BSC Attachment F (2008ab,bo,ao,bd).

DOE listed design bases for the ITS SSCs in BSC Table 6.9-1 (2008ac). The NRC staff notes that, in many cases, DOE ensured that design features used to screen initiating events were apparent in the nuclear safety design bases. For example, in the list of nuclear safety design bases provided in BSC Table 6.9-1 (2008ac), DOE included the requirement that the site transporter fuel tank will be designed to preclude explosions, thereby precluding consideration of cask breaches due to explosion. This list included columns to indicate how the design basis was derived from the safety analysis. However, the NRC staff notes that events are sometimes screened by design features that are not apparent in the nuclear safety design bases. Examples include the following.

- Damage to a DOE or HLW canister inside either a transportation cask or a codisposal waste package is screened out in BSC Table 6.0-2 (2008ac) on the basis of design features that would prevent a dropped object from contacting the canisters, but the associated safety function is identified only for the transportation cask.

- Damage to the canister transfer machine slide gate and supporting structures because of vertical drop of an object is screened out on the basis that the canister transfer machine slide gate and supporting structures would be designed to withstand a 30-cm [12-in] vertical drop of the heaviest canister (DOE, 2009dy).

- Transportation cask or shielded transfer cask falling into the decontamination pit is screened out on the basis that the decontamination pit cover would be strong enough to prevent a transportation cask or shielded transfer cask from penetrating the cover and falling into the pit, used to screen consideration of higher drop heights, as described in BSC Attachment A (2008bq).

Nonetheless, the NRC staff notes that DOE provided safety functions (see discussion in TER Section 2.1.1.6.3.1). TER Section 2.1.1.6.3.1 states that as part of the detailed design process, DOE should confirm that the safety functions and the assumptions regarding passive and active systems relied on to screen out initiating events are consistent with the design.

The NRC staff notes that, in many cases, screened events (were they to be included) would not be capable of significantly changing the frequency or consequences of event sequences. For example, DOE provided a rationale to show that the probability of dropping a canister into the Cask Unloading Room or Waste Package Positioning Room with no waste package present would require a series of human failures and mechanical failures that makes the initiating event unlikely.

The NRC staff notes that, in many cases, initiating events were reflected in the specific fault trees or system response trees to which they are mapped, and conversely, that events identified in fault trees and in HAZOP deviations were reflected in the list of events provided in SAR Table 1.6-3. The NRC staff was able to determine how initiating events were carried forward into the fault tree analyses.

2.1.1.3.3.2.2.2 Quantification of Initiating Events

DOE's approach for quantifying initiating events is discussed in SAR Section 1.7.2 and in the surface facility event sequence reliability and categorization analysis documents in BSC Sections 4.3 and 6.2.1 (2008ac,as,be,bq). DOE quantified initiating events by developing fault trees for groups of the initiating events listed in Table 1.6-3 rather than quantifying each individual initiating event. These groups are identified in the surface facility event sequence development analysis documents in BSC Attachment F (2008ab,bo,ao,bd). Each of the groups is mapped to a fault tree or basic event in BSC Attachment A (2008ac,as,be,bq).

DOE estimated equipment reliability either by a direct probability assignment or by modeling using fault trees parameterized by empirical data derived from standard equipment reliability databases or estimates of human error on the basis of a human reliability analysis. DOE stated that it used fault trees, rather than direct probability assignments, to model 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 BSC Sections 4.3.2.1 and 6.2.2 (2008ac,as,be,bq), DOE developed fault trees following the process described in NUREG–0492 (NRC, 1981ab). Fault trees for particular components were provided in BSC Attachment B (2008ac,as,be,bq).

As discussed in SAR Section1.7.2.2 and in BSC Sections 4.3.3 and 6.3 and Attachment C (2008ac,as,be,bq), DOE estimated equipment reliability (either directly assigned or modeled using fault trees) using databases such as NUREG–1774 (NRC, 2003ai), NRPD–95 (Denson, et al., 1994aa), and others. DOE estimated the probability of failure for various components and then used these estimates to develop initiating event probabilities as part of its PCSA.

SAR Section 1.7.2.2 described how DOE 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. DOE described, in BSC Section C1 (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. DOE then used 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 included with the supporting documents. In applying the Bayesian techniques, DOE used a parametric empirical Bayes method (Siu and Kelly, 1998aa; Droguett, et al., 2004aa). Additionally, DOE used the alpha factor method to quantify common-cause failure. DOE 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. DOE explained that this approach was selected when the use of Bayesian analysis with multiple similar estimates would yield an unrealistically narrow distribution. DOE used this approach to quantify, for example, the interlock failure on demand. DOE 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). DOE 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), DOE 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 BSC Attachment C (2008ac,as,be,bq) were computed and documented using a combination of spreadsheets and Mathcad files that DOE included as BSC Attachment H (2008ac,as,be,bq).

DOE described the approach to assessing human reliability in SAR Section 1.7.2.5 and BSC Attachment E and Sections 4.3.4 and 6.4 (2008ac,as,be,bq). The list of resulting human failure events was included in BSC Tables 6.4-2 and 6.4-1 (2008ac,as,be,bq). DOE 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–Sb–2005 [American Society of Mechanical Engineers Section 4.5.5 (2005ad)], and NUREG–1624 (NRC, 2000ai). The approach DOE outlined is an iterative process that begins with a definition of the scope of the analysis, works through an identification of potential human failure event, 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. DOE 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 (NUREG–1624) (NRC, 2000ai). DOE discussed the general human reliability assessment method selection in BSC Attachment E (2008ac,as,be,bq), including selecting four human failure event quantification methods to treat operator errors. DOE'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 Evaluation: The NRC staff reviewed the SAR and supporting documents to determine whether DOE reasonably determined the frequency of occurrence of internal initiating events. To conduct its review, the NRC staff conducted two types of audits. First, the NRC staff reviewed a subset of event sequences selected to cover a range of initiating event types and facilities and examined how the initiating events for these sequences were quantified. In addition, the NRC staff audited the initiating event identification for CRCF (BSC, 2008ab,ac) and for WHF (BSC, 2008bo,bq). The NRC staff selected these two facilities to provide a representative sample of handling both canistered wastes and canistered and uncanistered spent fuel. The NRC staff mapped selected events from the master logic diagram and HAZOP deviation tables into the fault trees. The NRC staff examined how these events were included in the fault trees developed to quantify the initiating event group represented by the small bubbles in BSC Attachment F (2008ab,bo,ao,bd). In particular, the NRC staff examined how DOE quantified the basic events used to quantify these fault trees, and then examined the frequencies within the initiating event groups (as the NRC staff developed in TER

Section 2.1.1.3.3.2.2.1) to determine whether the results of the quantification were reasonable and consistent with industry experience.

NRC Staff Evaluation of Equipment Reliability

DOE documented the quantification of active component reliability in SAR Section 1.7.2.2 and in BSC Sections 4.3.3 and 6.3.1, Table 6.3-1, Attachment C, and the supporting files in Attachment H (2008ac,as,be,bq). The NRC staff reviewed these documents to determine whether empirical analyses and modeling techniques were used reasonably to estimate equipment reliability and whether uncertainty in the reliability estimates had been addressed. The NRC staff examined DOE's approach to determine selected system failure probability distributions for the CRCF, intrasite, and the subsurface. Because DOE used the same approach to develop failure probability distributions for active systems and components for all surface facilities, the NRC staff only reviewed component failure distributions for the CRCF in detail.

For most of the cranes used in the surface facilities {i.e., the equivalent [200-ton] cask handling cranes, the waste package handling cranes, jib cranes, and the spent fuel transfer machine}, a direct quantification of the failure probability was performed based on analysis of empirical data on crane drops taken from NUREG–1774 (NRC, 2003ai) and NUREG–0612 (NRC, 1980aa). The failure frequencies DOE estimated for these cranes, as documented in BSC Attachment C.1.3 (2008ac,as,be,bq), are in the range of 10^{-6} to 10^{-4} per transfer. The NRC staff considers this to be in broad agreement with the values reported in NUREG–0612, Appendix B.1.1.2.3 (NRC, 1980aa) of approximately 10^{-5} and 1.5×10^{-4} per lift for non-single-failure-proof cranes on the basis of U.S. Navy experience in 1977 and the values for very heavy load drops (3 drops out of 54,000 lifts, or approximately 6×10^{-5} per transfer) from NUREG–1774 (NRC, 2003ai). The spent fuel transfer machine drop rate is estimated to be in the range of 5×10^{-6} per transfer, which the NRC staff considers to be in broad agreement with the data on fuel handling drops from NUREG–1774 (NRC, 2003ai).

DOE modeled the reliability of equipment for which no historical data exist at the system level (e.g., the canister transfer machine, the cask transfer trolley) by developing fault trees to a level of detail regarding components that allows the use of industry experience with similar components. For example, failure of pressure sensors used in crane load cells was quantified using a selection of pressure sensors from the NRPD–95 (Denson, et al., 1994aa) database. The NRC staff notes that databases used by DOE, such as NRPD–95 (Denson, et al., 1994aa), are widely used for reliability engineering.

The NRC staff notes that DOE used Bayesian analysis to combine information from multiple data sources to quantify the uncertainty in reliability estimates for active systems and components. DOE documented these analyses in Mathcad files included with the supporting documents. Bayesian analysis techniques are standard techniques identified in NUREG/CR–6823 (Atwood, et al., 2003aa). For cases with only one data source with no uncertainty data, DOE used the single data source and then updated the value using a Jeffreys noninformative prior distribution to estimate the uncertainty in the reliability estimate. The NRC staff notes this approach is consistent with NUREG/CR–6823 (Atwood, et al., 2003aa). The NRC staff notes that in some cases, DOE used a single data source to quantify the uncertainty in equipment reliability even when more than one data source was present. DOE explained that this approach was taken to avoid unrealistically narrow uncertainty estimates. DOE stated that, in such cases, it selected the data source that would yield the greatest uncertainty and used the median of the data sources as representative of the mean failure rate (DOE, 2009dy). DOE's approach is reasonable because this approach results in more realistic uncertainty estimates. In addition, the NRC staff notes that DOE used the alpha factor method to quantify common cause failure. This method is a standard method capable of handling various levels of redundancy.

The NRC staff notes, on the basis of examination of the spreadsheets provided in support of the active component reliability database in BSC Attachment H (2008ac,as,be,bq), that DOE generally selected data sources that reflected comparable component types and failure modes from the selected database. DOE identified the necessity of evaluating the similarity between the Yucca Mountain Project operating environment and that represented in each generic data source to ensure data appropriateness. However, documentation of this evaluation was not included in the attachments to the event sequence reliability and categorization analysis documents. Although DOE used failure data from a holding brake used in sonar systems to quantify the canister transfer machine holding brake failure and used data on pressure sensors from a hotel HVAC system to quantify failure of the canister transfer machine load cell pressure sensor, DOE did not explain why these environments were similar to the Yucca Mountain Project operating environment. Likewise, in estimating the reliability of the missing equivalent needed [200-ton] crane, DOE treated data from NUREG–0612 (NRC, 1980aa) that appear to reflect the same underlying piece of equipment as independent, distinct data sources, with no discussion of why this is appropriate. Also, the NRC staff notes that DOE modeled the reliability of the canister transfer machine on the basis that the canister transfer machine is a crane for which no historical data exist at the system level but that the unique features of the canister transfer machine that would distinguish it from a more conventional crane have not been specifically identified and examined. However, the NRC staff determined that DOE selected component reliability numbers that result in overall estimates of system reliability and are in a broad agreement with industry experience, as provided in NUREG–1774 (NRC, 2003ai).

NRC Staff Evaluation of Human Reliability

The NRC staff examined whether empirical analyses and modeling techniques were used reasonably to estimate human reliability and whether uncertainty in the reliability estimates have been addressed. The NRC staff reviewed the information on quantification of human failure events that DOE provided using the guidance in the YMRP and HLWRS-ISG–04 (NRC, 2007ad) to evaluate DOE's treatment of operator errors to determine whether (i) the methodology DOE used to assess the potential for operator errors was reasonable and (ii) DOE reasonably implemented its methodology.

The NRC staff examined DOE's methodologies documented in the SAR and BSC Attachments A, B, and E (2008ac,as,au,be,bk,bq). To determine whether DOE's overall human reliability analysis process is reasonable, the NRC staff examined the detailed description of the process in BSC Attachment E (2008ac,as,au,be,bk,bq) and notes that DOE used a comprehensive human reliability analysis process, including steps not typically addressed in detail for nuclear power plant human reliability analysis analyses. The NRC staff considers this reasonable because it is consistent with other accepted human reliability analysis processes and HLWRS–ISG–04 (NRC, 2007ad). For example, the process DOE used was the one developed for the ATHEANA human reliability analysis method (NRC, 2000ai). As part of this process, DOE included a detailed search process to identify human failure events, which is not often an important focus for nuclear power plant human reliability analysis. Furthermore, DOE'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.

To determine whether DOE selected the human failure event quantification methods appropriately, the NRC staff examined BSC Attachment E, Appendix E.IV (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. DOE 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. DOE's choices are reasonable because relevant factors to these decisions were identified and discussed and were generally consistent with the NRC guidelines (2007ad). For example, NUREG–1792 (NRC, 2005ae) recommends 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. BSC Appendix E, Attachment E.IV (2008ac,as,au,be,bk,bq) documented how DOE matched the four selected methods with characteristics of the repository facilities and its operations that are relevant to potential operator errors.

Because the NRC staff considers that DOE's overall human reliability analysis process and selection of human failure event quantification methods are reasonable, the NRC staff notes that DOE selected reasonable methodologies for human reliability analysis.

To determine whether the results of DOE's process for identifying human failure events to include in its PCSA are reasonable, the NRC staff examined the results documented in BSC Table E7–1 (2008ac,as,au,be,bk,bq), as well as qualitative analyses of potential human failure events in BSC Appendices A, B, and E (2008ac,as,au,be,bk,bq). DOE considered a broad range of potential operator errors to include in the PCSA. The NRC staff notes that this is reasonable because the identification process described was thorough and the justifications provided for excluding selected operator actions were logical. For example, in BSC Section E.6.0 (2008ac,as,au,be,bk,bq), DOE provided an overview of the identification process, including a description of the role of HAZOP and other types of hazard search approaches, and examples of how such analyses were used to make decisions about including certain crosscutting human failure events in the PCSA.

To determine whether the results of DOE's qualitative analyses—both for the facility as a whole and for specific activities, locations, and environments—are reasonable, the NRC staff examined the results of the general qualitative analysis documented in BSC Sections E.4 and E.5 (2008ac,as,au,be,bk,bq), as well as the results of the qualitative analysis for specific human failure events documented in the various subsections of BSC Section E.6 (2008ac,as,au,be,bk,bq). BSC Section E.5 (2008ac,as,au,be,bk,bq) described the various operator roles and the potential operator vulnerabilities that would be generally applicable to the repository operations. Then the qualitative inputs used in human reliability analysis quantification were discussed for each specific human failure event. The NRC staff notes that the qualitative analysis produced a variety of general and human failure event-specific results and, for the current level of design, the results of DOE's qualitative analyses are reasonable. As part of the detailed design process, DOE should confirm that its human reliability analyses (e.g., task analyses) identified potential vulnerabilities for the repository facilities and associated activities.

To determine whether DOE's treatment of dependencies was reasonable, the NRC staff examined the general discussion of dependency treatment in BSC Section E.3.3 (2008ac,as,au,be,bk,bq), as well as the treatment for specific human failure events, as applicable. DOE used a traditional approach for treating dependencies and, in discussing the quantification for specific human failure events, DOE identified when and why dependencies should be modeled. The DOE approach is reasonable because traditional mechanisms for potential dependencies were identified and addressed. For example, DOE used the dependence approach documented in Swain and Guttman (1983aa), which is the traditional approach for treating dependencies.

To determine whether DOE's selection of human reliability analysis quantification methods for specific human failure events was reasonable, the NRC staff reviewed the detailed discussion of the qualitative analysis inputs associated with each human failure event, as well as any generic qualitative inputs, and compared these inputs with how DOE represented these factors by selecting and using specific human reliability analysis detailed quantification methods. DOE 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. Therefore, DOE's approach is reasonable because results of qualitative analyses matched well with the required inputs for the quantification method and the quantification method addressed the relevant error modes for the operator action. For example, in BSC Section E.6.5.3.4.4.5 (2008ac,as,au,be,bk,bq), DOE first described the scenario (or elements that contribute to the human failure event) and then described, in storylike fashion, how the scenario might occur. The elements of this discussion were then related to a specific method (e.g., NARA) 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 reasonable, 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. DOE explicitly described the analyst choices for specific inputs to detailed human reliability analysis quantification methods, and the NRC staff notes this is reasonable because relevant and expected error mechanisms were reflected, the methods were applied consistently and as intended, qualitative analysis inputs were reflected, and, generally, the SAR was consistent with HLWRS–ISG–04 (NRC, 2007ad). For example, in BSC Section E.6.5.3.4.4.5 (2008ac,as,au,be,bk,bq), DOE 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.

NRC Staff Evaluation of Overall Initiating Event Frequency

The NRC staff examined the results of DOE's analyses of initiating event frequency. In particular, the NRC staff examined the SAPHIRE files included in Attachment H to evaluate the numerical estimates of the initiating event fault trees identified in BSC Attachment F (2008ac,as,be,bq).

DOE 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 Yucca Mountain facilities. DOE modeled failure rates using fault trees that were developed and quantified using SAPHIRE to model systems that it considered to have no direct analog. The models combined estimates of equipment and human errors that could lead to unintended radiological exposures. The NRC staff considers the use of fault trees, as described in NUREG–0492 (NRC, 1981ab), for quantifying initiating events to be a standard industry practice used by NRC. In particular, SAPHIRE has been developed for NRC use and is a reasonable software platform for modeling initiating event frequencies.

DOE 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 a few events out of every thousand transfers). Data provided by DOE indicate that the reliability of the waste containers against such challenges is very high (failure frequencies less than 1 in every 100 million challenges), such that the probability of a breach would be very low. The estimated frequency of these events tended to be dominated by estimates of human errors. The NRC staff considers that frequencies of this magnitude are consistent with estimates of human error.

DOE estimated frequency of tipovers, drops of heavy objects onto canisters, and flat bottom drops of a waste container (all of which would result in higher energy mechanical impacts against which the canisters are less resilient) to occur less frequently (a few events out of every 100,000 transfers). The NRC staff notes that these estimates were dominated by estimates of crane reliability. These frequencies are broadly consistent with empirical data on crane reliability from industry experience as documented in NUREG–0612 (NRC, 1980aa) and NUREG–1774 (NRC, 2003ai).

The NRC staff notes that events associated with shearing-type impacts to a canister were estimated to occur with very low frequencies (on the order of 7×10^{-9} per transfer) and were associated with multiple human and equipment failures. The NRC staff considers that events associated with multiple human and equipment failures could be reasonably expected to have a low frequency of occurrence. Furthermore, DOE has included design bases specifically intended to limit the occurrence of shearing-type impacts.

The NRC staff notes that DOE estimated event frequencies in WHF that were similar to those estimated for the CRCF for analogous canister handling operations. DOE estimated a frequency of damage due to drops during spent fuel transfers in WHF on the order of 5×10^{-6} per transfer. NRC staff notes that these frequencies are broadly consistent with empirical data from industry experience as documented in NUREG–0612 (NRC, 1980aa) and NUREG–1774 (NRC, 2003ai).

The NRC staff notes that events involving a loss of shielding due to equipment or human error were estimated to occur relatively infrequently (on the order of only a few events every 100,000 transfers). DOE estimates of the frequency of these events tended to be dominated by multiple human errors resulting in loss of shielding. Because these events did not have passive mitigation features, the probability of the initiating event was the probability of the event sequence, potentially leading to generally relatively high frequency Category 2 event sequences. The NRC staff notes that DOE stated that it will add an interlock function to an existing interlock on the canister transfer machine in the CRCF to reduce the probability of an operator error leading to a direct exposure (DOE, 2009dx). DOE also stated (DOE, 2009dx) that it will review direct exposure event sequences in the CRCF, RF, WHF, and IHF and make changes, if necessary, to achieve margins to the Category 1 threshold similar to those achieved in the event sequences involving direct exposure during canister transfer in CRCF [CRCF–ESD–18, as discussed in BSC Section 6.2.19 (2008ab) and in DOE (2009dx)].

2.1.1.3.3.2.3 Quantification of Initiating Event Frequency for Subsurface Operations

DOE provided information and analysis on hazards and initiating events at the subsurface facilities in SAR Section 1.6.3.1 and BSC Section 6.2 (2008bk). DOE described the operations within the subsurface facilities that are used in the MLD and HAZOP analyses in BSC Section 6.1.2 (2008bj) and discussed the results in BSC Section 6.1.3 (2008bj). DOE identified 29 initiating events at the subsurface facilities developed from these analyses, as outlined in BSC Table 11 (2008bj) and SAR Table 1.6-3. Additional information and analyses of these initiating events were provided in BSC (2008bk) and DOE's response to the NRC staff's RAIs (DOE, 2009dy,ey). DOE used this assessment of initiating events as the basis for evaluating whether any of these initiating events would develop into an event sequence. DOE further detailed the results of the MLD and HAZOP processes in BSC Attachments D and E (2008bj). DOE aggregated the individual initiating events into ESDs and described the results of this aggregation in BSC Section 6.2 (2008bj), where each resulting ESD was discussed. DOE performed a screening analysis on the aggregated ESDs, as described in SAR Section 1.7.1.2 and in BSC Section 6.0 (2008bk). To quantify the probability of occurrence of the event sequence, DOE used fault tree analyses as described in BSC Sections 4.3.2 and 6.2.2 and Attachment B (2008bk). DOE also considered the human error and used the results of passive reliability analysis for equipment failure, as given in BSC Attachment D (2008bk) in developing the initiating event analysis, as detailed in BSC Attachment E (2008bk).

Consistent with the audit approach discussed in NRC (2001aa), the NRC staff selected a subset of the initiating events for detailed review. 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 reasonably considering site-specific and facility information.

Use of System Information and Methodology

For the initiating event "TEV Impact During Transit," DOE indicated that collision with another object can 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 Table 11 (2008bj) and BSC Section B1.4.4 (2008bk). Using this facility-specific information, DOE developed the fault tree model for the initiating event. DOE 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.

DOE modeled the initiating event "TEV Stops for an Extended Period of Time" in a fault tree, SHIELD–STOP, as described in BSC Section B1.4.5.4 (2008bk) 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. Additionally, DOE used the alpha-factor method to analyze common-cause failure of the speed sensor of the motors.

NRC Staff Evaluation: The NRC staff reviewed DOE's assessment of initiating events of "TEV Impact During Transit" and "TEV Stops for an Extended Period of Time" using the guidance in the YMRP. The fault tree approach for assessing the "TEV Impact During Transit" initiating event is reasonable because it models the scenarios likely to be encountered. In addition, the NRC staff notes that DOE reasonably 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.”

Data Use

In conducting the fault tree analysis for the initiating event, labeled as TRANSIT–IMPACT (BSC, 2008bk), DOE used the reliability data for component failure from NPRD–95 (Denson, et al., 1994aa). For the initiating event “TEV Stops for an Extended Period of Time,” detailed in BSC Section B1.4.5 (2008bk), DOE identified a requirement for the TEV shielding to be designed to sustain the thermal loading for all waste package loadings over an extended period without significant degradation of the shielding function, as described in BSC Section B1.4.5.3 (2008bk). DOE indicated in BSC Section B1.2.4 (2008bk) that the TEV shielded enclosure includes a layer of synthetic polymer (NS–4–FR) with a maximum continuous operating temperature of 150 °C [302 °F].

NRC Staff Evaluation: The NRC staff reviewed the reliability data DOE used in the fault tree analysis for the initiating event, labeled as TRANSIT–IMPACT, using the guidance in the YMRP. DOE used information on mechanical component reliability from an authoritative reference (Denson, et al., 1994aa). Therefore, the NRC staff considers that DOE used the TEV system information and reasonable reliability data from the fault tree analysis. In addition, DOE stated that it will use a layer of synthetic polymer in the TEV enclosure. As discussed next, this polymer will help keep the operating temperature within the shielded enclosure of the TEV below 150 °C [302 °F]. As all possible failure modes for TRANSIT–IMPACT were reasonably considered in the analysis, the NRC staff notes that DOE used reasonable information and data to identify initiating events.

Estimation of Annual Frequency

DOE indicated that the frequency of TEV impact during transit is primarily controlled by an operator driving another vehicle into it (the operator fails to yield to the TEV at a crossing on the surface), which is approximately 99 percent of the contribution. To reduce the probability of TEV impact during transit, DOE proposed to install special crossing barricades and signals at all surface intersections. DOE stated that it will restrict all traffic from the area of a loaded TEV in the subsurface facilities. Additionally, the TEV travels slowly {roughly 3.2 km/hr [2 mph]} and an operator will watch via camera, as outlined in BSC Table E6.2-2 (2008bk).

NRC Staff Evaluation: The NRC staff reviewed the information DOE used to determine the annual frequency of occurrence of the initiating event “TEV Impact During Transit” using the guidance in the YMRP. DOE determined that the frequency of TEV impact during transit is primarily controlled by an operator driving another vehicle into it, which is approximately 99 percent of the contribution. The NRC staff notes this is reasonable because the human-induced hazard was reasonably identified and most mechanical or electrical failure has a significantly low probability of occurrence. Furthermore, DOE’s proposed actions to reduce the probability of TEV impact during transit are reasonable because these actions included significant redundancy for many mechanical components of the TEV (e.g., eight motors of the wheel system), installation of special crossing barricades and signals at all surface intersections, restriction of all traffic from 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 Table E6.2-2 (2008bk).

Technical Basis for Screening

DOE estimated the annual frequency of TEV impact during transit to be 3.03×10^{-4}, on the basis of a point estimate, as detailed in BSC Figure B1.4-7 (2008bk). DOE also considered the uncertainties associated with the parameters. Following the standard approach, DOE (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 2.94×10^{-4} and 7.36×10^{-4}, respectively, as shown in BSC Figure B1.4-7 (2008bk). Consequently, this initiating event was retained for event sequence analysis (SAR Table 1.6-3). Note that the mean is not same as the point estimate in the BSC analysis (2008bk).

DOE estimated approximately 8.5 occurrences of extended TEV stoppage during the preclosure period, as outlined in the Event Tree SSO-ESD04 (BSC, 2008bk), but screened out SSO–ESD–04 on the basis of a zero probability for loss of shielding. This event was screened out because DOE 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, DOE 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 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. DOE plans to 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. Because the neutron shielding material NS–4–FR would degrade over time (DOE, 2009ey), DOE stated that it will implement a preventive maintenance program, which would routinely assess the effectiveness of the shielding materials and replace them as necessary.

NRC Staff Evaluation: The NRC staff reviewed the technical basis DOE used to screen the initiating events “TEV Impact During Transit” and “TEV Stops for an Extended Period of Time” using the guidance in the YMRP. To determine whether DOE provided a reasonable technical basis for screening initiating events, the NRC staff examined the initiating event “TEV Impact During Transit.” The NRC staff notes that the difference between the mean and the point estimate in the BSC analysis is because DOE did not perform sufficient Monte Carlo sampling. The NRC staff’s independent verification showed that the mean would be the same as the point estimate if sufficient Monte Carlo sampling is performed.

The NRC staff therefore notes that DOE showed that extended TEV stoppage would not initiate an event sequence, because DOE stated that it 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 and the effectiveness of this material would be routinely assessed. Therefore, on the basis of this discussion, DOE characterized the potential hazards and initiating events at the subsurface facilities during operations.

2.1.1.3.3.2.4 Quantification of Initiating Event Frequency for Fire Hazards

With respect to potential fire hazards at the repository facilities during the preclosure period, DOE presented information in SAR Sections 1.6 and 1.7 and BSC Attachment F (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 reviews the fire hazard at the repository facilities in this TER section. However, potential hazards from wildfires are evaluated in TER Section 2.1.1.3.3.1.3.5.4.

Methodologies Used

DOE identified fire as a potential initiating event and employed the probabilistic risk assessment methodology outlined in Science Applications International Corporation (2002aa) to assess the fire potential at the repository facilities. DOE described the methodology in BSC Attachment F (2008ac,as,au,be,bk,bq). The methodology requires DOE to determine (i) an overall ignition frequency for a particular facility or area, (ii) the distribution of that ignition frequency to develop definitive fire event sequences, and (iii) the likelihood fire will propagate from the area of origin to other areas of the facility.

DOE used an overall methodology described in Science Applications International Corporation (2002aa) to determine the frequency of fire-related initiating events for GROA facilities. This methodology begins with developing an overall building or facility ignition frequency. The overall frequency was derived using two different approaches and data sets. The ignition frequency for surface facilities [e.g., CRCF, IHF, RF, WHF, and LLW facility (LLWF)] was derived on the basis of historical fire data from comparable industrial facilities, as Tillander (2004aa) described. In addition, DOE used a scoring methodology, described in Electric Power Research Institute (2005aa), to determine the ignition frequency on a per-room basis. The ignition frequencies for the intrasite operations and waste storage areas (e.g., subsurface areas and aging pads) were determined on a per-facility basis using historical data from the U.S. Census Bureau (2000ab).

NRC Staff Evaluation: Using the guidance provided in the YMRP, the NRC staff reviewed information DOE provided on its methodologies to access the fire potential at the repository facilities. In addition, the NRC staff referenced Science Applications International Corporation (2002aa), Electric Power Research Institute (2005aa), and Tillander (2004aa) for the methodology used in the CRCF, IHF, RF, WHF, and LLWF.

The NRC staff evaluated the reasonableness of the method used to develop initiating events arising from fires that could affect operations at the waste handling facilities, intrasite operations, LLWF, and subsurface operations. The NRC staff examined DOE's overall approach and subsequently selected specific facilities for detailed review. The fire analysis for the CRCF, as described in BSC (2008ac), was selected for detailed review because (i) the activities at this facility 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 (e.g., 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 represent a different analysis methodology. As appropriate, the NRC staff reviewed the consistency among facility layout described in SAR Section 1.2.1.2, operations described in SAR Section 1.2.1.3, and fire ignition frequency estimation described in BSC (e.g., 2008ac,as,au,be,bk,bq). The NRC staff's review focus was to determine whether DOE (i) used reasonable methodologies to assess fire hazard, (ii) applied the methodologies reasonably, (iii) used facility-specific data and system information reasonably, and (iv) quantified fire-initiating event frequencies reasonably.

DOE showed that similarities exist in the process, operational characteristics, and fire vulnerabilities between the GROA and facilities handling hazardous chemicals. Therefore, it is reasonable to use the methodology in Science Applications International Corporation (2002aa) to assess fire-related hazards at the waste handling facilities.

Facility Information and Data Sources

The ignition frequency for the intrasite and subsurface facilities was derived using historical data the U.S. Census Bureau (2000ab) provided. DOE 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. DOE considered intrasite operations and subsurface operations from separate facilities to determine ignition frequency. DOE 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, DOE 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) had sufficient potential to expose waste forms in the subsurface facility. DOE opted to assume 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 in the event sequence analysis, as reviewed in TER Section 2.1.1.4.3.1.3.

NRC Staff Evaluation: The NRC staff reviewed information DOE provided on data sources using the guidance provided in the YMRP. The NRC staff referenced U.S. Census Bureau (2000ab) for the number of a given type of facility and Ahrens (2007aa) for historical information on fire in radioactive materials handling facilities. The NRC staff also referenced Ahrens (2000aa) for fire data in industrial chemical, hazardous chemical, and plastic manufacturing facilities.

The NRC staff examined the facility information and data sources DOE used to determine whether DOE used site-specific data and system information reasonably. Ignition frequency was based on a number of data sets (e.g., data from U.S. Census Bureau, NFPA, and Finnish databases). The data sources were relevant to the types of operations being performed at the GROA because they capture historical information at either radioactive material handling facilities or other industrial facilities. Furthermore, DOE employed conservative assumptions and error factors regarding the use of these differing data sources. The ignition frequency distribution was based on historical data on equipment present at radioactive material working facilities and nuclear energy plants of noncombustible construction. The NRC staff considers equipment at radioactive material working facilities and nuclear energy plants to represent the types of equipment present at the GROA. With the exception of heat-generating equipment (none of which is present at the GROA), the NRC staff notes that the ignition categories outlined in Ahrens (2007aa) were reasonably applied.

DOE’s determination of fire propagation probabilities at intrasite and subsurface facilities is reasonable because DOE used data from radioactive material working facilities and nuclear energy plants of noncombustible construction and conservative assumptions regarding flame extent, automatic suppression, and passive fire protection in the analysis.

Application of Methodologies

DOE 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 m² [10,764 ft^2], as Tillander (2004aa) reported. DOE also estimated the confidence limits, as provided in BSC Table F.III-2 (2008ac,as,au,be,bk,bq). According to DOE, the fire ignition frequency was multiplied by the appropriate 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), DOE 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. DOE 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 Electric Power Research Institute (2005aa), to determine the ignition frequency on a per-room basis.

NRC Staff Evaluation: The NRC staff reviewed information DOE provided on application of methodologies to access the fire potential at the repository facilities using the guidance provided in the YMRP. In addition, the NRC staff referenced Science Applications International Corporation (2002aa), Electric Power Research Institute (2005aa), and Tillander (2004aa) for the methodology used in the CRCF, IHF, RF, WHF, and LLWF. The NRC staff referenced U.S. Census Bureau (2000ab) on the number of a given type of facilities and Ahrens (2007aa) on historical information on fire in radioactive materials handling facilities. The NRC staff also referenced Ahrens (2000aa) for fire data in industrial chemical, hazardous chemical, and plastic manufacturing facilities.

To determine whether DOE applied the fire hazard assessment methodologies reasonably, the NRC staff assessed whether (i) the overall facility fire frequency was evaluated reasonably, (ii) the ignition frequency by ignition category was reasonably estimated, (iii) the ignition sources were reasonably distributed between 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. DOE's estimation of fire-related facility initiating event frequencies is reasonable because the calculations are consistent with facility-specific information, and the relationship between ignition frequency and facility floor area contained in Tillander (2004aa) is consistent with types of operations being performed at the GROA. When the NRC staff considered the operations of lifting, welding, packaging, and transporting, it recognized that these physical activities would be commonly found in industrial facilities (e.g., heavy equipment manufacturing, materials processing). The material actually being handled at the facility generally has little effect on the likelihood of an ignition. Therefore, the use of industrial facility data is reasonable. The NRC staff also notes that use of Tillander (2004aa) yields a conservative ignition frequency when compared to historical U.S. data.

The NRC staff reviewed the classification of ignition frequency by ignition sources (e.g., number of fires attributed to welding equipment, electrical equipment, vehicles) and notes that the distribution of ignition frequency by equipment involved is reasonable because the distribution is based on historic fires observed at radioactive material handling facilities and nuclear power plants from 1980 through 1998, as documented in Ahrens Table 36 (2007aa). The use of information in Ahrens (2007aa) to distribute ignition frequency is reasonable as the types of equipment present at the GROA facilities do not differ substantially from the types of equipment used in other radioactive materials handling facilities and nuclear energy plants. With the exception of heat-generating equipment (not present in GROA surface facilities), all of the ignition categories were represented.

The NRC staff reviewed how DOE distributed the overall building ignition frequency to each room of each facility and notes that ignition sources were appropriately distributed among the rooms within the facilities. The approach DOE used is consistent with methodology proposed in NUREG/CR–6850 (Electric Power Research Institute, 2005aa) and Science Applications International Corporation (2002aa), which are accepted methods for fire hazard assessment. The NRC staff also notes that DOE used site-specific inventories of the types of equipment expected in each room to distribute the ignition frequency.

The NRC staff also reviewed how DOE assessed the propagation probabilities of fire inside a waste handling facility. DOE reasonably estimated the probability of fire affecting a waste form at a particular location within the facility, because DOE used conservative approximations to adapt historical information on fire propagation, as documented in Ahrens (2000aa), inside the waste handling facilities. The historical data presented in Ahrens (2007aa) are based on fire propagation, as defined by the maximum extent of flame travel inside a facility. Although the NFPA data (Ahrens, 2007aa) do not provide information on the intensity of the fire once it reaches this maximum extent of flame travel point, the NRC staff notes that DOE conservatively assumed that the fire would have sufficient intensity to become an initiating event.

To determine whether DOE reasonably applied the fire hazard assessment methodology for intrasite and subsurface operations, the NRC staff examined whether DOE (i) reasonably estimated the frequency of fire-related initiating events and (ii) reasonably distributed the fires to functional areas within the GROA. DOE estimated the fire frequency in a manner that is consistent with the approach outlined in Science Applications International Corporation (2002aa). DOE estimated the fire ignition frequency using data of historical fire events Ahrens provided (2000aa) and U.S. Census Bureau (2000ab) data from chemical, plastic, or petroleum products plants (Category Codes 324, 325, and 3261). DOE justified the use of information on chemical, plastic, or petroleum products plants for the waste handling facilities (DOE, 2009fj). The fire hazard analysis used this information only to estimate the fire ignition frequency outside the facility and, consequently, the type of materials being handled inside generally does not significantly influence the estimated fire frequency. The NRC staff understands that ignition sources outside the facilities (in the intrasite and subsurface areas) will be similar to the potential ignition sources identified in the intrasite operations of chemical, plastic, or petroleum plants. The NRC staff notes that the number of fires Ahrens Section 5, Table 1 (2000aa) used is conservative as several of the fires documented in the 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 information on chemical, plastic, or petroleum products facilities would be reasonable for the subsurface repository facilities. Subsurface repository facilities would be similar to noncoal mines in operational aspects. 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 fires annually in noncoal mines. 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 2.2×10^{-3} fires/facility-yr. The NRC staff notes that by considering chemical, plastic, or petroleum products facilities, the assessment in BSC (2008bk) provided a conservative estimate of annual facility fire frequency for subsurface fire events.

For fires in the intrasite and subsurface areas, DOE assumed that a waste form was 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 credible initiating events. The NRC staff notes that both assumptions are conservative.

Initiating Event Frequencies

DOE 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, DOE 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 estimation process also considered the residence time of the waste form in each room or area to assess the likelihood a waste form will be present in various parts of the building during a fire. DOE discounted the performance of passive-fire-resistance-rated wall construction within the facility. DOE 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. DOE summarized the results of these analyses in SAR Section 1.7.1.2.2.

DOE 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 is lognormal; however, normal distribution was selected for the other two parameters. In addition, DOE conducted Monte Carlo simulations with 10,000 samples for each initiating event to estimate mean, standard deviation, maximum, and minimum values using Crystal Ball software.

NRC Staff Evaluation: The NRC staff reviewed information DOE provided on quantification of fire event frequencies using the guidance provided in the YMRP. DOE 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, DOE was able to determine which fires had the potential of reaching a waste form and serving as an initiating event. DOE used facility-specific throughput data to determine residence times for waste forms in various facility locations.

DOE assessed “large fires” as separate events. The large fire events are based on the Ahrens (2007aa) data that showed in roughly 16.9 percent of the recorded fires in radioactive material working facilities and nuclear energy plants, flame propagated throughout the entire floor of origin or beyond. Although the information provided in Ahrens (2007aa) did not explicitly state whether any of these fires were capable of breaching multiple 3-hour rated fire barriers or indicate the level of intensity the fire had during its progression, the NRC staff notes that DOE conservatively assumed that these fires had sufficient intensity to affect a waste package in an adjacent fire area. As a result, the estimated probability of exposure from a large fire is reasonably high and conservative.

DOE 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), and multiplied by a residence time fraction for a particular waste form within the building. The NRC staff notes this is a reasonable approach to estimate the fire-initiating event frequency because it uses data from reliable sources on nuclear facilities and power plants. As stated earlier, the aging pad and subsurface initiating event determination did not include any throughput or propagation probabilities. DOE assumed that waste forms would always be present during potential fire conditions and that fires in these areas would be of sufficient intensity to affect waste forms, which is again a conservative assumption.

The analyses used to quantify uncertainties in fire-initiating event frequencies are reasonable because DOE identified statistical distributions for the parameters of the analysis and used conservative estimates of the error factor for these distributions. For example, DOE assumed an error factor of 15 for the ignition frequency in subsurface and intrasite operations (BSC, 2008au,bk) because two different databases were used to estimate this parameter. The NRC staff notes this large error factor is reasonable because it will include the large uncertainty expected. Additionally, DOE's justification (DOE, 2009fj) of using an error factor of 2.0 for LLWF is reasonable 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. In summary, DOE quantified fire-initiating event frequencies reasonably because it applied the methodology described in Science Applications International Corporation (2005aa) reasonably and quantified uncertainties reasonably in the estimation of fire-initiating event frequencies.

2.1.1.3.3.2.5 Screening of Initiating Events Related to Internal Flood Hazards

DOE identified internal flooding initiating events and provided its technical bases for screening out internal flooding. For the surface facilities, DOE 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, DOE screened out internal flooding as an initiating event. However, DOE 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 addressed in TER Section 2.1.1.4.3.3.2.2 under moderator intrusion control.

DOE screened out internal flooding in the surface facilities on the basis of design (SAR Section 1.7.1.2.3; DOE, 2009fn). DOE stated that a waste form container exposed to water will not lose its structural integrity or shielding capability. DOE also stated that ITS equipment would be located so as not to be wetted or submerged, local barriers would be used, and some components are designed for a wetted or submerged environment. In SAR Section 1.7.1.2.3, DOE addressed the potential for criticality by identifying that there are no water sources sufficient for decreasing boron concentration in the WHF pool to a level that criticality would be a concern. DOE 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 an NRC staff RAI (DOE, 2009fm) and in BSC Section 6.0.4 (2008bk), DOE screened out internal flooding in the subsurface on the basis of insufficient volume to rise to the level of the emplacement drift and insufficient volume to contact a waste package in the TEV. In addition, DOE screened out internal flooding on basis of the TEV design such that the floodwater would not adversely affect the waste package, result in degradation of TEV shielding, nor result in release of radionuclides.

NRC Staff Evaluation: The NRC staff reviewed the identification of internal flooding and DOE's technical bases for screening it out using the guidance in the YMRP. The NRC staff notes that casks, canisters, and waste packages would be sufficiently robust to protect the waste form against exposure to water when they are sealed, as stated in DOE's response to an NRC staff RAI (DOE, 2009fn), and that this is reasonable. In addition, on the basis of the NRC staff's evaluation in TER Section 2.1.1.3.2.7.5 for criticality hazards, the NRC staff notes that criticality is not a concern for internal flooding.

However, DOE did not clearly link the technical bases for screening out internal flooding in the surface facilities to nuclear safety design bases. For example, the NRC staff does not have information to ensure that local barriers could perform their intended safety function to protect equipment, nor does the NRC staff have information to ensure that canister transfer machine ITS interlocks could be designed for environmental conditions involving water spray as stated in DOE's response to this RAI (DOE, 2009fn). However, the NRC staff is able to determine that although some safety functions may not have been specified as part of a nuclear safety design basis, DOE provided safety functions (see discussion in TER Section 2.1.1.6.3.1). Therefore, on the basis of this review and the review results in TER Section 2.1.1.6.3.1, DOE reasonably screened out internal flooding in the surface facilities.

On the basis of its review of SAR Section 1.7.1.2.3 and DOE's response to an NRC staff RAI (DOE, 2009fm), DOE provided reasonable technical bases to screen out internal flooding in the subsurface. The water level would not reach the elevation of a waste package in an emplacement drift. In addition, the NRC staff notes that the water level would not reach the level of a waste package being transported in the TEV. On the basis of its review of DOE (2009fm), the NRC staff notes 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.

2.1.1.3.3.2.6 Screening and Quantification of Initiating Event Frequency for Criticality Hazards

With respect to criticality hazards at the repository facilities during the preclosure period, DOE provided information in SAR Sections 1.6.1.6, 1.7, and 1.14 and BSC (2008ba,bq) and responses to RAIs (DOE, 2009dy,ey). DOE 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 screen out all criticality-related initiating events during the preclosure period, DOE identified seven parameters as important to criticality in SAR Table 1.14-2. For each parameter, DOE performed criticality sensitivity analyses to evaluate the impact on reactivity caused by variations in those parameters. In SAR Section 1.14, DOE provided results from these analyses. These criticality sensitivity analyses showed that each of the parameters (i) needs to be controlled, (ii) does not need to be controlled, or (iii) needs to be conditionally controlled (i.e., needs to be controlled if another parameter is not controlled).

On the basis of 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, DOE identified, developed, quantified, and categorized event sequences that impact the criticality control parameters that were established as needing 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 PSCs were employed to prevent such event sequences. Potentially critical configurations that could occur without a breach or require the introduction of moderator are 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, DOE used the MLD method supplemented by an HAZOP evaluation (e.g., BSC, 2008bo). The screened-in initiating events were listed in SAR Table 1.6-3, and those screened out were listed in SAR Table 1.7-1. DOE stated in DOE Enclosure 8, Section 1 (2009ey) that there were no Category 1 or Category 2 event sequences that required crediting fixed neutron absorbers. Therefore, DOE concluded that material selection errors during manufacturing of such absorbers need not be considered for preclosure criticality safety.

DOE screened out 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 Enclosure 9, Section 1.1 (2009ey). DOE provided the technical basis for screening the neutronic interaction of more than four DOE SNF casks/canisters by referencing criticality calculations (BSC, 2008cm). DOE stated that 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 screened out by relying on a combination of human actions and design solutions, as shown in SAR Table 1.7-1.

Boron dilution was screened out in the HAZOP evaluation and MLDs as an initiating event because boron dilution, in the absence of other independent initiating events, does not initiate a sequence of events that could potentially lead to a criticality, as outlined in DOE Enclosure 5, Section 1.2.1 (2009dy). The only water source DOE considered in the boron dilution initiating event screening was the deionized water system because it was the only source of unborated water available during normal operations. Other sources of water would be available only during event sequences and were considered pivotal events within those sequences, not initiating events, as discussed in DOE Enclosure 5, Section 1.1 (2009dy). In DOE Enclosure 5, Section 1.3.3 (2009dy), DOE stated that procured soluble boron would be accompanied by the necessary material data sheets and each shipment would be tested upon receipt to check its enrichment.

DOE screened out boron dilution/moderator introduction in the HAZOP and MLD evaluations of the DPC fill water, because the boron dilution initiating event was screened out (SAR Table 1.7-1). The DPC fill water was drawn from the pool water via the borated water treatment system. Other water sources cannot be connected to the fill water piping. Therefore, the only source of water available for DPC fill operations is the pool borated water, as described in DOE Section 1.2.4 (2009dy).

NRC Staff Evaluation: The NRC staff conducted its review by evaluating the information DOE provided in SAR Sections 1.6, 1.7, and 1.14 and BSC (2008ba,bq) to assess the criticality-related hazards at the repository facilities during the preclosure period, using the guidance in the YMRP. In addition, the NRC staff reviewed supporting documents (BSC, 2008ai,bj,bo,cm) and DOE’s responses to NRC staff RAIs (DOE, 2009dy,ey).

The NRC staff evaluated the rationale DOE used to assess the criticality-related hazards originated outside the GROA. The NRC staff considered the main type of external event that could impact preclosure criticality to be an error resulting in an absence of neutron absorbers. DOE’s technical basis for screening out neutron absorber manufacturing errors is reasonable because the decision is supported by criticality analyses, which showed that lack of neutron absorber plates would not cause a criticality if the pool boron concentration and enrichment is maintained, as detailed in BSC Section 6.3.4 (2008cm). Screening out boron underenrichment by testing boron when it is received is reasonable because it provides a way to check the material data sheets to ensure that the boron is of the correct enrichment.

The NRC staff examined the rationale DOE used to assess criticality-related initiating events at the IHF and the interaction among more than two naval canisters in the IHF, which was screened out from further consideration in the PCSA by crediting the design of the mechanical handling capabilities of the IHF. The NRC staff notes that this approach is reasonable because the handling equipment within the IHF physically precluded 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 screening out criticality, resulting from the interaction of more than four DOE SNF canisters at the CRCF, as initiating events. The DOE determination is reasonable because it was supported by criticality analyses (discussed in SAR Section 1.14.2.3.2.3.4), which showed that a criticality event would only occur for unlikely configurations (most reactive canisters, most reactive reflectors, and close-packed configuration). The NRC staff notes that crediting physical prevention of critical configurations and human actions is in accordance with standard industry practice and, therefore, is reasonable.

The NRC staff examined the rationale DOE used to screen out the boron-dilution-related criticality initiating events at the WHF. DOE developed PSC–9 to ensure sufficient concentrations of enriched boron in the pool. DOE stated that this control provided 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). Due to the PSC-9 requirements, the required boron concentration fraction needed to maintain subcriticality is less than 15 to 53 percent of the boron normally available in the pool, as discussed in DOE Enclosure 5, Section 1.2.2-5 (2009dy). DOE stated that even if enough unborated water were added to fill the pool to the brim, the concentration fraction would remain above 91 percent, as described in DOE Enclosure 5, Section 1.2.2 (2009dy). DOE 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 Enclosure 5, Section 1.1 (2009dy). Neither of these systems would be connected directly to the pool or pool piping, and the only flow path is through runoff into the pool. Therefore, once the pool is full, water flow would follow the path of least resistance away from the pool. DOE’s basis for screening out boron dilution is reasonable because DOE reasonably considered the current design of the facility in screening this initiating event and, even when the pool is filled with nonborated water, increasing the concentration fraction to 91 percent, the dilution is still not sufficient to allow a criticality event in the WHF pool.

DOE reasonably screened out the introduction of unborated water into a DPC because physical controls were used to prevent the fill water piping from being connected to unborated water sources and because boron dilution was screened out. Additionally, the fuel burnup and fixed absorber panels in the DPCs were not credited.

2.1.1.3.4 NRC Staff Conclusions

The NRC staff notes that DOE’s identification, screening, and quantification of external initiating events are consistent with the guidance in the YMRP. The NRC staff also notes that DOE reasonably identified, screened, and grouped internal initiating events and DOE reasonably quantified these grouped initiating events as discussed in this chapter.

DOE (2008ah, 2009dx) stated that it will (i) develop controls to restrict maneuvering and other activities while transiting the flight-restricted air space (TER Section 2.1.1.3.3.1.3.3); (ii) add an interlock function to an existing interlock on the canister transfer machine to reduce the probability of an operator error (TER Section 2.1.1.3.3.2.2.2); and (iii) review direct exposure event sequences (TER Section 2.1.1.3.3.2.2.2). As part of the detailed design process, DOE should confirm that its human reliability analyses (e.g., task analyses) identified potential vulnerabilities for the repository facilities and associated activities (TER Section 2.1.1.3.3.2.2.2).

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CHAPTER 4

2.1.1.4 Identification of Event Sequences

2.1.1.4.1 Introduction

This chapter discusses the U.S. Nuclear Regulatory Commission (NRC) staff's review of information on identification of event sequences the U.S. Department of Energy (DOE) provided for preclosure safety analysis (PCSA). The objective of the review is to assess DOE's technical basis for developing, quantifying, and categorizing event sequences used in 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 DOE provided in response to the NRC staff's requests for additional information (RAIs) (DOE, 2009bl,dq,dx,dz,ed,ej,fg,fk,fl,fr,ft–fz,ga–gi).

The evaluation presented in this chapter considers information reviewed in other Technical Evaluation Report (TER) chapters: site description in TER Section 2.1.1.1; the description of structures, systems, and components (SSCs), operational process, and throughput analysis in TER Section 2.1.1.2; identification of hazards and initiating events in TER Section 2.1.1.3; and design of SSCs ITS in TER Section 2.1.1.7. The output from this chapter includes event sequences and their associated categorizations that will be used in TER Section 2.1.1.5.

2.1.1.4.2 Evaluation Criteria

The regulatory requirements for identifying and categorizing event sequences for the preclosure period are in 10 CFR 63.21(c)(5), 63.111(c), and 63.112(b). 10 CFR 63.21(c)(5) requires a PCSA of the geologic repository operations area (GROA) for the period before permanent closure. For the purposes of this analysis, it is assumed that GROA operations will be carried out at the maximum capacity and rate of receipt of radioactive waste. 10 CFR 63.112(b) requires an identification and systematic analysis of naturally occurring and human-induced hazards at the GROA, including a comprehensive identification of potential event sequences. An event sequence, as defined in 10 CFR 63.2, includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. As defined in 10 CFR 63.2, Category 1 event sequences are those that are expected to occur one or more times before the permanent closure of the GROA and Category 2 event sequences are those that have at least 1 chance in 10,000 of occurring before permanent closure.

The NRC staff reviewed DOE's identification of event sequences and the categorization using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The relevant acceptance criteria are (i) providing adequate technical basis and justification for the methodology used and assumptions made to identify event sequences and (ii) adequately identifying Categories 1 and 2 event sequences. The NRC staff also used additional guidance such as NRC standard review plans, interim staff guidance (ISG), regulatory guides, and codes and standards when applicable. These additional guidance documents are discussed in the relevant sections that follow.

2.1.1.4.3 Technical Evaluation

The NRC staff's review focused on evaluating technical bases and justification for methods selected, assumptions made, and site-specific data DOE used 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.

DOE's identification of event sequences stems from the identification of naturally occurring and human-induced external and internal hazards (SAR Figure 1.7-1). DOE developed a list of internal and external events in SAR Section 1.6. The NRC staff evaluates, in TER Section 2.1.1.3, DOE's screening of hazards and initiating events and the associated frequency of occurrences for event sequence analyses. As discussed in TER Section 2.1.1.3, the NRC staff performed a risk-informed review concentrating on a subset of initiating events selected on the basis of risk potential, proximity of frequency to the categorization limits, and experience with facilities of similar operations. Evaluation of event sequence development, quantification, and categorization in this chapter relies on the reasonableness of screening and frequency of occurrence of initiating events reviewed in TER Section 2.1.1.3.

On the basis of the initiating events identified for event sequence analysis, the review in this chapter 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) the methodology for event sequence development and categorization, (ii) event sequence development, (iii) reliability of SSCs, and (iv) event sequence quantification and categorization.

2.1.1.4.3.1 Methodology for Development and Characterization of Event Sequences

The methodology DOE used to develop and categorize event sequences in the PCSA is evaluated in this section. DOE summarized the methodology for analyzing event sequences for the surface, intrasite, and subsurface facilities in SAR Section 1.7.1.

Consistent with the guidance in the YMRP, HLWRS–ISG–01 (NRC, 2006ad), and HLWRS–ISG–02 (NRC, 2007ab), the NRC staff evaluated DOE's methods for event sequence identification. The NRC staff reviewed DOE's event sequence development analyses reports (BSC, 2008ab,ao,at,bd,bj,bo), and event sequence reliability and categorization documents (BSC, 2008ac,as,au,be,bk,bq), which contained supporting information on event sequence development methodology and calculations for the various facilities. The NRC staff considers the methodology appropriate and consistent with the site and facility design and operations if (i) the overall approach is reasonable, (ii) methods for developing event sequences are reasonable, (iii) methods for modeling event sequences are reasonable, (iv) the model reasonably represents events sequences, and (v) the methodology for categorization for event sequences is reasonable. The review is structured into the four following subsections: methodology for internal events (TER Section 2.1.1.4.3.1.1), seismically initiated events (TER Section 2.1.1.4.3.1.2), fire-initiated events within the GROA (TER Section 2.1.1.4.3.1.3), and event sequence categorization methodology (TER Section 2.1.1.4.3.1.4).

2.1.1.4.3.1.1 Internal Events

In SAR Section 1.7.1, DOE described the methodology for development of event sequences for internal events initiated by random failure of equipment and human errors during preclosure operations. The methodology for development of event sequences using event sequence diagrams (ESDs) and models was illustrated in SAR Figures 1.7-2 to 1.7-5. These figures collectively showed DOE's approach to considering preclosure-operations-related initiating events and their progression leading to potential consequences or end states. DOE developed ESDs or block flow diagrams 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 potential outcome or end states of each event sequence in an ESD is associated with potential radiological consequences, such as filtered and unfiltered radiological release to public, direct exposure to workers, and important to criticality. DOE developed internal event ESDs specific to each facility and operations.

For quantitative analysis, ESDs were modeled using event trees consisting of an Initiator Event Tree and System-Response Event Tree as shown in SAR Figures 1.7-4 to 1.7-5. DOE performed event tree analyses for each type of waste form configuration related to an ESD. Event sequences were developed for each waste containers [e.g., dual purpose canister (DPC); transportation, aging, and disposal (TAD) canisters; aging overpack (AO)] in each of the four types of handling facilities [Canister Receipt and Closure Facility (CRCF), Initial Handling Facility (IHF), Receipt Facility (RF), and Wet Handling Facility (WHF)] and in intrasite and subsurface facilities. In addition, at the WHF, event sequences were analyzed for transfer of spent nuclear fuel (SNF) assemblies. Thus, all internal event sequences were quantified for a specific waste form configuration. Initiator event trees (SAR Figure 1.7-4) modeled a group of initiating events associated with the ESD and accounted for the number of operations over the preclosure period associated with the waste form. The progression of initiating events was delineated in the system response event trees (SAR Figure 1.7-5), which modeled the failure and success of the containment, shielding, confinement, and criticality control functions of the SSCs in the appropriate pivotal events as a response to the initiating events. Each branch of the system response tree represents an event sequence terminating into a definite end state.

For quantification of internal event sequences, DOE modeled the initiator event trees and system response event trees using the SAPHIRE computer software (Version 7.26 for nonseismic event sequences and Version 7.27 for seismic event sequences). The reliability of active systems was modeled using the fault tree approach (SAR Figure 1.7-8). The fault trees were also used to model initiating events, and all fault trees were linked to event trees at the pivotal nodes in the SAPHIRE models. For passive systems, engineering calculations were performed to estimate the passive reliability and used as input to the pivotal events.

NRC Staff Evaluation: The NRC staff reviewed the methodology for event sequence development for internal events using the guidance in the YMRP. The NRC staff evaluated DOE's overall approach to develop event sequences. DOE's overall methodology for internal events is reasonable because the approach included one or more initiating events and associated combinations of repository system component failures that could potentially lead to exposure of individuals to radiation, consistent with event sequences.

The NRC staff reviewed the methods for developing internal event sequences and notes that each ESD is associated with a specific hazard (e.g., structural and mechanical challenges to the waste forms or direct exposure resulting from handling the waste form during specific operations in the surface and subsurface facilities). The methods for developing internal event sequences using ESD, including the grouping of initiating events, are reasonable because the approaches are consistent with the standard practices in probabilistic risk analysis (PRA) for nuclear power plants (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa).

The NRC staff evaluated DOE's methods for modeling and quantifying internal event sequences. The NRC staff determined that DOE's application of methods for quantifying the likelihood of individual event sequences using the event tree analysis and modeling system reliability using the fault tree analysis is reasonable because DOE used approaches in the PRA that are consistent with the NRC guidance and industry practice (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa; NRC, 2007ab; American Society of Mechanical Engineers, 2005ad) and integrated safety analysis for chemical and other industries (American Institute of Chemical Engineers, 1992aa). In addition, the NRC staff notes that DOE's use of SAPHIRE software for event tree and fault tree analyses and event sequence quantification is reasonable because SAPHIRE is widely used and accepted within the nuclear industry.

2.1.1.4.3.1.2 Seismic Events

DOE provided information on the methods used to identify seismically initiated event sequences in SAR Sections 1.7.1.4 and 1.7.2.4 and supporting documentation (BSC, 2008bg). DOE screened in seismicity as a credible natural hazard for PCSA. DOE used a four-stage approach that includes (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. DOE evaluated potential seismically induced initiating events and analyzed event progression by assessing the subsequent failure or success of preventive or mitigative features that could lead to radiological dose consequences to public or workers. A seismically induced event sequence initiated failure of individual SSCs. The 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. DOE developed site-specific seismic hazard curves for the surface and subsurface repository block on the basis of probabilistic hazard analysis. 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. DOE summarized, in SAR Section 1.7.2.4, the methodology used to develop mean fragility curves, which represent probability of failure or unacceptable performance of a SSC as a function of peak horizontal ground acceleration [this was shown in SAR Figure 1.7-9 for the canister transfer machine (CTM)]. The mean fragility curves are represented by a lognormal probability distribution controlled by two parameters: (i) median and (ii) logarithmic standard deviation as a measure of dispersion or uncertainty. DOE developed fragility parameters for facility structures, as shown in BSC Table 6.2-1 (2008bg), and mechanical systems and equipment, as shown in BSC Table 6.2-2 (2008bg); the failure of these structures, systems, and equipment potentially could initiate event sequences. By convolving fragility and seismic hazard curves, DOE evaluated the mean annual probability of failure of SSCs, which is the initiating event frequency.

DOE 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 in each of the four types of handling facilities, and intrasite and subsurface facilities. The event sequences, in DOE's analysis, elicited pivotal events similar to the internal random initiating events. The event trees consisted 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. DOE did not credit confinement of the heating, ventilation, and air conditioning (HVAC) system in the seismic event sequence analysis. DOE used the residence time factor or total exposure time of the waste form (expressed in years) and the total number of waste forms handled during the preclosure period to obtain the expected number of event sequences. The exposure or residence time is the time the waste form is involved in waste handling operation with specific equipment.

NRC Staff Evaluation: The NRC staff reviewed DOE's methodology for seismically induced event sequences using the guidance in the YMRP. DOE's methodology for developing seismically initiated event sequences is reasonable because the overall approach is consistent with the guidance in HLWRS–ISG–01 (NRC, 2006ad).

DOE's approach to develop and quantify event sequences is reasonable because (i) DOE's assumption of lognormal distribution to define the mean fragility curve for SSCs is a standard practice in seismic PRA (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa); (ii) 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); and (iii) 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. In addition, DOE's use of the exposure time for evaluating the expected number of occurrences is reasonable because the impact to the waste form is conditional to the exposure to a specific hazard (e.g., CTM collapse on TAD canister).

2.1.1.4.3.1.3 Fire Events

DOE discussed fire initiated event sequences in SAR Section 1.7.1.2.2. DOE coupled the fire hazards analyses (BSC, 2007ab,aw,bb,bf; 2008ae,ai,ap,bp) and event sequence development documents (BSC, 2008ab,ao,at,bd,bj,bo) with fire-related reliability analyses to quantify event sequences (BSC, 2008ac,as,au,be,bk,bq).

DOE developed the fire-initiated event sequences for the CRCF, IHF, RF, WHF, intrasite, low-level radioactive waste (LLW), and subsurface facilities on the basis of exposure to each potential waste form (e.g., DPC, TAD, AO) in that facility. DOE identified areas in each facility where fires could play a role in either directly exposing a waste form or affecting an SSC. DOE developed initiating event probabilities for local fires that impact particular waste forms 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) and also developed a large fire scenario to capture an event sequence that assumes a substantial fire propagates through a facility and impacts waste forms in any location within the facility. These initiating event frequencies were provided on a per waste form or package basis, and were evaluated in TER Section 2.1.1.3.3.2.3.4.3.

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. This event considers the potential of a breached canister as a result of a fire. The derivation of this pivotal event probability involved an analysis of canister reliability under fire conditions as discussed in TER Section 2.1.1.4.3.3.1.2.3. The probability for the loss of shielding pivotal event was also consistent for all response trees and was derived using heat transfer and basic material properties data of the shield materials. The moderator pivotal event was common to response trees and was used to discern between radionuclide releases with or without criticality potential. The moderator intrusion probability was derived from the reliability of systems to contain moderators (e.g., sprinkler system water and mechanical system lubricating oil). These events are discussed in TER Section 2.1.1.4.3.3.2.2. Event sequences taking place in buildings with HVAC confinement capabilities included an additional "confinement" probability in their event sequence development.

DOE used SAPHIRE to model the event sequences. ESDs were compiled for each facility and included applicable throughput data, initiating event probability, and pivotal event probability data. The SAPHIRE model for each facility was divided into ESDs pertaining to individual waste forms and, according to DOE, incorporated the appropriate initiating event and pivotal event probabilities for that particular waste form and facility.

Because DOE's fire-related event sequence development methodology was based on a per unit probability, DOE used throughput data in SAR Table 1.7-5, BSC (2008ac,as,au,be,bk,bq), and DOE (2009ga) to convert event frequencies into a total number of occurrences related to that waste form. DOE quantified each event sequence outcome using the methodology outlined in BSC (2008ac,as,au,be,bk,bq).

NRC Staff Evaluation: The NRC staff reviewed the methodology for developing fire-initiated event sequences using the guidance in the YMRP and notes that the overall methodology DOE used is reasonable because this methodology is consistent with industry-recognized PRAs. DOE employed standard PRA techniques utilizing event tree and fault tree analyses. DOE selected pivotal events that accurately reflected the intended safety functions of the SSCs on the basis of descriptions of their designs. The NRC staff further notes that the methods used for developing event sequences are reasonable because they are based on a rational analysis of the potential fire-related failure mechanisms that could play a role in exposure consequences.

DOE's use of SAPHIRE to propagate initiating event frequencies through the established pivotal events and derive final event sequence frequencies is reasonable because it reflects basic PRA practices. The NRC staff reviewed the models established in SAPHIRE and notes that the models reasonably represented the established fire-related event sequences.

2.1.1.4.3.1.4 Event Sequence Categorization Methodology

DOE described its overall approach to event sequence categorization in SAR Section 1.7.5. For internal, seismic, and fire events, DOE quantified each event sequence in an ESD by calculating an expected number of occurrences. DOE categorized event sequences into three bins on the basis of quantitative thresholds for the expected number of occurrences over the preclosure period as follows: (i) Category 1—number of occurrences equal to one or more during the preclosure period; (ii) Category 2—an expected number of occurrences during the preclosure period of less than one but greater than 10^{-4}; and (iii) Beyond Category 2—an expected number of occurrences of less than 10^{-4} during the preclosure period.

NRC Staff Evaluation: The NRC staff reviewed DOE's methodology for categorizing event sequences using the guidance in the YMRP and notes that DOE's approach to categorize

Category 1 event sequences is consistent with the event sequence category definition. For Category 2 event sequences, DOE derived the threshold value of the expected number of occurrences using a Poisson probability distribution where the probability limit of 1 chance in 10,000 (i.e., 10^{-4}) is used as an input parameter. DOE's approach for using a Poisson distribution to calculate threshold values of expected number of occurrences of the event sequences over the preclosure period is reasonable because the Poisson distribution reasonably relates random event probabilities to the expected number of occurrences within a specified time period. DOE applied the Poisson distribution in a manner consistent with the probability of Category 2 event sequences.

2.1.1.4.3.2 Event Sequences Development

DOE discussed development, quantification, and categorization of event sequences initiated by naturally occurring and human-induced hazards. This section documents the NRC staff's evaluation of DOE's technical basis for event sequence development and modeling. The scope of this section includes a review of the reasonableness of event sequence development for operational, seismic, and fire hazards at the surface, subsurface, and intrasite facilities. The NRC staff's review in this section is divided into three main sections: TER Sections 2.1.1.4.3.2.1, Internal Events; 2.1.1.4.3.2.2, Seismic Events; and 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 TER Section 2.1.1.4.3.1, the NRC staff's review focused on whether (i) initiating events were appropriately included in ESDs, (ii) initiating events and pivotal events in the model were consistent with the design and operations, (iii) safety functions of the SSCs relied on to prevent or mitigate exposure were clearly identified, and (iv) end states were consistent with the success or failure of the SSCs safety functions.

2.1.1.4.3.2.1 Internal Events

DOE 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 DOE's event sequence development.

The review of internal events is segregated by similarity in operations and facility. The review of event sequence development for the canister and cask handling operations in the CRCF, WHF, IHF, and RF and during intrasite operations is discussed in TER Section 2.1.1.4.3.2.1.1. The review of wet handling operations is discussed in TER Section 2.1.1.4.3.2.1.2 and handling of waste packages during subsurface operations is evaluated in TER 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

DOE discussed developing 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 in BSC (2008ab,ao,at,bd,bo). The ESDs form the basis for the specific event trees delineated in Attachment F, while BSC Table G–2 (2008ab,ao,at,bd,bo) summarized the relationship among the ESDs, the initiator event trees, and the system response trees.

At the CRCF, IHF, RF, and WHF, DOE identified initiating events mainly related to (i) structural challenges to various waste form configurations causing radiological consequences to the public and workers and (ii) temporary loss of shielding causing direct exposure to the workers. DOE evaluated event sequences for structural challenges to (i) the transportation cask with waste canisters and SNF assemblies during receipt operations inside the facilities that were loaded onto the transfer trolley and transferred to the cask unloading room at the surface facilities; (ii) the AO loaded with the TAD canister or DPC during closure, transferred to the unloading room, and exported from the CRCF; (iii) bare waste canisters [TAD, high-level radioactive waste (HLW), DPC, naval, U.S. Department of Energy (DOE) standardized] during transfer operations with CTM; and (iv) waste packages during transfer, closure, and loading onto the Transport and Emplacement Vehicle (TEV). The direct exposure to workers was associated with cask preparation activities and CTM activities inside the canister transfer room. The initiating events associated with intrasite operations involved structural challenges to the transportation cask, AO, and shielded transfer cask (STC) during transport, and placement and retrieval activities at the aging facility.

In further developing event sequences, DOE modeled a group of initiating events, referred to as a “small bubble” (BSC, 2008ab), in an ESD as an initiator event tree and the progression of the event sequences by system response tree. DOE further evaluated event sequence frequencies and the associated end states for each waste form container handled in the facility. For structural challenges to waste form containers, the pivotal events in the system response tree consistently addressed the success/failure of SSCs relied on to provide containment, shielding, confinement, and moderator control functions to prevent or mitigate event sequences. The direct exposure resulted directly from the initiating events, and no response tree was involved with these event sequences.

The initiator event trees and response trees for each ESD in BSC Attachment F (2008ab, ao,at,bd,bo) were shown in BSC Attachment G (2008ab,ao,at,bd,bo). Each branch of the initiator event tree consists of a group of initiating events with the probability of occurrence and the throughput or number of waste containers or waste form containers with a structural challenge or direct exposure event. DOE developed six categories of system response trees for structural challenges in CRCF: RESPONSE–TCASK1, RESPONSE–TCASK2, RESPONSE–AO1, RESPONSE–CANISTER 1, RESPONSE–WP1, and RESPONSE–WP2. The structural challenges to the transportation cask with canisters inside were modeled in RESPONSE–TCASK1 before unbolting and RESPONSE–TCASK2 after unbolting of the cask. In RESPONSE–TCASK1, the pivotal events consisted of shielding functions for the transportation cask, containment of the transportation cask with canisters inside, confinement from the HVAC, and moderator control. In RESPONSE–TCASK2, the containment function was credited to canisters. For AO (system response tree RESPONSE–AO1), only the waste canisters were assumed to provide containment. The initiating events associated with structural challenges to canisters in the transfer room linked to the response tree RESPONSE–CANISTER1, which examined the reliability of the canisters for containment, shield bell for the shielding, the HVAC for confinement, and moderator exclusion for criticality control. The event sequences for structural challenges to the waste package after closure were analyzed using system response tree RESPONSE–WP2, which examined the containment capability of the waste package and the canister inside. Before waste package closure, the system response tree RESPONSE–WP1 was used to analyze structural challenges to waste packages in which only containment of the canister was modeled. Response trees in other facilities during structural challenges to casks, canisters, and waste packages are similar; however, to deal with structural challenges to the transportation cask with bare SNF assemblies in the WHF, DOE credited the containment function to the transportation cask, as indicated in the system response tree RESPONSE–TCASK–CSNF. DOE modeled the initiating events as 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 and failure branch in the response tree, DOE determined that the outcome of event sequences is radiological consequences. The resulting end state consisted of direct exposure from degradation or loss of shielding, filtered radiological release, unfiltered radiological release, filtered and unfiltered radiological release important to criticality, or a safe state with no radiological consequence.

NRC Staff Evaluation: The NRC staff reviewed information on event sequence development for internal events during operation of canister and casks in the CRCF, IHF, RF, and WHF and intrasite facilities using the guidance in the YMRP and HLWRS–ISG–02 (NRC, 2007ab). DOE modeled the event sequence in a two-step approach. In the first step, an initiator event tree was developed representing multiple initiating events associated with an ESD. In the second step, each branch was further expanded in a systems response tree where the success/failure of the preventive or mitigative systems was modeled as a pivotal event. DOE reasonably included initiating events in ESDs, because grouping of initiating events shown in an ESD in BSC Attachment F (2008ab,ao,at,bd,bo) for all the facilities is consistent with the initiator event trees shown in Attachment G of the same documents. The NRC staff verified that DOE reasonably included initiating events in the SAPHIRE models to quantify the event sequence probability because DOE considered relevant hazards and initiating events for identification of event sequences.

The NRC staff notes that the number of waste form configurations (e.g., casks, canisters, and waste packages) at the surface and underground facilities and SNF at the WHF used in the initiating event trees is consistent with the throughput numbers in SAR Table 1.7-5. DOE indicated that the SAR did not support quantification of event sequences involving multicanister overpacks (DOE, 2009bl). Therefore, the NRC staff did not evaluate event sequences associated with handling of multicanister overpacks.

The NRC staff evaluated the transfer of branches from the initiator event trees to response trees and notes that the pivotal events are reasonable because the models are consistent with facility design and operations. In addition, the safety functions of the SSCs relied on to prevent or mitigate radiological exposure in the pivotal events were identified in the event sequence development. For example, DOE relied on canisters (TAD, HLW, DOE standardized, naval, and waste package) for containment functions, the transportation cask, AO, and STC for shielding functions, and the HVAC system for confinement functions.

The NRC staff evaluated whether end states are consistent with the success or failure of the safety functions of the SSCs. The NRC staff does not distinguish between the end states for filtered and unfiltered consequences in the event of a criticality. The NRC staff used the end state related to importance to criticality with no failure of containment to evaluate DOE's event sequence categorization. DOE reasonably postulated end results for event sequences because the end states are consistent with the success or failure of the safety functions of the SSCs relied on to prevent or mitigate event sequences.

The NRC staff notes that the ESDs DOE developed for the CRCF, RF, IHF, and WHF and for the intrasite operations represented the event sequences. The NRC staff further reviewed the implementation of the ESDs in SAPHIRE software in BSC Attachment H (2008ac) and notes that DOE included the initiating events (small bubbles) (BSC, 2008ab) as a group in the initiator event trees. DOE modeled the linkage between the initiator event tree and response tree using linking rules in SAPHIRE and the linking rule connects the fault trees with the pivotal event. In addition, the linking rules and the end states are consistent with the representation of the event trees documented in BSC Attachment G (2008ab). The NRC staff also notes that DOE aggregated the frequencies to categorize event sequences using partition rules in the SAPHIRE models.

2.1.1.4.3.2.1.2 Wet Handling Operations

DOE discussed event sequence development for the wet handling operations in the WHF in SAR Section 1.7.5.4, in its event sequence development analysis (BSC, 2008bo), and in its reliability and event sequence categorization analysis (BSC, 2008bq). In addition, DOE included ESDs in BSC Attachment F (2008bo) and included event trees in BSC Attachment G (2008bo) and Attachment A (2008bq). It cross-referenced ESDs to event trees in BSC Table G–1 (2008bo).

The WHF is the only surface facility that handles uncanistered SNF. Wet handling operations in this facility involve the transfer of casks containing uncanistered SNF to and from the WHF pool, the transfer of commercial SNF (CSNF) assemblies in the pool, transportation cask preparation activities (e.g., sampling, filling), DPC cutting activities, and TAD canister closure activities.

DOE included the event sequence development for pool activities associated with the transfer of fuel assemblies in the ESD WHF–ESD–22, as shown in BSC Figure F–22 (2008bo). In this ESD, DOE described initiating event WHF–1809 as a fuel drop and initiating event WHF–1808 as the drop of a heavy load onto a staging rack or a TAD canister. DOE stated that these initiating events were intended to allow consideration of drops onto the racks and other drops not onto the racks (DOE, 2009gc). DOE further stated (DOE, 2009gc) that the drop of a heavy-load-initiating event was screened out because the spent fuel transfer machine (SFTM) only handles CSNF; DOE included drops from the SFTM with the fuel-drop-initiating event.

The event tree WHF–ESD20–CSNF, as shown in BSC Figure G–32 (2008bo), involved the structural challenge to a transportation cask (e.g., the drop of a cask) containing CSNF during its transfer from the preparation station to the pool ledge. The event tree WHF–ESD24–TAD, as outlined in BSC Figure G–39 (2008bo), involved the structural challenge to an STC containing an unsealed TAD canister during its transfer from the pool ledge to the TAD canister closure station. For both event trees, DOE credited the cask for maintaining containment if an event such as a drop or tipover occurs, and DOE identified that a cask could drop outside of the pool (i.e., onto the floor) or over the pool. In its response to an NRC staff RAI (DOE, 2009fk), DOE specified Procedural Safety Control–6 requiring a minimum number of installed fasteners on the casks and DOE described analyses it performed to determine the minimum number of installed fasteners.

For operations involving the drop of an object onto a cask in the pool, DOE responded to an NRC staff RAI (DOE, 2009gc) describing how it accounted for the drop of heavy loads in six event trees involving the movement of casks to and from the pool and the movement of casks between the pool ledge and the bottom of the pool. As shown in DOE Table 1 (2009gc), for the movement of a transportation cask containing CSNF from the pool to the bottom of the pool (i.e., WHF–ESD21–CSNF), DOE accounted for two potential object drops, and one of these involved the cask handling yoke. For WHF–ESD20–CSNF involving the movement of the cask from the preparation station to the pool ledge and WHF–ESD24–TAD involving the movement of a cask from the pool ledge to the TAD canister closure station, DOE identified no heavy load drops onto a cask in the pool. However, BSC Sections 6.1.2.20 and 6.1.2.22 (2008bo) specified the use of a yoke in these operations as well. In addition, DOE described a heavy load drop from the jib crane in its response to an RAI (DOE, 2009gc) for WHF–ESD19–DPC. However, BSC Figure F–19 (2008bo) showed initiating event WHF–709 involving the drop of a heavy load onto a cask, and the description for this initiating event in BSC Table 10 (2008bo) identified the Cask Handling Crane—not the jib crane. Similarly for WHF–ESD20–CSNF, initiating event WHF–705 referred to the cask handling crane, as shown in BSC Table 10 (2008bo), and not a jib crane, as indicated in DOE's response (DOE, 2009gc).

For event sequences involving direct exposure during pool operations, DOE included lifting a fuel assembly too high, exposure from the splash of pool water, and improper decontamination of empty transportation casks or DPCs, as shown in BSC Figure F–30 (2008bo). DOE showed in BSC Table 6.0-2 (2008bq) that improper decontamination was screened out as an off-normal event. In addition, in response to an NRC staff RAI (DOE, 2009fk), DOE stated that the splash of pool water was screened out as an off-normal event and used HLWRS–ISG–03 (NRC, 2007ac) as its basis.

For event trees involving structural challenges to casks when transferring them to or from the pool, as shown in BSC Figures G–32 and G–39 (2008bo), DOE used different response trees depending on whether the event (e.g., drop) occurs over the pool. For a drop over the pool, DOE considered an unfiltered radionuclide release of gases including those important to criticality. For a drop over the floor, DOE 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 surface nuclear confinement HVAC system. DOE identified in BSC Section 6.3.2.5 (2008bq) that, for containers having both containment and shielding functions, containment failure was considered to result in a concurrent loss of shielding. DOE included the direct exposure from the shielding loss end state in BSC Figure G–9 (2008bo) for an STC being transferred from the pool. DOE also included direct exposure from shielding degradation for the case when containment is not lost for a transportation cask being transferred to the pool and an STC being transferred from the pool in BSC Figures G–3 and G–9, respectively (2008bo).

NRC Staff Evaluation: The NRC staff reviewed DOE's WHF event sequence development using the guidance in the YMRP. The NRC staff used a vertical slice approach to focus on event sequences that were more risk significant or resulted in frequencies close to a categorization boundary. In terms of operations, the NRC staff notes that DOE, in its response to an NRC staff RAI (DOE, 2009fk), reasonably accounted for the configuration of the STC and transportation cask in the containment pivotal event for WHF–ESD20–CSNF and WHF–ESD24–TAD by specifying Procedural Safety Control–6 and the analyses it performed. In addition, on the basis of the review results in TER Section 2.1.1.3.3.2.1 pertaining to the consistency of ESDs with operations, DOE represented initiating and pivotal events consistent with design and operations.

In terms of end states, the NRC staff notes that DOE identified the end state for lifting a fuel assembly too high as a direct exposure and that this end state is consistent with a safety function for the SFTM identified in BSC Table 6.9-1 (2008bq). In addition, the end states involving the drop of a cask during transfer to or from the WHF pool are consistent with the success or failure of SSCs. In particular, the end states are consistent with the success or failure of the surface nuclear confinement HVAC system and the success or failure of the cask to maintain containment and shielding.

2.1.1.4.3.2.1.3 Subsurface Operations

DOE provided information in SAR Section 1.7.5.6 and a supporting document (BSC, 2008bj) regarding the development of potential event sequences that could occur during loading of waste packages onto a TEV in a surface waste handling facility, transport of waste packages to the subsurface facility, and waste emplacement underground, as shown in BSC Figure 6 (2008bj). SAR Table 1.7-17 and BSC Attachment F (2008bj) summarized the event sequences DOE identified that could occur during the operations.

DOE grouped the event sequences as outlined in BSC Attachment F (2008bj): structural challenges to the waste package at the surface facility (while being loaded onto or on board the TEV), in transit to the subsurface, or during or after emplacement underground; potential loss of shielding; and thermal challenges due to fire. DOE considered that event sequences which could challenge the structural integrity of a waste package may arise from mechanical impact from a collision with a shield door, other structure, or equipment; a drop or dragging of a waste package; or TEV derailment. DOE also considered event sequences that could result in loss of radiation shielding may arise from (i) a violation of an administrative or physical control (such as inadvertent entry into an emplacement drift, proximity to a loaded TEV, or inadvertent opening of a TEV door) or (ii) TEV shielding degradation due to overheating. DOE stated that the TEV shielding may degrade if a layer of polymer material in the shielding overheats; this could occur if a loaded TEV is disabled due to derailment or loss of power.

DOE considered operations needed to install drip shields over waste packages, as shown in BSC Figure 16 (2008bj), toward the end of approximately 100 years of subsurface operations, but did not identify any event sequences associated with drip shield installation. DOE relied on the subsurface structures and systems (e.g., network of underground openings and the invert structures and rails, power distribution infrastructure, and subsurface ventilation) functioning within the serviceability limits needed for subsurface operations through the preclosure period, as described in BSC Section 3 (2008bj). SAR Sections 1.3.3.3.2 and 1.3.4.4.2 stated that DOE will use monitoring and inspection programs to assess the need for maintenance to assure reasonable functionality of the subsurface structures and systems. In an RAI, the NRC staff requested DOE to clarify its approach for preventing or mitigating potential event sequences related to subsurface structures or systems failure, such as failure of the invert structure due to corrosion, thermal expansion, or loss of rock support; collapse of an emplacement drift, exhaust main, or exhaust shaft; loss of operating envelope due to wall convergence; ventilation failure due to blockage of an exhaust conduit, such as ventilation raise or exhaust main or shaft; or 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 this NRC staff's RAI (DOE, 2009ed), DOE stated that it will establish design criteria and bases to ensure stability of the structures and systems and implement a monitoring, inspection, and maintenance program to ensure any deterioration of the structures and systems will be detected and corrected in a timely manner.

NRC Staff Evaluation: The NRC staff reviewed the event sequence development for subsurface operations using the guidance in the YMRP to determine whether DOE reasonably considered potential occurrences that could result in radiation exposure or release of radioactive materials during loading, transport, and emplacement of waste packages; drip shield installation; and other subsurface operations such as waste package inspection. To review DOE's event sequence development, the NRC staff examined how the initiating events identified in BSC Tables 10 and 11 (2008bj), which were evaluated in TER Section 2.1.1.3.3.2.3.4.6, were assigned to event sequences, as outlined in BSC Attachment F

(2008bj). The NRC staff also examined the progression of initiating events in the response event trees to determine whether the initiating events were carried over into the branches of the individual event trees. Also, the NRC staff examined the event sequences in the context of the subsurface operations as described in the process flow diagrams in BSC Figure 15 (2008bj), ESDs in BSC Attachment F (2008bj), and the initiator event tree and response trees provided in BSC Attachment F (200bj). The NRC staff notes that the event sequences were developed and modeled consistent with the methodology because (i) DOE included all initiating events identified from the hazards analysis in the ESDs and in the event sequence quantification, (ii) the initiating events and pivotal events in the model are consistent with the design and operations, (iii) DOE identified the safety functions of the SSCs relied on to prevent or mitigate exposure in the pivotal events, and (iv) the end states are consistent with the success or failure of the SSC safety functions. The NRC staff notes that DOE relied on monitoring, inspection, and maintenance to prevent or mitigate potential event sequences related to subsurface structures or systems failure. On the basis of the NRC staff's review in TER Section 2.1.1.2.3.7.3, DOE's statement to monitor and maintain the performance of the subsurface structures and systems is reasonable to prevent or mitigate subsurface structures or systems failure that could trigger event sequences. Therefore, the event sequences DOE developed for subsurface operations represented potential occurrences that could result in radiation exposure or release of radioactive materials during surface operations.

2.1.1.4.3.2.2 Seismic Events

DOE provided information on the development of seismically induced event sequences for the GROA in BSC (2008bg). DOE developed seismically initiated event sequences for the CRCF, IHF, RF, WHF, and intrasite and subsurface operations. The NRC staff reviewed the information to determine whether DOE presented reasonable combinations of seismically induced initiating events and the associated combinations of repository SSCs failure that could lead to exposure of individuals to radiation.

<u>Collapse of Surface Structures</u>

DOE considered collapse of all surface facility structures as an initiating event potentially causing breach of waste form containment and loss of confinement leading to unfiltered radionuclide release. Thus, DOE did not transfer building collapses to a seismic system response tree.

NRC Staff Evaluation: The NRC staff reviewed the information on event sequences resulting from seismically induced surface building collapse using the guidance in the YMRP. The NRC staff notes that DOE's approach of assuming that collapse of all surface facility structures potentially causes breach of all waste form containment and loss of confinement leading to unfiltered radionuclide release is conservative.

<u>Waste Handling in Waste Handling Buildings</u>

DOE discussed development of seismically induced event sequences during waste handling operations in surface facilities in BSC Section 6.0 (2008bg). During handling of 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 its seismically induced failure modes. The seismic initiator event trees were developed on the basis of operations and waste form. For example, the initiator event tree during transfer of TAD in the AO to the waste package was shown in BSC Figure 6.6-4 (2008bg). DOE typically identified events initiated by seismically induced collapse of mechanical structures (e.g., entry door, shield door, mobile or cask prep platform, welding robot arm) on the waste containers when the waste containers are in the proximity of these structures. The seismically induced initiating events were also identified as resulting from failure of equipment and systems during handling of waste containers similar to internal events. However, the failure modes are conditional to the seismic events instead of occurring randomly. The mechanical handling equipment [e.g., cask and waste package handling cranes, cask and waste package transfer trolleys (WPTTs), CTM, TEV, and site transporter] has several seismic failure modes induced by seismic load and potentially impacting waste containers.

For each initiating event, DOE developed a fault tree model. A typical fault tree in DOE's analysis consisted of the exposure time factor of a structure or equipment and its potential failure modes that contributes to the failure. For example, the initiating event "CTM seismic failure," as shown in BSC Figure C1.1-7 (2008bg), was caused 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 Figure C1.2-4 (2008bg). The failure probability for each failure mode was quantified by convolving the fragility curve defined by the parameters given in BSC Table 6.2.2 (2008bg) and the seismic hazard curve given in BSC Section 6.1 (2008bg). The exposure/residence time factors in DOE's analysis accounted for the amount of time the waste container is exposed to the seismic hazard. DOE's calculation of exposure time was based on waste processing activity with equipment represented as "years per single waste container."

For event sequence analysis, the initiating events in the seismic initiating event tree are transferred to seismic system response trees. Similar to the internal events, a typical seismic response tree, as shown in BSC Figure C1.1-5 (2008bg), 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 into 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 to structural challenges from drop or other impacts. DOE considered HVAC to fail if the seismic event caused breach to the waste canister (BSC, 2008bg), therefore taking no credit for the "Confinement" pivotal event. DOE attributed piping system failure and intrusion of moderator into a breached canister to a criticality end state.

In addition, at the WHF where bare fuel assemblies are handled, DOE 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 WHF) causing unfiltered release. For seismically induced initiating events caused by failure of the SFTM, transfer station, cask handling crane, auxiliary pool crane during cask handling, TAD, and fuel assemblies in the pool, DOE relied on the pool integrity to provide shielding and unfiltered radionuclide release.

NRC Staff Evaluation: The NRC staff reviewed the seismically induced event sequences related to waste handling in the CRCF, IHF, WHF, and RF using the guidance in the YMRP. When the structural integrity of the facility is maintained during a seismic event, the NRC staff notes that the identification of seismically induced initiating events is consistent with the facility design and operations because DOE considered failure of equipment during the handling of waste. In addition, the initiating event includes seismic interaction or seismic failure of mechanical components impacting the waste form containers, commonly known as the two-over-one issues in a seismic PRA for nuclear power plants (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa). DOE addressed the collapse of several nearby mechanical structures on the waste containers during specific operations. Thus, the initiating events are consistent with the facility and SSC design.

For quantification of event sequences, DOE modeled the event sequences using initiator event trees and seismic response trees in a two-step approach. The initiator event tree contains multiple initiating events and the throughput number of the waste form containers. Each initiating event is modeled using a fault tree for evaluating frequency of occurrence. DOE identified the seismically initiating events by considering the seismic failure modes of mechanical systems handling waste form containers and failure of mechanical structures onto the waste form containers. Therefore, the seismic failure modes of the equipment handling waste and those of the SSCs that affect the waste form causing seismic initiating events are reasonable.

DOE used similar seismic system response trees for all the surface facilities associated with handling of the waste container. 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 form containers. DOE relied on the passive reliability of the waste form canisters for containment and transportation and aging casks for shielding. The seismic response trees used in the event sequence analysis are reasonable because DOE considered the pivotal events consistent with the safety functions of SSCs. The end state of the seismic event sequences relates to potential radiological consequences from loss of containment and direct exposure from loss or degradation of shielding. Because the HVAC system containment function is not credited, the loss of containment leads to unfiltered release. The system response tree for wet handling operations at the WHF shows that DOE relied on the availability of the pool for shielding, scrubbing of radioactive release, and boration. The NRC staff notes 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.

Intrasite Operations

The intrasite operations involve movement and storage of AOs containing TAD canisters and horizontal transportation casks containing DPCs at the Aging Facility, storage of LLW in the Low-Level Radioactive 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 are initiated by failure of AO, Horizontal Aging Module structure failure, horizontal transporter and site transporter failures associated with railcar and trucks at the yard and during movement, and LLW building collapse. The initiating events were not transferred to a response tree, and the 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, DOE stated that failure of an earth slope near the aging pad would not result in a credible event sequence, because (i) a slope failure would have no effect on the aging pad structure because of the distance of the pad from adjacent cut or fill slopes and (ii) the frequency of canister failure due to a seismically induced slope failure is beyond Category 2 (DOE, 2009gg). To explain case (i), DOE indicated that the aging pad foundation would be located approximately 22.9 m [75 ft] from the edge of adjacent cut or fill slopes, as shown in DOE

Enclosure 2, Figures 1 and 2 (2009gg). For case (ii), DOE's assessment of the frequency of canister failure included an assumption that the slope design would be stable under a Design Basis Ground Motion (DBGM)–2 earthquake. DOE discussed this assumption on the basis of an analysis provided in DOE (2009ej), and the NRC staff reviewed this assumption in TER Section 2.1.1.1.3.5.4.

NRC Staff Evaluation: The NRC staff reviewed the seismically induced event sequences related to intrasite operations using the guidance in the YMRP. The NRC staff notes that DOE's identification of initiating events in BSC Section 6.0 (2008bg) is reasonable because it is consistent with the facility design and operations and the delineation of end states from these events.

On the basis of TER Sections 2.1.1.1.3.5.4 and 2.1.1.7.3.1.3, the NRC staff notes that the design of earth slopes at the surface facilities would ensure stability of the slopes during a DBGM–2 earthquake. The slopes include cut and fill slopes near the aging pads and along transportation routes (TER Section 2.1.1.1.3.5.4) and side slopes of flood control dikes and channels (TER Section 2.1.1.7.3.1.3). On the basis of the aging pad layout design (DOE, 2009gg), the NRC staff notes 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 slope failure. The NRC staff evaluation regarding stability of cut and fill slopes (TER Section 2.1.1.1.3.5.4) and flood control dikes and channels (TER Section 2.1.1.7.3.1.3) identifies the design criteria that DOE relied on to exclude potential event sequences due to slope failure along a transportation route or flood control dike or channel. Therefore, the NRC staff notes DOE explained that failure of an earth slope near the aging pad would not result in a credible event sequence. Also, on the basis of TER Sections 2.1.1.1.3.5.4 and 2.1.1.7.3.1.1, DOE's design is reasonable to prevent or mitigate potential event sequences due to slope failure along the transportation routes or flood control dikes and channels. Consequently, DOE reasonably considered seismically induced event sequences for intrasite operations.

Subsurface Operations and Other Issues

The seismic event sequences for subsurface operations were presented in BSC Section 6.9 (2008bg). The subsurface operations involved movement of the TEV with a waste package from surface facilities to the subsurface emplacement drift, storage of a waste package until permanent closure, and installation of drip shields before permanent closure. The initiating events included TEV derailment, entry door collapse on the TEV, rockfall on the waste package in the emplacement drift, and drift instability burying the waste package under rock rubble. Other initiating events considered were drip shield and gantry failure on the waste package and impact to the waste package during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's development of event sequences for the subsurface resulting from seismically initiated events using the guidance in the YMRP. The NRC staff notes that for initiating events caused by TEV tipover, DOE relied on the waste package containment function and TEV shielding function in the pivotal events to prevent event sequences, while other initiating events (rockfall and drift instability) lead to unfiltered radionuclide release. Thus, the seismic event sequences for subsurface operations are consistent with the facility design and operations and safety functions of the SSCs relied on for containment and shielding functions. Therefore, DOE reasonably considered seismically induced event sequences for subsurface facilities.

2.1.1.4.3.2.3 Fire Events

DOE described fire-initiated event sequences (SAR Section 1.7), the supporting fire hazards analyses (BSC, 2007ab,aw,bb,bf; 2008ae,ai,ap,bp), and the resulting event sequence development documents (BSC, 2008ab,ao,at,bd,bj,bo). The NRC staff's review assessed whether DOE provided a systematic analysis of naturally occurring and human-induced, fire-related hazards at the GROA, including a comprehensive identification of potential event sequences.

DOE screened out external fire and explosion-related events and focused its analysis on internal fire events. The NRC staff reviewed DOE's screening bases in TER Section 2.1.1.3.3.1.3.5 and notes that DOE's screening bases are reasonable. DOE stated that its administrative controls, such as a vegetation-free buffer zone, controlled vehicle operation and parking, and safe separation distances to potential explosion sources, would prevent significant SSC damage from fire and explosion-related event sequences. Separation distance to a fire or explosion event reduces the impact of incident heat flux (fire) or overpressures (explosion) on an SSC.

DOE began its development of fire-related event sequences with the aggregation of fire-related initiating event probabilities, as discussed in TER Section 2.1.1.3.3.2.3.4.3. DOE based the number of fire-related initiating event trees for a particular event sequence on the number of different types of canisters anticipated at a facility and the number of locations where a particular waste form could be found within a facility. The event sequences were established on a per waste form/per facility basis. These event sequences were given ESD numbers (e.g., ESD–20 for the CRCF, ESD–09 for intrasite) with varying suffixes to represent the waste form (e.g., DPC, TAD) so that they could be reconciled between initiating events and the corresponding event sequences.

DOE propagated the initiating events through response trees to obtain end state probabilities. The response tree diagrams used to develop the fire-related event sequences for each facility were similar to response trees for other internal events. These response trees shared common pivotal events including containment, shielding, confinement, and moderator; however, the development of the fire-related event sequences involved an assignment of pivotal event probabilities on the basis of the individual SSC's response to hypothetical fire events. The determination of SSC reliability under fire conditions and the corresponding pivotal event probability was based on information provided in BSC (2008ac,as,au,be,bk,bq).

DOE 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). For loss of low melting temperature shielding material during a fire in shielding pivotal events and loss of HVAC confinement during a large fire for confinement pivotal events, DOE opted to estimate failure probabilities in the presence of a thermal challenge by assuming an SSC failure probability of 1.0 or success probability of 0.0.

DOE developed the LLWF as a single initiating event that involved all combustible waste at the LLWF. There was one response tree for the LLWF because DOE identified only one initiating event for the entire facility.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's development of event sequences resulting from internal fire events using the guidance in the YMRP. DOE included initiating events and canister throughput data to develop initiating event trees. This information was reviewed in TER Section 2.1.1.3.3.2.3.4.3, where the NRC staff notes that DOE reasonably represented this information in the development of fire-related initiating events. The NRC staff reviewed DOE's development of the corresponding fire-related response trees for event sequence analysis in this TER section.

The NRC staff notes that DOE (i) reasonably developed ESDs because corresponding response trees illustrate the SSCs used to mitigate the event sequences and (ii) included the performance of these SSCs as pivotal events. The NRC staff further notes that DOE identified the safety functions of the SSCs relied on to prevent or mitigate exposures. DOE has identified the pivotal events where SSCs could be assigned conservative basic probabilities (e.g., failure probability of 1.0 and a success probability of 0.0) and has identified the pivotal events where the event probability was driven by a more detailed fault tree analysis.

The NRC staff notes that, although the end states varied on the basis of the facility or the container type that was being modeled, the end states are consistent with the success or failure of SSCs under fire challenges. For example, facilities that included HVAC confinement pivotal events also included filtered radionuclide release and filtered radionuclide releases that may be important to criticality.

2.1.1.4.3.3 Reliability of Structures, Systems, and Components

DOE relied on 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. DOE presented the methodology for estimating the SSC reliability and the role of passive reliability for active systems, passive systems, and seismic fragilities of structural and mechanical systems in SAR Sections 1.7.2.2, 1.7.2.3, and 1.7.2.4, respectively. Additional information on DOE's approach and evaluation of the reliability of SSCs was addressed in BSC (2008ac,as,au,be, bg,bk,bq). The focus of NRC staff's review is to assess whether DOE (i) selected and implemented the methods for estimating the reliability of passive and active systems, (ii) estimated reliability using the data consistent with the design description DOE provided, (iii) estimated reliability based on engineering practices and consistent with design methodologies and analysis, and (iv) addressed uncertainty in the reliability estimate.

The review presented in this section is organized by passive systems (TER Section 2.1.1.4.3.3.1) and active systems (TER Section 2.1.1.4.3.3.2). The passive system 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

DOE's determination of reliability of passive systems can be categorized into two classes: (i) waste containers (TAD canisters, DOE standardized, DPC canisters, HLW canisters, waste package, transportation cask, and AO) subjected to structural and thermal challenges and (ii) seismic fragility of facility structures, and mechanical systems. Structural challenges to a container result from drops and impacts, while thermal challenges to a container arise during fire events in the facility. DOE estimated reliabilities (failure probabilities) of the containers to provide containment and shielding functions and used these failure probabilities as input to containment and shielding pivotal events in the system response event trees for internal and seismic event sequence quantification. DOE estimated seismic fragility of surface structures and equipment to develop seismically initiated events.

The NRC staff's review of passive reliability for structural challenges is presented in TER Section 2.1.1.4.3.3.1.1, seismic fragility for structural and mechanical systems is presented in TER Section 2.1.1.4.3.3.1.2, and thermal challenges is presented in TER Section 2.1.1.4.3.3.1.3.

2.1.1.4.3.3.1.1 Passive Reliability for Structural Challenges Resulting From Internal Events

DOE provided information on reliability of passive SSCs in SAR Section 1.7.2.3. DOE presented the passive equipment failure analyses (PEFA) and summarized the failure probabilities for each container in BSC (2008ac,as,au,be,bq,) for surface facilities (CRCF, RF, IHF, WHF) and intrasite operations. In addition, DOE presented passive reliability of containers used in seismic event sequences in BSC Table 6.3-2 (2008bg) and reliability analysis was discussed in BSC Section 6.3.3 and Attachment H (2008bg).

The containers relied upon to provide containment were the waste packages, TAD canisters, DPCs, and HLW canisters. The containers providing shielding functions included the transportation cask and AO. DOE 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. DOE used the first approach to determine passive reliability of HLW canisters in which statistical analyses was performed on the drop test results. The probability of loss of containment for TAD, DPC, DOE standard canisters and loss or degradation of shielding for transportation casks and AOs was performed by computing the demand from drop or impact on the containers by finite element modeling and evaluating the capacity by the experimental testing of the material. Loss of containment and shielding of these containers subjected to structural challenges during preclosure operations is discussed next.

Loss of Containment

Structural challenges causing potential loss of containment include drop and slapdown of containers, collision of containers with other structures or objects, and drop of objects onto waste containers. In its event sequence analysis, DOE 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), DOE evaluated the probability of failure of HLW canisters for drops from operational and beyond operational height. DOE'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) and 9 m [30 ft] (considered as beyond operational height) 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 DOE 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 canisters made of stainless steel did not show any ruptures or surface cracks.

DOE 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, DOE used a Bayesian approach to estimate failure probabilities separately for the two drop heights. DOE based the Bayesian analysis on a beta-binomial conjugate distribution, which led to a beta posterior failure probability distribution. DOE then used the drop test results to estimate the mean and standard deviation for the beta posterior failure probability distribution. Using this approach, DOE determined that for the 7- and 9-m [23- and 30-ft] drop heights, the mean failure probability posterior distribution was 3.4×10^{-2} and 6.7×10^{-2}, respectively. DOE 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 3×10^{-2} for a drop from the operational height and 7×10^{-2} for a drop from greater than operational height, as shown in BSC Table 6.3-7 (2008ac). DOE estimated the HLW canister failure probability for the case of a 14-m [45-ft] drop through extrapolation using BSC Section 6.3.2.2, Equation 17 (2008ac).

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of HLW canisters using the guidance in the YMRP and notes that the design of the HLW canisters used in the experimental tests is consistent with the design detail descriptions presented in TER Sections 2.1.1.2.3.4.1 and 2.1.1.7.3.9.3.2.

The NRC staff notes that using full-scale experimental drop tests is a reasonable approach to assess HLW canister failure because the use of similar impact tests to evaluate structural integrity of other canisters is well documented in the literature (Morton, et al., 2006aa). The use of drop tests and statistical analysis of the accompanying data are standard methodologies for estimating canister reliability.

DOE's use of helium leak and liquid dye penetrant tests to evaluate HLW canister failure is reasonable because these tests are standard test methods used in the industry. The NRC staff evaluated the drop test results of 27 HLW stainless steel canisters {14 from 7 m [23 ft] and 13 from 9 m [30 ft]} and the results indicated no failure (BSC, 2007de). The NRC staff notes that the experimental test data DOE used to estimate the HLW canister failure reliability are reasonable.

DOE's approach for estimating failure probability using the Bayesian methodology is reasonable because this is an industry-accepted methodology. The NRC staff notes that, consistent with Siu and Kelly (1998aa), DOE computed the estimated mean and standard deviation of the failure probability posterior beta distribution and interpreted the experimental results using a beta-binomial conjugate distribution Bayes analysis. Furthermore, consistent with Siu and Kelly (1998aa), DOE computed the estimated means and standard deviations for the failure probabilities for the HLW canisters dropped from 7 and 9 m [23 and 30 ft], respectively.

The NRC staff reviewed the failure probability values in BSC Table 6.3-7 (2008ac) for several different cases used in the event sequence analyses and notes that DOE used a failure probability of 3×10^{-2} in the event sequence analysis for a canister drop from an operational height {7 m [23 ft]} and 7×10^{-2} for a canister drop greater than operational height {9 m [30 ft]}. Regarding determining the failure probability of the HLW dropped from a 14-m [45-ft] height, DOE's use of BSC Section 6.3.2.2, Equation 17 (2008ac) is reasonable because, as will be discussed further in the section on the Transportation Cask, this equation is based upon the standard engineering principle of energy balance.

Waste Package

In SAR Section 1.7.2.3 and BSC Sections 6.3.2.2 and D1.4 (2008ac), DOE discussed the calculation of the waste package passive reliability.

DOE defined the waste package as a passive component that may fail when it is subjected to loads that exceed its capacity (i.e., strength). Moreover, DOE stated that, because the waste package is designed in accordance with the provisions of American Society of Mechanical Engineers, 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 TER Sections 2.1.1.2.3.5.1 for more details), DOE based the waste package passive reliability only on the capacity of the Alloy 22 outer corrosion barrier.

DOE 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 containment 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. DOE used explicit, nonlinear finite element analyses (i.e., LS–DYNA) to determine the demand on the waste package when subjected to different structural challenges.

From the finite element models, DOE 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, DOE estimated the waste package demand as a wall-averaged expended toughness (BSC, 2007cq).

DOE modeled the capacity of the waste package Alloy 22 outer corrosion barrier using a material toughness and determined the waste package capacity through calculating the material toughness index (BSC, 2007bi,cq). DOE stated (BSC, 2007cq) that vendor-averaged properties were used for the mean strength properties of the Alloy 22 outer corrosion barrier; thus, DOE took into account the variability of the Alloy 22 material properties. Further, DOE 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, DOE 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, DOE 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). DOE assumed that waste package damage occurs for values of $ETF \geq 1$. DOE used *ETF* to compute the probability of containment failure using BSC Section D1.4, Equation D–3 (2008ac). The equation is based on a normal distribution assumption for *ETF* and, for computational purposes, transforming *ETF* to a standardized normal value. DOE summarized the waste package failure probabilities in BSC Tables D1.4-1 and 6.3-7 (2008ac).

DOE provided the waste package failure probability values that were used for event sequence quantifications for different structural challenges in BSC Tables D1.4-1 and 6.3-7 (2008ac). The tables provided waste package failure probabilities for different impact conditions. DOE reported a failure probability of 10^{-5} for the 1.8-m [6-ft] horizontal drop, 9,072-kg [10-T] drop on a container, the 4 km/hour [2.5 mph] end-to-end collision, and the 14.5 km/hour [9 mph] end-to-end collision. For the 4 and 14.5 km/hour [2.5 and 9 mph] flat side impacts, DOE used a failure probability of 10^{-8} in the event sequence analysis.

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of waste packages using the guidance in the YMRP. The NRC staff notes that all waste package configurations consist of an Alloy 22 outer corrosion barrier and a 316 stainless steel inner vessel, and that DOE determined the waste package passive reliability on the basis of the capacity of the Alloy 22 outer corrosion barrier. Use of an Alloy 22 outer corrosion barrier is reasonable because DOE's approach is conservative in that it does not take credit for the waste package inner vessel to provide containment of the waste form if an outer corrosion barrier breach occurs.

The NRC staff reviewed DOE's approach for calculating the demands on the waste package due to structural challenges and notes that using the maximum Von Mises strain and stress in the failure calculations is reasonable because it is the most commonly used stress/strain measurement for ductile materials (e.g., metals). The data utilized in the finite element analyses (BSC, 2007cn,cq,cr) are consistent with the information on design description and design of the waste package and its components reviewed in TER Sections 2.1.1.2.3.5.1 and 2.1.1.7.3.9.1. In these analyses, DOE 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 notes that it is reasonable for DOE to model the capacity of the waste package Alloy 22 outer corrosion barrier using a toughness index because the toughness index is a measure of the material energy absorption capacity. DOE 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 notes that this is reasonable because it follows standard engineering practice (Bathe, 1996aa). In addition, use of *ETF* as a form of damage measure is reasonable because *ETF* is based on the energy-absorbing capacity of Alloy 22, which is a highly ductile material.

The NRC staff reviewed BSC Section D1.4, p. D–21, Equation D–3 (2008ac) to evaluate the probability of failure of waste packages using *ETF*. To calculate the failure probability for Alloy 22, DOE assumed a normal distribution for *ETF* to account for the relative variability of the capacity (i.e., material strength). The NRC staff notes that this assumption is reasonable because the low waste package failure probabilities demonstrate that there is a high likelihood that the waste package would not fail for these types of drops and impacts. The NRC staff also notes that the formula for computing the probability given in BSC Section D1.4, p. D–21, Equation D–3 (2008ac) is reasonable because a basic statistical procedure for computing probabilities for a normal distribution was utilized.

The NRC staff reviewed the waste package failure probability data presented in BSC Table 6.3-7 (2008ac). The NRC staff notes that DOE's use of a higher probability value (10^{-5}) in the event sequence quantification for the 1.8-m [6-ft] drop, 9,072-kg [10-T] 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 actual calculated failure probability of 10^{-8} is conservative. The failure probability of 10^{-8} for the cases of 4 and 14.5 km/hour [2.5 and 9 mph] flat side impact is reasonable because the impact at these velocities, as DOE determined, would correspond to a drop height that is lower than the 1.8-m [6-ft] height and would result in a lower failure probability.

Transportation, Aging, and Disposal Canisters and Dual Purpose Canisters

In SAR Section 1.7.2.3.1 and BSC Sections 6.3.2.2 and D1.1 (2008ac), DOE discussed the passive reliability of the TAD canisters and DPCs. DOE's methodology for estimating the reliability of the representative canister was accomplished by establishing the relationship between demand and capacity defined in terms of strain in the canister material. DOE calculated the demand in terms of the maximum effective plastic strain from each finite element drop simulation analysis. DOE also determined the structural capacity of TAD canisters and DPCs on the basis of tensile elongation at failure data obtained from canister material tests. DOE performed a statistical analysis of the tensile elongation test data 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. Finally, DOE calculated the probability of canister breach by relating the strain obtained from a finite element analysis to the fragility curve.

Although DOE provided performance specifications for the TAD canisters (SAR Section 1.5.1.1.1.2.1.3), no specific TAD canister design information was presented. Therefore, DOE did not directly evaluate the TAD canister's reliability under the structural challenges. Instead, DOE utilized a representative canister to evaluate the failure probability. DOE defined the representative canister on the basis of the available information on existing SNF canisters (i.e., DPCs and the naval canisters), as shown in BSC Table 4.3.3-2 (2008cp). The key structural features are the loaded weight, total length, diameter, and shell and plate thickness of the representative canister. DOE chose these dimensions to be close to the average of a DPC and a naval canister, as shown in BSC Table 4.3.3-2 (2008cp). The material properties used for the representative canister were those of SS304 stainless steel. The failure probability of the representative canister was used for TAD canisters and DPCs in the PCSA.

DOE used the nonlinear explicit finite element code LS–DYNA to perform a number of finite element analyses to estimate the demand on the representative canister. The structural challenges analyzed were (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 [10-metric-ton] load onto the top of the canister. DOE presented details of the LS–DYNA finite element models in BSC Section 6.3.3 (2008cp). DOE modeled the canister shell with multiple solid brick (three-dimensional) finite elements. DOE 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. DOE determined the demand due to impact using the maximum effective plastic strain of a single brick element through the thickness of the shell.

The development of the canister capacity (fragility) curve was based on the material used for the representative container, which was SS304. The fragility curve, which represents probability of failure as a function of true strain, was determined by fitting a probability density function to engineering tensile strain data for SS304. A frequency histogram, as outlined in BSC Figure 6.3.7-2 (2008cp), of the tensile elongation failure data was constructed from a sample of 204 tensile failure tests. DOE stated that the data are not normally distributed and the data 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 Figure 6.3.7-3 (2008cp), DOE shifted this initial fragility curve by 8.3 percent to a lower value of minimum elongation, because the original data were for an annealed SS304, which has a larger elongation (strain) at failure than the unannealed SS304 proposed for the canister. This shift resulted in a more conservative (i.e., higher) estimate of the failure probability. DOE also used this fragility curve for assessing the capacity of the DOE standardized canister and the representative canister contained within the transportation cask and AO.

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 Figure 6.3.7-1 (2008cp). DOE used a canister failure probability of 1×10^{-5} for event sequence analyses related to canister drop from heights of 9.9 , 12, and 13.7 m [32.5, 40, 45 ft], and a 3-m [10-ft] drop of a 9,072-kg [10-T] object on the canister, as shown in BSC Table 6.3-7 (2008ac). The reliability of the representative canister was used for TAD canisters, DPC, and naval canisters in the event sequence analyses, as stated in BSC Attachment D (2008bq).

DOE included the probability of failure for a 4° off-vertical drop in BSC Table D1.2-3 (2008ac). DOE asserted in SAR Section 1.7.2.3.1 that the TAD canister and DPC would undergo a flat-bottom drop during transfer operations because the canisters would fit tightly inside the CTM shield bell and a canister guide sleeve would ensure a flat bottom drop. In its response to an NRC staff RAI (DOE, 2009fy), DOE presented information on guide sleeve functions.

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of the TAD canister and DPC using the guidance in the YMRP. to assess the methodology used to estimate the reliability of the TAD canister and DPC. DOE's reliability estimates based on evaluating demand and capacity are consistent with the methodology described in NRC guidance given in HLWRS–ISG–02 (NRC, 2007ab).

The NRC staff reviewed DOE's use of a representative canister in the reliability estimate for DPC. The staff notes that the thickness of the canister shell and top and bottom plates used in the analysis are, on average, thinner than the corresponding DPC. DOE demonstrated that the maximum effective plastic strain is a function of the shell thickness, as shown in BSC Figure 6.3.3.6-1 (2008cp). The NRC staff notes that the use of a thin shell for the representative canister will produce larger effective plastic strains. Therefore, the use of a representative canister for DPC is reasonable. In absence of a final TAD canister design, DOE used the same representative canister to determine the failure probability. The NRC staff reviewed DOE's use of a representative canister in the reliability estimate for the TAD and notes this is reasonable for a final TAD canister design with similar dimensions and weight.

The NRC staff notes the selection of LS–DYNA for the nonlinear finite element analysis for estimating demand is reasonable because it is an industry-accepted code that is commonly used for highly nonlinear, transient impact (drop) simulations. The canister shell is modeled with multiple solid brick (three-dimensional) finite elements, which the NRC staff notes reasonably 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 use of the maximum effective plastic strain is reasonable 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 multiaxial state of strain in the material. In addition, 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, DOE's use of maximum effective plastic strain is conservative. The NRC staff also notes that reasonable engineering modeling techniques were applied to the finite element analyses for estimating demand because DOE (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 DOE's development of the canister capacity (fragility) curve as shown in BSC Figure 6.3.7-3 (2008cp), on the basis of material used for the representative container. The NRC staff notes that using the tensile elongation (failure) data to construct the fragility curve is conservative because both compressive and tensile strains are present. The NRC staff reviewed the uncertainty in the fit to the chosen probability distribution, which naturally leads to uncertainty in the fragility curve. On the basis of this review, the NRC staff notes that the uncertainty is small because the fragility curve was modeled with a goodness of fit test having a 95 percent confidence level. Therefore, the transmitted uncertainty is not significant, because of the high confidence level in the fit. The NRC staff notes it is reasonable to use a fragility curve to determine canister capacity because it relates the magnitude of strain (demand) obtained from the simulation to the probability of failure of the canister's stainless steel material. The NRC staff reviewed the effects of strain rate and temperature on the canister failure probability. In its response to an NRC staff RAI (DOE, 2009fv), DOE stated the effects of strain rate and temperature were not included in the fragility curve used in BSC (2008cp). The NRC staff reviewed DOE's response and notes the fragility curve without the strain rate and temperature effects is conservative because it results in more localized material failure.

The NRC staff reviewed failure probability values for several different cases in BSC Table 6.3-7 (2008ac) for the bare representative canister. The failure probability value of 1×10^{-5} for cases related to canister drop 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-T] object on the canister are consistent with results obtained from the finite element analysis. The NRC staff notes that DOE reasonably used a failure probability of 1×10^{-5} for event sequence analyses, as shown in BSC Table 6.3-7 (2008ac), instead of 1×10^{-8}. DOE's determination of probability of failure for a 13.7-m [45-ft] canister drop is reasonable because the approach of extrapolation of strains to different drop heights is based on the conservation of energy in which this impact energy is proportional to drop height.

The NRC staff reviewed calculations DOE provided for determining the maximum canister drop angle that could be achieved when the guide sleeve is present (DOE, 2009fy). The NRC staff notes that for a canister with dimensions similar to the TAD canister, the maximum drop angle would be approximately 0.9° and the corresponding failure probability less than 10^{-8} as long as the guide sleeve fulfills its intended function, as described in DOE's RAI response (DOE, 2009fy).

On the basis of the NRC staff's evaluation discussed in this section, DOE's reliability analyses of TAD canisters and DPC due to structural challenges are reasonable because DOE used industry-accepted methodologies to estimate failure probabilities and the capacity and demand of the canisters.

<u>DOE Standardized Canisters</u>

DOE discussed the determination of failure probabilities of the DOE standardized canisters in BSC Section D1.2 (2008ac). DOE's methodology for estimating the reliability of the DOE standardized canister was accomplished by establishing the relationship between demand and capacity defined in terms of strain in the canister material. This methodology is the same as that described in DOE's analysis of the representative canister.

DOE calculated the demand in terms of the maximum effective plastic strain from each finite element drop simulation analysis. 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. DOE stated that a series of full-scale, experimental drop tests were performed on 30- and 61-cm [18- and 24-in]-diameter DOE standardized canisters at Idaho National Laboratory. The purpose of these tests was to validate the finite element simulations of the corresponding experimental drop tests. DOE compared the numerical results obtained from the finite element analyses with experimental observations (measured in terms of permanent deformation) (SAR Figures 1.5.1-23 through 1.5.1-28). DOE showed in SAR Figures 1.5.1-23 through 1.5.1-28 that the nonlinear finite element analyses can accurately capture the highly nonlinear deformation response of the canister when subjected to a drop test.

DOE determined that the structural capacity of the canister depended on tensile elongation at failure obtained from canister material tests. DOE utilized the same stainless steel fragility curve as was used to determine the capacity of the representative TAD and DPCs, as outlined in BSC Figure 6.3.7-3 (2008cp).

DOE 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 Table D1.2-6 (2008ac) for both the 30- and 61-cm [18- and 24-in]-diameter canisters. Using the canister capacity curve (i.e., fragility for the stainless steel material), DOE calculated the failure probabilities using the maximum equivalent plastic strains.

DOE summarized the failure probabilities for the DOE standardized canister in BSC Tables D1.2-7 and 6.3-7 (2008ac). For the case of vertical container drop from normal operating height {7 m [23 ft]}, DOE stated the failure probability is equal to 10^{-8}, as shown in BSC Table D1.2-7 (2008ac). However, DOE used the failure probability of 10^{-5}, as detailed in BSC Table 6.3-7 (2008ac), for event sequence quantification. For the 9-m [30-ft] vertical drop case, DOE extrapolated the amount of strain from the 7-m [23-ft] drop case following the procedure in BSC Section D1.5 (2008ac) and estimated the failure probability to be 10^{-5}. For the cases of 4 and 14.5 km/hour [2.5 and 9 mph] end-to-end collisions, DOE reported a failure probability of 10^{-5}.

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of the DOE standardized canister using the guidance in the YMRP. to assess the methodology DOE used.

The NRC staff notes that using the ABAQUS/Explicit finite element code to obtain the demand on the canister resulting from drop (impact)-induced structural challenges is reasonable because it is well established in the industry for performing nonlinear, highly transient analyses. The finite element analysis approach is considered reasonable because the finite element models predict similar deformations when compared to the experimental results. Because of this good correlation, the NRC staff also notes that the finite element analyses used for the reliability estimates are based on reasonable engineering modeling techniques. In addition, the models utilize data which are consistent with the design, as evaluated in TER Sections 2.1.1.2.3.5.2 and 2.1.1.7.3.9.3.1.

DOE used the same fragility (capacity) curve for the representative canister. This is reasonable because DOE's standardized canister is fabricated from a 316 stainless steel and the representative canister was also specified to be constructed of stainless steel. The NRC staff notes the original fragility curve [BSC Figure 6.3.7-3 (2008cp)] was based on SS304. DOE accounted for the different steel by using the shifted curve to obtain a more conservative (i.e., higher) estimate of the failure probability. Therefore, there is conservatism in the fragility curve for the DOE standardized canister.

The NRC staff notes that although DOE estimated the failure probability for the cases of vertical container drop from operational height {7 m [23 ft]} and 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} to be 10^{-8}, as outlined in BSC Table D1.2-7 (2008ac), it used the failure probability of 10^{-5}, as shown in BSC Table 6.3-7 (2008ac), for event sequence quantification. The NRC staff notes that this is reasonable because using 1×10^{-5} adds conservatism.

Transportation Cask

DOE discussed loss of containment of transportation cask due to drops and impacts in BSC Section 6.3.2.2 (2008ac) with the associated failure probabilities given in Table 6.3-2 of the same document. DOE's methodology for estimating the transportation cask reliability is the same as that for the representative canister. DOE established a relationship between demand and capacity defined in terms of strain in the canister material.

DOE stated in BSC (2008cp) that the transportation cask provides shielding but not containment. The internal representative canister is relied upon to provide containment and the breach of the container "system" can occur only when the internal representative canister material fails. Thus, maximum equivalent plastic strains in the internal representative canister (inside the transportation cask) were calculated using finite element drop simulations to determine the demand. 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. DOE used the fragility (capacity) curve developed for representative canister as shown in BSC Figure 6.3.7-3 (2008cp).

DOE 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 LS–DYNA was performed, as described in BSC Section 6.3.2 (2008cp). BSC Table 4.3.3-1a (2008cp) listed all of the cases analyzed for the transportation cask. BSC Figure 6.3.2-1 (2008cp) showed the structural components that were included in the finite element model that was used to perform the drop analyses. The components that were modeled include 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. DOE listed all dimensions of the components in BSC Table 6.3.2-1 (2008cp), and the necessary material property data were given in BSC Table 6.3.2-2 (2008cp). The finite element model was shown in BSC Figure 6.3.2-2 (2008cp).

DOE calculated the failure probabilities corresponding to the low velocity events (collisions) using the principle of conservation of energy to convert the low speeds into an equivalent drop height (BSC, 2008ac). The failure probabilities were determined using BSC Section 6.3.2.2, Equation 17 (2008ac), which used the known failure probabilities from the LS–DYNA (BSC, 2008cp) analyzed drop heights, as shown in BSC Table 6.3-2 (2008ac). BSC Section 6.3.2.2, Equation 17 (2008ac) is based on the concept that the strain is approximately proportional to the impact energy, which directly relates to the drop height.

DOE summarized the failure probabilities for the transportation cask in BSC Table 6.3-7 (2008ac) and BSC Table 6.3.7.6-2 (2008cp). DOE stated in BSC Table 6.3.7.6-2 (2008cp) that, for all of the cases considered (including slapdown), the corresponding failure probabilities are less than 10^{-8}. However, DOE reported failure probabilities of 10^{-5} in BSC Table 6.3-7 (2008ac) for use in the event sequence quantifications. DOE stated that this was done to add additional conservatism. For the low velocity impacts (which correspond to very small drop heights), the failure probabilities remained at 10^{-8}, as outlined in BSC Table 6.3-7 (2008ac).

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of the transportation cask using the guidance in the YMRP Section. The methodology used for estimating the reliability of the transportation cask is reasonable because it is a commonly used methodology in the industry to estimate reliability of a mechanical system.

DOE utilized nonlinear finite element analysis (i.e., LS–DYNA) to determine the demand on the internal representative canister (inside the transportation cask). DOE's use of the nonlinear finite element analysis for modeling the drop (impact) analysis is reasonable because this approach is appropriate for impact analysis and LS–DYNA is commonly used in industry for performing highly nonlinear, transient impact analyses. In addition, the NRC staff notes that reasonable engineering modeling was used in finite element models. For example, the finite element mesh appears to be of reasonable refinement, especially through the shell thickness. The mesh refinement is reasonable because it follows standard engineering practice in that it will allow a gradient of plastic strain to develop through the thickness.

DOE followed the same methodology for determining structural capacity of the representative canister as detailed in BSC (2008cp). The use of the fragility curve in BSC Figure 6.3.7-3 (2008cp) is reasonable because DOE utilized the same representative canister inside the transportation cask.

The NRC staff reviewed DOE's approach of using the principle of conservation of energy to convert travel speeds into equivalent drop heights for calculating the failure probabilities corresponding to the low velocity events (collisions) reported in BSC Table 6.3-4 (2008ac) using BSC Section 6.3.2.2, Equation 17 (2008ac). The NRC staff notes that this approach is reasonable because the strain is approximately proportional to the impact energy, which directly relates to the drop height. On the basis that strain is proportional to drop height, it is reasonable to expect that height would have a normal distribution because DOE assumed a normal distribution for strain. For a normal distribution, the mean and median drop heights are equivalent in BSC Section 6.3.2.2, Equation 17 (2008ac). Therefore, use of this approach for calculating the failure probabilities for the low velocity (collision) events is reasonable.

The NRC staff reviewed the failure probabilities listed in BSC Table 6.3-7 (2008ac) for the case of the transportation cask. The NRC staff also reviewed the corresponding analyses for the cases of 9,072 kg [10 T] dropped on the container, vertical drop of the container, and vertical drop followed by slapdown. For these cases the failure probability of 10^{-5} was calculated from the LS–DYNA analyses given in BSC Section 6.3.2 (2008cp). All corresponding failure probabilities are less than 10^{-8}, as outlined in BSC Table 6.3.7.6-2 (2008cp). However, BSC Table 6.3-7 (2008ac) reported a value of 10^{-5}. The NRC staff notes this is consistent with other failure probabilities in which DOE used 10^{-5} to add conservatism. In addition, DOE's use of failure probabilities of low velocity impacts at 10^{-8} is reasonable because these low velocities correspond to small drop heights (some almost negligible), as shown in BSC Table 6.3-4 (2008ac).

On the basis of its review of BSC Tables 6.3-7 and 6.3-8, PEFA.xls (2008bq), the NRC staff notes for the probability of loss of containment of the transportation cask with bare SNF assemblies, DOE used a probability value of 10^{-5} (BSC, 2008ac,bq). The transportation casks, which are classified as ITS, are certified under 10 CFR Part 71 as DOE stated in SAR Section 1.2.8.4.5.1. In accordance with NRC guidance (NRC, 2000aj), the containment system should be designed and constructed conforming to Section III, Division 3, ASME Boiler and Pressure Vessel (B&PV Division 3) Code. In addition, the certified transportation cask would demonstrate that it has the structural integrity to maintain containment, shielding, and subcriticality when subjected to a free drop from a height of 9 m [30 ft] onto an unyielding, flat, horizontal surface, striking the surface in a position for which maximum damage is expected (NRC, 2000aj). Because the transportation cask would be structurally intact after a 9-m [30-ft] drop, the NRC staff notes that assuming a failure probability of 10^{-5} is reasonable.

<u>Aging Overpack</u>

DOE presented the failure probabilities for loss of containment for AOs in BSC Table 6.3-7 (2008ac). DOE 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 discussed in BSC Section 6.3.1 (2008cp), 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, DOE 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.

DOE summarized in BSC Table 6.3-7 (2008ac) the 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, DOE calculated the probability of AO containment failure as 10^{-5}. In addition, DOE specified a failure probability of 10^{-8} for the cases of low velocity impact/collisions.

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of loss of containment for the AO using the guidance in the YMRP. The NRC staff reviewed the LS–DYNA analyses DOE performed to determine 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. DOE's use of LS–DYNA is reasonable because it is commonly used in the industry for highly nonlinear, transient impact analyses.

The NRC staff reviewed failure probability results from the LS–DYNA analyses as given in BSC Table 6.3.7.6-1 (2008cp) and notes the AO failure probabilities to be less than 10^{-8} for both cases. However, the NRC staff notes DOE used a failure probability value of 10^{-5} for the event sequence quantifications, as detailed in BSC Table 6.3-7 (2008ac), which the NRC staff notes is conservative. For the remaining failure probabilities associated with the low velocity impacts, these low velocities would have equivalently small drop heights as shown in BSC Table 6.3-4 (2008ac). Therefore, because the drop heights are small, the failure probabilities of 10^{-8} are reasonable.

Loss of Shielding

Structural challenges causing potential loss of shielding include drop of a container (including slapdown), collision of a container with other structures or objects, and drop of an object onto waste container. The probability of loss of shielding or failure of casks under structural challenges was directly used as a point estimate in the pivotal event, SHIELDING, in the system response tree model RESPONSE–CANISTER1.

Aging Overpack

DOE discussed the probabilities of the loss of shielding function of AOs in BSC Section D3.4 (2008ac). The AO transports a canister from the 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

DOE's basic 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 AO shielding does not fail and the conditional probabilities of canister success given AO shielding fails, as provided in BSC Equation D–26 (2008ac). DOE rearranged the terms in BSC Equation D–26 (2008ac) and made some simplifying assumptions to obtain BSC Equation D–29 (2008ac), which expresses the probability of AO shielding failure as a function of the internal canister failure.

To calculate the demand on the internal representative canister contained within the AO, DOE followed the methodology in BSC (2008cp). DOE 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. DOE 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 Section 6.3.1 (2008cp). The finite element model consisted of an AO and the internal representative canister contained within the AO. DOE stated that the internal SNF canister was the same as that used for the representative canister.

DOE considered two different loading scenarios in the analyses. Case 1 analyzed the AO/canister model for a 0.9-m [3-ft] vertical drop (normal operating height) onto a rigid surface. Case 2 analyzed the AO/canister model when subjected to a slapdown from a vertical orientation while also having a 4 km/hour [2.5 mph] horizontal velocity.

DOE summarized the AO failure probabilities used for event sequence quantifications in BSC Table 6.3-7 (2008ac). For the 0.9-m [3-ft] vertical drop, DOE calculated the AO shielding failure probability as 5.0×10^{-6}. In addition, DOE specified a failure probability of 10^{-5} for the cases of low velocity impact/collisions and slapdown.

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of loss of shielding for the AO using the guidance in the YMRP. The NRC staff reviewed DOE's methodology for determining the probability of AO shielding failure due to a structural challenge and notes that DOE's approach of using BSC Equation D–26 (2008ac) is consistent with the established methodology for estimating probabilities (Harris, 1966aa). DOE's use of BSC Equation D–29 (2008ac), which expresses the probability of AO shielding failure as a function of the internal canister failure, is reasonable because of AO robustness against impact loads and that likelihood of AO breach is lower than likelihood of AO success for values exceeding the drop and impact speed conditions.

The NRC staff reviewed the finite element analyses DOE performed to determine demand, and the analyses are reasonable because the mesh for the AO appears to be of reasonable refinement. The NRC staff also notes that the use of LS–DYNA is reasonable because it is commonly used in the industry for highly nonlinear, transient impact analyses.

DOE followed the same methodology for determining structural capacity of the representative canister as detailed in BSC (2008cp). DOE's use of a fragility curve in BSC Figure 6.3.7-3 (2008cp) is reasonable because DOE utilized the same representative canister inside the AO.

The NRC staff reviewed the failure probabilities listed in BSC Table 6.3-7 (2008ac) for the AO and notes the AO failure probability of 5.0×10^{-6} is reasonable for the case of vertical drop from normal operating height, because it is based upon the internal canister failure probability of 10^{-5}. The NRC staff notes this is conservative on the basis of its review of failure probability of the representative canister as described previously. For the case of AO slapdown, an AO failure probability of 10^{-5} was also used, which is conservative.

For the remaining failure probabilities associated with the low velocity impacts, these low velocities would have equivalently small drop heights, as shown in BSC Table 6.3-4 (2008ac). Therefore, because the drop heights are small, failure probabilities of 10^{-5} are reasonable.

Transportation Cask

DOE discussed in BSC Section D3 (2008ac) the degradation of shielding for a transportation cask when subjected to a structural challenge due to impact.

DOE's methodology for estimating the failure probability utilized finite element analysis to determine the demand on the transportation cask subjected to transportation accident impacts. DOE 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.

DOE used the finite element analyses 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. DOE stated in BSC Section D3 (2008ac) 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, DOE stated that lead slump occurs mainly for the end-impact orientations and, to a lesser extent, for corner impacts. For side impacts, DOE stated that there is no significant reduction in shielding. Thus, DOE focused only on the steel/lead/steel casks with the primary orientation being the end-impact condition. DOE listed various impact speeds and the resulting maximum plastic strains for impacts onto an unyielding surface, as outlined in BSC Table D3.2-1 (2008ac). DOE also listed the equivalent speeds from impacts onto real surfaces, such as soil and concrete, as shown in BSC Table D3.3-2 (2008ac), and established a damage threshold for lead slumping, as described in BSC Sections D3.1 and D3.2 (2008ac). DOE stated that for maximum effective plastic strain levels exceeding 2 percent, lead slumping is likely. Using BSC

Figures D3.2-2 and D3.2-3 (2008ac), DOE estimated the threshold velocities in which the loss of shielding (lead slumping) would occur. DOE 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 Table D3.3-1 (2008ac). Thus, DOE used an estimate of a median threshold for the failure drop height as 41 m [133 ft] (i.e., 2 percent plastic strain).

To calculate the transportation cask failure probability as a function of height, DOE utilized BSC Equation 17 (2008ac). DOE based the failure probability upon a median threshold of a 41-m [133-ft] drop, as discussed previously. In addition, DOE applied a normal distribution to drop height on the basis that strain and drop height is approximately proportional, as described in BSC Section 6.3.2.2 (2008ac). DOE presented the failure probabilities for a number of different drop heights and collisions in BSC Table 6.3-7 (2008ac).

The previous discussion focused on the steel/lead/steel sandwich type of transportation cask. DOE 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 Section D3.4 (2008ac).

NRC Staff Evaluation: The NRC staff reviewed DOE's reliability analysis of loss of shielding for the transportation cask using the guidance in the YMRP.

The NRC staff notes that the 2 percent maximum plastic strain threshold DOE established would correspond to a 101 km/hour [63 mph] impact on a concrete surface for a truck cask using data from BSC Figure D3.2-3 (2008ac) and linear interpolation. The speed of 101 km/hour [63 mph] translates into an equivalent drop height of 41 m [133 ft] using the data from BSC Table D3.3-1 (2008ac). Therefore, the threshold for lead slumping is reasonable for a 41-m [133-ft] drop. In addition, the NRC staff notes that applying a normal distribution to drop height is reasonable because it is consistent with DOE's argument that strain and drop height are approximately proportional, as described in BSC Section 6.3.2.2 (2008ac).

The NRC staff reviewed the failure probabilities listed in BSC Table 6.3-7 (2008ac) for the transportation cask and verified that BSC Equation 17 (2008ac) was used to calculate the failure probabilities. The NRC staff also verified that the probability for loss of shielding due to an impact of a lead slump from a 4.6-m [15-ft] drop height was less than 10^{-8}. The NRC staff notes the DOE-estimated failure probability of 10^{-8} for the low speed impacts is reasonable because these low velocities correspond to small drop heights [see BSC Table D3.3-1 (2008ac)]. As discussed previously, such small drop heights produced low failure probabilities. The NRC staff notes that DOE conservatively used a failure probability of 10^{-5} for event sequence analysis, as shown in BSC Table 6.3-7 (2008ac) instead of 10^{-8}.

The NRC staff notes that both the transportation cask and the STC were considered as a representative cask in the PCSA (SAR Table 1.9-4) and the previous discussions for the transportation cask are also applicable to an STC. DOE's estimation of the failure probability of 10^{-5} for loss of shielding of a STC, as shown in BSC Table D3.4-1 (2008ac), is reasonable because this value is conservative.

2.1.1.4.3.3.1.2 Passive Reliability for Structural Challenges Resulting From Seismic Events

This TER section contains the NRC staff's review of reliability of the passive ITS SSCs for structural challenges resulting from seismic events. The NRC staff's review is presented in TER Section 2.1.1.4.3.3.1.2.1 for surface civil structures, and mechanical systems are covered in TER Section 2.1.1.4.3.3.1.2.2.

2.1.1.4.3.3.1.2.1 Surface Structural Civil Facilities

DOE provided information on seismic performance of surface structural civil facilities in SAR Sections 1.7.1.4 and 1.7.2.4 and BSC (2008bg). DOE described the methodology for seismic performance evaluation of surface structures in SAR Section 1.7.1.4. The methodology to develop the fragility of surface facilities was summarized in SAR Section 1.7.2.4. DOE's approach was to demonstrate that the probability of structural failure is beyond Category 2 event sequences and therefore precludes radiological dose computation. DOE 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 Table 6.2-1 (2008bg).

The ITS surface facilities reviewed in this section are the IHF, CRCF, WHF, and RF. The NRC staff reviewed the information presented in the SAR with respect to (i) the approach used to generate fragility curves for ITS structural facilities, (ii) the capacity and uncertainty data used to develop fragility curves, and (iii) evaluation of annual probability of unacceptable performance.

For the ITS surface facilities, DOE stated that seismic loading controls the structural performance (TER Section 2.1.1.7.3.1.1) and evaluated the structural performance only at the collapse limit state [i.e., Limit State A, according to American Society of Civil Engineers, Table 1-1 (2005aa)]. DOE defined four stages to conduct seismically initiating event sequences: (i) development of seismic event sequences, (ii) development of seismic hazard curves, (iii) evaluation of seismic fragilities, and (iv) quantification of event sequences. To evaluate event sequences associated with structure failure, DOE assumed that unfiltered radionuclide release will occur after the structure collapses (e.g., BSC, 2008bg). Therefore, the NRC staff's review focused on generating the seismic fragility curves for the collapse limit state and using the curves to calculate the probability of collapse of ITS surface facilities.

Probability of Failure or Unacceptable Performance

DOE assessed the structural performance of ITS surface facilities by computing the probability of unacceptable performance (or probability of failure) of the ITS surface facilities. To obtain the probability of failure, DOE combined seismic hazard curves at the site with fragility curves for the evaluated failure mode (SAR Section 1.7.1.4; DOE, 2007ab). The probability of failure for each facility was compared to the probability threshold for the mean frequency of collapse due to seismic events of 2×10^{-6} (SAR Table 1.2.4-4). DOE obtained this probability threshold by assuming that operational activities will be performed for no more than 50 years (SAR Section 2.2).

The probability of failure for ITS buildings was presented in BSC Table 6.2-1 (2008bg). DOE also provided a table summarizing the computation of the probability of failure, P_f, for the CRCF in DOE Enclosure 3 (2009dz), which DOE considered representative of the ITS buildings.

NRC Staff Evaluation: The NRC staff reviewed the methodology DOE provided for computing the probability of unacceptable performance (or probability of failure) using the guidance in the YMRP. The NRC staff's review is based on the assumption that the receipt and emplacement operations at the surface facilities are projected to span 50 years as stated in SAR Section 2.2 and as evaluated in TER Section 2.1.1.2.3.6.1. The NRC staff notes that DOE's seismic performance calculation methodology is reasonable for the performance evaluation of ITS structural facilities and consistent with common industry practice. Regarding convolution of the hazard and fragility curves to obtain the probability of failure of the ITS buildings, the method is appropriate because it is based on common industry practices and NRC guidance (NRC, 2006ad).

Methodology for Generation of Fragility Curves

DOE used a simplified methodology for developing seismic fragility curves for ITS surface facilities in SAR Section 1.7.2.4 and DOE (2007ab). In this methodology, DOE based seismic fragility curves on an approximation of the capacity at 1 percent conditional probability of failure, $C_{1\%}$, and a composite logarithmic standard deviation, β_c, assumed by engineering judgment and described in DOE Sections 4.2 and 4.4.2 (2007ab). 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 β_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 TER Section 2.1.1.7.3.1.1 for a detailed discussion). Additionally, in its response to an NRC staff RAI (DOE, 2009dz), DOE provided the fragility calculation only for the CRCF (BSC, 2007df).

NRC Staff Evaluation: The NRC staff reviewed the methodology DOE provided to generate the fragility curves for the ITS surface facilities using the guidance in the YMRP. The NRC staff notes that the methods used to generate the fragility curves for the ITS surface facilities are reasonable because they are consistent with the standard industry practice for the performance evaluation of similar risk nuclear facilities.

Computation of $C_{1\%}$

To obtain $C_{1\%}$, DOE 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 $C_{1\%}$. DOE computed the HCLPF capacity on the basis of the Conservative Deterministic Failure Margin method. For low-rise shear walls, DOE computed the HCLPF capacity on the basis of American Society of Civil Engineers Equation 4-3 (2005aa) and BSC Section B4.3, Step 3 (2007ba).

DOE designed the ITS surface facilities to withstand DBGM–2 seismic levels, which correspond to a mean annual probability of exceedance (MAPE) of 5×10^{-4}, as detailed in DOE Section 3.1.1 (2007ab), and a horizontal PGA of 0.45 g (BSC, 2007ba). This design was based 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 TER Section 2.1.1.7.3.1.1. DOE 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, DOE used a finite element model, but soil–structure interaction effects were not included in the analysis of the superstructure.

On the basis of the structural configuration obtained from the design for DBGM–2 events, DOE performed linear elastic analyses based on Tier #1 models using BDBGM seismic events with an MAPE of 1×10^{-4} and horizontal PGA of 0.91g as detailed in BSC Section B.4.2 (2007ba). The results from analyses for BDBGM events were used to compute $C_{1\%}$ of the fragility curves. DOE stated that the structural response under BDBGM seismic events is expected to exhibit inelastic behavior, or at least be close to the inelastic threshold.

In its response to an NRC staff RAI regarding the technical bases on using Tier #1 models instead of more realistic finite element models, DOE stated that Tier #1 models provided realistic forces and moments that were sufficient for the initial design and evaluation of seismic performance for use in the PCSA and mentioned several sources of conservatisms embedded in Tier #1 models (DOE, 2009gh).

In its response to an NRC staff RAI in DOE Enclosure 8 (2009gh), DOE stated it will not provide nonlinear structural analyses of the surface facilities. Instead, DOE stated that nonlinear time history analyses were performed for the Diablo Canyon Turbine Building (Kennedy, et al., 1988aa), and that the results were compared to those obtained from a simplified approach (Electric Power Research Institute, 1991aa), showing excellent agreement (Kennedy, et al., 1988aa). Because the Diablo Canyon structure exhibited structural irregularities, DOE considered that the recommended approach for surface facilities with structural irregularities is reasonable.

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided about the Tier #1 structural analysis used to obtain the probability of unacceptable performance, using the guidance in the YMRP. The NRC staff notes that structural models similar to the Tier #1 models have been used in the nuclear industry. The NRC staff notes that the simplified structural analyses DOE used provided a reasonable estimate of the structural response of the ITS buildings when subjected to DBGM–2 events. The computation of $C_{1\%}$ based on Tier #1 analysis at BDBGM ground motion may be affected by the simplified elastic structural analyses because the buildings exhibit structural irregularities that may not be captured in these analyses. The NRC staff notes that the nonlinear performance of specific facilities may not compare with nonlinear evaluations of other structural systems (e.g., Diablo Canyon), which exhibit a different structural response (because of different failure modes, modes of vibration, soil conditions, and structural irregularities) and are subjected to a different seismic hazard. Although DOE stated in BSC Section 8.4 (2007ba) that the demand-to-capacity ratios for components of reinforced concrete structures would be less than 0.5–0.6, the NRC staff notes that on the basis of the review in TER Section 2.1.1.7.3.1, the maximum elastic demand-to-capacity ratios for the ITS facilities were usually higher than 0.7–0.8 under DBGM–2 seismic events [e.g., BSC Tables 10 and 11 (2007bl)].

The methodology used to approximate $C_{1\%}$ on the basis of an estimate of the HCLPF capacity for shear walls with boundary elements is reasonable. However, the NRC staff notes that DOE computed the HCLPF capacity based solely on the seismic response of shear walls and slabs and did not include the effect of soil and foundation behavior on the seismic performance, as outlined in DOE Enclosure 3 (2009dz) and BSC (2007df). On the basis of the review of DOE's calculations in TER Section 2.1.1.7.3.1, the bearing pressures for the WHF and IHF buildings would lead to soil demand-to-capacity ratios exceeding 0.80[1] (about 0.8 to 0.88) under DBGM–2 seismic events for a PGA of about 0.45 g; DOE did not present similar analysis of the

[1]The soil demand-to-capacity ratio assumes that the large bearing capacity of 2,394 kPa [50 ksf] DOE proposed is adequate.

foundation at BDBGM (0.91 g) and its effect on the system capacity. The elastic analysis results for seismic accelerations of 0.45 g suggest that soil failure is likely to occur if the accelerations are increased to BDBGM affecting the intended safety function of the structure (e.g., exceeding the limit state A deformation). As part of the detailed design process, DOE should confirm that the elastic spring constants to model soil at the BDBGM seismic level of 0.91 g for evaluation of $C_{1\%}$ are reasonable.

DOE used the capacity equation for low-rise rectangular concrete shear walls in American Society of Civil Engineers (2005aa) for all cases at the CRCF (BSC, 2007df). The ASCE equation, detailed in American Society of Civil Engineers Section 4.2.3 (2005aa), is only applicable to shear walls with boundary elements[2] or end walls, 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, DOE indicated in DOE Enclosure 4 (2009dz) that where there are no end or cross walls (e.g., a pier between openings), the vertical reinforcement displaced by the opening is placed as additional reinforcement on the two sides of the opening including providing confinement reinforcement to enhance ductility when needed and the final reinforcement details will be developed in the detailed design for construction.

The NRC staff notes that for the current level of design, DOE's estimation of $C_{1\%}$ for the CRCF facility is reasonable. The NRC staff recognizes that $C_{1\%}$ defining the fragility curves could change as the design evolves. Consequently, as part of the detailed design process, DOE should conduct, as indicated in DOE (2009gh), seismic structural and foundation analyses to confirm $C_{1\%}$ for defining the fragility curves of the CRCF, WHF, RF, and IHF as shown in BSC Table 6.2-1 (2008bg).

<u>Estimation of β_c</u>

DOE selected β_c by judgment from an interval ranging from 0.3 to 0.5 for ITS surface facilities, as described in DOE Section 4.4.2 (2007ab) and American Society of Civil Engineers Section C2.2.1.2 (2005aa). In response to an NRC staff RAI in DOE Enclosure 8 (2009ge), DOE indicated that β_c estimates higher than 0.52 were not credible and that β_c estimates exceeding 0.4 were the result of simplified evaluations. DOE indicated in DOE Section 4.4.2 (2007ab) that an estimate of β_c is sufficient to obtain the annual probability of failure, which is relatively insensitive to β_c variations. DOE recommended using the lower bound value, $\beta_c = 0.3$, to obtain conservative estimates of the probability of failure when generating the fragility curves, as detailed in BSC Section B4.3 (2007ba). The applicant, however, used a β_c value of 0.4 to evaluate probability of failure, as outlined in BSC Section 6.2.1 (2008bg). DOE stated that using $\beta_c = 0.3$ increased the P_f by about 50 percent as compared with using $\beta_c = 0.4$; this is still less than the Category 2 event sequence lower threshold of 2×10^{-6} per year, as described in DOE Enclosure 8 (2009ge).

NRC Staff Evaluation: The NRC staff reviewed the information DOE provided about the Tier #1 structural analysis used to obtain the probability of unacceptable performance, using the guidance in the YMRP. DOE selected the range of β_c consistent with the proposed values in ASCE 43–05 (American Society of Civil Engineers, 2005aa). DOE's calculations for the CRCF show that using $\beta_c = 0.3$ instead of $\beta_c = 0.4$ increases the probability of failure by about

[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.

50 percent, as described in DOE Enclosure 8 (2009dz). In the CRCF example, the probability of failure is slightly less than the probability threshold for Category 2 event sequences when the fragility curve is based on β_c = 0.3. On the basis of this evaluation, DOE's assessment of probability of failure of structural systems is reasonable.

2.1.1.4.3.3.1.2.2 Mechanical Equipment and Systems

DOE provided information on seismic fragilities of the mechanical equipment in SAR Section 1.7. DOE used probabilities of failure or unacceptable performance of the mechanical equipment to determine the probability of seismically initiated event sequences for categorization in the PCSA. 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. DOE used the separation of variable method to develop a mean fragility curve of mechanical equipment by estimating the median capacity and composite variability (Electric Power Research Institute, 1994aa). The methodology is based on quantifying design margins in the capacity and the seismic demand to develop median capacities and the logarithmic standard deviations. DOE considered a number of variables to quantify design margins and standard deviations (BSC, 2008bg).

DOE developed fragility parameters for several mechanical structures (e.g., cask preparation platform, mobile platform, shield door, entry door) and handling equipment (e.g., crane, CTM, transfer trolley, and TEV), as shown in BSC Table 6.2.2 (2008bg). DOE 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 Table 6.2.2 (2008bg). The annual probability of failure was calculated by convolving the fragility curve and the seismic hazard curve.

DOE indicated that the equipment design is preliminary and its seismic capacity was evaluated on the assumption that the design stress was equal to the code allowable stress (BSC, 2008bg). DOE stated that it expects the final equipment design will provide a margin exceeding the preliminary design.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's approach for assessing seismic fragility of mechanical systems using the guidance in the YMRP. DOE's methodology for developing seismic fragilities of mechanical equipment is reasonable because it is consistent with the methodology used for the safety-related equipment in nuclear power plants (Kennedy, et al., 1980aa; Kennedy and Ravindra, 1984aa).

The equipment fragility evaluation was based on a preliminary design (BSC, 2008bg). DOE stated (DOE, 2009bl) that it will verify that the final equipment design and its associated fragility will satisfy the results in BSC Table 6.2-2 (2008bg).

2.1.1.4.3.3.1.2.3 Passive Reliability for Structural Challenges Resulting From Fire Events

DOE 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; 2008ae,ai,ap,bp).

DOE 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 probability of maintaining shielding. DOE 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.

Thermal Challenges and Loss of Containment

DOE characterized the thermal demands on a canister as the canister wall temperature resulting from a fire exposure of a certain temperature for a certain duration. To quantify the demands, DOE postulated fire exposure conditions and calculated the expected wall temperatures that could result from those exposures. DOE used the fire data from large-scale tests conducted by different laboratories to develop a reasonable distribution of fire durations. As the automatic sprinkler system was not classified as ITS, the analysis used an assumed fire duration in the absence of any automatic fire protection. DOE concluded that, in an unsprinklered building, 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. A lognormal distribution was assigned to the fire duration.

DOE also quantified the intensity of the fire upon arrival at a cask containing a waste form, so that the reliability or fragility under a set of exposure conditions could be evaluated (BSC, 2008ac,as,au,be,bk,bq). DOE took the fire temperature as the effective blackbody temperature of the fire. DOE used solid fuel (e.g., wood, paper, or plastic) and liquid fuel (e.g., hydrocarbon pools) fire temperatures from Society of Fire Protection Engineers (2002aa). DOE also used flammable liquid fire data obtained from large-scale hydrocarbon fires involving railcars (Birk A.M. Engineering, 2005aa). DOE 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. DOE used these two data points 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), DOE indicated that 99.9 percent of all fires would have a temperature lower than or equal to approximately 1,330 °C [2,426 °F].

DOE 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. DOE utilized 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). DOE screened out the failures of bare canisters during transfer operations because of very low probability that the bare canister would be outside of a cask, a waste package, or the confines of the CTM or shield bell during a fire event.

DOE evaluated the canister response to a thermal challenge on the basis of the canister's ability to withstand stresses induced by the elevated temperature. DOE 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. DOE evaluated both failure modes independently, and the lower of the two failure temperatures was assumed to govern.

Once the range of thermal demands and responses were identified, DOE 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 million 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. DOE assigned a failure probability of less than 10^{-6} for cases with no observed failures after 1 million trials.

DOE calculated failure probabilities of six canister configurations and also calculated the failure probability of bare fuel in a standard GA–4/GA–9 transportation cask (BSC, 2008ac,as,au, be,bk,bq). The resulting failure probabilities for canisters in different configurations ranged from 1.0×10^{-6} to 3.2×10^{-4}, as shown in BSC Table D2.1-8 (2008ac,as,au,be,bk,bq). DOE estimated the failure probability of 5.4×10^{-4} for the bare fuel in a transportation cask on the basis of a conservative failure temperature of 700 °C [1,292°F], as shown in BSC Table D2.1-10 (2008ac,as,au,be,bk,bq).

NRC Staff Evaluation: The NRC staff reviewed DOE's evaluation of passive reliability of canisters under thermal challenges using the guidance in the YMRP. The NRC staff notes that the methods DOE selected and implemented for estimating the reliability of a canister during a fire were based on representative test data and fundamental heat transfer equations. These methodologies are reasonable for the anticipated fire exposure and canister configurations.

In addition, DOE used solid fuel (e.g., wood, paper, or plastic) and liquid fuel (e.g., hydrocarbon) to estimate the distribution of fire temperature. DOE reasonably used fire data consistent with the fuel density and burning rates of commodities expected in the GROA facilities for demand calculations. The NRC staff also notes that it is conservative for DOE to use test data based on fires where sprinklers were not assumed to operate, as shown in BSC Table F.11-2 (2008ac). Although the fire temperature generally rises to a peak and decreases once the combustible materials are consumed in the fire, DOE modeled a more conservative "steady-state" exposure. On the basis of the normal distribution of fire temperature DOE proposed (BSC, 2008ac), 99.9 percent of all fires would have a temperature lower than or equal to approximately 1,330 °C [2,426 °F]. Furthermore, the NRC staff verified the thermal demands DOE assumed and the thermal demands are consistent with previously established exposures derived from large-scale testing and industry-accepted data (e.g., Society of Fire Protection Engineers, 2002aa).

Although DOE did not provide any technical justification for the assumed mean fire temperature 799 °C [1,470 °F] and standard deviation of 172 °C [342 °F], assuming a normal distribution, the NRC staff notes that this mean temperature is consistent with the fire exposure temperature for hypothetical accident conditions outlined in NUREG–1617 (NRC, 2000aj) and therefore reasonable. The NRC staff also notes that the assumed standard deviation would encompass a range of ordinary combustibles (e.g., paper, wood and plastic) and also represent the potentially higher temperatures that concentrated amounts of plastics or flammable liquids cause.

The overall methods and data DOE used to determine canister response to thermal events are reasonable 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. The NRC staff notes that DOE reasonably used regression analysis to describe temperature dependency of the strength parameters and associated uncertainties.

Thermal Challenges and Loss of Shielding

DOE indicated that the thermal challenges of transportation casks may degrade the shielding by melting any low melting temperature shield materials present in the cask. For concrete shielding in AOs, degradation in shielding is realized through spalling, cracking, or other physical damage to the concrete encasement. In contrast, the 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.

DOE described the loss of shielding in transportation casks due to a thermal challenge in BSC Section D2.2.1 (2008ac,as,au,be,bk,bq). DOE indicated that all transportation casks have separate gamma and neutron shields. Because the neutron shield is typically fabricated from a low melting point polymer, DOE 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 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 selected as the design most likely to result in loss of shielding due to fire. DOE stated that the mode of failure would be 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, DOE assigned a probability of failure of 1.0 for loss of transportation cask shielding due to a thermal challenge. DOE also conservatively applied a failure probability of 1.0 to all transportation casks that do not use lead for shielding, as shown in BSC Table D3.4-1 (2008ac,as,au,be,bk,bq).

DOE described the loss of shielding in AOs due to a thermal challenge in BSC Section D2.2.3 (2008ac,as,au,be,bk,bq). Because concrete thickness provides AO shielding, the primary concern is loss of concrete thickness due to spalling. DOE discussed the predicted dose as a function of concrete loss due to spalling and demonstrated that up to 20 percent of the AO concrete thickness could be lost due to spalling without resulting in any unreasonable doses. As a result, DOE assumed a loss of shielding probability of 0.0 for AOs.

DOE also considered loss of shielding during transfer operations using the CTM and indicated that loss of shielding in the CTM was based on the failure of a shield bell that encompasses the waste form. DOE assumed that loss of CTM shielding probability was 0.0 because the shield bell components (primarily depleted uranium) have high melting temperatures {3,400 °C [6,152 °F]}. Additionally, DOE stated that absence of combustible materials in proximity to the CTM made it unlikely that these melting temperatures would be achieved.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's evaluation of the reliability of passive systems providing shielding under thermal challenges using the guidance in the YMRP. The NRC staff notes that DOE used reasonable methods and implemented these methods for estimating the reliability of passive systems subject to fire exposure.

DOE made conservative estimates of the reliability probabilities for the shield materials consistent with the proposed shielding design and did not include detailed heat transfer models. For example, data used in this analysis were based on the configuration of various shield materials (lead, concrete, depleted uranium), coupled with material properties for each shield material. The analysis focused on the melting temperature of transportation casks and shield bell components and AO spalling characteristics. The data used to determine reliability are consistent with the design descriptions and are consistent with material property data for the various shield components.

DOE's assumption of a failure probability of 1.0 for loss of transportation cask shielding regarding the failure mode of low melting temperature shielding materials for the transportation cask is reasonable because this assumption makes a bounding case. The NRC staff further notes the probability of loss of transportation cask shielding due to thermal challenge is reasonable because these assumptions are conservative. In addition, DOE made reasonable assumptions regarding the CTM shield bell performance because the shield bell has high melting temperature components and there are no combustible materials in the vicinity of canister transfer operations.

For the AO, DOE assumed various thicknesses of concrete to spall in a fire. The NRC staff did not independently verify the AO spalling calculations DOE used to support its assignment of a 0.0 failure probability for AOs. However, the NRC staff notes that worker dose during a fire event due to loss of concrete shielding is minimal because fires that are large enough to challenge the concrete AO shielding will take a substantial amount of time to develop. It is therefore reasonable to assume that workers will not be present in the area during these fire exposures and workers who are present as part of the firefighting effort will be properly protected against potential exposure.

2.1.1.4.3.3.2 Active Systems

The NRC staff's review of reliability of active systems includes the 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

DOE discussed 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). DOE 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 CRCF and WHF. DOE developed fault trees to quantify the failure to maintain confinement, which it characterized in these fault trees as a loss of delta pressure in the CRCF and WHF. DOE used the results from its fault tree analyses to specify the controlling parameters for its nuclear safety design bases. Therefore, in this section, the NRC staff evaluates the surface nuclear confinement ITS HVAC systems failure probability used in the confinement pivotal event and specified as a controlling parameter in its nuclear safety design bases. The NRC staff evaluates the design of the ITS HVAC systems in

TER Section 2.1.1.7.3.3 and the ability of the ITS HVAC systems to perform their safety functions in TER Section 2.1.1.6.3.2.8.2.2.

DOE 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). In addition, 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. Therefore, the NRC staff evaluated the nonconfinement ITS HVAC system in the EDGF to the extent that it supports the functioning of the emergency diesel generators and ITS electrical equipment, which provide power to ITS HVAC in CRCF and WHF in the event of loss of offsite electrical power. DOE described the surface nonconfinement ITS HVAC system for the EDGF in SAR Section 1.2.8.3. During this review, DOE updated SAR Sections 1.2.2.3, 1.2.4.4, 1.2.5.5, and 1.2.8.3 and supporting documents with regard to ITS HVAC systems. Therefore, the updated SAR (DOE, 2009av) and the updated reliability and event sequence categorization documents for the CRCF and WHF (BSC, 2009ab,ac) were also included in this evaluation. In addition, because the surface nuclear confinement ITS HVAC systems are similar in the CRCF and WHF, this section primarily focuses on the HVAC system in the CRCF.

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. This section primarily focuses on the ITS subsystem providing filtration, 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, on failure of this operating train, the standby train will automatically start. DOE 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). 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 annotated the differential pressure switches across the exhaust fan and HEPA filter plenums ITS and the flow instrumentation ITS. In addition, it specified the adjustable speed drive and interlock ITS but did not specifically annotate the radioactivity monitor ITS.

DOE provided a fault tree model for loss of delta pressure in the CRCF in BSC Section B7.4 (2009ab). This fault tree model accounts for both equipment failure and human error. DOE stated that the top level of the fault tree in BSC Figure B7.4-3 (2009ab) is incorrect for the CRCF and that BSC Figure B7.4-3 (2009ac) for the WHF is applicable to the ITS HVAC exhaust subsystem in the CRCF (DOE, 2009dq). DOE showed in its fault tree in BSC Figure B7.4-3 (2009ac) and in lower level fault trees that it accounted for the loss of confinement boundary and high winds and for a loss of HEPA-filtered exhaust from operating and standby trains. For the loss of confinement boundary, DOE identified a potential leak from a HEPA filter plenum, as shown in BSC Figure B7.4-4 (2009ab), and as a result of this leak, the failure of an operator to respond to a high radioactivity alarm as well as the failure of this alarm itself [e.g., BSC Figure B7.4-11 (2009ab)]. In addition, for the loss of confinement boundary, DOE accounted for an operator potentially opening more than one door resulting in a loss of flow through the HEPA filters, as outlined in BSC Figure B7.4-8 (2009ab). For the loss of HEPA-filtered exhaust, DOE showed failure of the operating train due to component failures, such as exhaust fan failure or tornado, backdraft, and isolation damper failures, and combined these operating train failures with standby train failure to startup and continue to operate. For exhaust fan failure, DOE showed failure of the fan itself, failure of the adjustable speed drive for the fan, and loss of AC power in the fault tree. For loss of AC power, DOE accounted for the loss of ventilation to the ITS electrical equipment and battery room in the CRCF and failure of an emergency diesel generator. For the standby train, DOE identified its unavailability due to maintenance and the failure of the standby exhaust fan to startup, as shown in BSC Figure B7.4-6 (2009ab), and continue to operate, as shown in BSC Figure B7.4-19 (2009ab). Additionally, for the failure of the standby exhaust fan to startup, DOE specified failure of the adjustable speed drive start logic and failure of the logic controller, as shown in BSC Figure B7.4-9 (2009ab).

DOE calculated a point estimate failure probability of 4.0×10^{-2} and mean failure probability of 4.5×10^{-2} for failure to maintain delta pressure in the CRCF, as outlined in BSC Figure B7.4-1 (2009ab). The associated controlling parameter in SAR Table 1.9-3 (DOE, 2009av) is 4×10^{-2}. DOE listed its dominant cut sets in BSC Table B7.4-3 (2009ab). In this table, DOE showed that the operator error associated with opening inner and outer doors is in a cut set having the greatest contribution to the failure probability. The second highest contributor was the failure of the exhaust fan in the operating train combined with a human error for which a control switch was left in the wrong position. This human error involving the control switch prevented the standby train from automatically starting up on failure of the operating train and appeared in several other dominant cut sets as well. This automatic start capability was part of DOE's surface nuclear confinement ITS HVAC exhaust system design criteria listed in SAR Table 1.2.4-4 (DOE, 2009av). DOE specified several other dominant cut sets including a single-event cut set accounting for wind speeds exceeding 48 km/hour [30 mph], diesel generator failure combined with loss of offsite power, and damper failure in one train combined with exhaust fan failure in the other train.

DOE provided its basic event probabilities and descriptions for these basic events in BSC Table B7.4-1 (2009ab). DOE included associated active component reliabilities in BSC Attachment C (2009ab) and in its active component database (i.e., file "YMP Active Comp Database_final 8 August 2008.xls") included in BSC Attachment H (2009ab). DOE's response to an NRC staff RAI (DOE, 2009fz) described how DOE quantified basic events used in its fault tree analysis. For a loss of delta pressure due to sustained wind speeds exceeding 48 km/hour [30 mph], DOE described its use of meteorological data pertaining to winds exceeding 64 km/hour [40 mph] during any 30-day period—not 48 km/hour [30 mph]. It also stated that the high wind speed is not an HVAC failure mode and does not need to be modeled. DOE also described its quantification for duct rupture with regard to one data source in which the length of the duct was needed, and explained why it removed a basic event involving the failure of Train B to start. DOE stated that it replaced the basic event by specific equipment failures. It also clarified, in its quantification of failure of the air handling unit to run, how it used data for standby air handling units regardless of whether the unit was in the standby or operating train. DOE stated that the data it selected most closely approximated the failure to run regardless of the unit being in the standby or operating train (DOE, 2009fz). In addition, DOE indicated that it included control circuitry pertaining to fan tripped detection and failure of the ITS interlock in its exhaust fan failure to start a basic event (DOE, 2009gi).

NRC Staff Evaluation: The NRC staff reviewed DOE's HVAC system failure probability quantification using the guidance in the YMRP. The NRC staff notes that DOE's use of fault tree analysis for quantifying the failure probability of the surface nuclear confinement HVAC systems is reasonable because fault tree analysis is a standard technique and DOE followed the guidance in NUREG–0492 (NRC, 1981ab). In addition, DOE reasonably accounted for human errors in its fault tree model on the basis of the descriptions it provided for human interactions with the system. Although DOE's use of component failures in the model was not transparent, in response to an NRC staff RAI (DOE, 2009dq), DOE explained how the differential pressure switch across the HEPA filter plenums [identified as ITS in SAR Figure 1.2.4-101 (DOE, 2009av)] was accounted for in its fault tree analysis. DOE stated that it encompassed this component failure into the start logic signal failure basic event in DOE (2009dq), and in DOE (2009gi), DOE identified other portions of the control circuitry, such as fan tripped detection, that it included in a different basic event. In addition, DOE relied on a radioactivity monitor to alert the operator to HEPA filter plenum leaks. DOE included this radioactivity monitor in the fault tree model and identified this radioactivity monitor as non-ITS in SAR Figure 1.2.4-101 (DOE, 2009av). Furthermore, DOE in SAR Table 1.9-1 (DOE, 2009av) identified the entire Radiation/Radiological Monitoring system as non-ITS. On the basis of this evaluation, DOE reasonably modeled the loss of delta pressure. As part of the detailed design process, DOE should confirm that the identification of ITS components and the associated nuclear safety design bases are consistent with the design.

In terms of DOE's quantification of basic events used in its fault tree analysis, the NRC staff notes that, in several cases, DOE did not describe all the failures it included in a basic event failure probability and the failure probability did not reflect the failures it did identify. For example, for the exhaust fan failure to start basic event (i.e., 060–VCTO–FAN00B–FAN–FTS) in BSC Table B7.4-1 (2009ab), DOE stated that it included control circuitry failure and interlock failure (DOE, 2009gi). The brief description for this basic event in BSC Table B7.4-1 (2009ab) did not specify either of these failures. In addition, the active component database specified motor-driven fan failure and included data source descriptions, but did not specify these failures.

Similarly, in its response to an NRC staff RAI (DOE, 2009gi), DOE stated that flow or differential pressure instruments were explicitly modeled in the HVAC system fault tree. DOE further described, in DOE (2009dq), that it included the failure of a differential pressure switch across the HEPA filter plenums in a basic event that involved failure of the Train B adjustable speed drive start logic signal (i.e., 050–VCTO–FANBASD–CTL–FOD). However, on the basis of the information provided by DOE, the NRC staff cannot determine what other signals are part of this basic event. DOE's quantification for the basic event included in its active component database used information from data sources corresponding to delta temperature, pressure, and level process logic components. However, DOE did not identify nor specify ITS, delta temperature, or level process logic components in SAR Figure 1.2.4-101 (DOE, 2009av) and, therefore, it is not apparent to the NRC staff whether they are applicable to this basic event.

In reviewing DOE's response to an NRC staff RAI (DOE, 2009fz), it is not transparent to the NRC staff how the data DOE selected for the air handling units' failure to run apply to the environment being modeled. However, the NRC staff notes that DOE's controlling parameter of failure probability of 4×10^{-2} in SAR Table 1.9-3 (DOE, 2009av) is reasonable for a surface nuclear confinement ITS HVAC system for use in PCSA because this failure probability is consistent with that of a surface nuclear confinement ITS HVAC system commonly used in the nuclear industry. In addition, DOE did include conservatism in its model. For example, BSC Section E6.0.2.3.4 (2008ac) shows that DOE included a human failure probability that did not credit the operator noticing HEPA filter leaks. Therefore, on the basis of this evaluation, DOE's quantification for the confinement pivotal event is reasonable.

The NRC staff does recognize that the detailed designs for ITS structures and systems will progress over time. As a result, fault tree modeling of ITS components within an ITS system may also change as the design evolves. As part of the detailed design process, DOE should confirm that the fault tree modeling specifies the components used to quantify its basic events.

2.1.1.4.3.3.2.2 Moderator Intrusion Control

DOE identified the moderator control areas in SAR Table 1.4.3-2 and discussed moderator intrusion in BSC Section 6.2.2.9 (2008ac), BSC Section 6.2.2.10 (2008bq), 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 Figure B9.5-1 (2008ac)]. DOE identified water from fire suppression systems, water from building service piping, and lubricating oil from overhead hydraulic equipment as potential moderator sources.

DOE 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. DOE identified these systems as ITS because of their role in preventing accidental moderator intrusion; a complete evaluation of the DIPA system is provided in TER Section 2.1.1.7.3.1.6. DOE described the 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 by spurious operation of the system allowing trapped water to be ejected into the moderator control area.

Other water sources, such as domestic water pipes, have failure probabilities based on historical data, which provide a failure rate per unit length of pipe. DOE used these data to derive a probability for other water sources introducing a moderator and included these potential sources in the moderator fault tree.

DOE 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 probability of moderator intrusion is a pivotal event in sequences that have potential moderator sources following a canister breach. The presence of moderator determines whether an important to criticality end state is realized. DOE 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. DOE 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 5.0×10^{-7} to 1.0×10^{-6}. These probabilities were facility dependent and primarily based on the number of sprinkler heads present in moderator-controlled areas. DOE also indicated that the failure probability of the overall system (including mechanical and human-induced failures) was actually 2.0×10^{-7} (DOE, 2009fr).

For moderator intrusion to occur, DOE stated that a canister has to be breached and is capable of accepting moderator. DOE assumed that breached canisters would be susceptible to moderator intrusion for a maximum of 30 days after a breach, stating that 30 days was a sufficient window to allow canister containment to be reestablished or other mitigation procedures for moderator control to be in place.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's evaluation of reliability of the moderator control system using the guidance in the YMRP. DOE's use of the failure probabilities of 1.0×10^{-6} to 5.0×10^{-7} as the design bases is conservative because DOE determined that the actual failure probability of the DIPA system (including mechanical and human-induced failures) was 2.0×10^{-7} (DOE, 2009fr).

DOE's determination that introduction of lubricating oil into a breached canister is unlikely is reasonable because this failure mode would involve a breach of the crane gearbox, a concurrent leak of the containment pan, and presence of a breached canister so lubricating oil would intrude from above. DOE reasonably indicated that this source of moderator can be screened out as an insignificant contributor to criticality.

Furthermore, DOE reasonably indicated (DOE, 2009fg) that the moderator intrusion event sequence was governed by the initial canister reliability. According to DOE, the probability of having a breached canister available for moderator introduction was approximately 1.0×10^{-4} (near the Category 2 limit). Once the combined probability of a second pivotal event (such as spurious sprinkler system operation, a leaking hydraulic unit, or an actual fire event) was introduced, DOE determined, and the NRC staff confirmed, that the overall event sequence probability quickly drops to beyond Category 2.

The NRC staff notes that DOE used conservative probabilities for assessing moderator intrusion into a breached canister. The conservatism results from the DIPA failure probability used in the SAPHIRE model (taken from the nuclear design basis) that was higher than actual failure probabilities expected for these systems (on the basis of a detailed fault tree analysis). The NRC staff also notes that screening out lubricating oil as a potential moderator [e.g., BSC Section 6.2.2.9.2 (2008ac)] is reasonable because consideration of lubricating oils as potential sources in the PCSA fault trees is conservative.

2.1.1.4.3.4 Event Sequence Quantification and Categorization

DOE discussed quantification and categorization of event sequences for its preclosure operations. DOE provided the results of event sequence quantification, categorization, and the end states in SAR Tables 1.7-7 through 1.7-18. The NRC staff considered whether DOE's evaluation of probability of occurrence of event sequences and categorization of event sequences (i) implemented the ESDs in the SAPHIRE model and (ii) used data to quantify the event sequences.

In its review of event sequence quantification and categorization, the NRC staff evaluated the the approaches for (i) categorization of event sequences in TER Section 2.1.1.4.3.1 and (ii) development of ESDs in TER Section 2.1.1.4.3.2. The NRC staff reviewed the data used in the event sequence quantification, which included (i) probability of initiating events in TER Section 2.1.1.3.3.2.3, (ii) throughput numbers of the waste form in TER Section 2.1.1.2.3.6.1, and (iii) reliability data of active and passive systems in TER Section 2.1.1.4.3.3.

Consistent with the organization of other sections in this chapter, the remainder of this section is organized into three subsections: Internal Events (TER Section 2.1.1.4.3.4.1), Seismic Events (TER Section 2.1.1.4.3.4.2), and Fire Events (TER Section 2.1.1.4.3.4.3).

2.1.1.4.3.4.1 Internal Events

The NRC staff's review of DOE's event sequence quantification and categorization for internal events focused on canister handling in surface facilities and intrasite operations, wet handling, and subsurface operations. These three topics are addressed separately in the next three subsections.

2.1.1.4.3.4.1.1 Canister and Cask Handling Operations

DOE listed the internal event sequences for the IHF, RF, CRCF, and intrasite operations in SAR Tables 1.7-7, 1.7-9, 1.7-11, 1.7-13, and 1.7-15, respectively. DOE reported event sequence quantification results in SAR Table 1.7-11 with a cutoff limit on the number of occurrences having a mean value of 10^{-7}. In addition, SAR Sections 1.7.5.1 through 1.7.5.3 summarized the categorization analysis. DOE provided additional information regarding the data used, modeling including the SAPHIRE models and supporting files, and calculation results in BSC (2008ab,ac, ao,as,at,au,bd,be,bo,bq). The NRC staff reviewed whether the data used in the event sequence analysis are consistent with the initiating event frequency, throughput values, and pivotal events. To evaluate DOE's implementation of ESDs in SAPHIRE and categorize event sequences, the NRC staff focused its review on the basis of end states: exposure and radionuclide release.

Data Used for Event Sequence Analysis

DOE's analysis of event sequence involved modeling of ESDs for each waste form. Each ESD, which contained a group of initiating events referred to as "small bubbles" (BSC, 2008ab), was modeled using initiator event trees and system event trees to examine the response of the SSCs providing containment, shielding, confinement, and criticality control safety functions resulting in several end states. DOE modeled the ESDs in SAPHIRE in an integrated fashion linking the initiator event tree, the system event trees, and the associated fault trees. DOE determined the frequencies and categorized each end state in an ESD by aggregating frequencies for the corresponding end states from each group of initiating events. The input data used for event sequence analyses were (i) initiating event frequency, (ii) throughput values of waste containers, and (iii) probability values in pivotal events. The data used for the event sequence analyses for the CRCF, WHF, IHF, RF, and intrasite operations were discussed in BSC Section 6 (2008ac,as,au,be,bq).

The NRC staff reviewed the data used for initiating event frequency, throughput values in initiator event tree, the passive probability of failure of waste containers in the containment and shielding pivotal events, and probability of failure of confinement and moderator control system. The NRC staff reviewed the data files for SAPHIRE models provided in BSC Appendix H (2008ac,as,au,be,bq) to evaluate DOE's event sequence analysis. The NRC staff focused its review on the categorization, especially when event sequence frequencies are close to Category 1 or Category 2 limits, and event sequences driven by high throughput numbers. Therefore, the NRC staff performed risk-informed and performance-based review and concentrated its review on structural challenges and direct exposure in the CRCF where the throughput values of waste containers are higher than in other facilities.

Initiating Events

DOE quantified each initiating event ("small bubble") in the ESD using fault tree analysis. The input data were used for the basic event of the fault tree analyses. The data used for the initiating event quantification for CRCF, WHF, IHF, RF, and intrasite operations in the SAPHIRE fault tree models were discussed in BSC Sections 6.2 and 6.3 (2008ac,as,au,be,bq).

NRC Staff Evaluation: The NRC staff reviewed the data DOE used for initiating events to evaluate internal event sequence quantification and categorization related to structural challenges and direct exposure using the guidance in the YMRP. The NRC staff reviewed the identification of initiating events and grouping of initiating events in ESDs in TER Sections 2.1.1.3.3.2.1 and 2.1.1.3.3.2.2.1 where the NRC staff notes that DOE's identification and grouping of initiating events are reasonable. In addition, evaluation of initiating event frequency quantification, including the data used for fault tree modeling, is reviewed in TER Section 2.1.1.3.3.2.2.2, where the NRC staff notes that DOE's event frequency quantification is reasonable. Therefore, on the basis of the evaluation in TER Sections 2.1.1.3.3.2.1 and 2.1.1.3.3.2.2.2, DOE's consideration of initiating events for event sequence quantification is reasonable.

Throughput Numbers

The number of waste form units or throughput was used for quantification of event sequences. The throughput of canisters, casks, and waste packages to be handled over the preclosure period in each facility was provided in SAR Tables 1.2.1-1 and 1.7-5. The throughput numbers include the number of handling operations associated with the waste form. In some instances, DOE factored in the number of operations in the initiating event fault tree.

NRC Staff Evaluation: The NRC staff reviewed the throughput data DOE used for event sequence quantification and categorization for internal events related to structural challenges and direct exposure using the guidance in the YMRP. The NRC staff notes that throughput numbers played a role in the CRCF event sequence quantification analysis and categorization. The throughput reflects the number of waste forms that will be handled over the preclosure period including the number of operations the waste containers will undergo during the handling process. The NRC staff reviewed DOE's throughput calculation in TER Section 2.1.1.2.3.6.1 and notes in that section that the throughput numbers listed in SAR Table 1.7-5 are reasonable. On the basis of comparison of the throughput numbers in SAR Table 1.7-5 and the throughput DOE used, the throughput numbers used in the SAPHIRE modeling for the CRCF, IHF, RF, and WHF are consistent with those listed in SAR Table 1.7-5.

The NRC staff reviewed event sequence CRCF–ESD09–TAD (BSC, 2008ac) to assess the throughput number for handling TAD canisters in the CRCF and consistency with the operational process described and notes that the throughput number used for TAD during transfer with the CTM is 15,120 and is consistent with the value in SAR Table 1.7-5 and the SAPHIRE model. The throughput included the TAD canister transfer planned for the CRCF and RF (BSC, 2007bh). Consistent with the review presented in TER Section 2.1.1.2.3.6.1, the NRC staff notes that DOE's approach to include TAD canister transfer operations in CRCF and RF facilities is reasonable for quantification and categorization of event sequences for similar waste forms and operations. However, DOE did not include the transfer of 1,165 TAD canisters involved in similar transfer operations using the CTM from a site transfer cask to the AO for export from the WHF. The NRC staff determined that an approximately 8 percent increase (i.e., the 1,165 transfers of TAD canisters) in throughput in CRCF–ESD09–TAD (BSC, 2008ac) would not change the event sequence categorization, because this increase does not cause a need to recategorize this event sequence. Therefore, DOE's use of throughput values and the numbers of operations are reasonably included in the event sequence quantification.

Pivotal Events

DOE calculated passive reliability of the canisters and casks and the probability of loss of containment and loss or degradation of shielding of the SSCs used in the containment and shielding pivotal events in the system response event trees. DOE relied on canisters of all waste forms to provide containment in the canister pivotal event (e.g., in the RESPONSE–CANISTER system event tree). In the WHF, DOE also credited the containment function of the transportation cask containing bare SNF. DOE evaluated the probability of loss or degradation of shielding under structural challenges for the cask system (e.g., transportation cask, AOs, and STC) and used these probability values as input in the shielding pivotal event.

DOE evaluated the probability of loss of confinement by developing a fault tree model of the HVAC system. The fault tree was linked to the confinement pivotal event in the system response tree. Similarly, the probability of loss of moderator intrusion control was evaluated by developing fault tree modeling for the failure of SSCs that could lead to moderator intrusion and linked the fault tree to the moderator pivotal event in the system response tree. The data utilized for the initiating event quantification for CRCF, WHF, IHF, RF, and intrasite operations and used in SAPHIRE fault trees and event trees as input were discussed in BSC Sections 6.2 and 6.3 (2008ac,as,au,be,bq).

NRC Staff Evaluation: The NRC staff reviewed the data DOE used for pivotal events for internal event sequence quantification and categorization related to structural challenges and direct exposure using the guidance in the YMRP. The NRC staff reviewed event sequence development in TER Section 2.1.1.4.3.2.1 and notes that the safety functions of the SSCs relied on to provide preventive and mitigative functions in the pivotal events are consistent with site information, facility design, operations, and human actions. The NRC staff reviewed the probability of containment and shielding failure of passive components in TER Section 2.1.1.4.3.3.1.1. DOE relied on TAD, HLW, DOE standardized canisters, DPCs, and waste packages to provide the containment function, and on transportation casks and AOs for the shielding function for structural challenges to the waste form during handling operations. On the basis of the evaluations in TER Section 2.1.1.4.3.3.1.1, the NRC staff notes that DOE provided information to support its reliability estimate of all canisters and casks and considered failure probability of containers in the “Containment” and “Shielding” pivotal events in the system event trees for event sequence quantification.

The NRC staff reviewed the reliability of HVAC system providing a containment function in TER Section 2.1.1.4.3.3.2.1 and, on the basis of the evaluation in that section, the NRC staff notes that DOE used data reasonably for the probability of loss of confinement in the confinement pivotal event for event sequence quantification. In addition, the NRC staff reviewed the loss of moderator intrusion control in TER Section 2.1.1.4.3.3.2.2.

In summary, the NRC staff notes that DOE reasonably quantified the pivotal events for the canister and cask handling operations using data for the probability of loss of moderator intrusion control in the moderator pivotal event.

Exposure to Workers

DOE stated that the event sequence frequencies that could potentially cause direct exposure to workers (SAR Table 1.7-11) are Category 2 or beyond Category 2 and thus are not included in the radiation dose calculation. Categorization of event sequences in DOE’s analysis can be classified into three groups: (i) direct exposure, (ii) loss or degraded shielding, and (iii) shield intact.

Direct Exposure

DOE identified event sequence frequency for exposure to workers during various waste handling operations at the CRCF, IHF, RF, and WHF. For example, the initiating events associated with cask preparation activities, CTM operations, and waste package closure and exporting in ESDs CRC–ESD–17 through 19 result in direct exposure to workers and were not transferred to system response event trees. On the basis of the frequency of the initiating events, DOE categorized the event sequences as Category 2.

NRC Staff Evaluation: The NRC staff reviewed DOE's event sequence quantification and categorization for internal events related to direct exposure using the guidance in the YMRP. The NRC staff reviewed the event sequences resulting from direct exposure identified for the CRCF (e.g., CRC–ESD–17, 18, 19). DOE identified similar event sequences for the IHF, RF, and WHF. As shown in SAR Table 1.7-11, the mean value of event sequence frequency for direct exposure resulting from operations with TAD, DOE standardized, waste package, and HLW ranged from 3×10^{-3} to 0.3. In the absence of pivotal events, the initiating event probability and the throughput values drive the event sequence frequencies close to the Category 1 threshold. On the basis of the NRC staff's review described in TER Section 2.1.1.3.3.2.2.2, the NRC staff notes DOE reasonably quantified and categorized these direct exposure event sequences because DOE reasonably quantified the initiating event probabilities. In addition, the throughput values used in these event sequence analyses were reasonable because they are consistent with those given in SAR Table 1.7-5.

Loss or Degraded Shielding

DOE evaluated exposure to workers from loss or degradation of shielding caused by structural challenges to the waste form containers. For example, during operations involving TAD canisters in the CRCF, DOE relied on the shielding capabilities of the transportation cask, AO, STC, CTM bell, or WPTT. DOE examined the loss or degradation of shielding, assuming no canister breach, under the pivotal event shielding in the system response event trees for estimating probability of end state DE–SHIELD–DEGRADE. DOE relied on the reliability of the SSCs to provide shielding functions and categorized these event sequences as Category 2 or beyond Category 2.

NRC Staff Evaluation: The NRC staff reviewed DOE's internal event sequence quantification and categorization related to loss or degraded shielding using the guidance in the YMRP. On the basis of the review, the NRC staff notes that initiating events were greater than one by considering initiating event frequency and throughput value for the preclosure period for some event sequences. For example, during receipt operations at the CRCF, a transportation cask containing TAD canisters, as shown in BSC Figure G–7 (2008ac) (CRCF–ESD01–TAD), could result in about 34 railcar collisions during the preclosure period. Those containing HLW canisters, as outlined in BSC Figure G–5 (2008ac) (CRCF–ESD01–HLW), resulted in about 9.5 Railcar/Truck Trailer (RC/TT) collisions. Similarly, side impacts of a transportation cask during removal of impact limiters, upending and transfer operations, and cask preparation activities (CRCF–ESD03–TAD and CRCF–ESD03–HLW) resulted in more than 41 events with TAD canisters inside and 11 events with HLW canisters inside. DOE's postulated incidents of the AO with TAD canisters inside ESDs (CRC–ESD–02, CRC–ESD–12, CRC–ESD–14, and CRC–ESD–16) showed about 39 collisions and 33 side impacts. DOE relied on the passive reliability for shielding capabilities of the transportation casks and the AO in pivotal event SHIELD in response trees RESPONSE–TCASK1 and RESPONSE–AO to drive the event sequences below the Category 1 threshold. The NRC staff notes that the low probability of loss of shielding of a transportation cask (10^{-8} for collision and 10^{-5} for derailment) and AO (10^{-5} for collision and side impact) reduced the overall event sequence frequency to Category 2 or beyond Category 2. Therefore, DOE reasonably evaluated and categorized the loss or degradation of shielding event sequences.

Shielding Intact

DOE identified event sequences associated with transfer of canisters inside the CTM shield bell and transfer of the waste package in the WPTT in ESDs (CRC–ESD–05, 08, 09, 10, 11, 13, and 15). These event sequences were caused by structural challenges to the waste canisters. DOE considered no shielding failure and used a value of zero probability in the pivotal event for response trees RESPONSE–WP1, RESPONSE–WP2, and RESPONSE–CANISTER1. Thus, DOE categorized the event sequences associated with exposure or direct radiation from loss or degradation of shielding to the workers as beyond Category 2.

NRC Staff Evaluation: The NRC staff reviewed DOE's quantification and categorization of internal event sequences for the conditions with shielding to remain intact using the guidance in the YMRP. The NRC staff reviewed selected event sequences to assess how they were categorized. Considering the number of waste packages with TAD and the collision frequency of WPTT could result in 24 collisions in ESD10–WP–TAD–SEQ2–DE. However, DOE determined that the shielding would remain intact, as described in BSC Section 6.3.2.5 (2008ac), because the WPTT travels at a slow speed {0.1 m/s [0.33 ft/s]} and an impact was not expected to damage the shielding (DOE, 2009ge). The NRC staff notes that, at a collision speed of 0.1 m/s [0.33 ft/s], WPTT shielding will not be damaged. DOE considered shielding to remain intact in the CRCF (e.g., CRC–ESD–09) because it indicated that the CTM was surrounded by shield walls and doors which would not be affected by structural challenges from internal random initiating events (BSC, 2008ac). DOE's statement that CTM shielding remained intact in the CRCF when the waste canister inside the CTM was structurally challenged is reasonable because the CTM was surrounded by shield walls and doors, which would not be affected by these structural challenges. On the basis of the evaluation presented in this section, DOE reasonably categorized the event sequences resulting from loss of shielding as Category 2 or beyond Category 2.

Radionuclide Release

DOE's results for event sequences depicted in SAR Table 1.7-11 for the CRCF showed three Category 2 event sequences, while the remaining event sequences are categorized as beyond Category 2. The identified Category 2 event sequences were associated with structural challenges to HLW and TAD canisters during transfer by the CTM. End states of filtered (ESD09–HLW–SEQ3–RRF) and unfiltered radionuclide releases (ESD09–HLW–SEQ5–RRU) involved breach of two sealed HLW canisters, while event sequence filtered release (ESD09–TAD–SEQ–RRF) resulted from breach of one TAD canister. Similarly, in SAR Table 1.7-7, one Category 2 event sequence involving an HLW canister was identified at the IHF. It was caused by structural challenges during transfer by the CTM. The remaining event sequences from structural challenges were categorized as beyond Category 2. At the RF (SAR Table 1.7-9) and during intrasite operations (SAR Table 1.7-15), the event sequences were beyond Category 2 from the internal events caused by structural challenges. In SAR Table 1.7-13, DOE identified several Category 2 event sequences for the WHF involving structural challenges (e.g., to the transportation casks with uncanistered SNF, DPCs during preparation activities and cutting operations, TAD canisters in STCs during transfer from pool and drying). DOE evaluated dose consequences in accordance with the end states for filtered and unfiltered radionuclide releases. DOE assigned dose consequence designators for the Category 2 event sequences involving structural challenges. The NRC staff evaluated DOE's dose calculations in TER Section 2.1.1.5.

NRC Staff Evaluation: The NRC staff reviewed DOE's quantification and categorization of selected internal event sequences related to radionuclide release using the guidance in the YMRP. These event sequences were associated with CRCF–ESD–09 involving structural challenges to TAD canisters during transfer from the AO to the waste package. The initiating events in the group were impact associated with lid removal, canister drop from operational height, impact to canister due to conveyance movement, side impact object drop on canister, canister drop inside the CTM shield bell, and canister drop above operational height, as outlined in BSC Figure A5–45 (2008ac). The response tree RESPONSE–CANISTER1 was shown in BSC Figure A5–21 (2008ac). The NRC staff reviewed results of the end states under the "Reports" menu in the SAPHIRE data file and notes the following end state event sequence frequencies: 1.14×10^{-4} for ESD09–TAD–SEQ3–RRF (filtered radionuclide release), 3.87×10^{-6} for ESD09–TAD–SEQ5–RRU (unfiltered radionuclide release), and 4.19×10^{-10} for ESD09–TAD–SEQ6–RRC (radionuclide release important to criticality). On the basis of the NRC staff's review, the cut set results for filtered release, ESD09–TAD–SEQ3–RRF, showed that the spurious movement of the CTM leading to shear failure of a TAD canister was the dominant contributor to the event sequence frequency, contributing about 93 percent. This was followed by an object drop on a canister and a canister drop, contributing about 6.5 percent of the overall frequency of occurrence. DOE used probability of canister failure as 10^{-5} for vertical drop. The NRC staff notes that DOE used the probability of canister failure as 10^{-5} for an object drop onto a canister and DOE conservatively did not take credit for the TAD canister for shear loading caused by untimely movement of conveyance. DOE determined that the unfiltered radionuclide release end state, ESD09–TAD–SEQ5–RRU, similar to the cut sets for the filtered release, had an additional probability of failure of an HVAC system failure in the cut set, resulting in an event sequence frequency categorized as beyond Category 2. On the basis of the evaluation presented in this paragraph, DOE's categorization of CRCF–ESD–09 is reasonable.

The NRC staff reviewed the event sequences for radionuclide release that resulted from high frequency initiating events evaluated in this TER section under loss of shielding or degraded shielding. DOE's analysis in the SAPHIRE data file showed that the filtered radionuclide release end state, ESD01–TAD–SEQ4–RRF, caused by structural impacts from railcar derailment and collision on a transportation cask containing TAD canisters at the CRCF during the receipt operations, was beyond Category 2. This event sequence was on the order of 10^{-7} mainly because of the probability of failure of 10^{-8} for the transportation cask providing containment. As discussed in TER Section 2.1.1.4.3.3.1.1, the probability of loss of containment of the transportation cask with a representative canister inside for low velocity impacts such as collisions is reasonable. As a result, the categorization of ESD01–TAD–SEQ4–RRF is reasonable. For structural challenges to the transportation cask with a TAD canister inside during removal of impact limiters, upending and transfer operations, and cask preparation activities at the CRCF, the filtered radionuclide release end state, ESD03–TAD–SEQ4–RRF, according to DOE, had an expected number of occurrences of a 2×10^{-5} event sequence frequency and was categorized as beyond Category 2 (SAR Table 1.7-11). For the initiator event tree CRCF–ESD03–TAD, as shown in BSC Figure A4–14 (2008ac) for this end state, the structural challenges to the transportation cask were caused by several initiating events including drop of cask, tipover, and side impact. On the basis of its review of end state results in the SAPHIRE data file, the NRC staff notes that the filtered radionuclide release resulted from loss of containment of the transportation cask. The failure probability for drop, drop on, and tipover was 10^{-5} and for side impact was 10^{-8}. Although the initiating event frequency of side impact was greater than one, as discussed earlier, the main contributors to the overall frequency were the tipover, drop on, and drop of transportation cask, resulting in the 2×10^{-5} event frequency. On the basis of its review in TER Section 2.1.1.4.3.3.1.1, the NRC staff notes that the use of the probability of failure of transportation casks with a representative canister inside is reasonable. Therefore, DOE's quantification and categorization of CRCF–ESD03–TAD is reasonable.

2.1.1.4.3.4.1.2 Wet Handling Operations

DOE described the WHF event sequence analysis in SAR Section 1.7.5.4. In addition, DOE included ESDs in BSC Attachment F (2008bo) and event trees in BSC Attachments G and A (2008bo,bq). Also, DOE included fault tree models in BSC Attachment B (2008bq) and SAPHIRE analyses and supporting files (e.g., data quantification files such as "YMP Active Comp Database.xls") in BSC Attachment H (2008bq). Finally, DOE provided the event sequence quantification summary tables in BSC Attachment G (2008bq).

Wet handling operations are activities (e.g., the transfer of fuel assemblies in the WHF pool) that are performed in the WHF involving uncanistered SNF. The WHF is the only surface facility that handles uncanistered SNF, and therefore the NRC staff evaluates event sequence quantification and categorization associated with uncanistered SNF handling in this TER section. The NRC staff reviewed event sequences associated with the transfer of fuel assemblies in the pool and the transfer of casks to and from the pool as well as event sequences involving cask preparation activities. The NRC staff used a vertical slice approach to focus on event sequences that were more risk significant or resulted in frequencies close to a categorization boundary. In this section, the NRC staff evaluates the quantification of pivotal events for selected event sequences, whereas the NRC staff evaluates the quantification of initiating events in TER Section 2.1.1.3.3.2.3.1.

SAR Table 1.7-13 showed a total of 48 event sequences that DOE identified with (i) no Category 1, (ii) 22 Category 2, and (iii) 26 beyond Category 2 event sequences. DOE performed bounding consequence analyses for 13 Category 2 event sequences. One of these involved a structural challenge to SNF assemblies, five involved a structural challenge to a transportation cask with uncanistered SNF assemblies, two involved a thermal challenge to a transportation cask with uncanistered SNF assemblies, two involved a structural challenge to a DPC, and three involved a structural challenge to a TAD canister.

For the transfer of fuel assemblies to a TAD canister in the pool (i.e., WHF–ESD22–FUEL), DOE showed its ESD in BSC Figure F–22 (2008bo) and its event trees in BSC Figures A5–36 and A5–37 (2008bq). It described these event trees in BSC Section A4.22 (2008bq). Each group of initiating events from the ESD in BSC Figure F–22 (2008bq) mapped to a branch of the initiator event tree in BSC Figure A5–36 (2008bq). In addition, failure to maintain reasonable boron concentration from the ESD was mapped to a pivotal event in DOE's response tree shown in BSC Figure A5–37 (2008bo). The end states in BSC Figure F–22 (2008bq) of the ESD specified a radionuclide release, one of which was also important to criticality. These end states were also shown in DOE's response tree as unfiltered gaseous radionuclide release end states, one of which was also important to criticality.

For the movement of a transportation cask from the preparation station to the pool ledge (i.e., WHF–ESD20–CSNF), DOE showed its ESD in BSC Figure F–20 (2008bo) and its event trees in BSC Figures A5–3, A5–31, and A5–32 (2008bq). It described these event trees in BSC Section A4.20 (2008bq). DOE 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. Each group of initiating events from the ESD mapped to two branches of the initiator event tree, as shown in BSC Figure A5–32 (2008bq), corresponding to the initiating event occurring over the pool or over the floor. For those initiating events occurring over the pool, a failure of the cask and the failure to maintain reasonable boron concentration were shown on the ESD and mapped to pivotal events in DOE's response tree, as outlined in BSC Figure A5–31 (2008bq). The end states of the ESD specified a radionuclide release, one of which was also important to criticality. These end states were also shown in DOE's response tree as unfiltered gaseous radionuclide release end states, one of which was also important to criticality. 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, failure to maintain confinement (i.e., HVAC failure), and failure to exclude moderator were mapped to pivotal events in DOE's response tree, as described in BSC Figure A5–3 (2008bq). In addition to end states involving radionuclide release, DOE specified a direct exposure in its ESD. These end states were shown in DOE's response tree as filtered and unfiltered radionuclide releases, as well as filtered and unfiltered radionuclide releases important to criticality. The response tree also included the direct exposure end state from the ESD accounting for degradation of the shielding with the cask remaining intact.

DOE described the movement of a cask from the pool ledge to the TAD canister closure station (i.e., WHF–ESD24–TAD). Similar to WHF–ESD20–CSNF, DOE mapped initiating event groups and pivotal events to branches in its associated event trees and reflected the end states from the ESD in its response trees. Because WHF–ESD24–TAD was similar to WHF–ESD20–CSNF, DOE's mapping of the ESD to event trees is not described further in this section.

For the transfer of casks to and from the pool (i.e., WHF–ESD20–CSNF and WHF–ESD24–TAD), DOE credited the cask for maintaining containment in the event of a drop, tipover, or impact over the pool or over the floor. There was one exception: DOE did not credit the cask for maintaining containment for a drop above the operational height over the pool. DOE clarified in its response to an NRC staff RAI (DOE, 2009fk) how the cask maintains containment with a lid held in place with a minimum number of installed fasteners. DOE indicated that it performed the analyses to determine the minimum number of installed fasteners required. DOE described the analyses for an inverted tipover of the cask and indicated that it will perform additional analyses during detailed design (DOE, 2009fk).

For transportation cask preparation activities (i.e., WHF–ESD16–CSNF), DOE showed its ESD in BSC Figure F–16 (2008bo) and its event trees in BSC Figures A5–26 and A5–27 (2008bq). It described these event trees in BSC Section A4.16 (2008bq). Each group of initiating events from the ESD mapped to a branch of the initiator event tree, as shown in BSC Figure A5–26 (2008bq). In addition, failure to maintain confinement (i.e., HVAC failure) and failure to exclude moderator were mapped to the pivotal events shown in DOE's response tree, as outlined in BSC Figure A5–27 (2008bq). The end states of the ESD specified a radionuclide release, one of which was also important to criticality. These end states were also shown in DOE's response tree as filtered and unfiltered radionuclide releases as well as filtered and unfiltered radionuclide releases important to criticality.

NRC Staff Evaluation: The NRC staff reviewed DOE's WHF event sequence quantification and categorization using the guidance in the YMRP. The NRC staff notes that DOE's event trees reflect the ESDs in terms of the higher level groups of initiating events (or small bubbles) that DOE showed on the ESDs, and the event trees reflect the pivotal events and end states from the ESDs. On the basis of the NRC staff's evaluation in TER Section 2.1.1.3.3.2.3.1, the NRC staff notes that DOE reasonably quantified initiating events. The NRC staff evaluated pivotal events involving failure to maintain containment (e.g., failure of a cask following a drop), failure to maintain confinement (i.e., HVAC failure), failure to maintain reasonable boron concentration in the pool, and failure to exclude moderator from entering a cask. These pivotal events are discussed next.

DOE quantified the confinement pivotal event consistent with the initiating events. The NRC staff also notes that the confinement pivotal event accounted for radionuclide release from the cask sampling and cooling process associated with cask preparation activities (i.e., WHF–ESD16–CSNF). In addition, on the basis of the NRC staff's review of HVAC in TER Section 2.1.1.4.3.3.2.1, the NRC staff notes that the confinement pivotal event was reasonably quantified.

However, on the basis of the NRC staff's evaluation of passive systems (e.g., casks) in TER Section 2.1.1.4.3.3.1, the NRC staff notes that the reliability of the transportation cask and the STC in the containment pivotal event was not quantified consistent with a tipover initiating event (e.g., WHF–ESD20–CSNF and WHF–ESD24–TAD). DOE quantified the tipover of a cask onto its side, but not the inverted tipover described in its response to an NRC staff RAI (DOE, 2009fk). However, DOE stated that it will perform analyses for drop and tipover scenarios for transportation casks and STCs (while holding uncanistered fuel, unsealed DPCs, and unsealed TAD canisters) to maintain containment with a minimum number of bolts left in place (DOE, 2009fk) as part of the detailed design activities. In other words, DOE identified the safety function for the transportation casks and STCs subject to the inverted tipover is to maintain containment and intended to show that this safety function could be achieved by performing analyses for drop and tipover scenarios. The NRC staff notes that this safety function provides a reasonable basis for DOE to categorize the inverted-tipover-related event sequences as beyond Category 2. Because DOE defined the nuclear safety design bases, including safety functions for transportation casks and STCs defined in the SAR, it is reasonable for DOE to perform analyses for drop and tipover scenarios for transportation casks and STCs as part of detailed design activities. DOE stated it would perform these analyses as part of the detailed design process (DOE, 2009fk).

On the basis of the evaluation performed in TER Section 2.1.1.3.3.2.7.5 related to criticality hazards, the NRC staff notes that maintaining reasonable boron concentration is quantified and consistent with the initiating event because it was quantified in terms of maintaining reasonable boron concentration such that a criticality condition cannot be created in the pool. For failure to exclude moderator from entering a cask, DOE specified a probability of failure of zero. On the basis of the NRC staff's evaluation in TER Section 2.1.1.3.3.2.7.5, the NRC staff notes that this pivotal event is quantified and consistent with the initiating events because borated pool water is used to fill the cask, other water sources cannot be connected to the fill water piping, and it is not credible that sufficient moderator could be introduced through a broken line (e.g., a sample line) to result in criticality.

On the basis of the review in this section and the evaluations in TER Sections 2.1.1.3.3.2.3.1 and 2.1.1.4.3.3.2.1, DOE reasonably quantified initiating and pivotal events and pivotal events are quantified consistent with initiating events. In addition, event sequences are reasonably categorized because event trees reflect ESDs, initiating and pivotal events are quantified, and pivotal events are consistent with initiating events.

2.1.1.4.3.4.1.3 Subsurface Operations

DOE provided information in SAR Section 1.7.5.6 and BSC Table G–1 (2008bk) regarding the quantification and categorization of potential event sequences associated with subsurface operations. DOE identified potential Category 2 and beyond Category 2 event sequences (SAR Tables 1.7-17 and 1.7-18) and stated that the Category 2 event sequences will not lead to any release. Also, DOE stated that although the Category 2 event sequences may result in radiation exposure, the potential dose to a member of the public is insignificant because of the large distance to the potential offsite receptors.

The event sequences that may result in direct exposure [labeled "SSO–ESD04" in BSC Table G–1 (2008bk)] are (i) inadvertent entry into an emplacement drift, (ii) worker proximity to a loaded TEV, (iii) inadvertent opening of a loaded TEV door, and (iv) loss of movement of a loaded TEV.

DOE relied on administrative controls to assign a beyond Category 2 frequency to the event sequence related to inadvertent entry into an emplacement drift. DOE stated that inadvertent entry into an emplacement drift loaded with waste packages will be prevented by means of access control using locked doors, interlocks, and a system of alarms (DOE, 2009ed). According to DOE, the emplacement drift access door cannot be opened from underground without the assistance of the surface control room operator; the access door will be monitored when open, and an emergency escape hatch within the access door only allows exit from inside the drift when the access door is locked.

DOE relied on administrative controls to assign a beyond Category 2 frequency to the event sequence related to worker proximity to a loaded TEV. DOE stated that worker proximity to a loaded TEV will be controlled by requiring a radiological work permit and by using cameras and alarms on the TEV to scan the surrounding area for nearby personnel and warn the operator and the personnel accordingly (DOE, 2009ed).

DOE estimated approximately 10^{-3} occurrences of the event sequence related to inadvertent opening of a loaded TEV door during the preclosure period using a fault tree analysis of related TEV components. DOE stated that the event sequence was Category 2. DOE stated that although this event sequence could result in radiation exposure, the potential dose to a member of the public is insignificant because of a large distance to potential offsite receptors.

DOE estimated 8.5 potential occurrences of loss of movement of a loaded TEV, but assigned a zero frequency of direct exposure because of a design requirement that the TEV shielding be able to sustain the thermal load for all waste package loadings, as described in BSC Section B1.4.5.3 (2008bj). DOE provided analysis results indicating that the operating temperature limit for the TEV would not be exceeded even if the TEV remained stationary for up to 30 days while loaded with a waste package of bounding thermal load (DOE, 2009ed).

NRC Staff Evaluation: The NRC staff reviewed DOE's quantification and categorization of the event sequence frequencies for subsurface operations except event sequences initiated by fire, which are reviewed in TER Section 2.1.1.4.3.4.3, using the guidance in the YMRP. The NRC staff notes that the administrative controls DOE described would prevent or mitigate the event sequences of "inadvertent entry into an emplacement drift" and "worker proximity to a loaded

TEV." The NRC staff notes that DOE's frequency categorization of "inadvertent opening of a loaded TEV door" is reasonable on the basis of an evaluation of the TEV design in TER Section 2.1.1.7.3.5.1. The NRC staff's evaluations in TER Section 2.1.1.7.3.5.1 on TEV include that (i) the design criteria for the design bases are reasonable, (ii) DOE's design methodology is reasonable, and (iii) DOE's design and design analysis for protection against tipover, runaway, derailment, and waste package ejection are reasonable. Also, the NRC staff notes, on the basis of its review of the surface facilities layout, that the potential dose to a member of the public due to the event sequence is insignificant because of a large distance to potential offsite receptors. Furthermore, direct exposure due to loss of TEV movement will be prevented or mitigated because DOE's analysis showed the TEV is not likely to overheat. Therefore, DOE's frequency categorization is reasonable for the potential event sequences related to inadvertent entry into an emplacement drift, worker proximity to a loaded TEV, inadvertent opening of a loaded TEV door, and direct exposure due to loss of movement of a loaded TEV.

2.1.1.4.3.4.2 Seismic Events

DOE addressed categorization of the seismic event sequences for the GROA facilities in SAR Section 1.7.5, with the seismic event sequence probability and category presented in SAR Tables 1.7-8, 1.7-10, 1.7-12, 1.7-14, 1.7-16, and 1.7-18. Seismic event sequences identified in the aforementioned SAR tables correspond to the surface facilities (IHF, RF, CRCF, and IHF), intrasite operations, and subsurface operations. DOE detailed the seismic event sequence analysis in BSC (2008bg).

DOE's analysis of seismic event sequences can be broadly divided into three groups: (i) collapse of facility structures, (ii) failure of equipment or mechanical components and (iii) tipover and sliding of transporters and transfer trolleys. In addition, DOE considered rockfall in the emplacement drift initiated by seismic events. DOE's tables in SAR Section 1.7 identified about 113 seismic event sequences, from which there were no Category 1 event sequences and 7 Category 2 sequences (1 in the CRCF, 4 in the WHF, 1 in the LLW building, and 1 in the subsurface). Six identified event sequences ended in unfiltered radionuclide releases, which required consequence analyses. The single Category 2 subsurface event sequence DOE identified involving direct exposure due to loss of TEV shielding by seismic failure required no dose calculation.

The NRC staff reviewed DOE's methodology for evaluation of seismically initiated events in TER Section 2.1.1.4.3.1.2 and development of event sequences in TER Section 2.1.1.4.3.2.2. The NRC staff also reviewed seismic fragility of SSCs in TER Section 2.1.1.4.3.3.2 and passive reliability of the containers in TER Section 2.1.1.4.3.3.1. In this section, the NRC staff reviews (i) the implementation of the methodology and the event sequences; (ii) whether DOE considered mean seismic hazard input and the mean conditional failure probabilities (i.e., fragility) of SSCs in calculating seismic initiating events; (iii) whether consideration of pivotal events and data used for throughput and resident time is consistent with the site, facility design, operations, and human actions; and (iv) whether the quantification and categorization of seismic event sequences are reasonable. The NRC review of seismic event sequences is divided into four major sections: (i) seismic hazard curve, (ii) structural collapse of facility structures, (iii) failure of equipment and mechanical systems, and (iv) subsurface.

<u>Seismic Hazard Curve</u>

DOE developed site-specific seismic hazard curves for the surface and subsurface repository block on the basis of probabilistic hazard analysis. The seismic hazard curve shown in SAR

Figure 1.7-7 represented the MAPE associated with the horizontal PGA for the surface facilities. DOE presented the acceleration data at the MAPE in BSC Tables 6.1-1 and 6.1-2 (2008bg) for surface and subsurface repository blocks and provided the discrete horizontal PGA and associated interval frequencies used for convolution analysis in BSC Tables 6.1-4 and 6.1-5 (2008bg).

NRC Staff Evaluation: The NRC staff reviewed the information on seismic hazard curves using the guidance in the YMRP. The NRC staff evaluated DOE's earthquake information including probabilistic seismic hazard analysis in TER Section 2.1.1.1.3.5.2, where the NRC staff notes that the seismic hazard curve is reasonable for PCSA. The NRC staff notes that the seismic hazard curve used in the seismically initiated event sequence analysis is consistent with TER Section 2.1.1.1.3.5.2 and, therefore, reasonable.

Structural Collapse of Facility Structures

DOE evaluated the collapse of the ITS CRCF, WHF, RF, and IHF structures as potential seismically initiated events. DOE presented the mean annual probability of failure or collapse of the surface facility structures in BSC Table 6.2-1 (2008bg). The mean annual probabilities of failure/collapse of the structures calculated between 3.8×10^{-7} and 8.7×10^{-7} are within DOE's nuclear safety design bases threshold of 2×10^{-6}/yr as indicated in SAR Tables 1.9-2 to 1.9-4. The collapse of surface structures could directly result in unfiltered radionuclide release. Thus, DOE did not transfer the initiating event to a seismic response tree, because DOE did not rely on any SSC to provide prevention or mitigation functions. DOE (i) quantified the event sequences by calculating the expected number of occurrence of building collapse on each waste form based on the residence time within the facility during specific operations and (ii) categorized the collapse of all surface facility structures as below Category 2. In addition, DOE categorized seismic collapse of non-ITS LLW building breaching multiple LLW containers to be a Category 2 event sequence. For this event, DOE further calculated the dose consequence, which is reviewed in TER Section 2.1.1.5.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's evaluation of structural collapse of facility structures using guidance in the YMRP. DOE included building collapse as an initiating event leading to potential dose consequences to the public because, as shown in SAR Figures 1.2.3.-18, 1.2.4-14, 1.2.5-18, and 1.2.6-13, multiple waste forms may be present in the surface facilities at any one time. Because all waste forms are likely to be impacted by structural collapse, the NRC staff's review approach is based on whether the mean annual probability of failure of each surface facility meets the Category 2 threshold. In SAR Section 2.2, DOE indicated the receipt and emplacement operations were projected to span 50 years and used a screening criterion of 2×10^{-6}/year for Category 2 event sequences for aircraft crash hazard based on a probability of occurrence of 1 in 10,000 over the 50-year period of preclosure operations at the surface facilities. Therefore, NRC notes that DOE's design basis threshold of 2×10^{-6}/year for annual probability of unacceptable performance of surface facility structures is at the Category 2 limit. The NRC staff reviewed the fragility evaluation and the seismic performance of the surface structures in TER Section 2.1.1.4.3.3.1.2.1, where the NRC staff notes that DOE's fragility evaluation and the seismic performance of the surface structures are reasonable. On the basis of the review of DOE's design basis threshold and fragility evaluation and the seismic performance of the surface structures, DOE's determination is reasonable that the seismic event sequences initiated by collapse of the surface structures are below Category 2.

Failure of Equipment and Mechanical Systems

The seismic event sequences initiated by the seismic failure of facility equipment and mechanical systems are discussed in this section. DOE presented the results of the event sequence analysis in SAR Tables 1.7-8 for the IHF, 1.7-10 for RF, 1.7-12 for the CRCF, and 1.7-14 for the WHF facilities. DOE's seismic event sequence analysis was detailed in BSC (2008bg), which includes SAPHIRE model data files for each facility indicated in BSC Appendix J (2008bg). DOE developed a seismic initiator event tree that included equipment failure during waste form handling or failure of other SSCs that can affect the waste form. The initiator event tree also included the number of waste containers and the seismic hazard data. The end state of the initiator event tree was linked to the seismic response trees, which included containment, confinement, shielding, and moderator control pivotal events. DOE's analysis did not take credit for the HVAC system and assigned a value of one to the pivotal event "Confinement." For the unfiltered radionuclide release end state involving structural challenges to other containers handled in all the surface facilities and intrasite operations, the event sequence quantification primarily relied on four parameters: probability of failure of equipment, passive reliability of waste form containers, exposure time factor, and throughput of the waste form.

DOE listed equipment used in the event sequence analysis in BSC Table 6.2.2 (2008bg). This list consisted of equipment that handles the waste form [e.g., CTM, cask handling crane, cask and waste package transfer trolley, SFTM and transporters (TEV and site transporters)]. DOE identified several failure modes under the seismic event for handling equipment. For example, failure modes for the CTM were structural collapse of a bridge girder or trolley platform, hoist failure causing drop of a canister or drop of an object onto a canister, and swinging of a canister inside or outside the shield bell. The list also contained equipment that was not used to handle waste containers directly but could initiate event sequences by impacting the waste form containers if it failed or collapsed during a seismic event. This equipment included rollup entry, emplacement, and equipment shield doors; mobile and cask preparation platforms; cask preparation cranes; CTM maintenance cranes; and jib cranes. Besides the equipment failure modes under a seismic event, this table also provided the fragility parameters. DOE calculated the annual probability of failure for each failure mode by convolution of the fragility curve and the hazard curve, as shown in BSC Table 6.2.2 (2008bg).

The failure probability of waste containers, TAD and other canisters, the waste package, and transportation and other casks used in the pivotal events of the seismic response tree was given in BSC Table 6.3.2 (2008bg). The table provided a probability of failure conditional to the severity of the structural challenges on the waste containers for different failure modes of the equipment. For example, all waste containers are assumed to breach for structural collapse failure mode of cranes (e.g., cask handling crane, CTM, jib crane, and equipment shield door), whereas for other failure modes (e.g., hoist failure mode causing drop of a load or drop of an object on the waste containers), DOE considered the probability of waste container failure to be 10^{-5}.

DOE used the exposure or residence time factor to assess preclosure safety in seismic event sequences. The exposure or residence time is the time the waste form is involved in waste handling operations with specific equipment. The exposure or residence time factors were given in BSC Tables 6.6.-1 for CRCF, 6.4-1 for IHF, 6.5-1 for RF, and 6.7-1 for WHF facilities (2008bg). Similarly, the TEV transit time was given in BSC Table 6.9-1 (2008bg), and the exposure time for intrasite operations and the aging facility was given in BSC Table 6.8-1 (2008bg).

NRC Staff Evaluation: The NRC staff reviewed the information on the event sequence quantification and categorization for seismic events using the guidance in the YMRP. The NRC staff reviewed DOE's basis for the fragility parameters for each failure mode and annual probability of failure in TER Section 2.1.1.4.3.3.1.2.2, where the NRC staff notes that DOE's approach to evaluate seismic fragilities of equipment is reasonable.

DOE's use of the probability of waste container failure to provide containment and shielding functions under drops, drops on, and collisions for use in the pivotal event in the seismic response tree is reasonable because the passive reliability of containers is consistent with the NRC staff's evaluation in TER Section 2.1.1.4.3.3.1.1. For event sequences that involved collapse of SSCs in the physical proximity of the waste containers (referred to as seismic spatial system interactions or two-over-one issues in the seismic probabilistic risk assessment for nuclear power plants), DOE assumed the waste containers breach in most cases. DOE assumed that the probability of failure for the waste containers was one for collapse of the crane, platforms, and staging racks. The NRC staff considers the assumption conservative and reasonable because with this assumption, DOE did not take credit for the structural strength of the waste containers that results in a bounding case. For collapse of the mobile platform, collapse of shield doors, and tipover of railcar trolleys on transportation casks, DOE considered a failure probability value of 10^{-5}. The NRC staff reviewed the transportation cask, loaded with a representative canister, subjected to different structural challenges. These structural challenges included a drop of a 9,072-kg [10-T] object onto the top of the cask, 9.14-m [30-ft] vertical and 1.83-m [6-ft] horizontal drops, and side impacts at different speeds. The probability of failure used in the analysis is reasonable because the evaluated structural challenges on the transportation cask are likely to bound impacts from collapse of these structures.

DOE's use of an exposure or residence time factor is a key component in estimating the frequency of seismic event sequences and the expected number of occurrences is sensitive to the variation of the exposure time of the containers. The expected number of occurrences of unfiltered release involving TAD and DOE standardized canisters during operations with the CTM and cask handling cranes and shield door impacts were within one order of magnitude of the Category 2 limit (SAR Table 1.7-12). DOE made similar observations in SAR Table 1.7-10 for the RF involving operations with TAD canisters and in SAR Table 1.7-14 for the WHF involving operations with spent fuel assemblies. The NRC notes that a deviation in the time required for operational sequences may elevate some of the event sequences to be within the Category 2 limit. Consequently, as part of the detailed design process, DOE should confirm that the exposure time of containers is consistent with the exposure time used in the PSCA for event sequence quantification and categorization.

<u>Subsurface</u>

DOE addressed the subsurface event sequences associated with rockfall impact in BSC Section 6.9.2.1 (2008bg). DOE screened out the rockfall impact on the waste package in the emplacement drift during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the information on DOE's evaluation of event sequence quantification and categorization for subsurface seismic events using the guidance in the YMRP. DOE's assessment of seismic event sequences in the subsurface is reasonable because (i) the waste package design includes an evaluation of a 20,000-kg [20-MT] rock block impact without causing breach, as reviewed in TER Section 2.1.1.7.3.9.1; (ii) based on SAR Figure 2.3.4-38 showing the distribution of the rock block mass for the seismic event with an annual probability of 10^{-6}, the maximum credible rock block impacting the waste package in the nonlithophysal area of the subsurface is not likely to exceed 20,000 kg [20 MT]; and (iii) the lithophysal rock units are heavily fractured with small-scale {lengths smaller than 1-m [3.3-ft]} fractures as reviewed in TER Section 2.1.1.1.3.5.4, so the size of potential rockfall blocks in the emplacement drifts in lithophysal rock areas is smaller than the nonlithophysal area.

Also, the NRC staff notes in TER Section 2.1.2.3.1 that potential overheating of waste packages due to rubble accumulation during the preclosure period is not of concern because (i) seismic ground motions strong enough to significantly damage an emplacement drift have a low likelihood of occurring during a 100-year preclosure period and (ii) on the basis of the NRC staff's review in TER Section 2.1.1.2.3.7.3, DOE stated that it will inspect, monitor, and maintain the emplacement drifts and invert structure during the preclosure period.

Intrasite

For AO on an aging pad, DOE considered in BSC Attachment E (2008bg) AO tipover, sliding of AO and impact with other AO, and aging pad displacement and tipover of AO. DOE categorized an event sequence caused by a seismically initiated event as beyond Category 2.

NRC Staff Evaluation: The NRC staff reviewed the information in DOE's evaluation of event sequence quantification and categorization for intrasite seismic events using the guidance in the YMRP. DOE presented the fragility parameters in BSC Table E1.3-2 (2008bg); however, in response to an NRC staff RAI (DOE, 2009dz), no supporting calculations were provided. The AO performance specification (BSC, 2007ac) requires an AO to remain upright and freestanding during and following a seismic event characterized by horizontal and vertical peak ground acceleration of 3 g. The horizontal and vertical peak ground accelerations are about 2.71 and 2.3 g respectively at a MAPE of 2×10^{-6} (BSC, 2008bg). Thus the AO is not likely to tip over for a MAPE 2×10^{-6} event. In addition, DOE conducted a finite element analysis of AO with a canister inside for (i) a 0.914-m [3-ft] vertical drop and (ii) a slapdown with a horizontal velocity of 1.117 m/s [2.5 mph] (BSC, 2008cp) on rigid (unyielding) ground. DOE used the analysis to estimate the maximum effective plastic strain of the canister inside as 0.16 percent for vertical drop and 0.82 percent for slapdown events, which resulted in a failure probability of about 10^{-8}, as described in BSC Table 6.3.7-1 (2008cp). The NRC staff notes that sliding of an AO on the aging pad during a seismic event could result in an impact with another AO; AOs are located 1.83 m [6 ft] apart. The NRC staff notes, because the maximum acceleration of the cask will be limited by the coefficient of friction between the aging pad and AO, the sliding velocity is less than the velocity at the impact from a 0.914-m [3-ft] drop. Thus, effective plastic strain will be bounded by the DOE drop and slapdown analysis. Consequently, DOE reasonably considered seismically induced event sequences for intrasite operations at the aging facility.

2.1.1.4.3.4.3 Fire Events

DOE quantified and categorized event sequences initiated by fires. DOE listed the fire-related event sequences for the 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 NRC staff reviewed the information to assess whether the data used to quantify the fire-related event sequences were used appropriately and whether DOE implemented the established fire-related ESDs correctly for the purposes of quantification.

Data Used in Fire Event Sequence Quantification

DOE determined the fire-initiated event sequence frequencies on the basis of the probability that a waste form is present in a location and the frequency that a fire originating in the facility can grow to a point where it affects the waste form. In the case of direct exposure due to loss of shielding, the initiating event frequency and number of canisters were the only inputs. In other end states, DOE relied on the derivation of passive reliabilities and other pivotal event probabilities to reduce the overall frequency of a particular event sequence.

DOE provided the throughput values in SAR Table 1.7.5 and fire-related initiating event frequencies in BSC (2008ac,as,au,be,bk,bq).

All direct exposure and radionuclide release event sequences relied on canister reliability in a fire as a pivotal event to reduce the overall likelihood of the event. DOE provided the canister failure probabilities due to fire in BSC Table D2.1-8 (2008ac,as,au,be,bk,bq) for various canister configurations (thin-walled or thick-walled canisters in waste packages, transport casks, and shield bells). A separate containment failure probability for bare fuels in casks was developed in BSC Section D2.1.5.3 (2008ac,as,au,be,bk,bq). In all cases, this canister failure probability was on the order of 10^{-4}.

In its event sequence quantification analyses, DOE assumed the loss of shielding probabilities to be either 0.0 or 1.0, depending upon the types of shield material expected for various waste forms. For example, DOE assumed that thin shields (polyethylene or lead) would fail when exposed to any thermal challenge and thick shields (e.g., concrete used for AOs or uranium used in the shield bell) would not fail under the range of expected thermal challenges. In the two end state probabilities that relied on shielding (no release and direct exposure due to shield loss), the event probability was driven by throughput and initiating event probability as described previously, because the canister reliability is high in DOE's analysis.

The fire-related fault trees typically assumed the HVAC had a probability of 1.0 to lose its confinement ability during a fire, except for cases where the HVAC systems were credited as ITS to either reduce event sequence frequency or mitigate consequences (e.g., CRCF ESD20, the fire-initiated event sequence in the CRCF, credited HVAC in the 060–DP–LOSS–CRCF pivotal event probability).

DOE assigned the probabilities of moderator intrusion to be either 0.0 or 1.0 for fire events. DOE's basic assumption was that a fire event would result in a normal actuation of the overhead sprinkler system, thus introducing moderator that either gains access to the fuel (1.0) or has no impact on the particular waste form under investigation (0.0). DOE identified a specific case for moderator intrusion in the CRCF (BSC, 2008ac) as extremely unlikely and assigned a probability value of 0.001 in the SAPHIRE models.

NRC Staff Evaluation: The NRC staff reviewed the data DOE used to quantify its fire event sequences using the guidance in the YMRP. The NRC staff evaluates DOE's quantification for loss of confinement and moderator intrusion control in TER Section 2.1.1.4.3.3.2. In addition, the NRC staff evaluates DOE's quantification for loss of containment (i.e., cask failure and loss of shielding) in TER Section 2.1.1.4.3.3.1.2.3. In those two TER sections, the NRC staff notes that DOE used reasonable data for the pivotal events in the event sequence analysis.

ESD Implementation and Event Sequence Quantification

For most facilities, a number of fire-related event sequences were developed for a particular waste form, depending on the processes surrounding that waste form. For example, a waste form that is handled in multiple areas of a building may have a higher frequency of exposure to a potential fire-initiated event sequence because this sequence can begin in a number of areas in the building that may ultimately affect the waste form. Other waste packages that undergo limited processing get limited exposure to fire-initiated event sequences. The effect of large fires was included in each ESD and incorporated the effect of a building-wide fire affecting a waste form anywhere in the facility.

CRCF–ESD–20 was the fire-initiated event sequence in the CRCF. In this event sequence, DOE assumed waste forms in transportation casks would suffer shielding failure during fire-initiated event sequences, while waste forms protected by the CTM shield bell were assumed to not fail. Furthermore, DOE's calculation of event sequence probabilities used a basic event probability for HVAC confinement failures (3×10^{-2}). DOE presented the moderator pivotal event in the CRCF ESD in BSC (2008ac) as extremely unlikely and assigned a lognormal distribution with a median of 0.001 and an error factor of 10.

IHF-ESD-13 was the fire-initiated event sequence in the IHF. This event sequence was applied to only naval SNF and HLW waste forms. Naval canisters or HLW canisters were assumed to experience shielding loss during a fire given the composition of their shield material. The calculation of event sequence probabilities assumed HVAC confinement failures. For the moderator pivotal event in the IHF ESD, DOE indicated that moderator had no impact on HLW; however, DOE assigned the moderator intrusion with a probability of 1.0 for naval canisters in the IHF.

RF–ESD–12 was the fire-initiated event sequence in the RF and applied to only DPCs and TAD canisters. DOE did not identify any direct exposure scenarios for fire events, because waste forms are in robust AOs and fires large enough to degrade shielding in these packages would also lead to personnel evacuation. DOE assumed that loss of shielding would occur (1.0) in transportation casks, consistent with assumptions made in BSC Attachment D (2008be). Event sequence probabilities used a basic event probability for HVAC confinement failures (3.7×10^{-2}), and the moderator pivotal event in the RF ESD was assumed to be 1.0.

WHF–ESD–31 was the fire-initiated event sequence in the WHF and applied to CSNF, DPCs, and TAD canisters. Loss of shielding was assumed to occur (1.0) in transportation casks, consistent with assumptions made in BSC Attachment D (2008bq). Event sequence probabilities used a basic event probability for HVAC confinement failures (3.5×10^{-2}). The moderator pivotal event in the WHF ESD was assumed to be 1.0 in the ESD quantification.

The intrasite quantification involved an event sequence for intrasite operations and an event sequence for a fire initiating at the LLWF. ISO–ESD–09 was evaluated for eight different waste forms for intrasite operations, and ISO–ES–D07 was evaluated for one event involving all combustible material at the LLWF. ISO–EDS–09 resulted in corresponding response trees for each waste form, where ISO–ESD–07 was a self-contained event without individual response trees.

NRC Staff Evaluation: The NRC staff reviewed the information on fire-related ESD implementation using the guidance in the YMRP and notes that the way DOE modeled the SDs in SAPHIRE and the data used to quantify the event sequences in the model are reasonable.

The CRCF–ESD–20 event sequence resulted in no Category 1 events and four Category 2 events as summarized in SAR Table 1.7-11. DOE categorized five event sequences as beyond Category 2 because they resulted in an expected number of occurrences between 2×10^{-5} and 10^{-6} over the preclosure period. The CRCF analysis produced one event sequence with a frequency of 2×10^{-1} (ESD20–TAD–SEQ2–DE). Consistent with SAR Section 1.7.5, events resulting in a frequency greater than 2×10^{-1} were screened for possible inclusion as a Category 1 event. A subsequent screening analysis of this event sequence was not provided. Upon review of the inputs to the event sequence, the NRC staff notes that there was a discrepancy in the derivation of cask ratios used in the event sequence analysis; however, the discrepancy is offset by other conservative assumptions and, therefore, the Category 2 classification is reasonable.

The IHF–ESD–13 event sequence resulted in no Category 1 events and two Category 2 events as summarized in SAR Table 1.7-7. The three event sequences that were beyond Category 2 resulted in probabilities between 4×10^{-6} and 9×10^{-6}. DOE's categorization of the IHF–ESD–13 event sequence is reasonable because DOE made a conservative assumption on the moderator intrusion with a probability of 1.0 for naval canisters in the IHF for this event sequence.

The RF–ESD–12 event sequence resulted in no Category 1 events and two Category 2 events as summarized in SAR Table 1.7-9. The two Category 2 event sequences for the RF were direct exposures due to shielding loss. ESD12–TAD–SEQ2–DEL resulted in a frequency of 2.0×10^{-1}. Consistent with SAR Section 1.7.5, events resulting in a frequency greater than 2.0×10^{-1} were screened for possible inclusion as a Category 1 event. A subsequent uncertainty analysis of this event sequence was not provided. Upon review of the inputs to the event sequences, the NRC staff notes that the categorization was driven by the high volume of TADs in the RF and the relatively high ignition frequency of a large fire. The NRC staff notes that the conservative assumptions made regarding the frequency of a large fire and the conservative assumptions made regarding loss of shielding indicate that a Category 2 classification is reasonable.

The WHF–ESD–31 event sequence resulted in no Category 1 events and five Category 2 events as summarized in SAR Table 1.7-13. No beyond Category 2 event sequences were described in the table. Only one event sequence approached DOE's screening threshold for Category 1 events (ESD31–TAD–SEQ2–DEL). Confinement probabilities varied in the WHF on the basis of the location of the initiating event. The NRC staff notes that the credit taken for HVAC confinement was either 0.0 or 3.5×10^{-2}. The NRC staff notes that the conservative assumptions made regarding the frequency of a large fire and the conservative assumptions made regarding loss of shielding indicate that the Category 2 classification is reasonable.

The fire-initiated events across the intrasite yielded identical failure probabilities for seven waste forms. Two event sequences (ISO09–UCSNF–SEQ3–RRU and ISO07–LLW–SEQ2–RRU) were identified as Category 2 events and were provided with consequence analyses. The remaining three events had frequencies of 10^{-6}, and they were categorized as beyond Category 2 events. DOE's estimates of failure probability for the seven waste forms are reasonable because DOE used conservative assumptions regarding shield failure—a failure probability of 1.0 (i.e., shielding protection is not functioning).

Fire Event Sequence Categorization

DOE indicated that fire-initiated event sequences resulting in radiation exposures would likely be either Category 2 or beyond Category 2 (SAR Tables 1.7-7, 1.7-9, 1.7-11, 1.7-13, and 1.7-15) and no Category 1 fire event sequence was identified. More specifically, DOE identified four Category 2 event sequences for the CRCF; two each for the IHF, RF, and intrasite; five for WHF; and one each for the LLWF and subsurface operations. The remaining fire event sequences were beyond Category 2.

NRC Staff Evaluation: The NRC staff reviewed DOE's fire event sequence quantification and categorization using the guidance in the YMRP. The NRC staff did not identify any events that could reasonably be considered Category 1, considering the conservative estimates that DOE made in the PCSA. Furthermore, the NRC staff notes that the assumptions that fire events resulting in radiation releases or direct exposures would not be spontaneous are reasonable. Fire-initiated events have a growth time associated with them and would require some time before the pivotal events of canister breach or shield loss were realized. DOE's calculation of event tree probabilities conservatively did not include this time delay (e.g., shields are assumed to fail, canisters are assumed to breach with some fixed probability). In the context of worker dose (Category 1 events), it is unlikely that unprotected workers would be present during a fire event that is large enough to cause loss of shielding or canister breach. Fires of these magnitudes would have already resulted in nonessential personnel evacuation from the facility, and such fires would only be attended by properly equipped Emergency Response Team personnel. This is regarded as an added factor of safety on DOE's final categorization.

2.1.1.4.4 NRC Staff Conclusions

The NRC staff notes that DOE's identification and categorization of event sequences in the GROA is consistent with the guidance in the YMRP. The NRC staff also notes that DOE reasonably identified and categorized event sequences as discussed in this chapter.

DOE stated that it will (i) verify that the final equipment design and its associated seismic fragility satisfy the conclusions in BSC Table 6.6-2 (2008bg) (TER Section 2.1.1.4.3.3.1.2.1) and (ii) perform analyses for drop and tipover scenarios for transportation casks and STCs (TER Section 2.1.1.4.3.4.1.2). As part of the detailed design process, DOE should (i) confirm that elastic spring constants to model soil at the BDBGM seismic level of 0.91 g for evaluation of $C_{1\%}$ are reasonable (TER Section 2.1.1.4.3.3.1.2); (ii) conduct seismic structural and foundation analyses to confirm the adequacy of $C_{1\%}$, which defines the fragility curves as shown in BSC Table 6.2-1 (2008bg) (TER Section 2.1.1.4.3.3.1.2); (iii) confirm the identification of ITS components and the associated nuclear safety design bases are consistent with the design (TER Section 2.1.1.4.3.3.2.1); (iv) confirm that the fault tree modeling specifies the components used to quantify its basic events (TER Section 2.1.1.4.3.3.2.1); and (v) confirm that the exposure time of containers is consistent with the exposure time used in the PSCA for event sequence quantification and categorization (TER Section 2.1.1.4.3.4.2).

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CHAPTER 5

2.1.1.5 Consequence Analysis

2.1.1.5.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's (DOE's) consequence analysis with respect to the preclosure operations of the repository. The objective of the review is to verify that DOE reasonably conducted consequence analysis to support its preclosure safety analysis (PCSA). The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.8 (DOE, 2008ab), DOE (2009av), and the NRC staff requests for additional information (RAIs) (DOE, 2009ek–eq). In addition, the NRC staff used the information in SAR Section 1.5 on the characteristics of spent nuclear fuel (SNF) and high-level radiological waste (HLW) to evaluate DOE's source term calculations.

SAR Section 1.8 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. DOE did not identify any Category 1 event sequence in the geologic repository operations area (GROA) (SAR Section 1.8.5). The NRC staff review and evaluation of DOE's consequence analysis includes (i) dose calculation methodology and input parameter data selection, (ii) source term and release fraction determination, and (iii) consequence analyses results that considered event sequences, which could lead to radiological consequences.

2.1.1.5.2 Evaluation Criteria

The regulatory requirements of 10 CFR 63.111 prescribe criteria for the preclosure safety analysis (e.g., dose to a member of the public calculated at the site boundary). Specifically, the preclosure safety analysis is required to evaluate

- The annual dose during normal operations and for Category 1 event sequences to any real member of the public located beyond the boundary of the site

- The aggregate radiation exposure and the aggregate radiation levels in both restricted and unrestricted levels for Category 1 events

- Radiation exposures to an individual located on, or beyond, the site boundary for any single Category 2 event

- The annual dose to a member of the public in the general environment

Normal operations are those DOE planned, routine activities in which monitored exposures are expected from the HLW processing at the 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 during the preclosure period. 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 during the preclosure period. The general environment is defined at 10 CFR 63.202 as everywhere outside the Yucca Mountain site, the Nellis Air Force Range, and the Nevada Test Site.

10 CFR 63.112 specifies requirements for the preclosure safety analysis for the GROA. In particular, the preclosure safety analysis must include

- A general description of the structure, systems, components, equipment, and process activities at the GROA

- An identification and systematic analysis of naturally occurring and human-induced hazards, including a comprehensive identification of potential event sequences

- Data used to identify naturally occurring and human induced hazards and the technical basis for either inclusion or exclusion of specific events in the safety analysis

- An analysis for the performance of structures, systems, and components to identify those that are important to safety

- A description and the discussion of the design, including the design bases and their relation to the design criteria

In its review, the NRC staff used the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The relevant acceptance criteria follow:

- DOE 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 DOE adequately assess the consequences to workers and members of the public from normal operations and Category 1 event sequences.

- Consequence analyses by DOE include Category 2 event sequences, as well as factors that allow an event sequence to propagate within the GROA.

- Consequence calculations by DOE adequately assess the consequences to members of the public from Category 2 event sequences.

The NRC staff also used additional guidance, such as NRC regulatory guides and interim staff guidance (ISG), to support the NRC staff review and evaluation. These additional guidance documents are discussed in the relevant sections that follow.

2.1.1.5.3 Staff Review and Analysis

The NRC staff review of SAR Section 1.8 focuses on (i) the methodology and input parameters used for the dose calculation, (ii) the consistency of the source terms of the dose calculation with those described in SAR Section 1.5, and (iii) the methodology for the worker and public dose determination.

DOE defined the radiation workers as those who are qualified and trained as radiation workers and who will receive occupational doses in performing their duties. Within the preclosure controlled area, referred to as the onsite areas, DOE 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. The Cind-R-Lite mining lease is located southwest of the surface facility GROA, within the preclosure controlled area, near the site boundary. The mining personnel who periodically access this area are considered to be onsite members of the public.

Offsite public is defined as individuals located at or beyond the site boundary of the preclosure controlled area (SAR Figure 1.8-1). Individuals located on the south and west boundaries are considered members of the public in the general environment. The general environment is the area outside the Yucca Mountain site, the Nellis Air Force Range, and the Nevada Test Site that allows public access. For the areas north and east of the site boundary, which are areas controlled by the Nevada Test Site (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."

2.1.1.5.3.1 Dose Calculation Methodology and Input Parameter Selection

DOE discussed the methodology used to calculate dose consequences to site workers and members of the public, both onsite and offsite (SAR Section 1.8). DOE considered the radiological doses to workers and the public for normal operations, off-normal events, and Category 2 event sequences for the GROA activities. DOE did not identify any Category 1 event sequence that required evaluation of dose consequences to workers or the public.

The NRC staff review and evaluation of DOE's dose calculation methodology and input parameter selection include (i) dose calculation methodology, (ii) atmospheric dispersion determination, and (iii) assumptions and input parameter selection, as discussed next.

<u>Dose Calculation Methodology</u>

DOE described the methodology for estimating doses to workers and the public and 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). DOE 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. DOE 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 consider 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. Inhalation and air submersion pathways were considered for atmospheric releases of radioactive material and resuspension of surface contamination. For onsite public locations, DOE indicated that direct exposure from the waste handling facilities was minor compared to DOE's 2.5 μSv/hour

[0.25 mrem/hour] shielding design limit. DOE stated that the onsite public doses were dominated by the transportation cask rail and truck buffer areas and the aging overpacks stored at the aging pads. DOE indicated 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.

DOE 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 (International Commission on Radiological Protection, 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, DOE calculated the external dose by multiplying the concentration to which the individual is exposed by a dose conversion factor for submersion from U.S. Environmental Protection Agency (EPA) (1993aa) and the exposure time. DOE calculated the TEDE for workers and the onsite public by summing the external dose due to direct radiation, the external dose due to submersion, and the internal dose due to inhalation. DOE stated that there are no agricultural activities in the onsite area and, thus, DOE did not include any dose due to ingestion of foodstuffs in the calculations.

To calculate the airborne exposures to the offsite public, DOE used the GENII Gaussian statistical model (Napier, 2007aa). GENII is a general-purpose computer code for estimating the consequences of radionuclides released into the environment. Air transport includes both plume and puff models. DOE assumed that releases were at ground level for the dose calculations at the site boundary.

For assessment of internal exposures, DOE used the methods proposed in EPA (1988aa, 1993aa, 1999aa). DOE used the GENII Gaussian statistical model, which implemented dosimetry models recommended by the ICRP and related guidance in EPA (1988aa, 1993aa, 1999aa). DOE applied both deterministic and stochastic approaches to model the impacts of GROA operations. A majority of DOE calculations used a combination of deterministic bounding values coupled with stochastic parameters characterized by a mean value and distribution. DOE's deterministic dose calculation methods used receptor characteristics that bound those of any offsite member of the public or a worker. DOE stochastic dose calculation methods used mean values and distributions for parameters including receptor-related parameters. Using sensitivity and uncertainty analysis techniques, DOE determined a dose distribution. DOE then determined the dose to an offsite member of the public by using the maximum value obtained from the calculated dose distributions.

NRC Staff Evaluation: The NRC staff reviewed DOE's dose calculation methodology using the guidance in the YMRP. On the basis of its review, the NRC staff notes that DOE'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 assuming use of TAD canisters is reasonable because these computer codes 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 notes that use of the GENII Gaussian statistical model, which implemented dosimetry models recommended by the ICRP and related guidance in EPA (1988aa, 1993aa, 1999aa) to assess internal exposures, is reasonable because the dosimetry model recommended by the ICRP and risk models used by GENII 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.

Atmospheric Dispersion Determination

DOE estimated airborne doses using the annual average onsite atmospheric dispersion coefficients (X/Q). To calculate the annual average onsite X/Q, DOE used the guidance in Regulatory Guide 1.194 (NRC, 2003ah) and performed calculations using the ARCON96 (NRC, 1977ab) computer code. DOE 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. DOE 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 95th percentile X/Q values were presented in SAR Table 1.8-12. The 95th percentile values were used for the Category 2 calculations. To estimate airborne exposures to the offsite public, GENII 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 Evaluation: The NRC staff reviewed DOE's atmospheric dispersion determination using the guidance in the YMRP and notes that DOE's use of ARCON96 to calculate the annual average onsite X/Q is reasonable. ARCON96 is an industry standard atmospheric dispersion computer code developed for NRC. ARCON96 is effective when release and receptor points are in close proximity, which was 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 typical Gaussian models overestimate concentrations in the vicinity of buildings. In addition, ARCON96 provides a model to account for building wake factors over short distances.

Assumptions and Input Parameter Selection

DOE 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, DOE assumed that the entire external surface area of each transportation cask was contaminated at the regulatory limit [49 CFR 173.443(a), Table 9]. DOE assumed that the contamination levels on the casks were 4 Bq/cm^2 [10^{-4} μCi/cm^2] beta/gamma and low toxicity alpha and 0.4 Bq/cm^2 [10^{-5} μCi/cm^2] for all other alpha following 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, DOE established a limit of 2.5 μSv/hour [0.25 mrem/hour] as a shielding design limit for the various waste handling facilities. DOE 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. DOE 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 (WHF).

DOE 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. DOE based its dose conversion factors on information presented in EPA (1988aa, 1993aa, 1999aa). As discussed previously, DOE performed the deterministic dose calculation using the receptor characteristics that bound those of any offsite member of the public or a worker. DOE selected individual parameter values at the 95^{th} percentile level for receptor-related parameters including food consumption rates and periods, and external and inhalation exposure times. DOE indicated that 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, DOE 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, DOE 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 Evaluation: The NRC staff reviewed SAR Sections 1.8.1 and 1.8.2 and supporting documentation using the guidance in the YMRP. DOE's assumptions and input parameter selections are reasonable because (i) DOE used applicable NRC guidance (i.e., NRC, 2000ag, 1977ac) for selection of dose modeling assumptions and input parameters and (ii) DOE's assumptions and selected input parameters are conservative when compared to the industry standards. The NRC staff notes that DOE's determination that the local outdoor worker group was conservative for evaluating exposures from airborne releases is reasonable because this group spends the most time outside and, therefore, has the highest exposure potential to airborne releases.

2.1.1.5.3.2 Source Term Evaluation

DOE described the kind, amount, and specifications of the radioactive material to be received and possessed at the GROA as part of DOE development of the source term. For conducting its PCSA, DOE 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 for normal operations included commercial spent nuclear fuel (CSNF), naval SNF, DOE SNF including a small amount of CSNF in DOE'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. DOE 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, DOE 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 review and evaluation of DOE's source term identification include (i) source term for dose calculation and (ii) cladding damage and leak path factor assumptions, as discussed next.

Source Term for Dose Calculation

In SAR Section 1.8.2, DOE 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, DOE 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. DOE 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 WHF 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.

DOE 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, WHF, and receipt facility (RF), 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.

DOE 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, DOE used maximum assembly inventories discussed in SAR Section 1.8.1.3. When developing the source terms, DOE 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, calculations assumed a full capacity of 2.1×10^7 kg [21,000 MTHM] of CSNF. For the SNF, representative and maximum inventory values were provided for both PWR and BWR

SNF (SAR Tables 1.8-2 and 1.8-3). The representative values were used for normal operations calculations. The maximum inventory values were used for Category 2 calculations.

DOE 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 the NRC staff RAI (DOE, 2009ep), DOE 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). DOE 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 Evaluation: The NRC staff reviewed the calculations of the source term using the guidance in the YMRP. The NRC staff notes that 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 will be handled at the GROA because to predict these characteristics DOE used proposed PWR and BWR SNF, naval spent fuel, and defense waste with reasonable assumptions and idealizations.

DOE’s identification of the dose-significant radionuclides is reasonable. The NRC staff evaluated the radionuclide concentrations DOE calculated using SCALE/ORIGEN-S and the data presented in published papers and NUREG/CR–6798 (Sanders and Gauld, 2003aa) and notes no significant divergences in radionuclide concentrations. In addition, the SCALE/ORIGEN-S software is reasonable for calculating radionuclide concentrations in the representative CSNF because it is a standard software widely used in the industry and the gamma and neutron source term calculated using the SCALE/ORIGEN-S accurately reflected the dependency on the two parameters, burnup and initial enrichment, as indicated in Gauld and Ryman Section 5 (2001aa) and Gauld and Parks Section 4.2.2 (2001aa).

Furthermore, the NRC staff reviewed DOE’s analysis and notes that DOE’s calculated isotopic compositions of high-burnup CSNF bounding are consistent with published papers and NUREG/CR–6798 (Sanders and Gauld, 2003aa). DOE’s assumptions on the parameters of the bounding 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, DOE’s use of SCALE/ORIGEN-S to calculate source terms for maximum SNF used in the PCSA for event sequences is reasonable.

Cladding Damage and Leak Path Factor Assumptions

In assessing dose consequences, DOE 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, DOE 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.

DOE provided CSNF in-pool release fractions for drop or impact events in the WHF SNF pool; pool decontamination factors; and LPFs for WHF 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, DOE stated that Regulatory Guide 1.183 recommendations in NRC Appendix B (2000ag) apply because the depth of water above the damaged fuel is at least 7.0 m [23 ft].

DOE 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). DOE used specific release fractions and respirable fractions for the dry active waste in drums, WHF 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, DOE assumed a respirable fraction of 1.0. DOE used bounding values of measured respirable fractions and airborne release fractions for uncontained waste.

NRC Staff Evaluation: The NRC staff reviewed DOE'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 the YMRP. The NRC staff notes 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 applicable NRC guidance documents as explained next.

DOE's assumptions of the fraction of the material at risk, release fractions, respirable fractions, and LPF values are reasonable and 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, DOE'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 notes 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 conservative and consistent with SFST–ISG–5 (NRC, 2000af)

- The assumption of the waste form respirable fraction of 10-μm [3.9×10^{-5}-in] aerodynamic equivalent diameter is within the cutoff limit of American Nuclear Society Appendix B (2006ab)

- Assumptions on cladding burst respirable fraction and oxidation respirable fraction for fuel fines and volatiles and low- and high-burnup SNF are conservative and consistent with SFST–ISG–5 (NRC, 2000af) and the published test results referred to 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) and is therefore reasonable

- The 0.0 LPF assumption for transportation casks and canisters designed and tested to be leak tight is reasonable because it is consistent with the recommendations in NUREG/CR–6672 (Sprung, et al., 2000aa) and SFST–ISG–5 (NRC, 2000af)

- The 0.1 LPF assumption for waste packages is reasonable 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 conservative and consistent with the recommendations in 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 is reasonable because these values are consistent with the recommendations in SFST–ISG–5 (NRC, 2000af)

In addition, DOE's assumptions for release fractions; pool decontamination factors; and LPF for the WHF pool for noble gases, halogens, and alkali metals are reasonable and conservative, because these fractions are consistent with the release fractions in Regulatory Guide 1.183 (NRC, 2000ag).

2.1.1.5.3.3 Public Dose Calculation

DOE 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. DOE 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 review and evaluation of DOE'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 next.

Features Limiting Onsite Public Exposures

DOE 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, DOE 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. DOE stated that 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. DOE also stated that throughout this process, exposures may occur that could affect the onsite public; in particular, the construction workers completing work on other portions of the site. DOE 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, DOE 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. DOE 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. DOE established shielding requirements using the point of maximum or peak radiation dose. Therefore, DOE stated that the overall general area radiation levels would be less than this maximum calculated dose. DOE performed dose calculations for the onsite handling facilities using the MCNP code.

DOE's shielding calculations were presented in a number of SAR sections, including Sections 1.10.3 and 1.8.4.1.3. DOE 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. DOE also stated that the WHF, 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, DOE's emergency plan includes provisions for warning site personnel and evacuating personnel to safe areas (SAR Section 5.7.2.2.3). DOE stated that distance attenuation between the waste handling facilities and the various onsite public areas further reduces the dose rates. DOE 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 DOE, 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 the YMRP. DOE's selection of these features follows common, industry standard approaches to incorporate the principles of time, distance, and shielding to limit radiological exposure. Therefore, DOE's selection of features limiting onsite public exposure is reasonable.

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.

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.

For direct radiation during normal operations, DOE 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 limits, DOE 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, DOE assumed that the dose rates for the transportation casks were bounded by the limits in 10 CFR 71.47. 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. DOE 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).

Airborne releases may occur from casks that require opening. DOE stated that this will involve only the CSNF handled in the WHF. 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 exposure limits, 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 (AF) 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 DOE, 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.

DOE considered normal subsurface radiological releases in the public dose calculations. Because the canisters placed inside the drifts are sealed, only contamination on the outside of the canister was assumed available for release. In addition, 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.

According to DOE, the highest estimated doses to the public were at the lower muck yard (Area 780) and the warehouse and nonnuclear RF (Area 230) due to the casks that will be temporarily stored at the rail and truck buffer areas (Areas 33A and 33B). DOE stated that the maximum dose rate limits for each cask were equal to rates provided in 10 CFR 71.47. This is 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, DOE indicated that the dose rates at this location become the predominant dose contributor to the onsite public. DOE 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 RF on the basis of an occupancy time of 2,000 hours/year. As discussed in SAR Section 1.1.9.3.2.12, DOE 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 Sections 6.1.2 and 6.1.3 (2007am). The basis for the warehouse and nonnuclear RF dose projection was discussed in DOE's response to the NRC staff RAI (DOE, 2009eq).

NRC Staff Evaluation: The NRC staff reviewed SAR Section 1.8.3 and supporting documentation using the guidance in the YMRP with respect to the onsite public dose assessment. DOE's approach is reasonable because it accounted for the significant radiological pathways, accounted for public exposure using industry standard codes and calculation methods, and followed NRC guidance for making these calculations.

<u>Features Limiting Offsite Public Exposures</u>

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, DOE incorporated the operational constraint for air emissions to any individual member of the public into its Operational Radiation Protection Program.

The AF 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.

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 to the site boundary where the public could establish permanent residence is approximately 18.5 km [11.5 mi] south. Currently, the nearest offsite member of the public lives 22.4 km [13.9 mi] from the surface facility GROA boundary. At these distances, DOE 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 Section 3.1.8.2 (2002aa). 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 Table 3-30 (2002aa). For Category 2 event sequences, even the worst case event results in only 0.1 mSv [10 mrem] at the site boundary within the general environment.

NRC Staff Evaluation: The NRC staff reviewed the facility features used to limit offsite public dose using the guidance in the YMRP. DOE's selection of the features limiting offsite public exposures follows common, industry standard approaches to incorporate the principles of time, distance, and shielding to limit radiological exposure. Therefore, DOE's selection of features limiting offsite public exposures is reasonable.

Offsite Members of the Public Dose Calculation

DOE performed a series of calculations to determine distances from the surface facility GROA and the subsurface exhaust shafts to the site boundary. The closest offsite member of the public to the surface facility GROA is currently 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 offsite member of the public is 21.5 km [13.3 mi] toward the south-southeast wind sector (BSC, 2007bp).

DOE did not use the distance to the offsite member of the public for the dose calculations, but instead conservatively determined the dose to a hypothetical member of the public located at the site boundary, closer than offsite members of the public. To determine X/Q values from the surface facility GROA boundary, DOE took the 8 wind sectors (of 16) 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), DOE used 11,000 m [6.8 mi]. DOE 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. DOE 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, DOE 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. DOE 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 Section 6.7 (2008ay) and provided in SAR Table 1.8-29. Source terms that DOE 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. DOE 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].

DOE stated that 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 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. DOE also stated that it evaluated the area south and west of the Yucca Mountain site boundary because this area is open to the general public. DOE 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, DOE used the minimum distances from the surface facility GROA boundary to the site boundary, as shown in BSC Tables 9 and 10 (2007bp) and SAR Table 1.8-10, to calculate the X/Q values for the offsite public not in the general environment. However, DOE used the distance values in BSC Table 18 (2007bp) to determine X/Q values for the general environment. BSC Table 34 (2007bp) provided the X/Q values shown in SAR Table 1.8-12 and referenced BSC Table 18 (2007bp) as its source.

Airborne exposures from normal operations were presented in SAR Section 1.8.3.1.2. DOE evaluated potential airborne release doses from inhalation, ingestion, resuspension inhalation, air submersion, and groundshine pathways as a continuous release throughout the year. DOE 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 WHF, the aging pads, and the subsurface exhaust shafts. For the offsite public in the general environment where food ingestion doses were evaluated, DOE 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. DOE calculated these doses on the basis of airborne releases. DOE did not include dose contributions from offsite transportation.

For Category 2 event sequences, only offsite doses to the public are calculated. DOE 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. Ground exposure and food consumption by the offsite public was established as 30 days. Because of the large distances to the site boundary, DOE calculated that direct radiation resulting from a Category 2 event sequence was insignificant.

The resulting doses for Category 2 event sequences were provided 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 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 doses were 0.3 mSv [30 mrem] TEDE. Of the two event sequences, according to DOE, 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 Evaluation: The NRC staff reviewed SAR Section 1.8.3 and supporting documentation using the guidance in the YMRP with respect to offsite public dose assessment. DOE's approach is reasonable because it accounted for the significant radiological pathways, accounted for public exposure using industry standard codes and calculation methods, and followed applicable NRC guidance for making these calculations.

DOE'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.

DOE's established 0.4 mSv/hour [40 mrem/hour] combined neutron and gamma dose design rate limit for the casks at the aging pad is reasonable because it is consistent with shielding design criteria in NUREG–1536 [NRC Section 5.0 (V)(1)(a) (1997ae)], which provides a range of 0.2 to 4.0 mSv/hour [20 to 400 mrem/hour].

In addition, DOE's calculation of 0.0011 mSv/year [0.11 mrem/year] TEDE for the offsite public not within the general environment and 0.0005 mSv/year [0.05 mrem/year] TEDE for the offsite public in the general environment are reasonable because DOE calculations accounted for the significant radiological pathways and public exposure using standard codes and standard calculation methods.

2.1.1.5.3.4 Worker Dose Calculation

DOE calculated radiological doses to radiation workers as part of PCSA (SAR Section 1.8.4). The radiation worker dose is assessed (i) for important to safety structures, systems, and components determinations and (ii) for radiation workers during normal operations and Category 1 event sequences. Radiation worker safety assessments are not performed for Category 2 event sequences. Because DOE'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 DOE's important to safety structures, systems, and components determination. DOE 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 0.013 Sv [1.3 rem] or less. SAR Table 1.8-25 showed that the direct external exposure to radiation emitted from radioactive waste in sealed containers during normal operations dominates worker dose. According to DOE, compared to other exposure pathways for workers (e.g., inhalation of airborne radioactive materials), worker external dose from emitted radiation is significant with respect to preclosure safety.

DOE based total annual doses to radiation workers on four 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] calculated by DOE corresponds to a radiation worker in the RF (SAR Table 1.8-25). DOE stated that this calculated dose was dominated by direct radiation from normal operations in the RF for an annual process rate of 273 casks.

DOE 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. BSC Tables 3, 6, and 7 (2008al) showed the estimated radiation worker doses at different facilities and indicated that direct irradiation during normal operations inside facilities represented the greatest contribution to dose for radiation workers.

The NRC staff review and evaluation of DOE's worker dose calculation includes (i) direct radiation calculation, (ii) airborne releases of radioactive material, and (iii) aggregation of worker doses, as presented next.

Direct Radiation Calculation

DOE'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 Table 7 (2008al). DOE'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). Estimated dose rates depend on the radiation emission rates from radioactive waste, direct radiation scaling factors DOE used, credit taken for shielding materials and distances of workers to direct radiation source terms, and flux-to-dose conversion factors. DOE assumed five work crews will be available to staff the three shifts of operations, as detailed in SAR Equation 1.8-26 and BSC Section 3.2.4 (2008bw).

The NRC staff organized its evaluation of DOE'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

DOE performed shielding calculations with the MCNP computer program (SAR Section 1.8.4.1.3). DOE used a maximum source term for establishing shielding design parameters and selected a design basis source term for calculating worker doses as shown in TER Section 2.1.1.5.3.2 (BSC, 2008cc). DOE 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 Evaluation: The NRC staff reviewed the information on radiation emission rates using the guidance in the YMRP. Because PWR fuel represents a greater source term for penetrating radiation (gamma ray and neutron) than other waste forms and these design basis characteristics are expected to overestimate the radiation emission rates compared to the average SNF assembly handled at the GROA, the NRC staff notes that this source term is conservative for external dose assessments and reasonably accounts for the preferential loading of SNF with a range of burnups. The NRC staff provided a detailed review on DOE's source terms in TER Section 2.1.1.5.3.2.

Direct Radiation Scaling Factors

DOE applied scaling factors for facility throughput and direct radiation source term to its worker dose calculations, as described in DOE Enclosure 1 (2009eo). DOE 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 DOE used a design basis source term instead of a maximum source term for calculating annual worker doses, DOE applied a dose reduction by a factor of 2.7, as detailed in DOE Enclosure 1, Section 1 (2009eo).

NRC Staff Evaluation: The NRC staff reviewed DOE's information on the scaling factor for the direct radiation source term, outlined in DOE Enclosure 1 (2009eo), using the guidance in the YMRP and notes that this adjustment 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 notes that the factor of 2.7 reduction in dose is reasonable 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 Enclosure 1, Section 1 (2009eo). 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 Enclosure 1, Section 1 (2009eo)], DOE's scaled doses for radiation workers are conservative. For these reasons, DOE's consequence analysis for radiation workers is reasonable.

Credit for Shielding Materials and Worker Distances

Although additional credit could have been taken for shielding materials in aging overpacks and shielded transfer casks, DOE used dose rate profiles for a TS125 rail transportation cask in the canister receipt and closure facility, as shown in BSC Table 3 (2007cl). DOE stated that dose rates from TS125 casks are bounding because of the higher dose rates compared to other cask configurations, as detailed in BSC Section 3.2.1 (2007cl). DOE used a similar approach for estimating external doses to workers in other facilities, as shown in BSC Table 1 (2008bw) and BSC Table 1 (2007ck).

In its calculations for estimating shielding design requirements, DOE assumed that each transportation cask received in the RF and canister receipt and closure facility will contain one canister of any type, as outlined in DOE Section 3.1.7 (2009en). DOE clarified in DOE Enclosure 5 (2009ek) 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 Enclosure 5, Section 1 (2009ek). DOE 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 Enclosure 5, Section 1 (2009ek).

NRC Staff Evaluation: The NRC staff reviewed the technical bases and input data used in DOE's worker dose analyses using the guidance in the YMRP and notes that the control and limitation of worker exposure during waste handling and transfer operations is reasonable because remote operations would limit time workers spend in elevated radiation areas. The NRC staff notes, in TER Section 2.1.1.5.3.2, that DOE's approach of using dose rate profiles for a rail transport cask adds conservatism and reasonably accounts for uncertainty, including any degradation during normal operations. The time-motion analysis DOE performed is a reasonable approach for establishing times, source-to-receptor distances, and dose rates for worker exposure. The NRC staff notes that, based on the details DOE provided on processing steps, the time DOE estimated it would take to complete tasks and the distances of radiation workers to radiation sources are reasonable [e.g., BSC Table 5 (2008bw)] because (i) DOE provided detail regarding processing steps and (ii) DOE external dose calculation accounted for processing steps that are important to the external dose calculation. The DOE time estimates used in the external dose calculation for radiation workers are consistent with DOE's anticipated times to complete individual processing steps.

<u>Flux-to-Dose Conversion Factors</u>

DOE 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 Section 6.1.2 (2008cc). 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 the internationally accepted quantity for effective dose equivalent that the ICRP proposed.

NRC Staff Evaluation: The NRC staff reviewed the information on flux-to-dose conversion factors using the guidance in the YMRP. The NRC staff notes that the 1977 version of the 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 ICRP–26 (International Commission on Radiological Protection, 1977aa) and 10 CFR 20.1003. 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, DOE's selection of flux-to-dose rate conversion factors is reasonable. The NRC staff also reviewed the dose conversion factors used by DOE in the dose assessments for airborne releases of radioactive material, as outlined in SAR Section 1.8.1.4.1 and BSC Section 6.1.2 (2008al). Because these dose coefficients are based on widely used dosimetric models from the International Commission on Radiological Protection and the U.S. Environmental Protection Agency, the dose conversion factors DOE used are reasonable.

The NRC staff also notes that it is reasonable for DOE 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 discussed previously, the resulting dose obtained using the effective dose equivalent of ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa) is approximately a factor of 2.7 less than the dose value obtained using the ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa) conversion factors. Therefore, DOE's use of ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa) is conservative because it results in a higher estimate of personnel exposures than would be calculated using ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa).

Airborne Releases of Radioactive Material

Airborne releases from handling individual assemblies of CSNF with cladding damage with pinhole leaks or hairline cracks in the WHF represented the largest airborne release source term from surface facilities during normal operations, as described in BSC Section 6.1.2 (2008al). 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. DOE clarified that worker doses from handling damaged fuel assemblies were included in its PCSA for normal operations, as detailed in DOE Enclosure 1, Section 1 (2009en). DOE described in DOE Enclosure 1, Section 1 (2009en) that potential airborne radioactive material from damaged fuel assemblies would be confined and routed through the heating, ventilation, and air conditioning system. DOE'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. DOE 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 WHF, as outlined in DOE Enclosure 1, Section 1 (2009en). DOE clarified in DOE Enclosure 4 (2009ek) that workers would access the pool room in the WHF 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 did not include damage to the assembly from a drop or collision, DOE determined in DOE Enclosure 4 (2009ek) that there would not be any additional radionuclide releases. Onsite ground contamination from estimated releases during normal operations was excluded in DOE's consequence analysis for radiation workers and onsite members of the public, as described in BSC Section 3.2.8 (2008ak). DOE indicated that such exclusion is reasonable because DOE would control onsite areas and monitor them for radionuclide contamination so that remedial actions could be taken. However, in BSC Appendix IV (2008ay), DOE 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.

DOE expects airborne releases of radionuclides during normal operations in the WHF (SAR Section 1.8.2.2.1). DOE 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 WHF would receive the greatest doses. SAR Section 1.8.2.2.1 described DOE's calculation of airborne releases in the WHF assuming 1 percent cladding damage to individual fuel assemblies. HEPA filters mitigate normal operation releases from the WHF, as described in BSC Section 6.2.1 (2007al). With regard to onsite workers located outside, DOE stated that this information provides a basis for consequence analyses from airborne radioactive material that originates inside the WHF and is subsequently released to the atmosphere during normal operations. DOE 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. DOE stated in DOE Enclosure 1 (2009em) that no unfiltered releases would occur directly into interior rooms of the WHF or to the atmosphere during normal operations.

NRC Staff Evaluation: The NRC staff reviewed the information on airborne releases of radioactive material using the guidance in the YMRP. DOE's accounting of radioactive particulates in the pool water from damaged fuel assemblies is reasonable 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 notes DOE's characterization of radioactive gases releases from pinhole leaks and hairline cracks is reasonable. 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). DOE's approach is reasonable because it accounts for the significant radiological pathways and accounts for radiation worker exposure both inside and outside of operational facilities. The NRC staff notes DOE's classification of off-normal events is consistent with HLWRS–ISG–03 [NRC Footnote 1 (2007ac)] and that these contributions do not represent significant elevations in worker exposure.

<u>Aggregation of Worker Doses</u>

In SAR Section 1.8.4.2 and Table 1.8-25, DOE aggregated the estimated dose contributions for the four major sources discussed previously to individual radiation workers. DOE calculated doses for each contribution by assuming worker exposure for 2,000 hours/year (100 percent occupancy). Because DOE's PCSA identified no Category 1 event sequences, contributions for Category 1 event sequences are zero (SAR Table 1.8-25). DOE also assessed worker doses from off-normal events in BSC Appendices IV and V (2008ay) and determined that these events did not provide significant contributions to the worker TEDE (SAR Section 1.8.4.2). However, these off-normal doses still were factored into DOE's dose aggregation.

NRC Staff Evaluation: The NRC staff reviewed the information on aggregation of worker doses using the guidance in the YMRP. DOE's dose aggregation approach is consistent with DOE's methodology for estimating doses to radiation workers and is reasonable because it accounts for the main sources of radiological exposure and does not underestimate the annual dose to an individual worker. The NRC staff notes that radiation worker exposure during waste handling and transfer operations, including the control and limitation of exposure duration, was reasonably characterized in DOE's consequence analyses because DOE will use remote operations to limit the time spent by workers in elevated radiation fields. Overall, DOE's technical bases and input data for worker dose analyses are reasonable.

2.1.1.5.3.5 Dose Consequences

DOE discussed potential public and worker dose consequences (SAR Section 1.8.6).

The NRC staff organized its evaluation of DOE's dose consequence information in the following five sections:

1. Interactions between hazard, event sequence, and consequence analyses
2. Facility throughput
3. Aggregation of annual doses

4. Dose consequence for normal operations and Category 1 event sequences
5. Dose consequence for Category 2 event sequences

Interactions Among Hazard, Event Sequence, and Consequence Analyses

Because there are no Category 1 event sequences to aggregate, DOE evaluated only the doses from normal operations for compliance with the preclosure performance objectives. SAR Section 1.8.6 summarized DOE's analysis of potential public and worker dose consequences for normal operations and Category 2 event sequences. In SAR Table 1.8-36, DOE listed the results of the public and worker dose consequences 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 DOE, consequence analyses were either performed by DOE or not needed. DOE also considered internal events that were not propagated into event sequences (SAR Table 1.7-1).

NRC Staff Evaluation: The NRC staff reviewed the dose compliance information using the guidance in the YMRP. The NRC staff notes that DOE reasonably treated interactions of the hazard on credited structures, systems, and components in the set of Category 2 consequence analyses, as detailed in BSC Table 49 (2008ay). For example, DOE did not take mitigation credit in the consequence analyses for event sequences involving fire. The NRC staff also notes 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 set of consequence analyses is reasonable with respect to the results presented for DOE's event sequence analysis. Detailed NRC staff technical reviews of DOE's consequence analyses for members of the public and radiation workers are documented in TER Sections 2.1.1.5.3.3 and 2.1.1.5.3.4, where the NRC staff notes that DOE's consequence analyses for members of the public and radiation workers are reasonable.

The NRC staff reviewed the credit taken for particulate filtration in DOE'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. NRC staff notes that mitigation of atmospheric releases by HEPA filtration is reasonable for these three Category 2 event sequences because nonseismic, internal structural challenges associated with the handling and transfer of the various containers are not expected to 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 3×10^6 and 3.6×10^6 kg [3,000 and 3,600 MTHM] per year, respectively, at the GROA. BSC Assumption 3.1.1 and Table 3 (2008al) stated that nominal worker doses, used for comparison to the regulatory limits, were based on expected nominal facility throughput of 3×10^6 kg [3,000 MTHM] (500 casks) of CSNF annually. DOE clarified in DOE Enclosure 3, Section 1 (2009el) that this throughput of 3×10^6 kg [3,000 MTHM] (500 casks) of CSNF reflects the repository maximum capacity and rate of receipt.

NRC Staff Evaluation: The NRC staff reviewed the throughput assumptions DOE used for calculating consequences to radiation workers using the guidance in the YMRP and notes that use of the maximum rate of receipt in DOE's atmospheric release calculations is reasonable because the maximum annual rate of receipt was determined by increasing the average rate of receipt by 20 percent. DOE's throughput assumptions for calculating doses to radiation workers is reasonable 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 Section 1.2.1.1.2 and outlined in DOE Enclosure 3, Section 1 (2009el).

Aggregation of Annual Doses

DOE aggregated doses for members of the public and radiation workers (SAR Section 1.8.1.2) by including four major contributions discussed in TER Section 2.1.1.5.3.4. Because DOE 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, DOE 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. DOE 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 DOE 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 Evaluation: The NRC staff reviewed DOE's methodology to aggregate annual doses for normal operations and doses from Category 1 event sequences using the guidance in the YMRP. Because onsite locations are nonresidential, the exposure times DOE used for onsite members of the public are conservative. The NRC staff also notes that the aggregation approach DOE used for onsite members of the public is reasonable. TER Section 2.1.1.5.3.4 provides the NRC staff evaluation of the dose aggregation approach for radiation workers and notes the dose aggregation approach is reasonable 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 Tables 43–46 (2008ay), where DOE showed that the aggregated offsite TEDE results equaled the sum of the highest TEDE estimates for airborne releases from the WHF, AF, and subsurface. The offsite public aggregation is reasonable because it accounts for the main sources of radiological dose and does not underestimate annual doses to offsite persons during normal operations.

Dose Consequences for Normal Operations and Category 1 Event Sequences

Because no Category 1 event sequences were identified in DOE'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. DOE assumed an annual exposure duration of 2,000 hours for an onsite member of the public. DOE 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, DOE 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 RF {0.76 mSv/year [76 mrem/year]}.

For the offsite public not in the general environment where access is controlled, DOE determined dose rates to be 0.0011 mSv/year [0.11 mrem/year] TEDE. For the offsite public in the general environment, DOE determined that, due to the long distance to the closest location in the general environment, direct radiation and skyshine contributions to dose were negligible. DOE discussed airborne exposures that result from normal operations in SAR Section 1.8.3.1.2. DOE indicated that the estimated TEDE was 0.0011 mSv/year [0.11 mrem/yr] (SAR Table 1.8-36). 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}.

In SAR Table 1.8-36, DOE presented consequence analysis results for radiation workers of 0.013 Sv/year [1.3 rem/year] TEDE. DOE 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 Enclosure 2, Table 4 (2009el). DOE reported a maximum lens dose equivalent of 0.013 Sv/year [1.3 rem/year] in DOE Enclosure 2, Table 4 (2009el).

NRC Staff Evaluation: The NRC staff reviewed DOE's dose consequences for workers and members of the public from normal operations and Category 1 event sequences using the guidance in the YMRP. 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, DOE's estimated dose consequences for normal operations are reasonable because DOE calculations are based on standard methods and reasonable assumptions.

DOE's approach for calculating TEDE is reasonable. DOE's approach for calculating the lens dose equivalent, as described in SAR Equation 1.8-11 and DOE Enclosure 2, Table 4, Notes on Formula for Column D (2009el), is reasonable because it is consistent with NRC regulatory guidance (refer to TER Section 2.1.1.5.3.1 for additional information). For a given calculated result of TEDE, the NRC staff notes that DOE's approach in SAR Equation 1.8-7 and DOE Enclosure 2, Table 4, Notes on Formula for Columns B and C (2009el) could underestimate the dose equivalent to the maximally exposed organ and shallow dose equivalent to the skin. DOE did not calculate organ dose equivalents for a dominant direct exposure pathway. Because organ dose equivalents can exceed the effective dose equivalent due to external irradiation, the NRC staff evaluated the potential uncertainty associated with DOE'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 investigation is described in following three paragraphs.

For external irradiations by gamma rays, the organ dose equivalent to the bone surface typically exceeds the effective dose equivalent, as detailed in EPA Table III.3 (1993aa) and International Commission on Radiological Protection Tables A.2 to A.20 (1996aa). 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 Tables III.1 and III.3 (1993aa) for these proxy sources to approximate the maximum uncertainty in organ dose equivalents from direct exposure. The NRC staff notes 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 recent information on effective dose by the International Commission on Radiological Protection Tables A.2 to A.20 (1996aa) 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 Tables A.26 to A.20 (1996aa). Using the energy-dependent radiation weighting factors for neutrons from International Commission on Radiological Protection Table 2 (1996aa), the NRC staff also compared the maximum organ doses to effective doses at several neutron energies between 0.001 eV and 10 MeV. 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 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 RF (SAR Table 1.8-36) with the bounding uncertainty for a maximally exposed organ of a factor of three for direct irradiation, the maximum organ dose equivalent would not exceed 0.04 Sv [4 rem]. On the basis of the previously described evaluation, DOE's radiation worker dose calculations are reasonable for normal operations.

<u>Dose Consequences for Category 2 Event Sequences</u>

DOE performed public dose consequence analyses for Category 2 event sequences. The GROA is designed to limit dose to any individual located on, or beyond, any point on the boundary of the site. For external exposure, DOE used the effective dose equivalent in place of the deep dose equivalent.

Because of the distances involved, DOE 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. DOE calculated the highest TEDE for the offsite public in the general environment to be 0.0001 Sv [0.01 rem].

NRC Staff Evaluation: The NRC staff reviewed DOE's dose consequence for members of the public from a single Category 2 event sequence using the guidance in the YMRP and notes that the Category 2 consequence analyses do not underestimate dose. The estimated consequences to offsite members of the public from a single Category 2 event sequence in SAR Table 1.8-36 are reasonable because DOE used bounding Category 2 event sequences.

2.1.1.5.4 NRC Staff Conclusions

The NRC staff notes that the information DOE provided relevant to the dose consequence analysis is consistent with the guidance in the YMRP. The NRC staff also notes that DOE used reasonable methodology and input parameters for dose calculations for public and radiation worker dose determinations as discussed in this chapter.

2.1.1.5.5 References

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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

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of DOE's identification of structures, systems, and components (SSCs) important to safety (ITS), safety controls, and measures to ensure availability and reliability of the safety systems. SSCs are identified as ITS if they are relied upon to provide safety of high-level radioactive waste (HLW) operations for Category 1 and Category 2 event sequences. Category 1 event sequences are those expected to occur one or more times before permanent closure, and Category 2 event sequences are other event sequences having at least 1 chance in 10,000 of occurring before permanent closure. The NRC staff evaluated the information provided in the Safety Analysis Report (SAR) Section 1.9 (DOE, 2008ab), supporting documents, and DOE's responses to NRC staff's requests for additional information (RAIs) (DOE, 2009dk,dq,fc,fm–fr).

To identify the SSCs as ITS, DOE performed a preclosure safety analysis (PCSA). DOE provided criteria used in ITS SSCs identification and listed SSCs ITS and associated nuclear safety design bases in SAR Section 1.9.1. SAR Section 1.9.1.14 summarized consideration of interactions between SSCs ITS and SSCs not important to safety (non-ITS). SAR Sections 1.9.3 and 1.9.4 discussed procedural safety controls (PSCs) and SSCs ITS risk significance categorizations.

2.1.1.6.2 Evaluation Criteria

The regulatory requirements for the PCSA to identify SSCs ITS, PSCs, and measures taken to ensure the availability of safety functions are contained in 10 CFR 63.112(e). Specifically, 10 CFR 63.112(e) requires that the PCSA must consider the means to limit the radioactive material concentration in air [63.112(e)(1)]; means to limit the time required to perform work near the radioactive materials [63.112(e)(2)]; shielding protection [63.112(e)(3)]; monitoring and controlling dispersal of radioactive contamination [63.112(e)(4)]; access control to high radiation or airborne radioactivity areas [63.112(e)(5)]; criticality control and prevention [63.112(e)(6)]; radiation alarm [63.112(e)(7)]; ability of SSCs ITS to perform their intended safety functions [63.112(e)(8)]; fire detection and suppression [63.112(e)(9)]; radioactive waste and effluent controls [63.112(e)(10)]; means to provide timely and reliable emergency power [63.112(e)(11)]; redundant systems [63.112(e)(12)]; and SSCs ITS inspection, testing, and maintenance [63.112(e)(13)].

The NRC staff reviewed DOE's SSCs ITS, safety controls, and measures to ensure availability of the safety systems using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The relevant acceptance criteria follow:

- DOE has provided a reasonable list of SSCs identified as being important to preclosure radiological safety, technical bases for the approaches used to identify SSCs ITS and safety controls, and a list and analysis of the measures to ensure the availability of the safety systems.

- Administrative controls or PSCs DOE identified to prevent event sequences or mitigate their effects are appropriate.

The review guidance related to the risk significance categorization of SSCs ITS in YMRP Section 2.1.1.6 is not applicable, because DOE did not perform the risk significance categorization of SSCs ITS and all SSCs ITS will be subject to the same level of quality assurance.

In addition, the NRC staff used additional applicable guidance, such as NRC standard review plans and regulatory guides, to support the NRC staff's review. These additional guidance documents are discussed in the relevant sections next.

2.1.1.6.3 Technical Evaluation

The NRC staff's review focuses on determining whether DOE's PCSA has reasonably identified the ITS SSCs. The NRC staff's focus was to determine that the ITS SSCs DOE relied upon to limit or prevent potential event sequences or mitigate their consequences and PSCs have been reasonably identified. The NRC staff also reviewed the information provided in the SAR, supporting documents, and DOE's responses to the NRC staff's RAIs to determine whether DOE's PCSA appropriately considered the means to limit the radioactive material concentration in air; the means to limit the time required to perform work near the radioactive materials; shielding protection; monitoring and controlling dispersal of radioactive contamination; access control to high radiation or airborne radioactivity areas; criticality control and prevention; radiation alarm; ability of SSCs ITS to perform their intended safety functions; fire detection and suppression; radioactive waste and effluent controls; the means to provide timely and reliable emergency power; redundant systems; and SSCs ITS inspection, testing, and maintenance.

The NRC staff organized its review of DOE's SSCs ITS, safety controls, and measures to ensure availability of the safety systems generally following the YMRP. The NRC staff's review of criticality control and prevention and of the ability of the ITS criticality controls to perform the intended safety functions appears in the Technical Evaluation Report (TER) Section 2.1.1.6.3.2.6. Consideration of fire detection and suppression is evaluated along with the review of ITS fire suppression systems to perform the intended safety functions in TER Section 2.1.1.6.3.2.8.4. Consideration of timely and reliable emergency power is evaluated along with the review of the ITS electrical power systems in TER Section 2.1.1.6.3.2.8.5. Consideration of redundant systems is evaluated in subsections of TER Section 2.1.1.6.3.2.8 because redundant systems are important aspects in assessing the ability of ITS SSCs to perform their intended safety functions.

The NRC staff also used the review results from TER Sections 2.1.1.3, 2.1.1.4, and 2.1.1.7, as appropriate, to support the review in this chapter.

2.1.1.6.3.1 List of Structures, Systems, and Components Important to Safety and Safety Controls

DOE presented the SSCs ITS, associated nuclear safety design bases, and risk significant categorization of SSCs ITS in SAR Sections 1.9.1 and 1.9.4. The NRC staff's review of this information is presented in this TER section, and its review of DOE's information on the means to limit the radioactive material concentration in air; the means to limit the time required to perform work near the radioactive materials; shielding protection; monitoring and controlling dispersal of radioactive contamination; access control to high radiation or airborne radioactivity areas; criticality control and prevention; radiation alarm; ability of SSCs ITS to perform their intended safety functions; fire detection and suppression; radioactive waste and effluent controls; the means to provide timely and reliable emergency power; redundant systems; and SSCs ITS inspection, testing, and maintenance is presented in TER Section 2.1.1.6.3.2.

<u>SSCs ITS Identification</u>

DOE performed a PCSA to identify SSCs ITS and define nuclear safety design bases associated with the SSCs ITS identified. The PCSA was conducted using site-specific information (external hazards including both natural and human induced) and facility-specific operational processes.

DOE developed and used four criteria to identify SSCs ITS. If one or more criteria were met, an SSC would be classified as ITS. Following its identification criteria, DOE indicated that an SSC would (i) reduce the frequency of an event sequence from a Category 1 to Category 2 event sequence, (ii) reduce the frequency of an event sequence from Category 2 to beyond Category 2 event sequence, (iii) reduce the aggregated dose of Category 1 event sequences by reducing the event sequence mean frequency, and (iv) perform a dose mitigation or criticality safety control function. The SSCs ITS were listed in SAR Table 1.9-1.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's criteria to identify SSCs ITS. The NRC staff notes that the identification criteria DOE developed are reasonable because these criteria are consistent with the definition of the SSCs ITS.

The NRC staff reviewed selected SSCs ITS DOE identified for its Canister Receipt and Closure Facility (CRCF) as listed in SAR Table 1.9-3 to determine whether the four criteria have been implemented to identify SSCs ITS. The NRC staff notes that DOE reasonably implemented the criteria to identify structures and systems ITS. For example, DOE classified the CRCF structure where various waste forms were handled to be ITS (SAR Table 1.9-1) because it is required to protect ITS SSCs inside the building from external events (SAR Table 1.9-3) and collapse of this structure could lead to an unacceptable consequence. Consequently, the CRCF structure is relied on to reduce the frequency of an event sequence to beyond a Category 2 event sequence. As another example, DOE classified the waste package system as ITS (SAR Table 1.9-1) because this system is required to provide confinement; otherwise, unacceptable consequences may result. The NRC staff notes that DOE reasonably implemented the criteria based on the NRC staff review of the initiating events and event sequences DOE developed and the safety function(s) an ITS structure or system is required to perform.

DOE identified ITS components for ITS systems. This information was provided in the figures for the process and instrumentation diagrams in SAR Section 1.2.4 for various ITS systems. The NRC staff was able to determine how DOE implemented its criteria to identify ITS components for some systems. For example, the NRC staff can associate ITS components, such as the grapple actuator and associated switches for a canister transfer machine (CTM) grapple (SAR Figure 1.2.4-64), with a safety function to prevent a grapple from dropping a load. However, for some ITS systems, it is not clear to the NRC staff how DOE implemented its criteria to identify the ITS components and interface of ITS equipment with non-ITS equipment. Even though the process and instrumentation diagrams provided in the SAR showed ITS components of the ITS systems, DOE did not discuss how these ITS components were identified. Although the NRC staff examined the corresponding fault trees, the NRC staff was not able to determine how the ITS components within the ITS heating, ventilating, and air conditioning systems (HVAC); instrumentation and control systems; and electric power systems were identified using DOE's criteria. DOE responded to the NRC staff's RAI (DOE, 2009dq) to address this concern regarding HVAC systems, but DOE's response did not provide a transparent description of how it applied its criteria for identifying ITS components or how it linked the components it did identify to basic events in fault trees (see additional discussions in TER Sections 2.1.1.4.3.2.1 and 2.1.1.6.3.2.8.2.2). This information is important for quantifying the reliability of ITS systems and for determining how DOE implemented interfaces (i.e., physical, electrical, or logical) between ITS and non-ITS equipment. The NRC staff notes that the key purpose of the ITS components is to ensure that the safety functions assigned for the ITS system that contains these ITS components can be achieved and maintained. So long as the safety functions for the ITS system are maintained, the selection or identification of the associated ITS components may change as the design evolves. The NRC staff further notes that the safety functions for the ITS system can be achieved and maintained through implementing equipment qualification and maintenance programs as DOE stated in SAR Sections 1.13 and 1.9.1.13. As part of the detailed design process, DOE should confirm that the identification of ITS components and the associated nuclear safety design bases are consistent with the design.

Nuclear Safety Design Bases

DOE developed the nuclear safety design bases for the SSCs ITS from the PCSA event sequence analyses. The design bases include the required safety functions of the SSCs ITS and the associated controlling parameters and values. These nuclear safety design bases stipulate that SSCs ITS are required to prevent occurrence or mitigate consequences of event sequences. SAR Tables 1.9-2 through 1.9-7 listed the design bases. DOE also provided safety functions for ITS controls in response to an NRC staff RAI (DOE, 2009dk).

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's nuclear safety design bases for the ITS SSCs. The NRC staff performed an evaluation to determine whether the design bases for SSCs ITS are developed on the basis of the PCSA results. The evaluation included review of DOE's SAR and several supporting documents related to reliability and event sequence categorization analyses (BSC, 2008ac,as,be). DOE developed these nuclear safety design bases through the process of PCSA and that SAR Tables 1.9-2 through 1.9-7 provided a representative event sequence for each design basis. The NRC staff notes that DOE reasonably identified nuclear safety design bases for most ITS SSCs and these nuclear safety design bases are appropriate to reduce event sequence frequencies or mitigate consequences. However, DOE missed identifying design bases for some ITS SSCs. For example, the NRC staff's risk-informed evaluation of DOE's identification of initiating events in selected areas performed in TER Sections 2.1.1.3.3.2.3.4.1, 2.1.1.3.3.2.3.4.4, and 2.1.1.4.3.4.1.1 notes that DOE did not provide nuclear safety design bases for several initiating events that it screened out on the basis of design or passive controls, including (i) waste packages to preclude a lid drop onto a DOE or HLW canister in unsealed waste packages, (ii) the CTM slide gate and supporting structures including guide sleeve to screen out initiating events related to canister drops, and (iii) CTM ITS interlocks to be designed against water spray to prevent workers from direct radiation exposure or prevent shield doors from inadvertent opening due to spurious signals caused by water spray. The NRC staff recognizes that the detailed designs for ITS SSCs are not currently completed. As part of the detailed design process, DOE should confirm that the safety functions identified in the PCSA for passive and active systems that are credited to screen out initiating events are consistent with the design.

2.1.1.6.3.2 Structures, Systems, and Components Important to Safety and Safety Controls

The NRC staff focused its review in this section on determining whether DOE appropriately considered the means to limit the radioactive material concentration in air; the means to limit the time required to perform work near the radioactive materials; shielding protection; monitoring and controlling dispersal of radioactive contamination; access control to high radiation or airborne radioactivity areas; criticality control and prevention; radiation alarm; ability of SSCs ITS to perform their intended safety functions; fire detection and suppression; radioactive waste and effluent controls; means to provide timely and reliable emergency power; redundant systems; and SSCs ITS inspection, testing, and maintenance in its PSCA.

Review of DOE's consideration on each aspect discussed previously is provided in individual sections with some exceptions discussed in TER Section 2.1.1.6.3. The NRC staff also used the review results from TER Sections 2.1.1.3, 2.1.1.4, and 2.1.1.7 to streamline and support the review.

2.1.1.6.3.2.1 Limiting Concentration of Radioactive Material in Air

DOE discussed equipment and facility designs that are intended to limit the concentration of radioactive material in air to assess the potential that such equipment may be designated as ITS in SAR Section 1.9.1.1. DOE relied on 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. DOE further indicated that the HVAC systems design is consistent with Regulatory Guide 8.8, Regulatory Position C.2.d (NRC, 1978ab). All surface facilities would be equipped with HVAC systems, but only those components of HVAC systems in the CRCF and Wet Handling Facility (WHF) that exhaust air from areas with a potential breach of the waste container were designated as ITS.

Other potential sources of radioactive material in air are subsurface releases from radioactive sources such as resuspension of external surface contamination from the waste packages and neutron activation of air and dust.

According to SAR Section 1.9.1.1, radiation-monitoring systems that DOE will use consist of monitors and alarms. These radiation-monitoring systems are for area radiation, continuous air, and airborne radioactivity effluent monitoring. Alarms are triggered by high radiation levels and are provided locally at the potential release point, at the Central Control Center, and on appropriate consoles in the facility operations room to alert operators of radiological releases or extreme radiation conditions. Radiation alarm systems are further discussed in TER Section 2.1.1.6.3.2.7. Airborne radioactivity effluent monitors, in designated release points in surface facilities, routinely monitor sampled air.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP and Regulatory Guide 8.8 (NRC, 1978ab) to review SAR Section 1.9.1.1. The NRC staff notes that the use of HVAC systems to limit the concentration of radioactive material in the air is reasonable because the design of the ITS HVAC systems follows Regulatory Guide 8.8, Regulatory Position C.2.d (NRC, 1978ab). In addition, the radiation is monitored at the source point, manned operating stations, or other locations in the surface facilities. Therefore, DOE's ITS designation is reasonable for those HVAC systems in the CRCF and WHF because DOE did not rely on the HVAC systems in the PCSA to mitigate a radioactive release in the other areas. Evaluation of the HVAC systems DOE proposed using to limit concentration of radioactive material in air is provided in TER Section 2.1.1.6.3.2.8.2.2.

2.1.1.6.3.2.2 Limiting Worker Exposure Time When Performing Work

DOE discussed equipment and facility designs that are intended to limit the exposure time when performing work in radiation areas to assess the potential that such equipment may be designated as ITS. DOE provided this information in SAR Section 1.9.1.2. DOE determined that limiting the time required for workers to perform activities in radiation areas during normal operations was not needed to prevent or mitigate any identified event sequence. Instead, it is part of DOE’s as low as is reasonably achievable (ALARA) principles and program. Therefore, these design features are non-ITS. TER Section 2.1.1.8 provides the NRC staff’s review of DOE’s ALARA program and notes that DOE reasonably described how it would incorporate the ALARA principles in the proposed operations.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review SAR Section 1.9.1.2 to determine whether DOE reasonably considered the means to limit the required time to perform work in radiological areas in its PCSA. The NRC staff notes that the ALARA program is part of normal operations, and there are no design features associated with limiting worker exposure time that are relied upon to prevent a Category 1 or Category 2 event sequence. Therefore, DOE correctly designates limiting worker exposure time when performing work as non-ITS.

2.1.1.6.3.2.3 Shielding Protection

DOE discussed equipment and facility designs that are intended to provide shielding protection to assess the potential that such equipment may be designated as ITS. DOE provided this information in SAR Section 1.9.1.3. Shielding is an important aspect of worker safety because the waste packages and canisters contain significant amounts of radioactivity that could expose personnel to lethal doses. DOE indicated that it relied on the shielding design for normal operations and Category 1 and Category 2 event sequences. DOE designated shielding as non-ITS, where the shielding is used exclusively during normal operations. Although no Category 1 event sequences were identified in DOE’s PCSA, DOE determined that some shielding features were 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, therefore designating it ITS. DOE did not include shielding features in the consequence evaluation of Category 2 event sequences, because it did not take credit for shielding during Category 2 event sequences in the PCSA.

NRC Staff Evaluation: The NRC staff reviewed DOE’s information in SAR Section 1.9.1.3 using the guidance in the YMRP and HLWRS ISG–03 (NRC, 2007ac). The NRC staff’s evaluation was to determine whether DOE reasonably considered the need for shielding to limit worker exposure in its PCSA. The NRC staff notes that the shielding features used exclusively for normal operations are non-ITS, because they are not relied upon to reduce the mean frequency of an event sequence or mitigate radiological consequences. Further, DOE’s designation of the shielding features that are required to reduce the probability of an event sequence to below Category 1 as ITS is reasonable because these features are needed to prevent Category 1 event sequences.

2.1.1.6.3.2.4 Radioactive Contamination Dispersal Monitoring and Control

DOE discussed equipment and facility designs that are intended to provide radioactive contamination dispersal monitoring and control to assess the potential that such equipment may be designated as ITS. DOE provided this information in SAR Section 1.9.1.4.

DOE described how it will continuously monitor release points in surface process facilities, where airborne radioactivity effluent monitors sample the effluent stream for airborne radioactivity particulates and gases. The monitors alert operators to off-normal conditions such as radiological releases or high levels of radiation. Because the radiation/radiological monitoring system does not initiate automatic actions required to reduce the event sequence frequency or mitigate the consequences of an event sequence, DOE designated radioactive contamination dispersal monitoring and control as non-ITS. DOE described the capability to monitor radioactive effluents in SAR Section 1.4.2.2.

DOE's HVAC systems are designed to minimize the spread of radioactive contamination by controlling air flows from areas of low potential contamination to areas of higher potential contamination. DOE designated the portions of the surface confinement HVAC system that exhaust from areas with a potential for breach in the CRCF and WHF as ITS. The HVAC systems in the IHF and RF and the subsurface ventilation systems are non-ITS. The subsurface ventilation system was non-ITS, because all event sequences involving a waste package breach are classified as beyond Category 2. DOE 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 and provided additional details on how radioactive contamination is monitored and controlled.

NRC Staff Evaluation: The NRC staff reviewed SAR Section 1.9.1.4 using the guidance provided in the YMRP. The NRC staff's evaluation focused on determining whether DOE reasonably considered radioactive contamination dispersal monitoring and control in its PCSA. The NRC staff notes that DOE appropriately designated those portions of the HVAC system that exhaust from areas with a potential for a breach in the CRCF and WHF as ITS. The NRC staff evaluates the adequacy of the HVAC systems controlling the dispersal of radioactive contamination in the surface facilities in TER Section 2.1.1.6.3.2.8.2.2. On the basis of the evaluations performed in this section and TER Section 2.1.1.6.3.2.8.2.2, the NRC staff notes that DOE does not rely on radioactive contamination dispersal monitoring and control systems to mitigate the effects or reduce the dose from Category 1 or 2 event sequences; therefore, DOE's non-ITS designation for the radioactive contamination dispersal monitoring and control systems is reasonable.

2.1.1.6.3.2.5 Access Control to High Radiation Areas and Airborne Radioactivity Areas

DOE discussed equipment and facility designs that are intended to provide access control to high radiation areas to assess the potential that such equipment may be designated as ITS. DOE provided this information in SAR Section 1.9.1.5.

DOE stated that controlling personnel access to normally unoccupied high radiation areas, very high radiation areas, or airborne radioactivity areas is part of normal operations and is not relied upon for prevention or mitigation of Category 1 or Category 2 event sequences. Therefore, DOE designated these design features as non-ITS. For those areas requiring periodic personnel access for waste handling operations where the radiation levels are subject to change as a result of Category 1 or Category 2 event sequences, DOE identified PSCs or

SSCs ITS to provide access controls. These access controls include interlocks or other positive controls on shield doors and administrative procedures.

With respect to airborne radioactivity areas, DOE's analyses of airborne radioactivity areas provided the technical basis for the inability to practically apply process or other engineering controls to restrict concentrations of radioactive material in air to values below those that define an airborne radioactivity area in accordance with guidance in Regulatory Guide 8.38 (NRC, 2006ac). DOE provided its plan for monitoring and limiting intakes of radiation. This plan will include controlling access, limiting individual exposure times, and using individual respiratory protection equipment in accordance with guidance in Regulatory Guide 8.38 (NRC, 2006ac).

NRC Staff Evaluation: The NRC staff reviewed SAR Section 1.9.1.5 using the guidance in the YMRP and Regulatory Guide 8.38 (NRC, 2006ac). The NRC staff's evaluation was to determine whether DOE reasonably considered access control to high radiation areas and airborne radioactivity areas in its PCSA. DOE's designation of access control to high radiation areas or airborne radioactivity areas as non-ITS is reasonable because the access control is not relied on to prevent any Category 1 or Category 2 event sequences or mitigate the consequences of these event sequences in the PCSA. Furthermore, the access-control-related PSCs or SSCs ITS DOE identified are reasonable for those areas requiring periodic personnel access for waste handling operations where the radiation levels are subject to change as a result of Category 1 or Category 2 event sequences.

2.1.1.6.3.2.6 Criticality Control and Prevention and Ability to Perform Safety Functions

DOE presented its criticality safety program in SAR Section 1.14 and BSC (2008ba). The goal of the program is to prevent criticality during the preclosure period for normal conditions and Category 1 and Category 2 event sequences.

DOE provided information in response to the NRC staff's RAIs (DOE, 2009az) about the organization of DOE's proposed criticality safety program. DOE described the use of the organizational structure to implement the program in SAR Section 5.3. In its response to an NRC staff RAI (DOE, 2009az), DOE stated that it will revise SAR Figure 5.3-1 to show that the radiation protection and criticality safety manager reports directly to the site operations manager and is therefore independent of operations. The waste handling manager ensures the design bases are maintained and implements the actual criticality safety measures (DOE, 2009az).

DOE stated that it followed the applicable portions of Regulatory Guide 3.71 (NRC, 2005ac), which endorses certain criticality-related industry standards. To show that the program is robust and is based upon industry standard practices, DOE listed the American National Standards Institute/American Nuclear Society (ANSI/ANS) standards used in SAR Section 1.14.3.1. For example, DOE's criticality safety training will be developed in accordance with ANSI/ANS–8.20–1991 (American Nuclear Society, 1991ab). DOE stated that criticality safety practices and procedures will be developed and criticality safety audits and assessments will also be performed in accordance with its quality assurance program described in SAR Section 5.1 and ANSI/ANS–8.19–2005 (American Nuclear Society, 2005aa). DOE also used other endorsed standards as shown in SAR Section 1.14.3.1 related to such things as moderator control, use of neutron absorbers, and validation of criticality safety analysis.

In BSC (2008ba), DOE presented a detailed preclosure criticality safety evaluation, which is used to show criticality prevention. DOE evaluated seven parameters to determine whether they need to be controlled to prevent criticality during the preclosure period: waste form characteristics, moderation, fixed neutron absorbers, soluble neutron absorbers, geometry, interaction, and reflection. For each parameter, DOE performed criticality sensitivity calculations for the different fuel types and conditions. DOE 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 upon its sensitivity analyses, DOE determined that moderation is the primary criticality control parameter for commercial SNF. DOE determined that none of the event sequence descriptions includes potential for moderator to come in contact with fissile materials (SAR Tables 1.7-7 through 1.7-17). Therefore, DOE screened out all initiating events important to criticality.

DOE identified the SSCs relied upon to maintain subcriticality by preventing moderator from coming into contact with the fissile material as ITS in the PCSA. SAR Tables 1.9-2 through 1.9-7 listed three ITS systems providing containment throughout the Geologic Repository Operations Area (GROA) facilities: the DOE and commercial waste package system, naval SNF waste package system, and mechanical handling system. In addition, SAR Tables 1.9-3 and 1.9-4 listed two ITS systems (Mechanical Handling System in the CRCF and WHF and Fire Protection System in the CRCF) that have a moderator control safety function. DOE also relied on PSC–9 to maintain the concentration of boron (enriched to 90 percent of isotope boron-10) to above 2,500 mg/L [0.02 lb/gal] in the WHF pool and cask/canister.

DOE identified two other SSCs (DOE canister staging racks in the CRCF and staging racks in the WHF pool) as ITS to prevent criticality. These rely on controlling spacing to perform their ITS functions. The staging racks in the WHF pool also contain non-ITS neutron absorbers that are used to provide extra margin.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and Regulatory Guide 8.5 (NRC, 1981ac) to determine whether DOE reasonably considered criticality control and prevention in its PCSA and the related SSCs ITS could perform their intended safety functions. In addition, the NRC staff used the standards endorsed in Regulatory Guide 3.71 (NRC, 2005ac) to review DOE's design. DOE relied primarily 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.

DOE's use of the ANSI/ANS–8 standards listed in SAR Section 1.14.3 is consistent with Regulatory Guide 3.71 (NRC, 2005ac). Therefore, DOE's program is robust and based on industry standard practices. The NRC staff reviewed DOE's statement (DOE, 2009az) to determine whether DOE's criticality safety program is administratively independent of operations and whether its functions and responsibilities, such as development of procedures, assisting with training, and audits of safety practices, are consistent with those of other safety programs DOE proposed to use for its surface facility operations such as radiation protection. DOE's program follows ANSI/ANS–8.19–2005 (American Nuclear Society, 2005aa), which is consistent with Regulatory Guide 3.71 (NRC, 2005ac). DOE's statement in SAR Section 1.14.1 of developing a personnel training program in accordance with ANSI/ANS–8.20–1991 (American Nuclear Society, 1991ab) makes crediting PSC–9 appropriate because personnel will be trained and familiar with the importance of the soluble neutron absorber to criticality safety.

DOE followed ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac) by controlling moderators through the use of ITS systems, such as the double-interlock preaction (DIPA) sprinklers in the fire protection system. In addition, the NRC staff reviewed DOE's technical basis for identifying fixed neutron absorbers as non-ITS. The NRC staff notes that the technical basis is reasonable because fixed neutron absorbers are used to provide an additional margin of safety.

2.1.1.6.3.2.7 Radiation Alarm System

DOE 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 goal of the system is to monitor the surface and subsurface areas and effluents from the GROA release points, and to provide alarms and radiation level indication to personnel.

In SAR Section 1.9.1.7, DOE identified the radiation monitoring system as non-ITS because it is not relied upon to alert the operator to take actions in response to an event sequence and does not initiate automatic actions required to prevent or mitigate an event sequence. The system will sound an alarm when a threshold radiation level is reached.

The three major components of the radiation monitoring system are area radiation monitors, continuous air monitors, and airborne radioactivity effluent monitors (SAR Section 1.4.2.2). An uninterruptible power supply (UPS) powers the radiation monitoring system. DOE referenced ANSI/ANS–HPSSC–6.8.1–1981 (American Nuclear Society, 1981aa) in designing and positioning the area radiation monitors. Continuous air monitors use the methods and practices of ANSI N42.17B–1989 (American National Standards Institute, 1989aa) and will be located throughout the surface facilities and subsurface waste emplacement area. The airborne radioactivity effluent monitors will be located in exhaust stacks. The alarms and data collected by the radiation detectors will be available both locally and in the Central Control Center.

NRC Staff Evaluation: The NRC staff reviewed the information on radiation alarm system using the guidance in the YMRP and Regulatory Guide 8.5 (NRC, 1981ac) with the focus on determining whether DOE reasonably considered the radiation alarm system in its PCSA. The NRC staff reviewed the information presented in SAR Sections 1.4.2.2 and 1.9.1.7. On the basis of this evaluation, the NRC staff notes that the design codes and standards DOE selected (SAR Section 1.4.2.2.2) are appropriate to the radiation monitoring system design. The NRC staff compared the guidance of ANSI/ANS–HPSSC–6.8.1–1981 (American Nuclear Society, 1981aa) to that of ANSI/ANS N2.3–1979 (American Nuclear Society, 1979aa), which NRC endorsed in Regulatory Guide 8.5 (NRC, 1981ac), and notes that it is reasonable for designing a radiation alarm system to warn of significant increases in radiation levels, concentrations of radionuclides in air, and increased radioactivity in effluents. DOE's use of a UPS and the guidance of the listed standards concerning location, inspection, maintenance, and testing provides reasonable means to ensure that the radiation monitoring system will be able to promptly notify personnel of an increase in radiation levels.

In addition, DOE's approach of not having fixed neutron radiation detectors as part of the alarm system is reasonable because, as stated in SAR Section 5.11.2, DOE would include surveying for and monitoring dose from neutron radiation in its Operational Radiation Protection Program.

2.1.1.6.3.2.8 Ability of Structures, Systems, and Components Important to Safety to Perform Intended Safety Functions

DOE discussed the reliability of the SSCs ITS and PSCs to perform their intended safety functions to prevent and mitigate potential event sequences on the basis of reliability assessments in SAR Section 1.7. To provide additional assurance that the SSCs ITS will perform their safety functions to an appropriate level of reliability, DOE stated that it will qualify these SSCs ITS for the range of environmental conditions anticipated at the time of functional demand. Additionally, DOE stated that it will implement a monitoring program to detect operation deviations of SSCs ITS to permit appropriate corrective actions. DOE also will develop reliability-centered maintenance, inspection, and testing programs for the ITS SSCs, as necessary, to ensure their continued functioning and readiness.

In the following subsections, the NRC staff evaluated several selected ITS SSCs DOE identified and listed in SAR Tables 1.9-2 through 1.9-8 and 1.9-10 to determine whether the SSCs ITS DOE identified through its PCSA will perform their intended safety functions. The results drawn from this limited evaluation will be equally applicable to other ITS SSCs because DOE used the same reliability assessment approach described in SAR Section 1.7 to show the ability of all ITS SSCs to perform their intended safety functions.

The NRC staff evaluated the following ITS SSCs: surface structural and civil facilities; mechanical systems; transportation systems; electrical components and emergency power systems (EPS); fire protection systems; transportation, aging, and disposal (TAD) canisters; and waste packages.

2.1.1.6.3.2.8.1 Surface Structural/Civil Facilities Important to Safety

DOE provided descriptions and design information for the surface structures in SAR Section 1.2 and discussions on reliability used in event sequence categorization in SAR Section 1.7. DOE also addressed the ability of ITS surface facility structures to perform their intended safety functions in SAR Section 1.9.1.8.

The ITS surface facility structures include the IHF, CRCF, WHF, and RF. DOE identified nuclear safety design bases and criteria for the ITS surface facility structures in SAR Tables 1.9-2 to 1.9-7 and stated that the process of determining design bases was based on the PCSA, as shown in SAR Figure 1.7-1. DOE relies 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 nuclear safety design bases require that the mean annual probability of building collapse should not exceed 10^{-6}/year (probability threshold for Category 2 event sequences) for (i) straight wind and (ii) volcanic ash fall. DOE sets the design basis straight wind at 193 km/h [120 mph], corresponding to a mean annual probability of 1×10^{-6}, and design basis volcanic ash load on the roof at 1.0 kPa [21 lb/ft^2] (SAR Table 1.2.2.-1), corresponding to a mean annual probability of 6.4×10^{-8}. Because the design bases straight wind and volcanic ash roof load are either at or less than probability threshold for Category 2 event sequences, wind and volcanic ash fall are not expected to initiate event sequences.

For preventing seismically initiated event sequences from collapsing buildings, DOE indicated that the mean annual probability of failure of a building collapse due to a spectrum of seismic events will be less than or equal to 2×10^{-6}. This threshold value is based on event sequences that have at least 1 chance in 10,000 of occurrence over the 50-year period. DOE assumed a preclosure period of 100 years and operational period of 50 years for the surface facilities involving SNF and HLW handling. The design basis ground motion for all the facility structures and the structural design was addressed in SAR Section 1.2.2 and the supporting documents. For each facility, DOE determined the seismic fragility or mean probability of unacceptable performance as a function of ground motion. The fragility for the IHF, CRCF, WHF, and RF was calculated for imminent collapse or Limit State A (American Society of Civil Engineers, 2005aa), and the fragility parameters were shown in BSC Table 6.2-1 (2008bg). The structural performance or annual probability of failure was evaluated by convolving the fragility curves and the site-specific seismic hazard curve. As shown in BSC Table 6.2-1 (2008bg), DOE estimated the frequency of failure (per year) to be 3.8×10^{-7}, 4.1×10^{-7}, 7.8×10^{-7}, and 8.8×10^{-7} for the IHF, RF, CRCF, and WHF, respectively (BSC, 2008bg). DOE stated that the annual probability of failure for all the surface facility structures is less than the design basis threshold of 2×10^{-6}.

NRC Staff Evaluation: The NRC staff reviewed the safety functions and basis for controlling parameters and their relations to the PCSA using the guidance in the YMRP. The NRC staff's assessment of the ability of the facility structures to perform the intended safety functions and meet DOE's nuclear safety design bases and criteria for wind, ash fall, and seismic events is based on the evaluation of the structural design in TER Section 2.1.1.7 and the structural performance in TER Section 2.1.1.4. In SAR Section 1.6.3.4.4, DOE estimated the 10^{-6}/yr 3-second gust straight wind to be 193 km/h [120 mph]. The maximum design tornado wind speed for structural facilities is 304 km/h [189 mph] (SAR Table 1.2.2-1). DOE indicated that even though the design tornado wind speed exceeds the mean frequency 10^{-6}/yr straight wind speed by a large margin, the damage probability will be below the screening probability of 10^{-6}/yr. The NRC staff's evaluation of the probability of tornado effect on structural damage is given in TER Section 2.1.1.3.3.1.2.3, where the NRC staff notes that the probability of structural damage caused by tornado wind speed DOE estimated is appropriate. DOE's justification for the design ash load is evaluated in TER Section 2.1.1.1.3.6 where the NRC staff notes that DOE's estimation of ash fall load is reasonable.

The NRC staff's evaluation of the ITS facility structural design is provided in TER Section 2.1.1.7.3.1.1, which notes that DOE'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 reasonable. The NRC staff's evaluation of DOE's seismic performance or probability of failure determination to support review of DOE's facility structure design and fragility parameters is provided in TER Section 2.1.1.4.3.3.4.2.1, where the NRC staff notes that the mean annual probability of unacceptable performance of structural collapse for all surface facility structures is less than the threshold value of 2×10^{-6}/year. On the basis of NRC staff's evaluation in those two TER sections, the NRC staff notes that DOE's assessment of the ability of ITS buildings to perform their intended safety function is reasonable because the seismic structural design is in accordance with codes and standards and seismic performance or probability of structural collapse is below the Category 2 threshold.

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

DOE discussed the ability of the ITS mechanical handling systems to perform their intended safety functions if an event sequence occurs. The NRC staff reviewed five major groups of ITS mechanical handling systems DOE identified [CTM, waste package transfer trolley (WPTT), cask handling crane (CHC), spent fuel transfer machine (SFTM), and canister transfer trolley (CTT)]. These five groups are representative of all ITS mechanical handling systems that will be used in the GROA facilities. These five ITS mechanical handling system groups were described in SAR Sections 1.2.3 (IHF), 1.2.4 (CRCF), 1.2.5 (WHF), and 1.2.6 (RF); start up, maintenance, inspection, and testing were detailed in SAR Sections 5.5 and 5.6.

Mechanical Handling Systems

DOE relies on five ITS mechanical handling systems to prevent or mitigate event sequences related to handling of commercial SNF assemblies, canisters, and casks. In general, all five systems limit movement speed. The CTM, WPTT, and CHC also have the ability to prevent spurious movement. DOE stated that the CTM, CHC, and SFTM will not collapse during a seismic event and will prevent a load drop. In addition, DOE stated that the WPTT will have the ability of preventing tipover or rocking during a seismic event and rapid tilt down. Furthermore, according to DOE, the CTM needs to protect personnel from direct exposure and the SFTM cannot lift a commercial SNF assembly beyond a safe limit. These safety functions (SAR Tables 1.9-2 through 1.9-5) are evaluated in TER Section 2.1.1.7.3.2.

DOE stated that it will use American Society of Mechanical Engineers (ASME) NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for Type 1 cranes in the design of all five ITS mechanical handling systems reviewed in this TER section. The single-failure-proof requirement of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), along with conservative safety margins for the load and speed of travel and specific requirements for Level I or II coatings to sustain reasonable radiation levels and temperature fluctuations DOE specified, provides the ITS mechanical handling systems with substantial design margin to perform their intended safety functions.

In several cases, DOE took additional measures to ensure safety of the ITS mechanical handling systems. For example, the CTM, CHC, and SFTM are designed with (i) integrated over speed switches to limit trolley/bridge over speeding; (ii) rope miss-pool 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 anticollision sensors to prevent collision during CTM operation (SAR Section 1.2.4.2.2.1). The WPTT is equipped with two redundant drive trains to rotate the shielded enclosure, either of which can support the enclosure. For the CTT, DOE used redundant air pressure regulators to control the air-bearing pressure so that the loss of one regulator does not cause lack of air supply to the entire CTT. The CHC is equipped with redundant lower and upper limit switches to ensure that the grapple cannot be raised or lowered beyond the safe limits. DOE also will use various interlocks to ensure safe operations of the ITS mechanical handling systems.

In addition to applying safe engineering practices and adhering to the safe margins of design in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), DOE will use PSCs for the ITS mechanical handling systems to prevent event sequences or mitigate their effects. DOE defined a PSC for the WPTT, verifying that personnel are outside the waste package positioning room and load-out area before the WPTT will move. To limit spurious CTT movement, operators are required 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, a procedure will 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, DOE included a corrective action program to document and evaluate equipment failures (SAR Section 1.13.2.6). DOE also described a corporate operating experience program using lessons learned at other facilities performing similar operations to develop operating procedures (SAR Section 5.6). Furthermore, DOE stated that it will develop comprehensive 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.

NRC Staff Evaluation: Using the guidance in the YMRP, the NRC staff reviewed the ability of the five mechanical handling systems to perform their intended safety functions if an event sequence occurs. The NRC staff notes that the codes and standards DOE planned to use to design and construct the ITS mechanical handling systems are applicable because these codes and standards have been used for related activities at other NRC-licensed facilities. DOE’s use of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) is reasonable because this standard recommends safe engineering practices and is commonly used by the nuclear industry and, in several cases, DOE included safe engineering principles that are above and beyond those recommended in this code for ITS mechanical handling systems. The NRC staff also notes that DOE’s PSCs and qualification program provide an additional measure of confidence that the mechanical handling systems will perform their intended safety functions if an event sequence occurs because they increase reliability of these systems. Finally, the NRC staff notes that DOE’s implementation of preventive and corrective maintenance, surveillance, and periodic testing of ITS mechanical handling systems, including instrumentation and controls, will help ensure that ITS SSCs will function as intended. DOE’s plan to develop reliability-centered maintenance programs for ITS SSCs is evaluated in TER Section 2.1.1.6.3.2.10, where the NRC staff notes that using the reliability-centered approach to develop maintenance, testing, and inspection programs for the ITS SSCs is reasonable. On the basis of the aforementioned measures described by DOE, DOE’s design information for the reviewed mechanical handling systems is reasonable to show that these systems will perform their intended safety functions.

Important to Safety Controls Linking the Mechanical Handling Systems and Shield Doors

DOE identified 29 key groups of ITS SSCs (SAR Table 1.4.2-1) that require ITS controls work properly to accomplish their safety functions. Included in the 29 key groups are safety controls (interlocks) for shield gates, doors, and other SSCs that are external to the mechanical handling systems and interact with them to protect personnel from inadvertent direct exposure to radiation.

In SAR Section 1.4.2, DOE indicated that all ITS controls will be made up of individual hardwired devices, instead of being driven by software or programmable devices. DOE further indicated that programmable components are limited to normal operating functions and the hardwired ITS controls will be integrated into the ITS SSC design to prevent normal-use, non-ITS controls from overriding any ITS control function. 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. When programmable logic controllers are used, their use is constrained by the operation of the hardwired ITS controls associated with the system under control.

DOE 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 shield gates, skirts, doors, and other exposure protection components. Although DOE cited the codes and standards for these ITS interlock controls [IEEE–308 (Institute of Electrical and Electronics Engineers, 2001aa); IEEE–379 (Institute of Electrical and Electronics Engineers, 2001ab); IEEE–603 (Institute of Electrical and Electronics Engineers, 1998ab); and IEEE–384 (Institute of Electrical and Electronics Engineers, 1998aa)], it stated that, on the basis of its PCSA, it will take exceptions to multiple criteria of these codes and standards for ITS interlock controls.

NRC Staff Evaluation: The NRC staff reviewed the ITS safety controls that link the mechanical handling systems and other ITS SSCs that are external to these safety controls, such as the shield doors and other exposure protection devices, using the guidance in the YMRP. DOE identified the codes and standards applicable to the ITS safety interlocks. The NRC staff notes that it is reasonable for DOE to exclude safety-related criteria from the cited codes and standards during iterative, detailed design of ITS controls because DOE showed through its PCSA that these criteria are not needed for the ITS interlocks to perform their safety functions.

2.1.1.6.3.2.8.2.2 Heating, Ventilation, and Air Conditioning Systems Important to Safety

DOE provided information to show how the ITS HVAC SSCs that are relied on to mitigate the consequences of a radionuclide release and support the ITS electrical function will perform 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 PSCs and measures to ensure the availability of ITS HVAC systems. DOE described 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 EDGF that are ITS and the portions of the surface nuclear confinement HVAC systems in the CRCF and WHF that are ITS. DOE 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) and in both of these tables for the ITS HVAC SSCs for the EDGF. DOE relies on the ITS HVAC SSCs to provide (i) cooling to ITS electrical equipment and battery rooms or (ii) nuclear confinement.

Procedural Safety Controls

DOE described PSCs in SAR Section 1.9.3, identifying that for HVAC systems, the PSCs may be included as part of normal operating procedures. DOE identified PSC–7 for the ITS HVAC subsystems in the CRCF and WHF in SAR Table 1.9-10. In PSC–7, DOE specified that one ITS HVAC train is required to be operating with the other one in standby before waste handling operations begin. DOE also specified PSC–7 in SAR Sections 1.2.4.4.4 (for the CRCF) and 1.2.5.5.4 (for the WHF). DOE described in SAR Section 1.2.8.3.1.4 that it did not identify any PSC for the ITS HVAC subsystems in the EDGF. DOE responded to an NRC staff RAI (DOE, 2009fq) on how PSC–7 ensures ITS components within the ITS HVAC subsystems would be available during operations and how the safety function is met for the ITS HVAC subsystems in the EDGF having not specified a PSC in SAR Section 1.2.8.3.1.4. DOE stated that for the surface nuclear confinement HVAC systems, PSC–7 extends to electrical distribution equipment required to operate ITS HVAC components and additionally specified its applicability to ITS components in those HVAC subsystems. Also, as part of this response, DOE stated that the EDGF ITS HVAC system is covered by PSC–8.

NRC Staff Evaluation: The NRC staff reviewed the identification of PSCs using the guidance in the YMRP. The NRC staff evaluated the PSCs DOE specified for ITS HVAC systems and notes that the detailed description of PSC–8 in SAR Section 1.4.1.2.4 addressed the support systems (including EDGF HVAC) for ITS diesel generators. On the basis of this review, DOE reasonably identified PSCs for the ITS HVAC subsystems in the surface facilities through the conduct of its PCSA. The NRC staff also notes that these PSCs are reasonable to ensure operational safety because the safety functions for these PSCs were developed from its PCSA.

Means To Limit the Concentration of Radioactive Material in Air

The HVAC systems in the surface facilities 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). In addition, the CRCF and WHF HVAC subsystems that exhaust from areas with the potential for a canister breach are credited as ITS in event sequences because they are used to mitigate the consequences of a radionuclide release.

DOE identified ANSI/ANS–57.9–1992 (American Nuclear Society, 1992aa) and ANSI/ANS–57.7–1988 (American Nuclear Society, 1988aa) for HVAC systems (SAR Table 1.2.2-12). Both of these standards specify a design that provides for air flow from areas with low potential for radioactive contamination to higher potential for radioactive contamination.

NRC Staff Evaluation: The NRC staff reviewed the means to limit the concentration of radioactive material in air using the guidance in the YMRP. The NRC staff notes that the air flow from areas with low to higher potential for radioactive contamination is appropriate because this approach is consistent with Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff evaluated DOE's HVAC design in TER Section 2.1.1.7.3.1.2.2 and notes that the ITS HVAC exhaust subsystem design with two stages of HEPA filtration is adequate to achieve the DOE-specified overall filtration efficiency. DOE reasonably 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 (American Nuclear Society, 1988aa) was withdrawn in October 2007 because it did not include maintenance provisions, the NRC staff notes that it provides appropriate guidance in designing HVAC systems. Therefore, DOE reasonably specified the means to limit the concentration of radioactive material in air. Note that the NRC staff's review of DOE's means to inspect, test, and maintain ITS HVAC SSCs is discussed later in this TER section.

Means To Control the Dispersal of Radioactive Contamination

The HVAC systems are relied on to control the dispersal of radioactive contamination, as identified in SAR Section 1.9.1.4. DOE 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. DOE 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. DOE stated that this designation is in accordance with DOE–HDBK–1169–2003, as described in DOE Section 2.2.9 (2003ae).

In response to an NRC staff RAI (DOE, 2009fo), DOE stated that confinement areas are designated as ITS if the area has an identified Category 2 event sequence in which a loaded canister may breach and an ITS HVAC system is used to mitigate the potential release. In addition, DOE indicated that it does not rely on seals through walls and slabs to maintain confinement and stated in its response (DOE, 2009fo) that it will update SAR Section 1.9.1.10 to reflect this intent. Furthermore, DOE also described in this response that air flows from non-ITS confinement areas into ITS confinement areas.

NRC Staff Evaluation: The NRC staff reviewed the means to control dispersal of radioactive contamination using the guidance in the YMRP; ANSI/ANS–57.9–1992 (American Nuclear Society, 1992aa); and ANSI/ANS–57.7–1988 (American Nuclear Society, 1988aa). Specifically, the NRC staff reviewed DOE Section 2.2.9 (2003ae) and notes that (i) this handbook is consistent with industry standard guidance and (ii) DOE's confinement zoning as described in the SAR is designated in accordance with this handbook (DOE, 2009fo). Therefore, DOE's confinement zoning approach discussed in its response to the NRC staff's RAI is reasonable. On the basis of the NRC staff's evaluation and DOE's statement to update SAR Section 1.9.1.10 (DOE, 2009fo), the HVAC system will control the dispersal of radioactive contamination.

Redundancy Within the ITS HVAC Systems

The ITS HVAC system is designed to have more than one HVAC train and redundant components within HVAC trains. As part of its design criteria for the ITS HVAC subsystems in the CRCF and WHF, DOE identified two full-capacity independent trains that are used to exhaust from areas where there is a potential for a canister breach (SAR Tables 1.2.4-4 and 1.2.5-3). DOE stated in SAR Section 1.9.1.12 that one train is in operation with the other one in standby and that the trains alternate between these modes. Independent trains are also a part of DOE's design criteria for the ITS HVAC subsystems in the CRCF, WHF, and EDGF that 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, DOE 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, DOE showed this redundancy in SAR Figure 1.2.4-104 for the subsystem that provides cooling to ITS electrical equipment and battery rooms in the CRCF where standby units start automatically if the operating units fail.

NRC Staff Evaluation: The NRC staff reviewed redundancy within the ITS HVAC systems using the guidance provided in the YMRP. Although DOE did not apply the single failure criterion to the CRCF HVAC control design (refer to TER Section 2.1.1.7.3.1.5 for a detailed discussion), DOE reasonably addressed redundancy in the ITS HVAC systems because it specified as part of its design criteria the use of independent trains and further stated in an RAI response (DOE, 2009fs) that the HVAC trains are independent because components in one train cannot cause failure of both trains.

Means To Inspect, Test, and Maintain ITS HVAC SSCs

DOE indicated that it will monitor and maintain ITS SSCs and, if required, take corrective actions to ensure the required reliabilities are achieved. The inspection, test, and maintenance programs were described in SAR Section 1.9.1.13.

For ITS HVAC systems, DOE 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 was designed with backup units in case 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, DOE did not show standby air handling units in SAR Figure 1.2.8-26. DOE responded to an NRC staff RAI (DOE, 2009fo) regarding how the maintenance on this system would be performed without backup air handling units and how the maintenance on an ITS subsystem in the EDGF would not adversely affect the reliability of ITS systems or subsystems in other facilities. DOE indicated that backup air handling units would not be required, because the EDGF ITS HVAC system is a nonconfinement HVAC system and the PCSA accounts for regularly scheduled maintenance. DOE identified basic events involving load center and ITS diesel generator maintenances and described that the diesel generator maintenance basic event accounts for ITS components or systems that would prevent the diesel generator from performing its function. DOE referred to BSC Table B8.4–1 and Figure B8.4–10 (2008ac), which described the basic event as “ITS DG A OOS Maintenance.”

NRC Staff Evaluation: The NRC staff reviewed the means to inspect, test, and maintain ITS HVAC SSCs using the guidance in the YMRP. The NRC staff notes that DOE’s basic event descriptions are not sufficiently detailed (e.g., “ITS DG A OOS Maintenance”) for the NRC staff to determine that the EDGF ITS HVAC system maintenance will not adversely impact other ITS systems. However, DOE stated in SAR Section 1.9.1.13 that it will develop inspection, testing, and maintenance programs for SSCs ITS using a reliability-centered maintenance approach and these problems would be subject to NRC review prior to receipt and possession of waste. Therefore, DOE reasonably addressed the means to inspect, test, and maintain ITS HVAC SSCs.

<u>Ability of the ITS HVAC SSCs To Perform Their Intended Safety Functions</u>

DOE specified nuclear safety design bases for the ITS HVAC systems in SAR Tables 1.9-3 (for the CRCF), 1.9-4 (for the WHF), and 1.9-3 and 1.9-4 (for the EDGF). As described in these tables, the ITS HVAC systems mitigate the consequences of a radionuclide release or support the ITS electrical function.

DOE identified certain component failures when quantifying the failure probability of a system or subsystem in its fault tree models. However, DOE did not explain how it designated components within an ITS system or ITS subsystem or non-ITS system and how these components are included in its quantification of controlling parameters in the SAR or the supporting documents. In its response to an NRC staff RAI (DOE, 2009dq), DOE stated that components are designated ITS in drawings, tables, and the text of the SAR but not specifically in fault trees. DOE further described how it specifically accounted for an ITS differential pressure switch in the CRCF ITS HVAC system in its response (DOE, 2009dq). It pointed out that the pressure switch is encompassed within a basic event.

NRC Staff Evaluation: The NRC staff reviewed HVAC information using the guidance in the YMRP. The NRC staff reviewed the nuclear safety design bases (including the safety functions) in TER Section 2.1.1.7.3.1.2.2 in terms of the HVAC system design to determine how the ITS HVAC systems 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 evaluated the controlling parameters for the nuclear safety design bases in TER Section 2.1.1.4.3.3.2.1 to determine whether DOE’s fault tree models support the controlling parameters specified such that the safety function could be accomplished in the mission time DOE specified.

The HVAC system failure probability was not clearly supported, because DOE’s use of component failures in the model was not always transparent (see detailed discussion in TER Section 2.1.1.3.3.2.3.4). In its response to an NRC staff RAI, DOE indicated that it specifically accounted for an ITS differential pressure switch in the CRCF ITS HVAC system by encompassing the pressure switch in a basic event (DOE, 2009dq). Although the description in this RAI response does not specifically refer to the pressure switch, but to the adjustable speed drive start logic, the NRC staff was able to use the description along with the drawings, tables, or sections of the SAR to relate this specific ITS component to a basic event in a fault tree. Therefore, the NRC staff notes that DOE identified ITS components and accounted for them when quantifying an ITS system's reliability. The NRC staff notes that this ITS system's reliability forms a part of the nuclear safety design basis to ensure the ITS system's ability to perform the intended safety functions. On the basis of the review in this section, the evaluation of DOE's RAI response (DOE, 2009dq), and the evaluation results in TER Section 2.1.1.6.3.1 pertaining to DOE's identification of ITS components and in TER Section 2.1.1.3.3.2.3.4 pertaining to transparency in specifying the components included in each basic event, the NRC staff notes that DOE reasonably addressed the ability of ITS HVAC SSCs to perform their intended safety functions.

2.1.1.6.3.2.8.3 Transportation Systems Important to Safety

DOE discussed the ability of the ITS transport and emplacement vehicle (TEV) and ITS intrasite transportation equipment (site transporter, cask tractor, cask transfer trailer, and site prime mover) to perform their intended safety functions in SAR Sections 1.9.1 and 1.9.1.8 and supporting documents.

<u>TEV ITS</u>

DOE identified five safety functions (SAR Tables 1.9-2 through 1.9-7) and one PSC (PSC–10) (SAR Table 1.9-10) that are needed to prevent an event sequence from occurring for the TEV. DOE indicated that these safety functions will be used to define specifications for the TEV design and validate TEV reliability to ensure that the TEV is functional and available through the preclosure period. For example, in SAR Section 1.3.3.5.2, DOE indicated that PSC–10 will serve as a reference for comparing the actual component failure rates and exposure times with the assumed values used in PCSA so deviations can be detected and analyzed for corrective actions.

DOE considered a spectrum of seismic events to address the TEV's ability to perform under seismic conditions. DOE conducted a study to assess risk and quantify the mean frequency of seismic-related event sequences (BSC, 2008bg). The study identified three potential TEV failure scenarios: (i) derailment, (ii) tipover, and (iii) ejection of waste package from the shielded enclosure. TEV fragility estimates were expressed as probabilities of unexpected performance as a function of a ground motion parameter. DOE provided an analysis (BSC, 2008co) to support the fragility calculations.

DOE stated that the TEV maintenance plan would be developed and implemented in accordance with SAR Section 5.6. DOE further indicated that the maintenance process will be centered on reliability and developed. Furthermore, DOE indicated that periodic tests will be performed at scheduled intervals to detect and replace parts subject to degradation before equipment deterioration reaches an unacceptable condition. DOE stated that it will design the TEV in accordance with the ASME NOG–1–2004 standard (American Society Mechanical Engineers, 2005aa). This nuclear-industry-accepted standard provides some guidelines for testing and maintenance of crane systems.

NRC Staff Evaluation: The NRC staff reviewed the TEV information using the guidance in the YMRP. To verify whether DOE used the PCSA results to identify critical components or subsystems within the TEV and measures incorporated in the TEV design for performing the safe functions, the NRC staff reviewed BSC (2008bz). On the basis of the NRC staff evaluation, DOE identified critical components required to meet design bases such as (i) the shielded enclosure for shielding personnel from waste package radiation in unrestricted areas, (ii) the braking system to control runaway conditions, (iii) the propulsion drive system to provide adequate drive power for emplacement and return to the surface, (iv) the ITS interlock switch for preventing inadvertent TEV shield door opening, (v) the restraint system for controlling TEV motion during seismic conditions, and (vi) the electrical power through a third rail to eliminate fuel use and potential for fire within the drift. Consequently, DOE reasonably used the PCSA results to identify these SSCs because DOE addressed each of the safety functions derived from the PCSA and included necessary redundancies in the design.

The NRC staff reviewed DOE's calculations, detailed in BSC Attachment H (2008co), for TEV to withstand seismic events. The NRC staff notes that the TEV cannot tip over at any credible ground motion level even after the TEV's seismic event restraints fail. The only credible condition where the waste package would be exposed to damage is during waste package transfer from the WPTT to the TEV at the docking station. Because this operation occurs during a very short period, any impact to the waste package due to TEV sliding, if it did occur, would be much less than the 6 m/s [20 ft/s] impact velocity the waste package would survive without breaching. As a result, DOE's calculations and results are reasonable because the calculations used reasonable methodologies and site-specific seismic data. In addition, the NRC staff notes that DOE provided reasonable justification that the TEV will perform under the designed seismic conditions.

The NRC staff also reviewed how DOE applied the component reliability assessment to show the TEV's overall ability to perform its intended safety functions (BSC, 2008bk) and notes that DOE reasonably represented the TEV design in the fault trees and included component reliability values on the basis of available component reliability databases. Therefore, the reliability estimates DOE documented are reasonable to show that the TEV's ability to perform the intended safety functions is achievable. The NRC staff notes that DOE's plan to perform a detailed reliability assessment of the TEV system (SAR Table 1.3.3-7) as the design evolves is reasonable because a more detailed analysis cannot be performed until DOE fully develops a more detailed TEV design. As described in BSC Section 3.1.1.1 (2008bz), DOE explicitly included in the detailed reliability analyses all identified ITS components such as the drive motors, drive shafts, wheels, gearboxes, door components (actuators, locks, and hinges), hardwired interlock circuitry, and seismic restraints to confirm the TEV reliability and its capability to perform the intended safety functions.

The NRC staff also reviewed SAR Sections 1.3.3.5.2 and 5.6 that described DOE's plans to develop the TEV inspection and maintenance plans. DOE considered maintenance in the TEV design, as detailed in BSC Section H6.2.2 (2008co). This consideration includes (i) placing the restraint system on the TEV chassis to facilitate maintenance and inspection at regular intervals and (ii) constructing 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. Finally, DOE indicated that if the TEV becomes contaminated, the maintenance and replacement activities on this system will consider ALARA principles. On the basis of the evaluation, DOE's implementation of testing and maintenance programs on the TEV provides additional confidence that the TEV will be able to perform its intended safety functions if an event sequence occurs.

<u>Surface Transportation Equipment ITS</u>

DOE identified 14 distinct safety functions (SAR Tables 1.9-2 through 1.9-7) and 2 PSCs (PSC–2 and PSC–10) (SAR Table 1.9-10) for the ITS surface transportation equipment, including the site transporter, cask tractor, cask transfer trailer, and site prime mover, that are needed to prevent occurrence of event sequences. DOE indicated that these safety functions will be used to define design specifications for the surface transportation equipment. DOE stated that it will require qualified vendors to identify and use applicable sections of codes and standards and define operational requirements and limits, as described in DOE Section 1.1 (2009ez,fg). DOE also stated that it will verify the final design and safety features of the site transportation systems through analysis to ensure their ability to perform the intended functions and to test these site transportation systems during startup operations to validate their functions, as outlined in DOE Section 1.2 (2009ez,fg).

DOE considered a spectrum of seismic events to address the ability of the site transporter to perform under seismic conditions. DOE conducted a study to quantify the mean frequency of seismic-related event sequences (BSC, 2008bg). The study identified two potential site-transporter failure scenarios: (i) tipovers (including those at locations of 5 percent grade in the direction of travel and 2 percent grade transversely) and (ii) sliding impacts. 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, DOE estimated the fragility related to tipover failure on the basis of conservative engineering judgments supported by general earthquake experience with railcars and truck trailers. DOE relied primarily on the transportation cask design to mitigate tipover events, and therefore, DOE concluded that tipover failure was not a major contributor to risk. Accordingly, DOE did not identify any safety function related to tipover for the cask tractor, cask transfer trailer, or site prime mover.

DOE identified PSCs to ensure the transportation equipment's ability to remain in a safe state under certain conditions. DOE determined that PSC–2 was necessary to limit spurious movement potentially causing a collision or tipover during the operation of the site transporter, cask tractor, cask transfer trailer, and site prime mover. DOE also identified PSC–10, which compares both residence times in a given process operation and the actual SSCs failure rates with the assumed values used in the PCSA. DOE will analyze any significant deviation to determine risk significance.

DOE described the plans for inspection, testing, and maintenance of the equipment to assess the availability of the surface transportation equipment to perform its intended safety functions. DOE briefly addressed the maintenance for the site transporter and site prime mover, but not for the cask tractor and cask transfer trailer. DOE indicated that the maintenance process will be centered on reliability and developed for the cask tractor and cask transfer trailer (DOE, 2009dk). In addition, DOE indicated that periodic tests will be performed at scheduled intervals to detect and replace parts subject to degradation before equipment deterioration reaches an unacceptable condition. DOE indicated 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, DOE discussed a comprehensive plan for environment, equipment, and seismic qualification programs that can validate the availability of the surface transportation equipment during the preclosure period. DOE stated that it will develop and conduct the programs following the guidelines in accepted industry standards such as IEEE 323–2003 and IEEE 344–2004 (Institute of Electrical and Electronics Engineers, 2004aa, 2005aa). DOE's plan included conditioning monitoring to determine whether the qualified equipment will remain in a qualified condition.

NRC Staff Evaluation: The NRC staff reviewed the ITS surface transportation equipment information using the guidance in the YMRP. After evaluating DOE's component reliability assessments to show the surface transportation equipment's overall availability to perform the intended safety functions (BSC, 2008au), the NRC staff notes that DOE reasonably represented 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 reliability estimates DOE documented provide reasonable technical bases to show that the intrasite transportation equipment's ability to perform the intended safety functions is achievable due to the large design margins.

The NRC staff reviewed DOE's calculations for the estimated fragility of tipover, sliding, and collision used to determine the ability of the site transporter to withstand the seismic events, as described in BSC Attachment G (2008co), and notes that reasonable technical basis was provided to support DOE's determination that the governing failure mode is the seismically induced sliding of the site transporter. For this scenario, the NRC staff notes that DOE provided supporting calculations showing an overall factor of safety of 4.56 indicating that seismically induced sliding of the site transporter is not plausible. Therefore, DOE's consideration for seismic conditions is reasonable. The NRC staff also notes that it is reasonable for DOE not to assign any safety function related to tipover for the cask tractor, cask transfer trailer, or site prime mover because DOE showed that seismically induced tipover, sliding, and collision of the cask tractor, cask transfer trailer, and site prime mover are not plausible.

The NRC staff reviewed the information related to PSC–2 and notes that PSC–2 is reasonable because it requires deactivating the surface transportation equipment with the brakes applied and detaching the site prime mover when performing waste handling. In addition, DOE augmented 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 notes that effective equipment testing and qualification programs can ensure that SSCs are functioning as specified in design documents. Therefore, the NRC staff notes that PSC–10 reasonably ensures the ability of the surface transportation equipment to perform its intended functions.

The NRC staff reviewed DOE's description of the plans for inspection, testing, and maintenance of the equipment to assess the availability of the surface transportation equipment to perform its intended safety functions and notes that DOE incorporated reasonable redundancy, alarms, safety measures, and qualification considerations to ensure proper detection, repair, and maintenance-related activities that will ensure the surface transportation equipment's ability to perform the intended safety functions.

2.1.1.6.3.2.8.4 Electrical Components and Emergency Power Systems Important to Safety

DOE provided information on electrical components and EPS ITS 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. DOE 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 necessary. In this section, the NRC staff focuses 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 UPS. The ITS distribution system distributes power to ITS loads within the GROA. The objective of the review is to determine whether the ITS EPS SSCs can perform their intended safety functions (i.e., power ITS HVAC) and the ITS diesel generators can provide reliable and timely emergency power, when required.

Means To Inspect, Test, and Maintain Electrical Power Systems ITS

DOE stated that it will use Regulatory Guide 1.9 (NRC, 2007ag) and IEEE 387–1995 (Institute of Electrical and Electronics Engineers, 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, DOE 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)] to the electrical distribution systems. Furthermore, DOE stated that it will use reliability-centered maintenance methodology to develop maintenance programs, including periodic inspecting and testing. This is important to assure that ITS electrical power distribution systems and ITS diesel generators are operated and maintained in accordance with the required performance and reliability pursuant to the PCSA (SAR Section 5.6.4).

NRC Staff Evaluation: The NRC staff reviewed the information on inspecting, testing, and maintaining the ITS EPS using the guidance in the YMRP. DOE's reliability-centered maintenance program provides a reasonable means to inspect, test, and maintain the ITS electrical power distribution systems and the ITS diesel generators. Furthermore, the NRC staff notes that it is reasonable for DOE to use the cited provisions of the codes and standards and regulatory guides for regular maintenance, inspections, and tests of the ITS diesel generators and periodic functional tests for the ITS electrical distribution systems because these provisions provide guidance to ensure that these systems function as designed.

Reliable and Timely Emergency Power

DOE provided a calculation that showed that the ITS HVAC HEPA filtration could be lost for up to 8 hours in the event of a loss of offsite power without resulting in unreasonable radionuclide release (DOE, 2009fp). Because of this reason, DOE 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. DOE stated that it will follow IEEE–387–1995 Section 4.1 (DOE, 2009fp) for determining 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. However, DOE stated that this time interval is expected to be less than 3 minutes (DOE, 2009fp).

DOE discussed the use of ITS UPS in SAR Section 1.4.1.3.1 and its response to the NRC staff's RAI (DOE, 2009gj). DOE stated that it did not identify any ITS SSCs whose safety functions would need a continuous power supply (DOE, 2009gj). In other words, DOE did not identify any safety functions for the ITS UPS. According to DOE, the purpose of using ITS UPS was to provide additional flexibility and capability, improve the voltage regulation of the ITS EPS SSCs, and provide a contingency power supply, if needed. DOE further stated that it will develop the ITS UPS maximum power requirements, allowing for future growth of the ITS UPS as part of the detailed design.

NRC Staff Evaluation: The NRC staff reviewed this information using the guidance in the YMRP. On the basis of a review of the design description and cited codes and standards provided in the SAR and supplemental materials, the NRC staff notes that the ITS diesel generator SSCs will be capable of starting and accepting intended loads before the 8-hour calculated allowable time window expires to provide reliable and timely emergency power.

System Redundancy

DOE indicated in SAR Section 1.9.1.12 that redundant, independent, and physically separated systems are available for ITS diesel generators. In addition, DOE indicated that each ITS diesel generator has a rated load-carrying capacity of 5 MVA and the estimated approximate demand is 3.9 MVA; hence, there is a 25 percent design margin (DOE, 2009dk). Redundant trains for ITS diesel generators; the 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 CRCFs, WHF, EDGF, and the non-ITS RF] were described in the SAR and supplemental information.

NRC Staff Evaluation: The NRC staff reviewed this information using the guidance in the YMRP. The NRC staff notes that DOE considered redundant systems, as necessary, to provide reasonable capacity and capability of utility systems ITS, as identified through its PCSA.

The NRC staff notes that the load-carrying capacity for ITS diesel generators is reasonable and the design provides spare capacity for future growth consistent with established engineering practice. In addition, the design configuration for the non-ITS RF HVAC is virtually identical to the ITS HVAC in the CRCFs and WHF. The EDGF includes identical ITS SSCs for distributing and controlling ITS power to the CRCFs, 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 is not automatically restored when normal or emergency power becomes available after a power outage. The NRC staff further notes that the design configuration for the redundant ITS EPS can be characterized as 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 detailed discussion, see TER Section 2.1.1.7.3.6). Therefore, DOE described redundant features for the major ITS EPS SSCs including the ITS diesel generators, ITS EPS main switchgear, and the ITS MCCs and ITS load centers located in each facility the ITS EPS serves.

Ability To Perform Intended Safety Function

DOE 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. DOE’s performance requirement for the SSCs that distribute electrical power to ITS surface nuclear confinement HVAC systems in the CRCF is 0.007 failures allowable during a 720-hour period following a radionuclide release event. Similarly, for the CRCF, 0.3 failures are allowable during a 720-hour period following a radionuclide release event for the ITS diesel generator SSCs to supply ITS electrical power.

NRC Staff Evaluation: The NRC staff reviewed this information using the guidance in the YMRP. The NRC staff notes that DOE’s inclusion of redundant systems for the ITS diesel generator increases the ability of the ITS diesel generators and electrical power distribution systems to provide electrical power to needed ITS SSCs to perform intended safety functions in the case of an event sequence and resulting loss of offsite power. The ability is further enhanced by DOE’s plans to follow the guidelines in Regulatory Guide 1.89 (NRC, 1984aa) and

IEEE 323–2003 (Institute of Electrical and Electronics Engineers, 2004aa) to seismically and environmentally qualify ITS active electrical equipment (SAR Section 1.13).

The NRC staff notes that inclusions of redundant systems and implementation of a seismic and environmental qualification program provide additional confidence of the ability of ITS electrical power distribution systems and ITS diesel generators to perform their safety functions.

2.1.1.6.3.2.8.5 Fire Protection Systems Important to Safety

DOE provided design descriptions and safety classifications for the DIPA sprinkler systems. The descriptions and safety classifications were provided in SAR Sections 1.6, 1.7, and 1.9.1.9 and in DOE's response to an NRC staff RAI (DOE, 2009fr).

DOE indicated that DIPA sprinkler systems are relied upon to protect moderator-controlled areas within the GROA (e.g., CRCF and 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 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 analysis and rationale DOE used to justify the safety classification of the detection and suppression systems. The NRC staff also evaluated the event sequence analyses to identify particular sequences and assess the role of the DIPA to prevent criticality events.

Selection of System

DOE classified the DIPA systems as ITS because they will be relied upon to protect areas of CRCF and WHF where moderator control is required. The DIPA systems were not selected on the basis of their ability to detect or suppress a fire, because these safety functions are not credited in the PCSA.

The PCSA showed that moderator could be introduced into a container following a canister breach and a subsequent spurious activation of the sprinkler system. DIPA systems 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. This feature makes DIPA systems more reliable against the accidental discharge of water during nonfire events.

NRC Staff Evaluation: The NRC staff reviewed the design using the guidance in the YMRP to evaluate the selection of the DIPA system as a suitable means to provide the designated ITS function. On the basis of this evaluation, the NRC staff notes that DIPA systems are commonly used in areas where spurious water delivery is undesirable. These systems are standard designs, using components that have been tested and listed for this intended function.

Ability To Perform Intended Safety Function

The design standards for the DIPA systems are found in National Fire Protection Association (NFPA) 13 (National Fire Protection Association, 2007ab) and NFPA 72 (National Fire Protection Association, 2007af). DOE applied the low probability of false water introduction from a DIPA system to achieve a low overall probability of moderator intrusion in criticality-related event sequences. As stated in SAR Table 1.4.3-2, DOE cited design failure probabilities of 1 × 10^{-6} over a 720-hour (30 days) period following radionuclide release in the CRCF, and 6 × 10^{-7} over a 720-hour period following radionuclide release in the WHF as its nuclear safety design bases. DOE selected a period of 720 hours following a canister breach when moderator introduction would be a potential hazard.

On the basis of the fault tree analysis provided in the response to an NRC staff RAI (DOE, 2009fr), DOE indicated that the mean probability of failure of spurious activation of the double-interlock sprinkler systems is 2 × 10^{-7} over a 720-hour period following radionuclide release. This probability is less than the design basis failure probability provided in SAR Table 1.4.3.2.

NRC Staff Evaluation: The NRC staff reviewed the design information using the guidance in the YMRP to assess the ability of the DIPA system to perform the intended safety functions. The NRC staff notes that the established design bases and the corresponding calculation of system reliability for the DIPA systems were developed through fault tree analyses. These analyses indicated that the proposed system will achieve the required reliability and satisfy the design bases. The 720-hour recovery period is reasonable, because it is assumed that appropriate actions can be taken to prevent moderator introduction within this time.

<u>Assessment of Continued Functionality and Means To Inspect, Test, and Maintain</u>

DOE indicated that the DIPA systems will be designed, installed, and maintained in accordance with National Fire Protection Association (NFPA) standards. The suppression system design guidelines outlined in NFPA 13 for installation of sprinkler systems (National Fire Protection Association, 2007ab), in conjunction with scheduled maintenance in accordance with NFPA 25 for the inspection, testing, and maintenance of water-based fire protection systems (National Fire Protection Association, 2008ac), will be used to ensure reliable suppression systems are provided. Fire detection systems that are used as components in the DIPA systems will be designed and inspected in accordance with NFPA 72 National Fire Alarm Code (National Fire Protection Association, 2007af). The use of appropriate national codes will provide an appropriate level of design and a degree of reliability that is consistent with other protected facilities.

NRC Staff Evaluation: The NRC staff reviewed the design information using the guidance in the YMRP to assess the functionality of the designed system and the ability of the designed system to be properly inspected, tested, and maintained. The NRC staff notes that DOE reasonably described standard DIPA systems that would be designed in accordance with nationally recognized codes. The NRC staff also notes that the installation, inspection, and maintenance procedures required in the referenced codes will assure continued functionality and reliable means to inspect, test, and maintain the systems.

2.1.1.6.3.2.8.6 Transportation, Aging, and Disposal Canisters

DOE provided a general description of its approach to show that TAD canisters can perform their intended safety functions assuming the occurrence of event sequences in SAR Section 1.9.1.8. In addition to SAR Section 1.9.1.8, the NRC staff also focused its review on DOE’s SAR Sections 1.5.1.1.1.2.1.3 and 1.7.2.3.1 and referenced reports (BSC, 2008ac,cp).

DOE classified the proposed TAD canister as ITS because the safety function for TAD canisters is to contain the SNF during the occurrence of an event sequence (SAR Tables 1.9-2 through 1.9-6). Note that currently only performance specifications are available for the TAD (DOE, 2008aa); therefore, in SAR Tables 1.9-2 through 1.9-6 a "representative canister" was used in place of the TAD.

SAR Section 1.7.2.3.1 described the methodology for determining passive component reliability and discussed loss of containment of a waste form container (e.g., TAD canister) due to structural challenges. The structural challenges consisted of vertical and off-axis drop, tipover, slapdown, and horizontal drops. Explicit finite element analyses were performed, using a model of a representative canister, to simulate the different structural challenges. As discussed in BSC (2008ac,cp), the representative canister utilized the average dimensions of several existing dual purpose canisters (DPCs), naval canisters, and the proposed TAD canisters. The material utilized for the representative canister was a stainless steel alloy consistent with that specified in the TAD canister performance specifications. These characteristics of the representative canister were used in making the numerical (finite element) models for evaluating the TAD canister's reliability (BSC, 2008cp). DOE 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.

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) for both structural and thermal challenges.

NRC Staff Evaluation: The NRC staff reviewed the TAD canister information using the guidance in the YMRP. In addition, the NRC staff utilized the review results from TER Section 2.1.1.4 to assist evaluation.

Because there is no design currently available for the TAD canister, the NRC staff was unable to evaluate the proposed design to determine whether the TAD canister was structurally capable of fulfilling its intended safety function. Therefore, the NRC staff's review is limited to the finite element analyses discussed in BSC (2008ac,cp) to evaluate the structural behavior of a representative SNF canister and its ability to provide containment when subject to different structural challenges. The NRC staff's evaluation of the finite element analyses discussed in BSC (2008ac,cp) is presented in TER Section 2.1.1.4.3.3.1.1, where the NRC staff notes that the finite element analysis results are reasonable because DOE used an industry-accepted code and reasonable engineering modeling techniques. The analysis results for the representative canister are applicable to the TAD canister because the TAD canister has dimension and weight requirements similar to those of the representative canister. Accordingly, the TAD canister has the ability to perform its intended safety functions.

2.1.1.6.3.2.8.7 Waste Packages

DOE provided information relative to waste package design and performance to show that the waste package has the capability to perform its intended safety functions, assuming the occurrence of event sequences. This information was presented in SAR Sections 1.5.2 and other applicable SAR sections (e.g., SAR Sections 1.2.1.4.1, 1.2.4.2.3.1.3, 1.3.1.2.5, and 2.3.6.7.4).

DOE proposed to use waste packages as an engineered barrier for disposal of commercial SNF, HLW, and DOE and naval SNF and classified the waste packages as ITS because they are relied upon to prevent radioactive gas or particulate release during normal operations and Category 1 and Category 2 event sequences.

DOE defined a list of safety functions (SAR Table 1.5.2-6) that waste packages are required to perform and evaluated the waste package performance using elastic-plastic finite elements analyses, conduction and radiation analyses, and analytical methods. For the structural analysis, DOE 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 provided in ASME 2001, Section III, Appendix F (American Society of Mechanical Engineers, 2001aa). For the thermal analysis, DOE determined time histories of the radial temperature distributions in the waste package and compared them to the temperature limits for accidental conditions. In addition, using energy absorption methodology in determining the outer corrosion barriers' capacity, DOE developed reliability estimates to calculate the probabilities of radionuclide release from waste packages assuming the occurrence of event sequences. Also, DOE did not identify any PSC needed for the waste package to prevent or mitigate event sequences.

NRC Staff Evaluation: The NRC staff reviewed waste-package-related information using the guidance in the YMRP. The NRC staff notes that the information presented in the SAR and other supporting documents reasonably described, assessed, and provided a basis for evaluating whether the waste package can perform its intended safety functions assuming the occurrence of event sequences.

According to the information in the representative finite element analysis for the 21–PWR/44–BWR TAD canister bearing, 5–DHLW/DOE short codisposal, and naval canistered SNF long waste package configurations DOE provided in response to an NRC staff RAI (DOE, 2009er), the NRC staff notes that (i) the calculated stresses in the waste package outer corrosion barrier satisfied the tiered screening criteria and (ii) the calculated temperature inside the waste package stayed below the temperature limit for accidental conditions (see TER Section 2.1.1.7.3.3.1 for detailed information on structural and thermal waste package analysis evaluation).

The NRC staff notes that DOE's structural and thermal analyses of the waste package performance provided reasonable technical basis to show that the postulated criteria for event sequences are met and the 21–PWR/44–BWR TAD canister bearing, 5–DHLW/DOE short codisposal, and naval canistered SNF long waste packages can perform their intended safety functions.

2.1.1.6.3.2.9 Radioactive Waste and Effluents Control

DOE provided information in SAR Sections 1.9.1.10 and 1.4.5.1.1 regarding radioactive waste and effluents control. DOE provided information to explain how PCSA addressed liquid and solid waste management systems to handle the expected volume of potentially radioactive liquid waste generated during normal operations and Category 1 and 2 event sequences and off-gas treatment, filtration, and ventilation systems for control of airborne radioactive effluents.

<u>Liquid Low-Level Waste Management</u>

DOE 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 LLW includes effluent from decontamination activities, actuation of a fire suppression system that generates water contaminated with radioactive material, and liquids deposited into drain or sump collection systems that gather water from any other activities that could generate LLW (SAR Section 1.4.5.1.1.2). DOE indicated 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. 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 LLW. In addition, DOE indicated that that all liquid waste facilities in which liquid LLW is detected will be decontaminated before normal activities are restored (SAR Section 1.4.5.1.1.2); DOE identified these liquid waste facilities as non-ITS.

DOE described the capacity of the effluent systems to contain the largest credible volume of fire-water discharge from fire suppression systems (DOE, 2009fo). A maximum of 34,069 L [9,000 gal] of effluent would be generated during a 30-minute fire that used 1,136 L/min [300 gpm] in fire suppression. The holding tanks for all five facilities are sufficiently large to accommodate design margins, freeboard capacity, a week's capacity for custodial maintenance and decontamination, sampling tank, and rounding errors. The collection tanks located outside the IHF, RF, and CRCF provide a cumulative working volume of 58,901 L [15,560 gal]. Similarly, the WHF has two collection tanks, each with a working volume of 57,917 L [15,300 gal]. The low-level radioactive waste facility (LLWF) has 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 working volume of each of these three tanks is 86,875 L [22,950 gal] after the addition of freeboard capacity, design margin, and rounding error.

NRC Staff Evaluation: The NRC staff reviewed DOE's descriptions of its liquid LLW management system using the guidance in the YMRP to determine whether DOE's PCSA included consideration of means to control radioactive waste and radioactive effluents and permit prompt termination of operations and evacuation of personnel during an emergency. The NRC staff also evaluated DOE's technical basis for managing liquid LLW discussed in SAR Section 1.4.5.1.1 to determine whether DOE presented a comprehensive plan for managing the liquid waste. Furthermore, the NRC staff reviewed whether analyses used to identify SSCs ITS, safety controls, and measures to ensure the availability and reliability of the safety systems reasonably considered liquid waste management systems that could handle the expected volume of potentially radioactive liquid waste generated during normal operations and Category 1 and 2 event sequences. Design features and procedures for these systems were also evaluated to determine whether they 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 LLW 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 LLW (SAR Section 1.4.5.1). The NRC staff also reviewed the fire suppression systems of the five facilities at the proposed repository (IHF, RF, CRCF, WHF, and LLWF) to determine whether sufficient capacity exists to contain and process the liquid waste generated by a 30-minute fire.

On the basis of these evaluations, the NRC staff notes that 1,136 L/min [300 gpm] for 30 minutes is sufficient to suppress a fire and that each facility has sufficient capacity to contain and process the liquid waste generated by a 30-minute fire in addition to normal decontamination and custodial maintenance activities. The NRC staff also notes that the collection, holding, and process tanks for liquid LLW management are non-ITS because these tanks handle liquid LLW only and damage to these tanks will not cause significant radiological releases.

Solid Low-Level Waste Management

DOE described the subsystem used to manage low-level radioactive solids and potentially radioactive solids in the waste handling facilities. Potential sources of solid (dry and wet) LLW are (i) water processing or decontamination activities that require some processing activity to meet waste disposal criteria at a disposal facility and (ii) the empty DPCs (SAR Section 1.4.5.1.1.3). Dry and wet solid LLW, except wet spent resins associated with the pool water treatment, are collected; transferred to the LLWF, which is non-ITS; and stored until processed to ensure that the final waste form meets DOE's criteria of the offsite disposal facility. Spent resin is dewatered at the WHF, then handled as dry solid LLW.

NRC Staff Evaluation: The NRC staff reviewed these descriptions using the guidance in the YMRP and Regulatory Guide 1.143 (NRC, 2001ab). The NRC staff evaluated DOE's technical basis for managing solid LLW and examined the capacity of the DOE-designed facilities and notes that the LLWF has reasonable 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 notes that (i) solid LLW generated during spent resin processing can be handled because the designed system has the capacity to manage reasonably expected solid LLW and (ii) DOE reasonably addressed any spillage or system leakage that may occur during this process.

In addition, DOE presented a comprehensive plan for managing the solid LLW and has waste management systems to handle the expected volume of potential solid LLW generated during normal operations consistent with Regulatory Guide 1.143 (NRC, 2001ab).

Finally, the NRC staff notes that DOE has the means to control the solid LLW generated and to permit prompt termination of operations and evacuation of personnel during an emergency because DOE provided a reasonable solid LLW management process.

Gaseous Low-Level Waste Management

DOE stated that surface facilities are designed to mitigate the potential release of radioactivity (if an event sequence includes a radionuclide release from casks or canisters containing HLW or SNF). HVAC systems pass exhaust from the confinement zones through HEPA filters before it is discharged to the atmosphere (SAR Section 1.4.5.1). TER Section 2.1.1.6.3.2.8.2.2 notes that the ITS HVAC system can perform intended safety functions. These confinement measures control airborne radioactive waste and effluents in the handling facilities. The radiation/radiological monitoring system (SAR Section 1.4.5), the digital control and management information system, and the communications system (SAR Section 1.4.2) facilitate a controlled termination of operations and evacuation of personnel, if required. DOE designed the normal repository operations to control gaseous low-level radioactive effluents to an appropriate level. As discussed previously, HEPA filters will remove radioactive particulates in gaseous effluent. Service gases, such as argon and helium, are discharged to the nuclear HVAC upstream from the filters. After the radioactive particulates are removed, the gaseous stream is discharged to the atmosphere through the HVAC exhaust.

NRC Staff Evaluation: The NRC staff reviewed these descriptions using the guidance in the YMRP. The NRC staff evaluated the information DOE provided in SAR Section 1.4.5.1.1.3, which described the potential sources of gaseous LLW and the proposed mechanisms and processes to capture these wastes. The waste streams expected to contribute to gaseous LLW at the proposed repository are from operations involving casks, TAD canisters, and DPCs. The NRC staff notes that DOE's use of HEPA and HVAC is appropriate to remove radioactive particulates from these waste streams because the off-gas treatment, filtration, and ventilation systems are specifically designed to handle the gaseous LLW anticipated at the proposed facility. In summary, DOE presented a reasonable plan for managing gaseous LLW and that, as a result of this plan, the off-gas treatment, filtration, and ventilation systems for control of airborne radioactive effluents are non-ITS.

2.1.1.6.3.2.10 Structures, Systems, and Components Important to Safety Inspection, Testing, and Maintenance

DOE provided information in SAR Section 1.9.1.13 on considerations of the means to inspect, test, and maintain SSCs ITS, if DOE relies on inspection, testing, and maintenance to ensure availability of the SSCs safety functions.

SAR Section 1.9.1.13 stated DOE will provide specifications that include the limiting conditions for operation of selected SSCs. According to DOE, the limiting conditions will include specific surveillance requirements, appropriate functional testing, and other inspections. DOE 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. Also, DOE 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's request to identify the inspection, testing, and maintenance needs for SSCs ITS, DOE (DOE, 2009dk) stated that the reliability-centered maintenance process will be used to develop plans and procedures for inspection, testing, and maintenance of SSCs ITS. According to DOE, the inspection, testing, and maintenance needs for each component will be based on manufacturer's recommendations, industry codes and standards, equipment qualification, and reliability requirements from the PCSA. DOE will use the reliability-centered maintenance process to ensure availability of safety functions of SSCs ITS or to detect degradation and adverse trends so that action can be taken prior to component failure.

NRC Staff Evaluation: The NRC staff reviewed the information using the guidance in the YMRP. The NRC staff notes that the maintenance programs DOE plans to develop would provide means to inspect, test, and maintain ITS SSCs to detect degradation and adverse trends so that actions can be taken prior to component failure. This is because DOE has stated it will develop the maintenance programs using a reliability-centered approach and DOE provided the reliability specifications as part of nuclear safety design bases. Therefore, the reliability-centered maintenance programs will provide a reasonable means to ensure availability of safety functions of SSCs ITS.

2.1.1.6.3.3 Administrative or Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

DOE described procedures that will be developed to prevent event sequences or mitigate their effects in SAR Section 1.9.3. DOE's description referred to the management controls and procedures that will be implemented to ensure that administrative controls and PSCs will function properly.

DOE indicated that the preclosure PSCs will be used to regulate human activities to ensure preclosure operations are maintained within the baseline conditions (limits). Preclosure PSCs were identified from the initiating event screening analyses, event sequence quantification analyses, consequence analyses, and criticality control analyses and are listed in SAR Table 1.9-10. According to DOE, preclosure PSCs will be implemented through individual procedures, normal operating procedures, administrative controls, or a radiation protection program.

NRC Staff Evaluation: The NRC staff reviewed the information using the guidance in the YMRP and Interim Staff Guidance–04 (NRC, 2007ad). The NRC staff verified that preclosure PSCs are derived from screening analyses of initiating events, event sequence quantification analyses, radiological consequence analyses, and criticality control measures. The preclosure PSCs will be relied on to (i) reduce the likelihood of an initiating event or of an event sequence or (ii) mitigate the consequences of an event sequence.

The NRC staff reviewed the PSCs that will be applied to ITS SSCs. The NRC staff reviewed the relevant facility reliability and event sequence categorization analysis reports and consequence analysis reports and verified that the proposed PSCs are reasonably achievable through routine management systems and procedures. In addition, the NRC staff verified that the PSCs are identified as ITS (i.e., are on the "Q" list). Regarding Interim Staff Guidance–04 (NRC, 2007ad), the NRC staff notes that DOE's approach of considering human factors evaluations to be elements of the management systems is reasonable because the management involvement is a key attribute to the effective implementation of PSCs.

2.1.1.6.4 NRC Staff Conclusions

The NRC staff notes that DOE's identification of structures, systems, and components (SSCs) important to safety (ITS), safety controls, and measures to ensure availability and reliability of the safety systems is consistent with the guidance in the YMRP. The NRC staff also notes that DOE reasonably identified SSCs ITS using the results of PCSA as discussed in this chapter.

DOE stated that it will develop a reliability-centered inspection, testing, and maintenance program for the ITS SSCs (TER Section 2.1.1.6.3.2.8.2.2). As part of the detailed design process, confirm that (i) identification of the ITS components and the associated nuclear safety design bases are consistent with the design (TER Section 2.1.1.6.3.1) and (ii) the safety functions identified in the PCSA for passive and active systems that are credited to screen out initiating events are consistent with the design (TER Section 2.1.1.6.3.1).

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NRC. 1978ab. 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. Washington, DC: NRC.

CHAPTER 7

2.1.1.7 Design of Structures, Systems, and Components Important to Safety and Safety Controls

2.1.1.7.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the proposed design of structures, systems, and components (SSCs) important to safety (ITS) and safety controls (SCs) in the geologic repository operations area (GROA). The objective of the NRC staff's review is to (i) determine whether the U.S. Department of Energy (DOE) reasonably defined the design bases, design criteria, and the relationship between the design criteria and the preclosure performance objectives and (ii) evaluate the capability of the proposed design of SSCs ITS and SCs to perform the design functions as intended for the full system lifetime. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Sections 1.2 through 1.5 (DOE, 2008ab), supplemental documents referenced in the SAR, and information DOE 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; 2010ak–an).

DOE provided design information for the ITS SSCs and SCs from its preclosure safety analysis (PCSA). These ITS SSCs are relied upon to prevent or mitigate an event sequence in the PCSA. These include surface structural and civil facilities where high-level radioactive waste (HLW) is handled; mechanical systems; heating, ventilation, and air conditioning (HVAC) systems; transportation systems used to move HLW; electrical power systems; instrumentation and control (I&C) systems; fire protection systems; canister systems; and criticality prevention and shielding systems. DOE did not identify any SSCs as ITS or SCs for the subsurface facilities in the GROA.

2.1.1.7.2 Evaluation Criteria

The regulatory requirements for the design of SSCs ITS and SCs, as they pertain to preclosure, are defined in 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f). Specifically, 10 CFR 63.21(c)(2) requires DOE to provide information for codes and standards that DOE proposes to apply to the GROA design and construction. 10 CFR 63.21(c)(3) requires DOE to describe and discuss the design, including (i) dimensions, materials properties, specifications, and analytical and design methods along with any applicable codes and standards; (ii) the design criteria and their relationship to the preclosure performance objectives; and (iii) the design bases and their relationship to the design criteria. 10 CFR 63.112(f) requires DOE to provide a description and discussion of the design, both surface and subsurface, of the GROA, including (i) the relationship between design criteria and the expected preclosure performance and (ii) the design bases and their relation to the design criteria.

The NRC staff reviewed DOE's design information using the guidance provided in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The relevant acceptance criteria follow:

- DOE adequately defined the relationship between the design criteria and the expected preclosure performance, the relationship between the design bases and the design criteria, and the design criteria and design bases for SSCs ITS.

- GROA design methodologies provided by DOE are adequate.

- Design codes and standards used by DOE for the design of surface facility SSCs ITS are identified and are appropriate for the DOE-selected design methodologies.

- The materials to be used by DOE for SSCs ITS related to surface facility design are consistent with the design methodologies.

- DOE design analyses related to surface facility design use appropriate load combinations for normal and Categories 1 and 2 event sequence conditions.

- DOE design analyses related to surface facility design are properly performed and documented.

- DOE adequately designed waste package engineered barrier system SSCs and their controls.

Nine other acceptance criteria in the YMRP are not listed here, because they are related to the non-ITS subsurface facility.

The NRC staff used additional guidance, such as NRC standard review plans and regulatory guides, when applicable. These additional guidance documents are discussed in the relevant sections that follow.

2.1.1.7.3 Technical Evaluation

The NRC staff focused its review on DOE's design of the SSCs ITS and SCs to determine whether the design information DOE provided in its SAR reasonably demonstrates that the SSCs ITS and SCs design, construction, and operation will meet the facility performance objectives. Specifically, the NRC staff's evaluation assesses whether the design information reasonably describes the relationship between the proposed design criteria and the GROA performance, and the relationship between the design bases and the design criteria. Furthermore, the NRC staff evaluated whether applicable codes and standards have been used for design and construction of the SSCs ITS and SCs in the GROA.

In addition, the evaluation assesses whether the design information presented by DOE reasonably provides information relative to the codes and standards for design and construction of the GROA and the design bases and their relationship to the proposed design criteria. The NRC staff's review also assesses whether the design methodologies, design analysis, and design are supported by reasonable technical bases and are consistent with established industry practices.

The NRC staff reviewed the design of the following SSCs ITS and SCs: (i) Structural and Civil Facilities; (ii) Mechanical Handling Transfer Systems; (iii) HVAC Systems; (iv) Other Mechanical Systems; (v) Transportation Systems; (vi) Electrical Power Systems; (vii) I&C Systems; (viii) Fire Protection Systems; (ix) Waste Package, Transportation, Aging and Disposal (TAD) Canister, and Other Canisters, Overpacks and Casks; and (x) Criticality Prevention and Shielding Systems.

2.1.1.7.3.1 Structural and Civil Facilities

DOE provided design information on the ITS surface waste-handling facilities, the aging facility, and flood control features. In general, the ITS buildings protect SSCs ITS located inside these buildings, aging pads provide a stable foundation for aging casks, and flood control structures protect surface facilities from flood hazards. DOE provided design information on the SSCs ITS in SAR Section 1.2.2. The NRC staff reviewed the structural and civil facilities design bases and design criteria to determine whether they are consistent with the safety functions identified in the PCSA and whether DOE used appropriate design methodologies to support its design analyses. The NRC staff's evaluation of the structural and civil facilities is described in the following subsections on the surface structural facilities, aging facility, and flood control features.

The NRC staff's review of the description of the surface facilities and the performance of the surface facilities to prevent or mitigate event sequences is provided in TER Sections 2.1.1.2 and 2.1.1.4, respectively.

2.1.1.7.3.1.1 Surface Structural Buildings

DOE presented the structural design information in SAR Section 1.2.2, which included design bases and design criteria, design methodologies, and design analyses. The ITS buildings [i.e., Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), and Receipt Facility (RF)] were described in SAR Sections 1.2.3 through 1.2.6.

<u>Design Bases and Design Criteria</u>

DOE summarized 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, that are based on Yucca Mountain site characterization information. The NRC staff evaluated this information in TER Chapter 2.1.1.1 and noted that DOE reasonably characterized natural and man-made hazards for the GROA design. These tables provided nuclear safety design bases for structural integrities of the buildings to protect ITS SSCs inside the buildings from external loads (e.g., seismic, wind loads) and against building collapse onto the waste containers. To meet these safety design bases, DOE specified in SAR Tables 1.2.2-1 through 1.2.2-3 that ITS surface buildings should be designed to withstand the external loads that are summarized next.

<u>Wind and Tornado</u>

The design wind load is a 3-second gust wind velocity of 145 km/hour [90 mph] (American Society of Civil Engineers, 2000ab) (SAR Section 1.2.2.1.6.1). The NRC staff evaluates wind design load in TER Section 2.1.1.3.3.1.2.3.1.

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 2068 Pa/s [0.30 psi/s] (NRC, 1976ab; Ramsdell and Andrews, 1986aa). DOE treated the effects of tornado missile impacts as live loads with impact. The NRC staff reviews tornado hazard in TER Section 2.1.1.3.3.1.2.3.2.

Explosion

Explosion hazard design values are based on a maximum no-damage overpressure of 6,895 Pa [1 psi] (SAR Table 1.2.2-1). The NRC staff evaluates the explosion hazard design values in TER Section 2.1.1.3.3.1.2.6.5.

Volcanic Ash

The roof live load caused by volcanic ash fall is 1,005 Pa [21 psf] (SAR Section 1.2.2.1.6.5). The NRC staff reviews ash load in TER Section 2.1.1.1.3.6.

Snow and Ice

The maximum design daily snowfall is 152 mm [6 in], and the maximum monthly snowfall is 168 mm [6.6 in] (SAR Section 1.2.2.1.6.4). The NRC staff reviews snow load in TER Section 2.1.1.1.3.3.

Seismic

The ITS surface buildings were designed for a site-specific seismic ground motion level [Design Basis Ground Motion 2 (DBGM–2)], associated with a mean annual probability of exceedance of 5×10^{-4} (SAR Table 1.2.2-3). The horizontal and vertical peak ground accelerations (PGAs) for DBGM–2 seismic events are 0.45 g and 0.32 g, respectively, where “g” is the acceleration due to gravity (SAR Table 1.2.2-3). Seismic ground motion response spectra for different seismic levels are shown in SAR Figures 1.2.2-8 to 1.2.2-13. The review of the seismic design spectra is provided in TER Section 2.1.1.1.3.5.2.

NRC Staff Evaluation: The NRC staff reviewed DOE’s design bases and design criteria for the ITS surface buildings using the guidance in the YMRP. On the basis of the evaluation of the design information presented in this chapter and the NRC staff’s evaluations of site-specific information, operations, and the PCSA performed in other chapters, the NRC staff notes that the design criteria used for the design of SSCs ITS are consistent with the site-specific information. The NRC staff also notes that the loads and loading parameters are consistent with the standard industry practice for the design of similar risk NRC-licensed nuclear facilities. DOE’s design criteria and design bases correspond to the safety functions identified in DOE’s PCSA. Therefore, the relationship between the design criteria and the preclosure performance was reasonably defined.

Design Methodologies

DOE determined that seismic loading bounds the design of the ITS surface buildings (DOE, 2009es). DOE’s methodology for seismic design of ITS buildings was based on elastic analyses developed in SAP2000 (Computers and Structures, 2005aa). DOE used the Tier #1 analyses, as outlined in BSC Section 7.1.3 (2007ba), for the 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 are based on lumped-mass, multistick models, in which the building walls are modeled as beam-column elements using cross section properties. The ends of the beams are constrained to a master node at each floor diaphragm level and, thus, the floors are considered to be rigid in all three directions, as described in BSC Section 7.2.1.1 (2007ba). In the Tier #1 analyses, soil springs were implemented to account for soil–structure interaction (SSI) effects, which impact design forces.

For the IHF design (SAR Section 1.2.2.1.6.3.2.4), DOE created finite element analysis models in SAP2000, in which the concrete components are modeled by shell elements and the components of the steel frame structure are modeled as beam-column elements. The IHF finite element model does not account for SSI, because it assumes a fixed base IHF structure.

NRC Staff Evaluation: The NRC staff reviewed DOE's modeling methodology discussed in the SAR and the response to the NRC staff RAIs using the guidance in the YMRP. The NRC staff notes that the numerical models and the computer codes used to support the design methodologies are reasonable because the models were developed based on standard procedures and DOE used standard computer codes. DOE's Tier #1 analyses are appropriate for the design of the ITS surface buildings subjected to DBGM–2 because they are industry-accepted standard methods. The implementation of the Tier #1 models to specific ITS surface buildings is evaluated later in this TER section. The NRC staff also notes that the finite element analysis used to model the IHF is a standard method used by the industry. Therefore, DOE's seismic modeling methodology is reasonable for the surface ITS building design.

Design and Analysis of Surface ITS Buildings

The first part of this section presents the NRC staff's evaluation of the general analysis and design procedures discussed in SAR Section 1.2.2.1. The second part presents the NRC staff's evaluation of specific aspects of analysis and design that are unique to each facility.

General Analysis and Design Procedures

Design Codes and Standards

DOE listed the codes and standards used for the structural design of surface buildings in SAR Section 1.2.2.1.8. DOE assigned live loads to the floors according to American Society of Civil Engineers ASCE 7–98 (American Society of Civil Engineers, 2000ab). Seismic analysis models of the buildings were selected in accordance with ASCE 4–98 (American Society of Civil Engineers, 2000aa). DOE used ASCE/SEI 43-05 (American Society of Civil Engineers, 2005aa) to compute the structural damping, as an alternative to Regulatory Guide 1.61 (NRC, 2007af). Reinforced concrete ITS SSCs were designed using the strength design method, as detailed in BSC Section 8.2 (2007ba) and specified in ACI 349–01 (American Concrete Institute, 2001aa). For steel design, DOE followed the allowable stress design method, as outlined in BSC Section 8.2 (2007ba) and specified in ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa).

NRC Staff Evaluation: The NRC staff reviewed DOE's proposed use of design codes and standards for surface facilities using the guidance in the YMRP. The NRC staff notes that these codes and standards are reasonable because they are the industry-accepted codes and standards and are applicable for the nuclear surface building design.

Consistency of Materials With Design Methodologies

DOE described the materials used for the surface facility design in SAR Section 1.2.2.1.7. DOE's calculations for the designs of shear walls, diaphragms, and foundations used a concrete compressive strength (f'_c) of 3.45×10^7 Pa [5,000 psi] and steel yield strength (f_y) of 4.14×10^8 Pa [60,000 psi].

NRC Staff Evaluation: The NRC staff reviewed the materials DOE proposed for the design of the reinforced concrete and steel structures and foundations, using guidance in the YMRP. The NRC staff notes that the proposed steel and concrete materials are reasonable because they conform to American Society for Testing and Materials (ASTM) specifications and are consistent with the design codes and standards used for nuclear facility design.

Load Combinations

DOE listed the load combinations to be used in ITS surface building design in SAR Section 1.2.2.1.9.2 and BSC Sections 4.2.11.4.5 and 4.2.11.4.6 (2007av). In response to an NRC staff RAI (DOE, 2009es), DOE stated that some of these load combinations are not applicable to the design of specific structures (e.g., fluid loads in the CRCF) and some individual loads are bounded by other load sources (e.g., wind, ash loads). Therefore, for most of the structural analyses of the ITS buildings, DOE used only one load combination that includes gravity (dead load and 25 percent of live load) and seismic loadings. For the seismic load combination, DOE considered several subcombinations using the 100-40-40 component factor method, which assumes that when the maximum acceleration in one direction occurs, the accelerations from the other two orthogonal directions are 40 percent of the maximum acceleration, as detailed in BSC Appendix A (2007ba); BSC Section 6.1 (2007af); and BSC Section 6.3 (2007ae).

In response to an NRC staff RAI (DOE, 2009es), DOE provided its rationale for using 25 percent of the live load for seismic load combinations, instead of the factors proposed in the load combinations presented in SAR Section 1.2.2.1.9.2. DOE stated that, on the basis of its analysis, the consideration of the full live load factor does not impact the overall design because the magnitude of the live load is relatively small compared to the other load sources.

NRC Staff Evaluation: The NRC staff reviewed the load combinations DOE proposed for the design of ITS surface buildings using the guidance in the YMRP. The proposed seismic load combinations are reasonable because DOE showed that seismic loads bound other loading sources that could be applied to the surface buildings. The use of a load combination that includes seismic loading, dead load, and 25 percent of the live load is appropriate because consideration of the full live load factor has a minor effect on the overall structural response of the surface facilities. Therefore, the seismic load combinations used for ITS surface building analyses are reasonable.

Seismic Analysis Methodology

For the seismic analyses of ITS surface buildings subjected to DBGM–2 events, as described in BSC Section 5.1 (2007ba), DOE used a modal analysis approach based on site response spectra (SAR Figures 1.2.2-8 to 1.2.2-13). To calculate total seismic loads, DOE used the NRC 10 percent method for modal combination and the square-root-of-sum-of-squares method for directional combination (NRC, 1987aa). In response to an NRC staff RAI (DOE, 2009es), DOE described the generation of site-specific ground motion and design spectra in SAR Section 1.1.5. DOE incorporated the uncertainty in the generation of the ground motions and generated envelope hazard curves.

The percentage of structural damping used in the numerical models (SAR Table 1.2.2-4) was based on ASCE/SEI 43–05 (American Society of Civil Engineers, 2005aa) recommendations. For the DBGM–2 seismic events, the structural damping used for the analysis and design of steel and concrete structures was 7 percent, as detailed in BSC Section 7.2.4.2 (2007ba).

For the soil, DOE used structural damping of 20 percent (see foundation analysis and design section that follows). To account for different percentages of critical damping values in the structural analysis, DOE generated “hybrid” spectra that combine the 20 and 7 percent damped spectra, as outlined in BSC Section 6.1 (2007af). To demonstrate the applicability of hybrid response spectra (DOE, 2009et), DOE compared the CRCF structural member forces obtained from a modal analysis based on a hybrid spectrum and from a time history analysis, where 4 percent structural damping and 20 percent soil damping were used. DOE determined that the forces obtained from time history analyses are about 11 to 13 percent higher on average than those calculated in the response spectrum analysis, even though the structural damping is lower (i.e., 4 percent).

Concrete cracking was not considered in the structural analyses. DOE determined that its primary interest is in the in-plane response and that concrete cracking does not significantly affect the in-plane response, as described in BSC Section 7.1.1 (2007ba). In response to an NRC staff RAI (DOE, 2009eu), DOE indicated that the shear wall stiffness corresponding to the uncracked section is generally satisfactory for determining wall design forces, which is consistent with ASCE 4–98, detailed in American Society of Civil Engineers Section C3.1.3.1 (2000aa).

NRC Staff Evaluation: The NRC staff reviewed the seismic analysis methodology, including the use of hybrid response spectra, damping values, and concrete properties for the shear walls, using the guidance in the YMRP. DOE’s seismic analysis methodology is appropriate for the evaluation of ITS surface buildings subjected to DBGM–2 seismic events because the methodology is an industry-accepted method.

On the basis of DOE’s responses to an NRC staff RAI (DOE, 2009et), the NRC staff notes DOE showed that the first three modes of structural vibration are controlled by soil deformations and the remaining modes by the structure. Therefore, the applicability of hybrid response for Tier #1 seismic analysis is reasonable.

The percentage of structural damping DOE proposed is reasonable for the analysis and design of ITS buildings because it is consistent with ASCE/SEI 43–05 (American Society of Civil Engineers, 2005aa). In addition, the NRC staff notes that excluding concrete cracking in the seismic analyses does not significantly affect the structural response, because the inclusion of concrete cracking in the model analysis would lead to a more flexible model that would result in lower spectral accelerations. Therefore, the seismic analysis methodology is reasonable for the surface ITS building design.

<u>Structural Design Methodologies</u>

The following subsections present DOE’s approach to evaluate the design of the structural components of the ITS buildings. DOE’s structural design of the surface buildings was based on a demand-to-capacity (D/C) ratio equal to or less than unity. Note that DOE tried to maintain the D/C ratio below 0.5–0.6 to meet the preclosure performance objectives for seismically initiated event sequences, as described in BSC Section 8.4 (2007ba).

<u>Shear Wall Design</u>

DOE’s methodology for shear wall design, as outlined in BSC Appendix D (2007ba), is based on ACI 349–01 (American Concrete Institute, 2001aa). DOE determined that the predominant load path of seismic load is through the diaphragms and shear walls to the base of the concrete slab.

DOE designed the shear walls for the combined effects of in-plane shear loads, axial loads, in-plane bending moments, out-of-plane bending moments, and transverse shear loads. Shear friction (i.e., the capacity of the wall to transfer horizontal loads into the base slab) was also considered in the design (e.g., BSC, 2007ba,cv).

DOE demonstrated, on the basis of its design calculations, that reinforced concrete shear walls (BSC, 2007ba,cy) have sufficient capacity to withstand the design loads. DOE included a torsional factor in the design forces [BSC Section 6.3 (2006ak); BSC Section 6.2 (2007cv)], as recommended by ASCE 4–98 (American Society of Civil Engineers, 2000aa). The torsional factor accounts for load eccentricity and results in an increase in the load forces used for shear wall design.

NRC Staff Evaluation: The NRC staff reviewed the shear wall design methodology of surface facilities using the guidance in the YMRP. The overall shear wall design methodology is reasonable because it is consistent with standard engineering practice. The NRC staff also reviewed the D/C ratios of the concrete shear walls and notes that these structures are reasonable to withstand DBGM–2 seismic events because DOE stated that it will maintain the D/C ratios below 0.5–0.6 and thus provide sufficient design margin.

However, the NRC staff notes that DOE's design of shear walls was not based on the structural analysis reports presented in the SAR. For instance, for the CRCF building, the response spectrum modal analysis presented in the SAR included soil springs representing 30.5 and 61-m [100 and 200-ft] depths of alluvium (BSC, 2007af). However, the shear wall design (BSC, 2007ba) was based on an earlier seismic analysis (BSC, 2006ak), in which soil springs represent alluvium depths of 10.7 and 33.5 m [35 and 110 ft]. Similar inconsistencies were detected in the RF (BSC, 2007az,cy,da) and WHF (BSC, 2007bm,cv,cx), and DOE did not provide any technical justification. In response to an NRC staff RAI (DOE, 2009eu), DOE stated that a comparison of the seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that an 11-m [35-ft]-thick alluvium controls the structural design because it provides the highest story shears and diaphragm accelerations for the CRCF and WHF. Therefore, DOE's use of the 11-m [35-ft]-thick alluvium depth is appropriate for the Tier #1 preliminary design analysis because this alluvium depth controls the structural design. In addition, DOE stated that it will perform Tier #2 analysis, which will include realistic soil columns, in support of the detailed design for construction, as described in BSC ACN02 (2007ba).

Slab Design

DOE presented the methodology for reinforced concrete slab design in BSC Appendix D (2007ba), which is in accordance with ACI 349–01 (American Concrete Institute, 2001aa). Most concrete slabs of the ITS buildings have a thickness from 0.46 to 0.61 m [1.5 to 2.0 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] steel corrugated deck only supports construction loads. Other than spanning between beams, no credit was taken for self-support of the concrete slabs, as described in BSC Section 6.6 (2007cz). For the Tier #1 analyses, the beam spans of these slabs were assumed to have the maximum span of 2.1 m [7 ft] as outlined in BSC Assumption 3.1.5 (2007ct). Shielded rooms were designed with thicker concrete slabs in which the steel support was 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 reinforcement requirements of the concrete slabs were computed 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 Tier #1 seismic analyses were amplified by a factor of two, as outlined in BSC Assumption 3.1.6 (2007ct). This amplification was used to account for the effects of vertical floor flexibility, given that the slabs were considered rigid diaphragms in Tier# 1 analyses. DOE determined that this factor is reasonable on the basis of a study performed for the Canister Handling Facility (BSC, 2005ao).

The reinforced concrete slabs were designed as one-way slabs (e.g., BSC, 2008cj, 2007ct). The reinforcing steel computed in the slab span direction was also provided in the orthogonal direction. For the in-plane loads, DOE analyzed multiple-span diaphragms as simple spans on the basis of the structural analysis of the largest span [e.g., BSC Assumption 3.2.2 (2007cz)]. DOE also stated that composite action was not considered between the concrete slabs and the supporting structural steel beams, as detailed in BSC Section 3.2 (2007cz).

The steel beams supporting the concrete slabs consisted of W-shaped members. The design considered that top flanges of the beams were laterally supported by the steel deck during construction and by the concrete slabs during service. The deflection limits of the structural steel members were in accordance with ANSI/AISC N690 (American Institute of Steel Construction, 1994aa) and International Code Council (2003aa) (see BSC, 2007cz). For design purposes, DOE assumed that the structural steel components provided support for the concrete slabs and superimposed loads for all applicable service and extreme load combinations. The design provided D/C ratios for bending moment, which was the controlling failure mechanism for this type of structural component.

NRC Staff Evaluation: The NRC staff reviewed the slab design methodology for ITS buildings using guidance in the YMRP. DOE's slab design methodology is reasonable for ITS buildings subjected to the DBGM–2 seismic events, because it is based on accepted codes and standards and is consistent with engineering practice. The NRC staff also reviewed the D/C ratios of the concrete shear walls and notes that these structures are reasonable to withstand DBGM–2 seismic events because DOE stated that it will maintain the D/C ratios below 0.5–0.6 and thus provide sufficient design margin.

The NRC staff notes that some of DOE's design assumptions underestimated the capacity of the slabs, such as not considering the effect of multiple spans. The NRC staff also notes that DOE utilized an amplification factor of two for the out-of-plane seismic force calculations of the Tier #1 analyses, which is reasonable for a continuous slab. In addition, the NRC staff notes that the designs of slabs of surface facilities were not based on the seismic analysis reports presented in the SAR. For instance, for the design of the CRCF diaphragms (BSC, 2007ct), DOE used vertical accelerations obtained from structural analyses in which the soil impedance functions were based on alluvium depths of 10.7 and 33.5 m [35 and 110 ft] (BSC, 2007ct), instead of the seismic analyses presented in the SAR that considered alluvium depths of 30.5 and 61 m [100 and 200 ft] (BSC, 2007af). Similar inconsistencies were detected in the RF (BSC, 2007az,da; 2008cj) and WHF buildings (BSC, 2007bm,cy). In response to an NRC staff RAI (DOE, 2009eu), DOE stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that an 11-m [35-ft]-thick alluvium depth controls the structural design because it provides the highest story shears and diaphragm accelerations for the CRCF and WHF. Therefore, DOE's use of the 11-m [35 ft]-thick alluvium depth is appropriate for the Tier #1 preliminary design analysis because this alluvium depth controls the structural design. In addition, DOE committed to perform

Tier #2 analysis, which will include realistic soil columns, in support of the detailed design for construction, as described in BSC ACN02 (2007ba).

Foundation Analysis and Design

The foundation design of the surface buildings was based on finite element method analyses of the mat foundation, as described in BSC Section 6.1 (2007ae), using SAP2000 (Computers and Structures, 2005aa). The numerical model of the ITS building foundations was coupled with the superstructure. However, the seismic analysis reports included in the SAR were not used for foundation design. In response to an NRC staff RAI (DOE, 2009eu), DOE stated that the forces and moments in the foundation obtained from a model that used the initial soil profiles bounded the forces and moments that would be obtained when using updated soil profiles. DOE did not provide additional calculations in its response.

The finite element analyses used a mesh size of 1.5 × 1.5 m [5 × 5 ft]. In response to an NRC staff RAI (DOE, 2009ev), DOE stated that the approximated mesh size of 1.5 × 1.5 m [5 × 5 ft] was not expected to significantly affect the calculated design forces and localized stress concentrations. DOE indicated that sensitivity analyses of the mesh refinement would be performed to support the detailed design.

To account for soil–structure interaction (SSI), DOE calculated soil spring constants on the bases of the elastic and shear modulus of alluvium and tuff. DOE computed critical soil damping as larger than 100 percent for surface buildings, as outlined in BSC Sections 6.1.3.3 and 7.1.5 (2008af), even after reducing the soil damping by 25 percent to account for the reduction in energy dissipation in nonhomogenous media (Hadjian and Ellison, 1985aa). DOE reduced the soil damping values to 20 percent, as detailed in BSC Section 7.1.5 (2008af), on the basis of its interpretation of ASCE 4–98 Section 3.1.5.4 (American Society of Civil Engineers, 2000aa). In response to an NRC staff RAI (DOE, 2009ev), DOE indicated that the damping ratios for the CRCF were determined using the SASSI 2000 software program and compared to the results from impedance functions (i.e., soil spring constants). DOE stated that the damping ratios obtained from SASSI were greater than 20 percent, but supporting calculations were not provided.

For the foundation seismic analyses, DOE modeled mat foundations using shell elements. Then the global soil springs obtained from the impedance functions were used to generate translational nonlinear (compression only) springs per unit area for each joint of the basemat foundation. Therefore, the analyses based on a foundation mat did not include the effects of global rotational springs. In response to an NRC staff RAI (DOE, 2009eu), DOE stated that the effect of not including rocking global soil springs is not significant, because of low contribution from rotational modes of vibration.

For the foundation seismic design, DOE estimated an allowable soil-bearing pressure of 2,394 kPa [50 ksf], as detailed in BSC Section 6.2.3 (2007ba). The stability against overturning was verified using a static evaluation approach of comparing forces and moments versus resistance. Stability against overturning was evaluated using the methodology provided in ASCE/SEI 43–05 Section 7.2 (American Society of Civil Engineers, 2005aa).

NRC Staff Evaluation: The NRC staff reviewed the foundation design methodology DOE proposed using the guidance in the YMRP. Specifically, the NRC staff reviewed DOE's use of the finite element analysis and notes this is reasonable because it is based on an appropriate industry standard method. DOE's foundation design methodology is consistent with codes and standards and is consistent with the standard engineering practice. On the basis of the characteristics of the soil–structure systems (e.g., effective height, equivalent foundation radius, shear wave velocity, and lengthening of the fundamental period), the soil-damping value of 20 percent is reasonable for the foundation design. Also, DOE's design should be consistent with the structural analysis reports presented in the SAR. For instance, for the CRCF foundation, DOE's models for seismic analyses are based on soil property profiles for alluvium depths of 30.5 and 61 m [100 and 200 ft], as described in BSC Section 4.3 (2008af), but the forces for foundation design (BSC, 2007ae) were derived from seismic analyses based on soil profiles of 10.7 and 33.5 m [35 and 110 ft] of alluvium depth (BSC, 2006ak). Similar inconsistencies were found for the RF (BSC, 2008bf, 2007ax) and the WHF (BSC, 2007bl,cx; 2008br). In response to an NRC staff RAI (DOE, 2009eu), DOE stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that an 11-m [35-ft]-thick alluvium depth controls the foundation design because it provides the highest story shears and diaphragm accelerations for the CRCF and WHF. On this basis, the NRC staff notes that use of the 11-m [35-ft]-thick alluvium depth is appropriate for the Tier #1 preliminary design analysis. In addition, DOE stated it would perform Tier #2 analysis, which will include realistic soil columns, in support of the detailed design for construction, as described in BSC ACN02 (2007ba). DOE did not provide foundation settlement calculations in the SAR. However, results from the Tier #2 analysis will provide a settlement of the foundation directly.

The NRC staff notes in TER Section 2.1.1.1.3.5.4 that the allowable bearing pressure under extreme loading (seismic) conditions of 2,395 kPa [50 ksf] is reasonable because this bearing pressure falls within an appropriate range of maximum allowable bearing pressure.

In summary, the NRC staff notes that the DOE evaluation of foundation stability and design is reasonable.

Facility-Specific Analysis and Design Procedures

Analysis and design information specific to each of the ITS buildings follows. The CRCF and RF are presented in the same section because their structural characteristics are similar and are analyzed using the same numerical methods. The WHF and the IHF are reviewed separately.

CRCF and RF

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 wide by 128 m long by 31 m high [392 ft wide by 420 ft long by 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.

The RF building footprint dimensions are approximately 96 m wide by 97 m long [315 ft wide by 318 ft long]. The part of the RF building that is considered to be ITS has dimensions of 61 m wide by 73 m long by 31 m high [200 ft wide by 240 ft long by 100 ft high] above grade. The superstructure is constructed of 1.2-m [4-ft]-thick exterior and interior concrete walls. The internal shielded rooms also have 1.2-m [4-ft]-thick concrete walls and roof slabs. The RF foundation mat is 2.1 m [7 ft] thick, and elevated floor diaphragm slabs are generally 0.46 m [1.5 ft] thick.

CRCF and RF Structural Analyses

The CRCF and RF buildings were analyzed using the general seismic analysis methodology evaluated previously in this TER section. The seismic analyses for the CRCF and RF were 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 Section 6.1 (2007bx) and BSC Section 6.1 (2007az).

For the CRCF building, DOE performed response spectrum modal analysis for six soil conditions corresponding to soil springs representing 31 and 61-m [100 and 200-ft] depths of alluvium for lower bound, median, and upper bound stiffness conditions (BSC, 2008af). DOE presented the story shears of the CRCF for the DBGM–2 seismic events in BSC Table 18 (2007af), showing that shear forces were controlled by the 31-m [100-ft] upper bound soil case. The fundamental period for this case is T_1 = 0.15 s. DOE also presented the interstory drifts for the upper bound soil condition in BSC Table 16 (2007bx), where the maximum drift is 1.22×10^{-4}. DOE compared this value to the allowable limit of 4×10^{-3}, as detailed in BSC Section 4.2.11.4.10 (2007av), indicating that the interstory drifts are significantly lower than the drift limit.

For the RF building, DOE performed a response spectrum modal analysis for soil conditions representing a 40-m [130-ft] alluvium depth for lower bound, median, and upper bound stiffness values (BSC, 2008bf). DOE indicated in BSC Section 7.1 (2007az) that the upper bound soil case provides the highest reactions and accelerations. The fundamental period of vibration for upper bound soil condition is T_1 = 0.15 s, as outlined in BSC Table C3 (2007az), whereas for the lower bound soil condition T_1 = 0.26 s. The largest interstory drift is 0.0127 percent, as shown in BSC Table C6 (2007az). This value is lower than the interstory drift threshold of 0.4 percent, which is recommended (American Society of Civil Engineers, 2005aa) for systems designed to experience limited permanent distortion (e.g., Limit State C).

NRC Staff Evaluation: The NRC staff reviewed the CRCF and RF seismic analyses using guidance in the YMRP. The NRC staff notes that the CRCF and RF seismic analyses and results are reasonable for the Tier #1 model analyses subjected to the DBGM–2 seismic events because the seismic analyses are based on an industry-accepted method applicable for nuclear surface building design.

CRCF and RF Shear Wall Design

The designs of CRCF and RF shear walls were based on the general design methodologies evaluated previously in this TER section. For the CRCF, the maximum D/C ratios of the shear walls DOE computed were 0.74 and 0.83 for in-plane shear and bending, respectively (BSC, 2007ba). For the RF shear wall design, the maximum D/C ratio for in-plane shear was 0.51, whereas the maximum D/C ratio for bending and axial loads was 0.76, as described in BSC Section 7 (2007cy).

NRC Staff Evaluation: The NRC staff reviewed the CRCF and RF design of the shear walls using guidance in the YMRP. The NRC staff notes that, for the given end forces and moments, the design of shear walls DOE provided is reasonable for the Tier #1 analyses under the DBGM–2 seismic events and consistent with current engineering practice.

CRCF and RF Slab Design

The designs of CRCF and RF slabs were based on the general design methodologies evaluated previously in this TER section. For the CRCF 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 Assumption 3.2.1 (2007ct). The design calculations were based on the ACI 349–01 (American Concrete Institute, 2001aa) recommendations. For the design of the steel beams that provide structural support to the slabs (BSC, 2007cu), DOE used ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa) and obtained maximum D/C ratios in flexure of 0.85.

DOE presented the RF slab design results in BSC Section 7 (2008cj) and concluded that a reasonable slab design is achieved for the imposed design loads. DOE presented the design of the RF structural steel framing that supports the reinforced concrete slabs (BSC, 2007cz). The results indicate that the maximum D/C ratio was 0.81 for the steel framing sections.

NRC Staff Evaluation: The NRC staff reviewed the CRCF and RF design of the slabs using guidance in the YMRP. The NRC staff notes that, for the given end forces and moments, the slab design DOE provided is reasonable for the Tier #1 analyses under DBGM–2 seismic events because it is consistent with current engineering practice.

CRCF and RF Foundation Analysis and Design

The CRCF and RF foundation design followed the general design methodologies evaluated previously in this TER section. For the seismic analyses of the CRCF and RF foundations, finite element models of the basemat were developed and coupled with the superstructure, as outlined in BSC Section 6.1 (2007ae) and BSC Section 4.3 (2007ax).

For the CRCF foundation, DOE created six models based on soil property profiles for alluvium depths of 30.5 and 61 m [100 and 200 ft] for median, upper bound, and lower bound soil stiffness conditions, as described in BSC Section 4.3 (2008af). However, the forces used for foundation design (BSC, 2007ae) were obtained from the results of the seismic analyses based on soil profiles of 10.7 and 33.5 m [35 and 110 ft] of alluvium depth (BSC, 2006ak). In response to an NRC staff RAI (DOE, 2009eu), DOE indicated that the analysis results for a model based on 10.7 m [35 ft] of alluvium depth bound the results of models based on updated soil profiles.

For the RF foundation, DOE estimated that the alluvium thickness varies from about 38 to 44 m [125 to 145 ft] (SAR Figure 1.1-130). DOE generated shear wave velocity data for a 40-m [130-ft]-thick alluvium underlain by tuff, based on soil data for 30.5 and 61-m [100 and 200-ft] alluvium depths (BSC, 2008bf). However, the foundation design was based on structural analyses in which the soil springs correspond to a 10.7-m [35-ft] alluvium depth, as detailed in BSC Section 6.1 (2007ax).

For the RF foundation flexural reinforcement, a standard rebar pattern was selected and contour plots were used to identify areas that require additional reinforcement. The shear capacity of the concrete (without any shear reinforcing) was computed, and shear contour plots were used 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 0.69 and 0.67, respectively, as outlined in BSC Section 6.5 (2007ae). The SAR did not include the design for the 4-m [13-ft]-thick mat foundation. For the RF foundation, the maximum moment and shear D/C ratios for the 2.1-m [7-ft] mat are 0.85 and 0.56, respectively. For the 1.2-m [4-ft] mat, the maximum moment and shear D/C ratios are 0.92 and 0.87, respectively.

For the design of CRCF and RF foundations, the maximum bearing pressure on the mat foundation was determined by dividing the maximum reaction force of the individual springs by the area of the corresponding shell element, as described in BSC Section 6.4.2 (2007ax). For the CRCF foundation, DOE calculated the maximum bearing pressure under the mat foundation as 545 kPa [11.4 ksf], as detailed in BSC Section 6.4.2 (2007ae). For the RF foundation, the maximum computed bearing pressure was 488 kPa [10.2 ksf]. For both foundations, the maximum bearing pressure was less than the allowable bearing pressure of 2,394 kPa [50 ksf] DOE proposed for the extreme seismic loading condition. DOE modeled the foundation as soil springs that can model linear elastic compression, but no tension forces.

For the CRCF foundation, DOE computed a safety factor of 3.1 against the structure overturning. For the foundation resistance against sliding, however, the computed safety factor was 0.68, which is less than the reasonable limit of 1.1. DOE computed the expected sliding using the reserve energy method (American Society of Civil Engineers, 2005aa). This analysis yielded a sliding displacement of 5.1 mm [0.2 in] during the 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 when the structure is subjected to the DBGM–2 seismic events (BSC, 2007ax). DOE calculated a sliding displacement of 5.1 mm [0.2 in] using the energy balance method (American Society of Civil Engineers, 2005aa). To obtain a safety factor of two for sliding displacement, DOE indicated that any utility connection that enters the CRCF and RF structures should have a flexibility to accommodate a sliding displacement of at least 10.2 mm [0.4 in], as outlined in BSC Section 6.6.1.2 (2007ae) and BSC Section 6.6.1 (2007ax).

NRC Staff Evaluation: The NRC staff reviewed the CRCF and RF foundation analyses and designs using the guidance in the YMRP. The NRC staff notes that the seismic analysis and design of the CRCF and RF foundations DOE provided are reasonable for the Tier #1 analyses under the DBGM–2 seismic events because the analysis and design are consistent with current engineering practice. However, DOE did not provide analysis results to demonstrate that the results based on analyses which considered a 10.7-m [35-ft] alluvium depth bound the results from analyses in which the alluvium depths were 30.5 and 61 m [100 and 200 ft]. In response to an NRC staff RAI (DOE, 2009eu), DOE stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that an 11-m [35-ft]-thick alluvium depth controls the foundation design because it provides the highest story shears and diaphragm accelerations for the CRCF and RF. Therefore, DOE’s Tier #1 preliminary design analysis with respect to the overturning of the structure is reasonable because use of the 11-m [35-ft]-thick alluvium depth controls the foundation design and the estimated safety factors against the structure overturning are greater than 2. In addition, DOE stated it will perform Tier #2 analysis, which will include realistic soil columns, in support of the detailed design for construction, as described in BSC ACN02 (2007ba). The NRC staff also notes that DOE’s Tier #1 preliminary design analysis with respect to sliding displacement is reasonable because connections entering the structure have sufficient flexibility to accommodate a sliding displacement of 10.2 mm [0.4 in], which provides for a safety factor of 2 for sliding displacement when compared with the DOE-calculated sliding displacement of 5.1 mm [0.2 in].

Wet Handing Facility

The WHF is an ITS reinforced concrete structure that consists of concrete shear walls, roof slab diaphragms, mat foundations, and a pool. The overall footprint of the WHF building is approximately 117 by 120 m [385 by 395 ft], and the ITS portion of the building is about 117 by 91 m [385 by 300 ft]. The maximum height of the building is 30.5 m [100 ft] above grade. The below-grade pool substructure is approximately 35 by 35 m [116 by 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 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 is constructed of 1.2-m [4-ft]-thick concrete walls. The floor diaphragm slabs are generally 0.46 to 0.61 m [1.5 to 2.0 ft] thick, except the internal shielded rooms that are 1.2 m [4 ft] thick.

WHF Structural Analysis

The seismic evaluation of the WHF building was based on a Tier #1 lumped-mass stick model. To incorporate the pool in the structural model, an additional set of global soil springs was attached to the pool foundation. The WHF pool was evaluated for sloshing of water due to a seismic event. This analysis determined water pressures imposed on the pool walls and the amount of freeboard required to prevent spilling of pool water.

The modal analysis of the WHF was performed with water as static mass in the pool using hybrid spectra, as shown in BSC Tables 1–12 (2007bm). Modal analyses were performed for six soil conditions corresponding to soil springs that represent alluvium depths of 9.1 and 30.5 m [30 and 100 ft] for upper, median, and lower bound stiffness conditions. The model with the upper bound 9.1-m [30-ft] alluvium soil case is the stiffest model, as shown in BSC Tables 1–6 (2007bm), resulting in a fundamental period of vibration of $T_1 = 0.157$ s. The maximum interstory drifts occurred for the 30.5-m [100-ft] lower bound soil condition, where $T_1 = 0.26$ s. The maximum interstory drift was 1.8×10^{-4}, as described in BSC Table 15 (2007bm), which is one order of magnitude lower than the interstory drift for Limit State C of 4×10^{-3} (American Society of Civil Engineers, 2005aa). DOE indicated in BSC Table 23 (2007bm) that the upper bound 9.1-m [30-ft] alluvium soil case controls the shear forces for most cases in the east-west direction, whereas the upper bound 30.5-m [100-ft] alluvium condition controls the structural response in the north-south direction.

NRC Staff Evaluation: The NRC staff reviewed the WHF seismic analyses using the guidance in the YMRP. The NRC staff notes that DOE's WHF seismic analysis based on a Tier #1 model is a reasonable representation of the structural response of the WHF subjected to the DBGM–2 seismic events because the analysis is based on an industry-accepted method applicable to the design of ITS surface buildings.

WHF Shear Wall Design

The WHF shear wall design used the general design methodologies evaluated previously in this TER section. For the shear wall design detailed in BSC Section 7 (2007cv), the maximum D/C ratio for in-plane shear was 0.71, whereas the maximum D/C ratio for bending and axial loads was 0.77.

NRC Staff Evaluation: The NRC staff reviewed the WHF shear wall design using guidance in the YMRP. The NRC staff notes that, for the given end forces and moments, the design of shear walls for the Tier #1 analyses under the DBGM–2 seismic events is reasonable because the design is consistent with current engineering practice.

WHF Slab Design

The WHF slab design used the general design methodologies evaluated previously in this TER section. DOE provided the design calculation of five representative reinforced concrete slabs in BSC Section 7 (2007cw), concluding that a reasonable slab design is achieved for the imposed design loads. DOE indicated that these results were preliminary and should only be used in the preliminary design phase of the project.

DOE presented the design of the structural steel framing that supports the reinforced concrete slabs in BSC (2007cu), which includes steel floor and roof decking, steel beams, trusses, and columns. Gravitational loads were assessed following the assumptions in BSC Section 3.1 (2007cu). The results of BSC (2007cu) indicated that the calculated maximum D/C ratio was 0.85 for a roof girder subjected to bending moments.

NRC Staff Evaluation: The NRC staff reviewed the WHF slab design using the guidance in the YMRP and notes that, for the given end forces and moments, the design of slabs for the Tier #1 analyses under DBGM–2 seismic events is reasonable because the design is consistent with current engineering practice.

WHF Foundation Analysis and Design

The WHF foundation consists of a 1.8-m [6-ft]-thick reinforced concrete slab at the grade level, a 2.4-m [8-ft]-thick pool base slab, and 1.8-m [6-ft]-thick retaining walls. The design is based on a finite element model analysis coupled with the Tier #1 model for the mat foundation and the pool structure, as outlined in BSC Section 4.3 (2007bl). DOE indicated that a detailed finite element model will supersede the results of the preliminary analysis, as described in BSC Section 3.1 (2007bl). Shear walls on top of the grade basemat were included to attain the stiffening effects. The applied loading combinations included dead, live, hydrostatic, lateral earth pressure, surcharge pressure, hydrodynamic, and seismic loads, as described in BSC Sections 4.3 and 6.3 (2007bl). To account for SSI, DOE calculated soil spring constants based on the Young's modulus and shear modulus of alluvium and tuff soil layers.

DOE indicated that soil properties for the WHF models were computed for 9.1 and 21.3-m [30 and 70-ft] soil conditions to envelope the potential seismic effects. However, the foundation was designed, following BSC Sections 3.1 and 6.1 (2007bl), on the basis of seismic accelerations from an earlier seismic analysis (BSC, 2007cx) for an upper bound 10.7-m [35-ft] soil condition. In response to an NRC staff RAI (DOE, 2009eu), DOE stated that the forces and moments obtained from the earlier seismic analysis (BSC, 2007cx) bound the results from the seismic analysis presented in the SAR (BSC, 2007bm). DOE did not provide a comparison in its response.

On the basis of the previous structural analyses, DOE generated moment and shear contour plots that were used to compute the shear and flexural reinforcing in the foundation mat. DOE proposed a typical rebar pattern, with additional reinforcement in critical regions. For the grade basemat, DOE computed maximum moment and shear D/C ratios of 0.85 and 0.82, respectively, as shown in BSC Tables 10 and 11 (2007bl). For the pool basemat, the maximum moment and shear D/C ratios are 0.79 and 0.83, as shown in BSC Tables 10 and 11 (2007bl).

To determine the maximum bearing pressure on the mat, the maximum vertical deflections of joints connecting the link elements were multiplied by the equivalent subgrade moduli, as described in BSC Section 6.5.3 (2007bl). The maximum bearing pressure based on linear elastic characteristics was estimated as 713 kPa [14.9 ksf] for the grade basemat and 2,006 kPa [41.9 ksf] for the pool basemat. These bearing pressures are smaller than the allowable bearing capacity of 2,394 kPa [50 ksf], as detailed in BSC Section 6.2.3 (2007ba), which is evaluated in TER Section 2.1.1.1.3.5.4. The soil springs representing the foundation were linear elastic under compression with zero capacity under tension. DOE did not demonstrate that soil springs that assumed linear elastic behavior in compression can be used for the foundation dynamic analysis under the DBGM–2 seismic events.

DOE evaluated the overall stability of the WHF, as outlined in BSC Section 6.7 (2007bl), obtaining an overturning safety factor of 2.7. However, the critical foundation resistance against sliding during a seismic event resulted in a safety factor of 0.363. Then DOE used the reserve energy method (American Society of Civil Engineers, 2005aa), obtaining a sliding displacement of 5 mm [0.2 in]. DOE concluded that sliding will not impact the intended safety function of the structure if the recommended flexible connections are implemented.

NRC Staff Evaluation: The NRC staff reviewed the WHF foundation analysis and design using the guidance in the YMRP and notes that the seismic analysis and design of the WHF foundation for the Tier #1 analyses under the DBGM–2 seismic events are reasonable because the design and analysis are consistent with current engineering practice. However, the NRC staff notes that the foundation design DOE presented is not consistent with the structural analyses included in the SAR (BSC, 2007bl,bm,cx). In response to an NRC staff RAI (DOE, 2009eu), DOE stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that an 11-m [35-ft]-thick alluvium depth controls the foundation design because it provides the highest story shears and diaphragm accelerations for the WHF. Therefore, DOE's use of the 11-m [35-ft]-thick alluvium depth is reasonable for the Tier #1 preliminary design analysis because this alluvium depth controls the foundation design. In addition, DOE stated it will perform a Tier #2 analysis, which will include realistic soil columns, in support of the detailed design for construction, as described in BSC ACN02 (2007ba). The NRC staff notes that DOE stated that connections entering the structure are designed to accommodate a sliding displacement of 10.2 mm [0.4 in], which provides for a safety factor of 2 for sliding displacement when compared with the DOE-calculated WHF sliding displacement of 5.1 mm [.2 in].

<u>Initial Handling Facility</u>

The IHF cask handling process area main structure is a braced-frame steel structure approximately 52 m wide, 57 m long, and 32 m high [170 ft wide, 187 ft long, and 105 ft high]. This main structure includes an internal reinforced concrete structure consisting of 1.2-m [4-ft]-thick walls and roof. 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 common basemat for the IHF main structure and waste package loadout room is a 1.8-m [6-ft]-thick reinforced concrete slab. The internal shielded rooms are made up of 1.2-m [4-ft]-thick concrete walls and common roof slab (i.e., the floor slab for the canister transfer area and for the waste package closure room).

IHF Structural Analysis

The analysis of the IHF steel frame and concrete structures followed the general design methodologies evaluated previously in this TER section. DOE analyzed the IHF structure using the response spectrum method with DBGM–2 ground motions, as outlined in BSC Section 1 (2008am). For the steel frame, DOE did not include SSI effects, because the structure supports are modeled as pinned connections at the basemat, as described in BSC Section 3.1.2 (2008am). In response to an NRC staff RAI (DOE, 2009ev), DOE indicated that SSI can be excluded based on ASCE 4–98, detailed in American Society of Civil Engineers (2000aa). The referenced ASCE section indicates that a fixed-base support may be assumed in modeling structures for seismic response analyses when the frequency obtained from an SSI analysis of a rigid structure (with soil springs representing the supporting soil medium) is more than twice the dominant frequency obtained from a fixed-base analysis of the flexible structure. DOE indicated that the IHF steel structure meets the requirements of ASCE 4–98, outlined in American Society of Civil Engineers (2000aa), but calculations were not provided. For the modal analysis, the mass source included gravitational loads, snow load, and the crane payload, as described in BSC Section 4.3 and 6.6 (2008am). The maximum accelerations of the steel frame (2.26 g) occur in components at elevations of 8.15 and 11.3 m [26.75 and 37 ft], described in BSC Section 7.1.1 (2008am), which shows the effect of higher modes in the structural response. The maximum displacement for the building is 50 mm [1.97 in] for a component at an elevation of 32 m [104.5 ft], as detailed in BSC Section 7.1.2 (2008am), which corresponds to 0.16 percent of the total height.

For the IHF reinforced concrete structures, DOE created a finite element model and performed dynamic analysis using DBGM–2 seismic design spectra data, as outlined in BSC Section 1.0 (2007aq). SSI effects were not included in the analysis and design of the concrete buildings (BSC, 2007aq). In response to an NRC staff RAI (DOE, 2009ev), DOE indicated that the in-plane shear forces will not be significantly affected by the inclusion of SSI effects, because the dominant fixed-base mode response is at or near peak spectral acceleration levels.

NRC Staff Evaluation: The NRC staff reviewed the analysis of the IHF steel frame and reinforced concrete components using guidance in the YMRP. DOE's IHF analysis is based on industry-accepted methods that are applicable to nuclear surface building design. The NRC staff notes that, on the basis of the site response spectra and the fixed-base frequencies of the concrete structures, SSI effects may increase the spectral acceleration. In response to an NRC staff RAI (DOE, 2009ev), DOE stated that (i) SSI effects are not expected to be significant, (ii) the initial design has a demand-to-capacity ratio of approximately 0.6 that provides a margin for demand increases, and (iii) finite element models to be used for the detailed design for construction will include SSI effects.

DOE's IHF structural analysis is reasonable because (i) the analysis is based on industry-accepted methods that are applicable to nuclear surface building design and (ii) the initial design provides a safety margin (i.e., demand-to-capacity ratio of approximately 0.6). The NRC staff notes that DOE stated SSI effects will be evaluated as part of the detailed design process (DOE, 2009ev). The NRC staff also notes that the information obtained by DOE during the detailed design process could be used to confirm DOE's calculated demand-to-capacity ratio for the IHF structure.

IHF Steel Frame Design

The design of the IHF steel frame followed the general design methodologies evaluated previously in this TER section. DOE presented design calculations in BSC Section 7.1.4 (2008am), in which the D/C ratios for columns were below 0.6 and the maximum D/C for the roof bracing group was 0.77.

NRC Staff Evaluation: The NRC staff reviewed the IHF steel frame design using the guidance in the YMRP. DOE's IHF steel frame design is reasonable because seismic design is based on industry-accepted codes and standards that are applicable to nuclear surface building design.

Design of IHF Shear Walls

The design of the IHF shear walls followed the general design methodologies evaluated previously in this TER section. The in-plane and out-of-plane forces and moments used for the reinforced concrete component design were based on integrated section cut forces calculated in SAP2000, as described in BSC Section 4.3.2.2 (2007aq). The design results indicated that the maximum D/C ratio for shear walls subjected to out-of-plane shear was 0.69.

NRC Staff Evaluation: The NRC staff reviewed the IHF shear wall design using the guidance in the YMRP. The IHF seismic design of the reinforced concrete components is reasonable because this design is consistent with current engineering practice.

Design of IHF Slabs

The analysis and design of the IHF slabs followed the general design methodologies evaluated previously in this TER section. The in-plane forces and moments used for the slab design were based on integrated section cut forces calculated in SAP2000. Out-of-plane shear and moments were obtained using the shell elements detailed in BSC Section 4.3.2.2 (2007aq). The results in BSC Section 6.6 (2007aq) indicated that the maximum D/C ratio for out-of-plane shear in the slabs was 0.68.

NRC Staff Evaluation: The NRC staff reviewed the IHF slab design using guidance in the YMRP. The IHF design of these reinforced concrete components is reasonable because the design of the reinforced concrete components is consistent with current engineering practice.

IHF Foundation Analysis and Design

The design of the IHF foundation followed the general design methodologies evaluated previously in this TER section. The IHF foundation consists of two individual basemats that were modeled using a finite element analysis. Unlike the analysis of the IHF superstructure, DOE included SSI effects in the foundation analysis. To account for SSI, DOE calculated soil spring constants on the basis of the elastic and shear modulus of alluvium and tuff.

For the IHF foundation analysis, DOE developed soil springs that represent alluvium depths of 9.1 and 30.5 m [30 and 100 ft], as outlined in BSC Section 4.3.1 (2008ar). DOE assumed in BSC Assumption 3.2.1 (2008aq) that the use of soil springs for the lower bound 30.5-m [100-ft] alluvium depth, which has the least stiffness, will generate the maximum bending moments and shear forces in the mat foundation due to greater deformation. Also, according to SAR Figure 1.1-130, the alluvium thickness for the IHF facility varies from about 9.1 to 27.4 m

[30 to 90 ft], a soil condition that is not necessarily represented by uniform soil conditions at 9.1 or 30.5 m [30 or 100 ft].

The computed global soil springs (BSC, 2008ar) were used to obtain springs per unit area, as detailed in BSC Section 6.2 (2008aq). The springs per unit area were obtained solely from the global translational springs, according to the tributary areas for each joint. DOE's calculations indicated, in BSC Section 6.5.2 (2008aq), that the large mat foundation is the critical mat, exhibiting maximum D/C ratios of 0.97 for moment and shear forces under the DBGM–2 seismic events.

DOE also stated in BSC Section 6.7 (2008aq) that the soil-bearing pressure for the small mat may reach 2,107 kPa [44 ksf] when subjected to the DBGM–2 seismic events. This result leads to a D/C ratio of 0.88, considering an allowable soil-bearing capacity of 2,394 kPa [50 ksf]. This allowable soil-bearing capacity is evaluated in TER Section 2.1.1.1.3.5.4.

NRC Staff Evaluation: The NRC staff reviewed the IHF foundation analysis and design using the guidance in the YMRP and notes that the seismic analysis and design of the IHF foundation is reasonable for the Tier #1 preliminary design analyses under the DBGM–2 seismic events because the seismic analysis and design of the IHF foundation is consistent with current engineering practice. DOE stated it selected representative alluvium depths as a means to bound the actual depths under the IHF (DOE, 2009ev) and that it will perform Tier #2 analysis, which will include realistic soil columns to evaluate the effects of the sloping alluvium, as part of the detailed design process, as described in BSC ACN02 (2007ba). The NRC staff notes that DOE's Tier #2 analysis provides information that could be used to confirm the DOE assumption that the representative alluvium depths of 9.1 and 30.5 m [30 and 100 ft] bound the effects of the sloping alluvium stratum on the foundation response and design (DOE, 2009ev).

2.1.1.7.3.1.2 Aging Facility

DOE provided information related to the design of the aging facility in SAR Section 1.2.7, including the design bases and design criteria, design methodologies, and design analyses. The aging facility consists of two areas of 0.91-m [3-ft]-thick reinforced concrete mat foundation at grade level, designed to support vertical aging casks and horizontal aging modules. The aging pad areas are designated as 17P {L-shaped 397 × 360 m [1,302 × 1,180 ft] with a cutout of 158 × 95 m [519 × 312 ft]} and 17R {rectangular shaped 506 × 274 m [1,661 × 900 ft]}, as depicted in SAR Figure 1.2.7-2. The aging facility was designed to accommodate (i) 2,400 vertical aging casks containing TAD canisters or dual-purpose canisters (DPCs) and (ii) 100 concrete horizontal aging modules containing only DPCs. The materials of the mat foundation are concrete with a minimum compressive strength of 1,034 kPa [5,000 psi] and Grade 60 reinforcing steel (ASTM International, 2006ad). DOE designed the reinforced concrete mat foundation in accordance with ACI 349–2001 (American Concrete Institute, 2001aa).

Design Bases and Design Criteria

In SAR Section 1.2.7.5 and Table 1.2.7-1, DOE provided the design bases and their relationship to the design criteria of the aging facility. DOE derived the design bases from site characteristics and the PCSA. SAR Table 1.2.7-1 provided nuclear safety design bases as (i) structural integrity of the aging pad to protect the ITS SSCs (the aging casks and aging horizontal modules) from external events such as earthquakes, extreme winds, and tornado winds and (ii) protection against aging overpack tipover and sliding. DOE determined that the aging facility must withstand a DBGM–2 seismic event.

NRC Staff Evaluation: The NRC staff reviewed the design bases and design criteria for the aging facility using the guidance in the YMRP. In particular, the NRC staff evaluated the consistency of the design information presented in this chapter with DOE's site-specific information (NRC staff review documented in TER Section 2.1.1.1) and DOE's PCSA information (NRC staff review documented in TER Section 2.1.1.6). The NRC staff notes that the design bases and design criteria for the ITS aging facility are reasonable because (i) the design bases and the design criteria are consistent with the site-specific information and the PCSA results and (ii) the relevant safety functions are addressed {structural integrity from external events to protect against sliding and tipover; location of at least equivalent [.5 mi] away from the heliport}.

Design Methodologies

In SAR Section 1.2.7.6, DOE described the design methodologies used for the structural design of the aging pads. Each aging pad slab is a reinforced concrete mat supported on grade. The pads are designed to withstand loads and load combinations imposed by natural phenomena, such as earthquakes, extreme winds, and tornado winds (SAR p. 1.2.7-11).

To design the concrete mats of the aging facility, DOE performed a finite element static analysis using SAP2000 (Computers and Structures, 2005aa). In response to the NRC staff RAIs, DOE provided a basis for certain assumptions and approaches used in the finite element analysis. In particular, DOE explained (i) the use of a small representative area {26.5 × 35 m [87 × 114 ft]} of the aging pad supporting 16 vertical aging casks to represent the behavior of the actual mats is reasonable because the design for the aging pad is repeating arrays of 16 vertical casks on a continuous concrete slab; (ii) horizontal aging modules were not modeled because the loadings on the pads for the vertical aging casks and horizontal aging modules are similar; (iii) modeling the concrete slab using shell elements with a 0.91 × 0.91-m [3 × 3-ft] mesh size is reasonable because the mesh size is sufficiently fine that the shear and moment contour diagrams reflect a gradual distribution of forces between supports, the areas of maximum positive and negative forces and points of inflection are clearly evident, and thus a finer mesh is not expected to significantly affect the computed forces; and (iv) soil stiffness properties were computed based on the lower bound values of moduli of subgrade reactions for the site-specific alluvium in the horizontal and vertical directions to provide a bounding deformation and a reasonable estimate of design forces (conservatively allows higher displacements in the concrete pad) (DOE, 2009ew).

DOE's analysis shows that the maximum demand-to-capacity ratios for the aging pad are significantly less than one and thus provide a margin for soil–structure interaction effects, including slab flexibility (DOE, 2009ew). DOE stated that analyses for detailed design will consider potential varying soil properties for the concrete foundation (DOE, 2009ew).

NRC Staff Evaluation: The NRC staff reviewed DOE's design methodologies for the aging facility using the guidance in the YMRP. DOE's design methodology for the aging pad is reasonable because (i) the design methodology considered external events that could affect the structural integrity of the aging pad and (ii) the results of the structural analysis result in a maximum demand-to-capacity ratio for the aging pad that are significantly less than one (i.e., a safety margin exists). Additionally, DOE stated it would consider the effects of soil variability on mat design forces as part of the detailed design process of the concrete mat foundation. The NRC staff notes that information obtained by DOE during the detailed design process could be used to confirm DOE's calculated demand-to-capacity ratio for the aging pad.

Regarding DOE's computer analysis, the NRC staff notes (i) DOE's rationale for the model size is reasonable because the continuity of the slab, which was not considered in the analysis, would reduce the design bending moments and shear forces; (ii) the concrete slab model for the vertical aging casks bounds the design bending moments and shear forces for the horizontal aging module mats, because the distributed load for the horizontal aging modules is smaller than that for the vertical aging casks; (iii) the mesh size in the finite element analysis is reasonable because the impact on the computed forces is minimal; and (iv) DOE's modeling of the soil as springs and the assigned stiffness values is reasonable because DOE used a standard industry practice.

<u>Design and Design Analyses</u>

DOE considered the effects of flooding loads due to high-intensity rainfall that could potentially impact the aging facility. In SAR Section 1.2.2.1.6.2.2, DOE stated that the aging facility 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 aging facility 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 aging facility or the planned slopes in the area. In response to an NRC staff RAI (DOE, 2009ew), DOE provided the results of its flood inundation analysis and the planned drainage channels. This information demonstrates that the aging facility concrete pads will be at higher elevation {more than 3.05 m [10 ft]} above the PMF level of approximately 1,138.5 m [3,735.4 ft mean sea level] in the vicinity of the aging facility. The NRC staff's evaluation of the flood protection barriers is discussed in TER Section 2.1.1.7.3.1.3.

DOE also analyzed dead loads (self-weight, aging casks or horizontal modules, and site transporter) and live loads {7.2 kPa [150 psf]} to account for other loads expected during the placement of aging cask, including loads from snow, wind, tornado, and volcanic ash and seismic loads associated with the DBGM–2. The seismic loads were based on the assumption of a PGA 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). For the aging casks on the concrete pad, seismic accelerations of 1.03 g and 0.716 g were used in the horizontal and vertical directions, respectively. The horizontal seismic forces from the aging casks were limited to 0.35 g because the casks would slide at accelerations beyond 0.35 g resulting from the DOE-specified coefficient of friction (COF) of 0.35 between the concrete pad and the aging cask. DOE determined that the wind and tornado loads on the aging casks were less than those from the seismic design basis event. Design bases loads, load combinations, and the design criteria are consistent with the standard industry practice and the site characterization parameters discussed in TER Section 2.1.1.1.

For the analysis of the concrete pad under seismic loads, DOE assumed the peak seismic ground acceleration for the concrete pad and the casks and did not consider the concrete pad flexibility and SSI effects of the pad with the casks and the supporting soils. The SSI effects may amplify the seismic accelerations and increase the mat design forces. In response to an NRC staff RAI (DOE, 2009ew), DOE recognized the potential effects of amplification of vertical seismic accelerations resulting from the SSI effects, but qualitatively considers this to be bounded by the design margins (i.e., the demand-to-capacity ratios are significantly less than one).

In determining the forces on the concrete pad from the aging casks during a seismic event, DOE assumed that the horizontal forces were limited to 0.35 g based on a COF of 0.35 between a cask and pad concrete. An NRC staff RAI (DOE, 2009ew) stated that the COF between the steel and concrete for aging casks, and between concrete and concrete for horizontal aging modules, could be as high as 0.8 and may result in increased concrete mat design forces. In response to this NRC staff RAI (DOE, 2009ew), DOE recognized that a higher COF for steel–concrete interface may be possible. However, DOE stated it will employ construction or engineering measures to achieve a low COF, if necessary, in the detailed design.

DOE assumed that the cask horizontal forces on the concrete pad are limited due to cask sliding at accelerations beyond those based on the COF between a cask and concrete pad. DOE also calculated cask displacement and rotation during a DBGM–2 seismic event to conclude that the casks will be stable and will not tip over and the sliding displacement will be small.

DOE calculated 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 (American Society of Civil Engineers, 2000aa). Considering various load combinations, DOE used the maximum forces (bending moments and shear forces) to design the flexural and shear reinforcing steel in accordance with ACI 349–01 (American Concrete Institute, 2001aa). DOE concluded that shear strength of concrete is greater than the demand and the shear reinforcement is not required. However, DOE proposes 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 Section 11.5.4.1 (American Concrete Institute, 2001aa). In response to an NRC staff RAI (DOE, 2009ew) on the amount and spacing of shear reinforcement of the concrete pad, DOE reiterated its position that it did not rely on the shear reinforcement for increasing the shear capacity of the pad for ACI 349 (American Concrete Institute, 2001aa) code compliance.

The maximum soil-bearing pressure was computed to be approximately 192 kPa [4 ksf], which is less than the bearing capacity of 2,394 kPa [50 ksf] evaluated in TER Section 2.1.1.1.3.5.4. The predicted maximum displacement is approximately 8.1 mm [0.32 in].

NRC Staff Evaluation: The NRC staff reviewed DOE's aging facility structural design using the guidance in the YMRP. DOE's modeling of the soil as springs with the assigned stiffness values is reasonable because it is a standard industry practice. The DOE approach of selecting a single value for the soil material property is reasonable because the single value (i) provides bounding deformations and a reasonable estimate of design forces and (ii) results in demand-to-capacity ratios significantly less than one. Also, DOE stated that it would consider the effects of soil variability on mat design forces as part of the detailed design process of the concrete mat foundation. The NRC staff notes that information obtained by DOE during the detailed design process could be used to confirm DOE's calculated demand-to-capacity ratio for the aging pad.

For the seismic loads analysis, the NRC staff notes that the structural design for the pad is reasonable because the design analysis (i) is consistent with standard industry practice and (ii) results in demand-to-capacity ratios significantly less than one. However, DOE should confirm that the SSI effects on the concrete foundation design of the aging pad for seismic loads do not adversely affect the demand-to-capacity ratios as part of the detailed design process.

DOE cited published information to support use of its proposed 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 Table 6-37 (2009aa). The NRC staff notes that the casks will be stable and will not tip over, and the sliding displacement will be small based on the results of parametric studies of the seismic behavior of dry cask storage systems conducted for NRC (Luk, et al., 2005aa). DOE should confirm the values of COF between steel and concrete, and between concrete and concrete as part of the detailed design process.

The NRC staff evaluated the amount of required reinforcing steel in the concrete pad and its maximum predicted settlement and bearing pressure by comparing the results to those for a facility with similar loads and dimensions and determined that the values were consistent with respect to reinforced concrete design and analysis.

In summary, DOE's design and design analyses methodology are reasonable.

2.1.1.7.3.1.3 Flood Control Features

DOE discussed the flood control features for the GROA in SAR Sections 1.2.2, 1.2.3, 1.2.7, and 1.6.3. The GROA surface facilities are located in two distinct areas: the North Portal pad area and aging facility area. Because of the steeply sloping terrain west of the North Portal pad area at Exile Hill and west to north of the aging facility, the GROA area is prone to flooding by storm runoff. DOE determined that without flood control measures, the design basis probable maximum precipitation (PMP) event and the resulting PMF could result in inundation of the GROA (SAR Section 1.6.3.4.5). DOE provided design information for the proposed flood control features credited with preventing inundation of the surface facilities from a PMF at the site in SAR Figure 1.2.2-7 and in its responses to NRC staff RAIs (DOE, 2009eh,fh). DOE also provided PMF and flood inundation analyses for the proposed flood control features (BSC, 2007db). DOE designated the flood control features as ITS in SAR Table 1.9-1.

DOE's proposed conceptual design includes the following features to control the PMF runoff: (i) a dike and channel system west, north, and east of the aging facility; (ii) a dike and channel system located between the North Portal pad and aging facility 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 southeast of the North Portal pad.

Design Basis and Design Criteria

In SAR Table 1.2.3-3, DOE provided the nuclear safety design bases and design criteria for the flood control features, which require that the flood protection features be located and sized to prevent the ITS structures from being inundated by a flood associated with the PMP event. On this basis, all the ditches, channels, and detention ponds are to be located and sized to convey or attenuate the design basis PMF flow with reasonable freeboard to prevent inundating the surface facilities. This criterion also requires all the flood control features (i.e., slopes of dikes, ditches, and channels) to be designed to withstand the design basis seismic event, design basis PMF flash flood event, and rapid drawdown condition following a flash flood. The performance of this safety function is required throughout the preclosure period.

NRC Staff Evaluation: The NRC staff reviewed the design criteria and design bases for the flood control features DOE proposed using the guidance in the YMRP. The NRC staff notes that the design bases and design criteria used in the design of the flood control features for the PMF are reasonable because the design basis and design criteria are based on site conditions for determining the PMF, the protection against flooding is consistent with Regulatory Guide 1.102 (NRC, 1976ac), and the flood control features prevent inundation of GROA surface ITS structures from a PMF event.

Design Methodologies

The design methods DOE used for the flood control features include estimation of the design basis PMP at the GROA site, which produces the PMF runoff. DOE specified that the flood control features will meet the design criterion discussed previously. The top levels along the dikes were established on the basis of the level of the PMF and the desired freeboard. DOE assumed a layout of channels, dikes, and diversion ditches and estimated the PMF water levels only for channel segments in the dike and channel system proposed in BSC (2007db). The determination of PMF water levels depends on the peak flows in the channel segments. BSC Section 7.2.1 (2007db) stated that the peak flow in the channel increases along the downstream direction because of the contribution from new drainage areas along the downstream direction of the channel. Also, DOE stated that, because it is not practical to calculate peak flow for each individual cross section along the channel, the PMF peak flows calculated by HEC-1 for subareas and concentration points were applied to the appropriate cross sections in the HEC-RAS model.

In response to an NRC staff RAI (DOE, 2010an), DOE stated that the diversion ditches will be sized to transport the PMF provided in BSC (2007db). Further, DOE stated that the detention ponds are downhill from ITS surface facilities of the GROA and significant land is available for locating the detention ponds (DOE, 2010ak). The final design parameters of storm water detention ponds (e.g., storage capacity, maximum flood detention time) will be determined as part of the detailed design (DOE, 2010ak).

SAR Figure 1.2.2-7 presented typical cross sections of the flood control features as a means of providing information on the geotechnical engineering aspects of the proposed flood control features. In response to an NRC staff RAI (DOE, 2010an), DOE provided a summary 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. DOE further stated it would follow 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).

NRC Staff Evaluation: The NRC staff reviewed the design methodologies for the flood control features DOE proposed using the guidance in the YMRP. The NRC staff notes that the design methodologies for flood control are reasonable because the PMF flow and water surface elevation calculations are based on site properties and used the HEC-1 and HEC-RAS models, which conform with established industry practice; the approach is consistent with Regulatory Guide 1.108, and the methodologies follow the U.S. Army Corps of Engineers guidance commonly used in the design of levees and flood control structures (U.S. Army Corps of Engineers, 2000aa, 1994aa).

Design and Analysis

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; however, DOE's analysis did not consider this. In response to an NRC staff RAI on the aging facility design described in SAR Sections 1.2.2 and 1.2.7 (DOE, 2009ew), DOE stated that final grading of the existing topography and associated cross sections through the aging facility site will be considered in the detailed design. DOE estimated a PMP of 335 mm [13.2 in] over 6 hours for the drainage basin encompassing the North Portal pad area and the Aging Facility area. If there were no flood control features, the PMF would result in flooding the North Portal pad area with water depths ranging between approximately 0.61 and 3.3 m [2 and 11 ft]. In response to an NRC staff RAI, DOE presented results of a flood inundation analysis using the HEC-RAS model indicating that the aging pad area would remain above the inundation surface (DOE, 2009ew).

On the basis of the estimated peak PMF flow of 1.42×10^6 L/s [50,219 ft^3/s], DOE stated that it will design the flood protection features to accommodate a flow of 1.56×10^6 L/s [55,240 ft^3/s] to provide 10 percent allowance for the bulking factor (BSC, 2007db). However, on the basis of an analysis with more conservative inputs, DOE estimated a peak PMF flow of 2.14×10^6 L/s [75,726 ft^3/s] (BSC, 2008cd). In response to an NRC staff RAI to clarify this discrepancy, DOE stated that its design basis PMF flow of 1.56×10^6 L/s [55,240 ft^3/s] exceeds the peak flood flow of 1.13×10^6 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).

SAR Section 1.2.2.1.6.2.2 stated that the protection against flooding is in accordance with Regulatory Guide 1.102 (NRC, 1976ac). DOE stated that ITS structures are located at or near the highest elevations of the North Portal and Aging Facility areas that are protected by engineered barriers for flood control and adequate slopes are provided in these areas to preclude inundation of any ITS structures (SAR p. 1.2.2-6). In response to NRC staff RAIs, DOE stated that flood detention pond design details are subject to the design bases for the sizing and placement of the diversion ditches (DOE, 2010ak, 2009fe).

NRC Staff Evaluation: The NRC staff reviewed the design and analysis for the flood control features DOE proposed using the guidance in the YMRP. The NRC staff notes that the design and analysis of the flood control features are reasonable because the design is based on Regulatory Guide 1.102 and the flood protection features are designed to accommodate a flow of 1.56×10^6 L/s [55,240 ft^3/s], which provides a margin of 38 percent over a Category 2 flood event.

2.1.1.7.3.2 Mechanical Handling Transfer Systems

DOE provided design information for the ITS mechanical handling equipment used at the GROA in SAR Sections 1.2.2, 1.2.3, 1.2.4, 1.2.5, and 1.2.6. The four main ITS mechanical handling systems are (i) Canister Transfer Machine (CTM), (ii) Waste Package Transfer Trolley (WPTT), (iii) Spent Fuel Transfer Machine (SFTM), and (iv) Canister Transfer Trolley (CTT). These mechanical handling transfer systems are located in multiple surface facilities; however, their design and functions are the same in all facilities. The NRC staff's review focused on the design bases and design criteria, design methodology, and design and design analysis.

The NRC staff's reviews on the description of the surface facilities and the ability of the mechanical handling systems to perform its intended safety functions are provided in TER Sections 2.1.1.2 and 2.1.1.6, respectively.

2.1.1.7.3.2.1 Canister Transfer Machine

The main function of the CTM is to transfer HLW and spent nuclear fuel (SNF) canisters from a transportation cask or an aging overpack, which arrive in a CTT, to a waste package or aging cask. The CTM is used in the canister transfer areas of the surface facilities and always located on the second floor of the IHF, CRCF, WHF, and RF. In SAR Table 1.2.2-11, DOE specified the rated capacity of the CTM to be 63,502 kg [70 tons] for all facilities. The design features of the CTM were described in SAR Section 1.2.4.2, and the mechanical envelope diagram was shown in SAR Figure 1.2.4-50. DOE also provided instrumentation and logic diagrams in SAR Figures 1.2.4-51 through 1.2.4-56.

Design Bases and Design Criteria

DOE presented the nuclear design bases for the CTM 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. DOE provided specific design criteria to meet each of the required safety functions, along with controlling parameters and bounding values.

DOE 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 American Society of Mechanical Engineers (ASME) NOG–1–2004 Type I (American Society of Mechanical Engineers, 2005aa), require (i) two hoist upper limit switches, (ii) hoist adjustable speed drive (ASD) at setpoints 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.

To prevent a canister breach, DOE's design criterion limits the load drop height. For example, the CTM design cannot 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.

To protect against unplanned movement and to limit travel speed, the following design criteria were imposed: (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, are required to have instantaneous overcurrent protection. The design criterion to prevent runaway of the CTM limits its speed to 6.1 m/min [20 ft/min].

To preclude a nonflat bottom drop of a canister, the design criterion for the CTM is to include guide features for DPCs and TAD canisters.

The design criterion addresses unacceptable radiation doses to workers 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 canister is not raised above the top of the shield bell. In addition, DOE described a procedural safety control (PSC) to mitigate radiation exposure to personnel. This PSC requires developing a procedure for closing the port slide gates when a canister transfer operation is complete. Safety evaluations of these ITS controls are provided in TER Section 2.1.1.7.3.7.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using the guidance in the YMRP. The NRC staff confirmed that the design criteria are derived from ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), which is an accepted code in the nuclear industry. For example, DOE's design criteria of limit switches and load cells are consistent with ASME NOG–1–2004 Sections 5459 and 6445 (American Society of Mechanical Engineers, 2005aa), respectively. The design criterion to prevent runaway of the CTM limits its speed to 6.1 m/min [20 ft/min], which is below the ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) recommended bridge speed of 30.5 m/min [100 ft/min] for the 63,502-kg [70-ton] CTM. The interlocks and limit switches of the CTM are consistent with ASME NOG–1–2004 Section 6440 (American Society of Mechanical Engineers, 2005aa). The PSC that DOE proposed to mitigate personnel radiation exposure is above and beyond the design requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). Therefore, the information DOE provided on the design bases and design criteria is reasonable.

Design Methodologies

DOE used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE considered seismic safety design through a combination of two important design aspects: (i) use of a conservative design code and standard and (ii) fragility assessments for the CTM to support the existence of reasonable capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of DOE's design methodology using the guidance in the YMRP and notes that the methodology is reasonable because DOE's design methodology for the CTM is consistent with that outlined in ASME NOG–1–2004 Sections 4200 and 5200 (American Society of Mechanical Engineers, 2005aa), which is widely accepted in the nuclear industry. Further, the fragility assessments for the CTM are reasonable to support the seismic capacity of the CTM during a seismic event because DOE's design methodology for the CTM is consistent with ASME NOG–1–2004.

Design and Design Analysis

The main CTM design features include load path redundancy, conservative design factors such as limited travel speed and high safety design margins for the grapple, overload protection, redundant braking systems, overtravel limit switches, and other protective devices to ensure safe operation of the CTM. These design features were based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE will also follow the design requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for the choice of materials for the CTM. ASME NOG–1–2004 Section 4210 (American Society of Mechanical Engineers, 2005aa) provides guidance on suitable materials for a Type I crane such as the CTM.

DOE considered static, dynamic, and environmental loads associated with normal operation as well as Categories 1 and 2 event sequences. The specific design loads that DOE used in the design and analysis of the CTM included those from (i) normal operation, (ii) seismic event, (iii) extreme wind conditions (only in the IHF), and (iv) collision. In the case of loads due to normal operation, DOE followed the design requirements of ASME NOG–1–2004 Sections 4140 and 5310 (American Society of Mechanical Engineers, 2005aa). In the case of loads due to a seismic event, DOE considered dead loads, live loads, and seismic loads of DBGM–2 levels. In the case of extreme wind loads, DOE considered the dead load and a nonseismic wind load.

In the case of a collision, DOE considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and the design recommendations of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The NRC staff verified that the CTM design is in accordance with the cited code, which is accepted in the nuclear industry. For example, the CTM braking systems and limit switches are consistent with ASME NOG–1–2004 Sections 5433 and 5443 (American Society of Mechanical Engineers, 2005aa). The design loads DOE considered are also in accordance with ASME NOG–1–2004 Sections 4130 and 4140 (American Society of Mechanical Engineers, 2005aa) and are, therefore, the applicable loads and load combinations for the CTM.

2.1.1.7.3.2.2 Waste Package Transfer Trolley

The WPTT is a trolley that 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 used in the surface facilities of the IHF, CRCF, WHF, and RF. The WPTT 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. It presents an empty waste package to the CTM when positioned under the waste package port in the vertical position. Before the loaded waste package is handed over to the TEV, it is rotated to the horizontal position.

The WPTT has a payload rating of 90,718 kg [100 ton] and is limited to a top speed of 4 km/hour [2.5 mph]. The design features were described in SAR Sections 1.2.3.2.4 and 1.2.4.2.4, and the mechanical envelope diagram was shown in SAR Figure 1.2.4-88. The instrumentation and logic diagrams were depicted in SAR Figures 1.2.4-89 and 1.2.4-90.

Design Bases and Design Criteria

DOE presented the nuclear design bases for the WPTT and their relationship with the design criteria in SAR Tables 1.2.3-3 (IHF) and 1.2.4-4 (CRCF). DOE also provided the specific design criteria for each of the safety functions, along with the controlling parameters and bounding values.

The nuclear safety design bases for the WPTT include preventing rapid tilt down. The design criteria, based on the redundancy design recommendation of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), require the WPTT to be designed with two drive trains to rotate the shielded enclosure. The two drive trains provide a redundant design feature, because one of the drive trains alone can handle the load. Further, electrical power is needed to rotate the load in either direction. Therefore, a loss in power will result in a stationary load because gravity-induced back driving of the drive train is precluded by minimizing gear backlash.

To limit travel speed and protect against unplanned movement, the WPTT design 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 and thereby halts the WPTT. The WPTT is limited to a travel speed of 4 km/hour [2.5 mph].

To protect the WPTT from tipping over or rocking during a seismic event while it is holding a loaded waste package, the design criterion requires that the WPTT be designed to ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for loads and accelerations associated with a DBGM–2 seismic event. Further, the rails on which the WPTT travels are designed with seismic restraints such that the WPTT will not rock during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using guidance in the YMRP In addition, the NRC staff verified that DOE based the WPTT design features on the ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) code and notes that the design will operate as intended. For example, the WPTT uses limit switches to limit WPTT travel in the forward and reverse directions and to limit rotational movement. Additional forward and reverse range detectors are also proposed to trip the WPTT translational motor in case a limit switch fails or if an object is in the path of the trolley. The NRC staff notes that the use of redundant limit switches and range detectors constitutes safe engineering practice and conforms to the design requirements of ASME NOG–1–2004 Section 6440 (American Society of Mechanical Engineers, 2005aa). Therefore, rapid tilt down of the load will be prevented because of the redundant, non-backlash, and non-backdrivable drive trains. DOE's approach to analyze motion during a seismic event and prevent seismic-induced rocking is reasonable and in accordance with ASME NOG–1–2004 Section 4136 (American Society of Mechanical Engineers, 2005aa), and the WPTT will, therefore, withstand seismic-induced rocking and tipover. On the basis of these evaluations, the design criteria DOE provided are comprehensive enough to provide design bounding limits for WPTT design, and the relationship between design bases and design criteria is clearly defined.

<u>Design Methodologies</u>

DOE used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE considered seismic safety design through a combination of two important design aspects: (i) use of a conservative design code and standard and (ii) fragility assessments for the WPTT to support reasonable capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of DOE's design methodology using the guidance in the YMRP. The NRC staff notes that DOE's design methodology for the WPTT is reasonable because it is consistent with that outlined in ASME NOG–1–2004 Sections 4200 and 5200 (American Society of Mechanical Engineers, 2005aa), which is widely accepted in the nuclear industry. Further, the fragility assessments for the WPTT are reasonable to support the WPTT capacity during a seismic event because DOE's design methodology is consistent with ASME NOG–1–2004.

<u>Design and Design Analysis</u>

The main design features of the WPTT are redundant drives, minimal gear backlash, and power interrupt interlocks. These design features are based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE will also follow ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for the choice of materials for the WPTT. ASME NOG–1–2004 Section 4210 (American Society of Mechanical Engineers, 2005aa) provides guidance on suitable materials for a Type I crane such as the WPTT.

DOE's analyses of design load combinations for the design of the WPTT include normal operating conditions, event sequences, and effects of natural phenomena (SAR

Section 1.2.4.2.4.9). In SAR Section 1.2.2.2.9.2.4, DOE specified the design loads related to (i) normal operation, (ii) a seismic event, (iii) extreme wind conditions (only in the IHF), and (iv) a collision. For loads due to normal operation, DOE followed the design requirements of ASME NOG–1–2004 Sections 4140 and 5310 and Table 5453.1(a)-1 (American Society of Mechanical Engineers, 2005aa). For loads due to a seismic event, DOE considered dead loads, live loads, and seismic loads of DBGM–2 levels. For extreme wind loads, DOE considered the dead load and a nonseismic wind load. For a collision, DOE considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and the design recommendations of ASME NOG–1–2004. The NRC staff notes that the design is consistent with the cited code, which is commonly used in the nuclear industry for these types of systems. The NRC staff notes that these design loads are consistent with ASME NOG–1–2004 Sections 4130 and 4140 (American Society of Mechanical Engineers, 2005aa) and are therefore reasonable loads and load combinations for the WPTT.

2.1.1.7.3.2.3 Spent Fuel Transfer Machine

SFTM is described as a bridge crane that operates over the WHF pool. The main function of the SFTM is to transfer commercial SNF (CSNF) assemblies to an empty TAD canister that was previously staged in the pool or, alternatively, to a staging rack in the pool. Human operators use a pendant to control the SFTM.

The rated capacity of the SFTM is 1,361 kg [1.5 tons]. The design features were described in SAR Section 1.2.5.2.2.1.3, and the mechanical envelope diagram was shown in SAR Figure 1.2.5-47. DOE also provided the process and instrumentation diagrams in SAR Figure 1.2.5-48 and the logic diagram in SAR Figures 1.2.5-49 and 1.2.5-50.

<u>Design Bases and Design Criteria</u>

The nuclear safety design bases and their relationship to the SFTM were presented in SAR Table 1.2.5-3. For each of these design bases, DOE presented multiple design criteria to meet the probability of failure of each design basis.

DOE provided two design criteria for the safety design basis to protect against drop of an SNF assembly or any other load SFTM transports. These criteria, based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), include one interlock to prevent the SFTM lifting device from operating if it is not connected to the hoisting system and another interlock to prevent the hoisting operation if the lifting device is not fully engaged or disengaged. To protect against lifting an SNF assembly above the limits for workers' safety, DOE includes a mechanical stop on the SFTM to limit the maximum lift height, using the guidance in ASME NOG–1–2004 Section 5458 (American Society of Mechanical Engineers, 2005aa).

To protect against SFTM collapse during a seismic event, the SFTM must 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.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and the design guidelines of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The NRC staff notes that the DOE design requirements meet the cited code and that the hoist design is more conservative than the specifications in ASME NOG–1–2004 Section 6320 (American Society of Mechanical Engineers, 2005aa). The DOE SFTM design reasonably precludes a load drop because the design is consistent with the hoist specifications of ASME NOG–1–2004 Section 6320 (American Society of Mechanical Engineers, 2005aa) and the design includes two interlocks to ensure safety of hoist operations. The NRC staff also notes that, by using the SFTM mechanical stop to control the lift height such that the vertical travel height does not exceed the water level, workers are protected from radiation. Therefore, (i) the design criteria DOE provided are reasonable to provide design bounding limits and (ii) DOE reasonably defined the relationship between design bases and design criteria.

Design Methodologies

DOE used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE considered seismic safety design through a combination of two important design aspects: (i) use of a conservative design code and standard and (ii) fragility assessments for the SFTM to support the existence of reasonable capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of DOE's design methodology using the guidance in the YMRP. The NRC staff notes that the SFTM is within the scope of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE's design methodology for the SFTM is consistent with that outlined in ASME NOG–1–2004 Sections 4200 and 5200 (American Society of Mechanical Engineers, 2005aa), which is widely accepted in the nuclear industry. The fragility assessments for the SFTM used to determine the seismic capacity are reasonable because the design methodology is consistent with ASME NOG–1–2004.

Design and Design Analysis

The main design features include conservative safety design factors based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) such as service factors of higher than one and keeping allowable stresses below 75 percent of the material yield strength, seismic safety design, overload protection, redundant hoist braking systems, and overtravel limit switches. The SFTM is standard equipment that is currently in operation in other NRC-licensed facilities. DOE will use ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as the principal design code and standard, including the use of materials. ASME NOG–1–2004 Section 4210 (American Society of Mechanical Engineers, 2005aa) provides guidance on suitable materials for a Type I crane such as the SFTM.

DOE's analyses of design load combinations for the SFTM design include normal operating conditions, event sequences, and effects of natural phenomena. In SAR Section 1.2.2.2.9.2.1, DOE specified the following design loads from (i) normal operation, (ii) a seismic event, (iii) extreme wind conditions (only in the IHF), and (iv) a collision. In the case of loads due to normal operation, DOE followed the design specifications of ASME NOG–1–2004 Sections 4140 and 5310 and Table 5453.1(a)-1 (American Society of Mechanical Engineers, 2005aa). For seismic consideration, DOE included dead loads, live loads, and seismic loads of DBGM–2 levels. For extreme wind loads, DOE considered the dead load and a nonseismic wind load. In the case of a collision, DOE considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The NRC staff verified that the design is consistent with the cited code, which is used in the nuclear industry for these types of systems. The design loads DOE proposed are consistent with the specifications in ASME NOG–1–2004 Sections 4130 and 4140 (American Society of Mechanical Engineers, 2005aa) and are therefore reasonable loads and load combinations for the SFTM.

2.1.1.7.3.2.4 Cask Transfer Trolley

The main function of the Cask Transfer Trolley (CTT) is to shuttle between the cask preparation area and the cask unloading room in the IHF, CRCF, WHF, and RF. In the cask preparation room, the CTT receives a loaded cask from the CHC, and in the unloading room, it hands over the loaded cask to the CTM. Once the cask is empty, it is transferred back to the cask preparation room. The CTT drive units and air bearings are controlled and monitored locally and powered by an onboard battery. The operator uses pendant controls to operate the trolley.

In the IHF, the capacity of the CTT is 2.5×10^5 kg [265 ton], whereas the CTT capacity in all the other surface facilities (CRCF, RF, and WHF) is 1.8×10^5 kg [200 ton]. DOE 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 the mechanical envelope diagram was shown in SAR Figure 1.2.3-20. DOE also provided the process and instrumentation diagram in SAR Figure 1.2.4-27.

Design Bases and Design Criteria

DOE 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. To limit the CTT speed, DOE's design criterion for the pneumatic-powered traction drives precludes travel speeds of greater than 4 km/hour [2.5 mph] using the shutoff valves in the air supply of the drive units.

To protect against unplanned movement, a design criterion requires disconnecting the pneumatic power supply during cask unloading so that the CTT is firmly on the floor.

To protect against waste container impact and to minimize seismically induced sliding or rocking, the CTT design includes energy-absorbing features to minimize the effect of seismically induced sliding impact or rocking. The energy-absorbing design features of the CTT are in accordance with ASME NOG–1–2004 Section 5458.1(1) (American Society of Mechanical Engineers, 2005aa).

Further, DOE proposed several other SCs to ensure safe operation of the CTT that are beyond the requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). For example, the CTT has a warning system that alerts 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. The trolley is designed not to tip over, but slide freely during a seismic event without encountering an obstruction. Finally, redundant systems, speed limitations, and protective features ensure that tipover, collision, or uncontrolled movements are avoided.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using the guidance in the YMRP. The NRC staff notes that the design criteria DOE provided are reasonable because the design criteria and Safety Controls (SCs) are derived from ASME NOG–1–2004 (American Society of Mechanical

Engineers, 2005aa). In addition to ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), DOE identified additional codes and standards to design the pneumatic components of the CTT. Further, the DOE description of the relationship of the design bases and design criteria for the CTT is reasonable because the information in SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3 provide the relationship between the design basis and design criteria with respect to limiting the speed and spurious movement of the CTT and impact protection during a seismic event, which are the safety functions identified for CTT.

Design Methodologies

DOE used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) except for unique features of the CTT associated with the pneumatic components. Because ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) does not cover pneumatic components, DOE identified specific design codes and standards that address the pneumatic valves, pressure relief valves, air cylinders, air bearings/casters, air motors, and piping (DOE, 2009dq). These codes and standards are 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, 2007aa) (for safety relief valves); ASME B31.3–2004 (American Society of Mechanical Engineers, 2004ab); American Petroleum Institute 526 and 527 (American Petroleum Institute, 2002aa, 1991aa; American Society of Mechanical Engineers, 2007aa, 2005ab, 2004ab).

Additionally, DOE considered seismic safety design through a combination of two important design aspects: (i) use of a conservative design code and standard and (ii) fragility assessments for the CTT to support the existence of reasonable capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of DOE's design methodology using the guidance in the YMRP. The CTT is within the scope of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) and other cited codes related to the pneumatic components of the CTT. DOE's design methodology for the CTT is consistent with that outlined in ASME NOG–1–2004 Sections 4200 and 5200 (American Society of Mechanical Engineers, 2005aa), which is widely accepted in the nuclear industry . The NRC staff notes that the fragility assessments for the CTT used to determine the seismic capacity are reasonable because the design methodology is consistent with ASME NOG–1–2004 and the other cited codes related to the pneumatic components of the CTT.

Design and Design Analysis

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. Some of these safety features, such as continuous monitoring of the CTT drives, are based on ASME NOG–1–2004 Section 6472.4 (American Society of Mechanical Engineers, 2005aa), which requires checking the motor thermal adequacy. Other safety features, such as onboard battery controls, are above and beyond the requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) because the CTT is a nonstandard, specialized trolley. DOE will use ASME NOG–1–2004 Sections 4200 and 5200 (American Society of Mechanical Engineers, 2005aa) for the construction materials used in the CTT design. ASME NOG–1–2004 Section 4210 (American Society of Mechanical Engineers, 2005aa) provides guidance on suitable materials for a Type I trolley such as the CTT.

DOE's analyses of design load combinations for the CTT design include normal operating conditions, event sequences, and effects of natural phenomena. In SAR Section 1.2.2.2.9.2.4, DOE specified the following design loads: (i) normal operation, (ii) a seismic event, (iii) extreme wind conditions (only in the IHF), and (iv) a collision. In the case of loads for normal operation, DOE followed the design requirements of ASME NOG–1–2004 Sections 4140 and 5310 and Table 5453.1(a)-1 (American Society of Mechanical Engineers, 2005aa) for the CTT design. For loads due to a seismic event, DOE considered dead loads, live loads, and seismic loads of DBGM–2 levels. In the case of extreme wind loads, DOE considered the dead load and a nonseismic wind load. For a collision, DOE considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and the design requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The NRC staff verified that the CTT design is consistent with well-accepted codes and standards for both the mechanical and pneumatic components of the CTT and will ensure safe operation. For example, the onboard battery systems and controls eliminate the need to drag a power line on the floor because this poses additional electrical hazards to workers. The NRC staff notes that control and monitoring of the air pressure used on the CTT for limiting the potential for the CTT to move in an unpredictable manner are reasonable because the pneumatic power used to power the air bearings to lift the CTT fails safe (i.e., comes to rest supported by the landing pads) on either loss of air or overpressurization and the CTT speed is limited to less than 4.0 km/h [2.5 mph] due to air supply throttle valve limits (SAR p. 1.2.4-12). In addition, the design loads are in accordance with ASME NOG–1–2004 Sections 4130 and 4140 (American Society of Mechanical Engineers, 2005aa) and therefore the applicable loads and load combinations were included for the CTT design analysis.

2.1.1.7.3.3 Heating, Ventilation, and Air Conditioning System

DOE provided information related to the materials of construction, codes and standards, and design of ITS HVAC systems. This information was provided in SAR Sections 1.2.2.3, 1.2.4.4 (CRCF), 1.2.5.5 (WHF), 1.2.8.3 [Emergency Diesel Generator Facility (EDGF)], and 1.9. The ITS HVAC systems in the CRCF and WHF provide temperature control, flow control, and filtration and support confinement. In the EDGF, the ITS HVAC system provides cooling to support the equipment in this facility.

The ITS HVAC systems include dampers, ductwork (including supports), fans, high-efficiency particulate air (HEPA) filters, moisture separators, prefilters, air handling units, electrical power supplies, and I&Cs. The NRC staff's review focused on the design bases and design criteria, design methodology, and design analysis for these systems, except for the electrical power supplies and I&C, which are reviewed in TER Sections 2.1.1.7.3.6 and 2.1.1.7.3.7, respectively. The NRC staff's review of the HVAC system general description is in TER Section 2.1.1.2.3.2.4 and ITS HVAC system performance is in TER Section 2.1.1.4.3.3.2.1.

<u>Design Bases and Design Criteria</u>

DOE listed the ITS HVAC systems nuclear safety design bases in SAR Tables 1.9-3 (CRCF) and 1.9-4 (WHF). The nuclear safety design bases for the EDGF ITS HVAC systems were also listed in these tables. DOE directly expressed 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 guidance [e.g., Regulatory Guide 1.52 (NRC, 2001ae)].

The nuclear safety design bases were developed based on the PCSA and include safety functions and controlling parameters for ITS HVAC systems performance during potential Category 2 event sequences.

DOE identified two safety functions for the surface nuclear confinement HVAC systems in the CRCF and WHF, and one safety function for the surface nonconfinement HVAC system in the EDGF: (i) mitigate the consequences of radionuclide release (CRCF and WHF) and (ii) support the ITS electrical function (CRCF, WHF, and EDGF). To mitigate the consequences of radionuclide release, the design criteria require two full-capacity, independent trains with automatic start capability if the operating train fails. To support the ITS electrical function, the design criteria require an independent train for the rooms associated with each ITS electrical train.

In terms of controlling parameters, DOE specified a probability of failure for the ITS HVAC systems over a mission time of 30 days following a potential radionuclide release. Additionally, for the WHF, DOE specified a probability of failure for the HVAC systems over a 1-day mission time following a radionuclide release involving the cask sampling and cooling process. DOE provided HVAC system fault trees to support the controlling parameter values used in the nuclear safety design bases. In response to the NRC staff RAI (DOE, 2009fd) on design bases and design criteria for the ITS HVAC systems, DOE provided additional information to address the requirements for overall filtration efficiency and the cooling requirements for ITS electrical equipment. DOE stated that HVAC systems will maintain indoor environmental conditions in accordance with American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) 2007 (American Society of Heating, Refrigerating, and Air Conditioning Engineers, 2007aa).

SAR Tables 1.2.4-4 and 1.2.5-3 identified 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, DOE stated that air is continuously exhausted from each battery room to maintain hydrogen concentrations well below the explosive limit (SAR p. 1.2.4-55). Additionally, the battery rooms are equipped with hydrogen gas detectors (SAR p. 1.4.1-15) and each group of electrical and battery rooms is served by redundant sets of HVAC supply and exhaust equipment (SAR p. 1.2.5-58). Thus, DOE stated hydrogen accumulation during battery charging was precluded during normal operations (SAR p. 1.2.4-60). In response to an NRC staff RAI (DOE, 2009fd), DOE indicated that it will perform 21 air changes per hour for the battery rooms; DOE also stated that 21 air changes per hour exceed the specification in ASHRAE 2007 (American Society of Heating, Refrigerating, and Air Conditioning Engineers, 2007aa) that exhaust systems should be designed to provide 5 volume changes per hour to preclude hydrogen accumulation when battery design information is unavailable.

NRC Staff Evaluation: The NRC staff reviewed DOE's design bases and design criteria for ITS HVAC systems using the guidance in the YMRP. The NRC staff notes that the cooling and filtration specifications are reasonable because they are based on accepted guidance and codes and standards (American Nuclear Society, 1997ad; American Society of Heating, Refrigerating and Air Conditioning Engineers, 2007aa). In addition, the generator room temperature is based on accepted industrial design goals (Cummins Power Generation, 2004aa) and is therefore reasonable. ASHRAE 2007 (American Society of Heating, Refrigerating, and Air Conditioning Engineers, 2007aa) is an applicable industry standard to determine volume change requirements to preclude hydrogen accumulation. DOE's RAI response (DOE, 2009fd) specifies design criteria to limit hydrogen concentration by performing

21 air changes per hour in the battery room, which exceeds the 5 volume changes per hour specified in ASHRAE 2007 by more than a factor of 4. The NRC staff notes that DOE's cooling and filtration specifications are reasonable; DOE specified hydrogen removal in SAR Section 1.2.4.4.1 and specified an applicable industry standard to determine volume change requirements (DOE, 2009fd). Therefore, DOE's design bases and design criteria are reasonable.

Design Methodologies

DOE specified that design methodologies for ITS HVAC systems are in accordance with the codes and standards identified in SAR Section 1.2.2.3. In response to an NRC staff RAI (DOE, 2009fd), DOE stated that it will design HEPA filters measuring 610 × 610 × 292 mm [24 × 24 × 11.5 in] in accordance with ASME AG–1–2003 (American Society of Mechanical Engineers, 2004ac). In addition, DOE stated that ASDs will be designed in accordance with National Electrical Manufacturers Association (NEMA) ICS 7–2006 (National Electrical Manufacturers Association, 2006ab). Prefilters and high efficiency filters for air handling units will be designed according to ASHRAE 2004 (American Society of Heating, Refrigerating and Air Conditioning Engineers, 2004aa) with their efficiency calculated using ANSI/ASHRAE 52.1–1992 (American Society of Heating, Refrigerating and Air Conditioning Engineers, 1992aa). Sizing criteria for filters and coils will be in accordance with ASHRAE 2005 (American Society of Heating, Refrigerating and Air Conditioning Engineers, 2005aa), and cooling coils and heating coils will be designed in accordance with ARI 410–2001 (Air Conditioning and Refrigeration Institute, 2002aa). In addition, DOE sized ducts on the basis of maintaining a fluid velocity of 12.7 m/s [41.68 ft/s] to minimize particulate settlement consistent with DOE-HDBK-1169–2003 (DOE, 2003ae).

NRC Staff Evaluation: The NRC staff reviewed the applicability of DOE's design methodology using the guidance in the YMRP. The NRC staff notes that the design methodology is comprehensive and appropriate because the methodology is based on codes and standards that are applicable to HVAC design, consistent with standard industry practice and NRC guidance for nuclear facilities (NRC, 2001ae).

Design and Design Analyses

In addition to the codes and standards described in the previous section, DOE used the 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) that provide guidance on the design, inspection, and testing criteria for air filtration and adsorption systems. DOE also used Regulatory Guide 3.18 (NRC, 1974ab), which provides guidance on design of confinement barriers and systems. DOE also indicated that it used ASME AG–1–2003, including ASME AG–1a–2004 (American Society of Mechanical Engineers, 2004ac), in lieu of ASME AG–1–1997 (American Society of Mechanical Engineers, 1997aa), and ASME N509–2002 (American Society of Mechanical Engineers, 2003ab), in lieu of ASME N509–1989 (American Society of Mechanical Engineers, 1996aa). In addition, DOE identified that it used DOE–HDBK–1169–2003 (DOE, 2003ae) in lieu of ERDA 76-21 (Burchsted, et al., 1976aa).

DOE stated that the construction materials for the ITS HVAC systems are in accordance with the codes and standards identified in SAR Section 1.2.2.3. DOE 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, DOE referred to ASME AG–1–2003, including ASME AG–1a–2004 (American Society of Mechanical Engineers, 2004ac) for the construction materials involving fans and HEPA filter housings.

DOE identified the use of independent trains in its design criteria for the ITS HVAC systems. For example, DOE identified two full-capacity, independent trains with automatic start capability on failure of the operating train for the subsystem that exhausts from areas with canister breach potential. For the subsystem that provides cooling for the ITS electrical equipment and battery rooms, DOE identified an independent train for the rooms associated with each ITS electrical train. Additionally, DOE described the physical separation of the trains. For example, for the CRCF, DOE identified Train A HVAC equipment located on the opposite end of the building from the Train B HVAC equipment (BSC, 2008ac). In response to an NRC staff RAI (DOE, 2009dw) on the independence of trains and the potential for a single point of failure, DOE stated that individual components such as interlocks and ASDs would not be a single point of failure as they may cause a spurious transfer of the operating train but will not cause system failure by themselves.

DOE's analyses of design 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, DOE identified the International Building Code 2000 (International Code Council, 2003aa) for the design of HVAC ducts and duct supports for seismic loads. DOE, however, did not credit the HVAC system for confinement following a seismic event.

DOE evaluated the thermal performance of waste forms and waste containers in the facility using standard simulation tools ANSYS® Version 8.0 and FLUENT® Version 6.0.12. DOE 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. DOE stated that the calculated peak cladding temperature remained below the established limit of 400 °C [752 °F] for normal and 570 °C [1,058 °F] for off-normal conditions. Using these calculations, DOE showed that the waste form can maintain the established temperature limit without the proper functioning of the HVAC system under off-normal conditions.

NRC Staff Evaluation: The NRC staff reviewed DOE's design information using the guidance in the YMRP and Regulatory Guides 1.140 and 1.52 (NRC, 2001ad,ae). While these regulatory guides refer to the use of older versions of ASME AG–1–1997 (American Society of Mechanical Engineers, 1997aa) and ASME N509–1989 (American Society of Mechanical Engineers, 1996aa), the NRC staff notes that the more recent versions of these ASME standards DOE used are appropriate because the recent versions include the guidelines and standards from the older versions. The NRC staff further notes that DOE's use of DOE–HDBK–1169–2003 (DOE, 2003ae), in lieu of ERDA 76-21 as recommended by Regulatory Guide 1.52 (NRC, 2001ae), is reasonable because DOE–HDBK–1169–2003 is based on ERDA 76-21 with updated information provided by industry and subject matter experts and is similar to the older version.

The thermal evaluation techniques DOE used are reasonable because DOE used standard simulation tools that are commonly used for numerical analyses. DOE's thermal analysis, which shows that the waste form will be able to maintain the established temperature limit without the functioning of the HVAC system, provides supplementary information regarding the significance of the HVAC system design relative to maintaining the temperature limit for the waste form.

DOE's design using physically separate HVAC trains is consistent with Regulatory Guide 1.52 (NRC, 2001ae). In addition, DOE specified as part of its design criteria the use of independent trains and further stated in an RAI response (DOE, 2009fs) that the HVAC trains are independent because failure of components in one train cannot cause failure of both trains. The NRC staff notes that the design using physically separate HVAC trains is applicable because it is consistent with NRC guidance (e.g., NRC, 2001ae). Because ITS HVAC systems are designed with independent trains, the NRC staff also notes in TER Section 2.1.1.6.3.2.8.2.2 that DOE reasonably addressed redundancy.

In summary, DOE's design and design analysis for the ITS HVAC systems are comprehensive and are reasonable. DOE used applicable techniques, and the design is consistent with industry standard codes and guidance.

2.1.1.7.3.4 Other Mechanical Systems

DOE provided design information for ITS mechanical systems other than the mechanical handling transfer systems evaluated in TER Section 2.1.1.7.3.2. This information was provided in SAR Sections 1.2.2 through 1.2.6 and SAR Tables 1.2.3-3, 1.2.4-3, 1.2.5-3, 1.2.6-3, and 1.9-2 through 1.9-5. Other mechanical systems are classified as follows: crane systems, special lifting devices, shield and confinement doors, rails, platforms, and racks. These mechanical systems are located in multiple surface facilities. However, their design and functions are the same regardless of the location. The mechanical systems reviewed in this section are standard equipment that is commonly used in other nuclear facilities. The NRC staff review focused on the design bases and design criteria, design methodology, and design and design analysis.

The NRC staff reviews on the description of the surface facilities are provided in TER Section 2.1.1.2, and the ability of the other mechanical systems to perform their intended safety functions is evaluated in TER Section 2.1.1.6.

2.1.1.7.3.4.1 Crane Systems

Crane systems are used in the CRCF, WHF, RF, and IHF. The main function of the crane systems is to upend a cask to a vertical position or move casks from one location to another. Examples include the 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, DOE specified the load ratings of 2,721 to 2.7×10^5 kg [3 to 300 ton] for these cranes.

<u>Design Bases and Design Criteria</u>

DOE presented the design bases and design criteria for 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). Similar information for auxiliary pool and jib cranes was presented in SAR Table 1.2.5-3 (WHF). DOE presented the design bases and design criteria for the CTM maintenance crane of the RF in SAR Table 1.2.6-3.

DOE used the design requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for Type I cranes to protect against a load drop. These criteria include the design recommendations of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), such as load path redundancy, conservative design factors, overload protection, redundant braking systems, and overtravel limit switches to limit the possibility of a load drop. In particular, the cranes (e.g., the cask handling crane) that handle critical loads are designed following Type I requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). ITS cranes that do not handle critical loads (e.g., CTM maintenance crane) are designed following Type II requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). Finally, non-ITS cranes that do not handle critical loads are designed following Type III requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa).

To limit the drop height, DOE's design criterion limits the hoist height. For example, the crane design precludes lifting the cask 9.1 m [30 ft] above the floor when the crane hoisting system is in a two-block condition.

To limit travel speed of the trolley and bridge, DOE imposed a speed limitation of 6.1 m/min [20 ft/min].

To protect against crane collapse onto a waste container, DOE's design criterion is based on the Type I crane requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for loads and accelerations associated with a DBGM–2 seismic event.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using guidance in the YMRP. The NRC staff determined that the design criteria for the crane category of mechanical systems are consistent with Type I, II, and III requirements of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as well as ASME NUM–1–2004 (American Society of Mechanical Engineers, 2005ac) for the design of the jib cranes. DOE's classification of cranes into Type I, II, and III classes, on the basis of handling critical loads, is consistent with that described in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). In addition, the NRC staff notes that reasonable seismic stability was considered in the design of the cranes because DOE accounted for loads and accelerations associated with a DBGM–2 seismic event. Therefore, the information DOE provided on the design bases and design criteria is reasonable.

<u>Design Methodologies</u>

For crane systems, DOE's design methodology for overhead bridge cranes was based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for Type I, II, and III cranes. DOE's design methodology for jib cranes used to handle loads over waste containers was based on ASME NUM–1–2004 (American Society of Mechanical Engineers, 2005ac) for Type IA cranes. In addition, DOE considered seismic safety design by accounting for loads and accelerations associated with a DBGM–2 seismic event as described in SAR Section 1.2.2.1.6.3.

NRC Staff Evaluation: The NRC staff reviewed DOE's design methodology using the guidance in the YMRP and design guidelines of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) and ASME NUM–1–2004. The NRC staff notes that both the overhead bridge cranes and the jib cranes are within the scope of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) and ASME NUM–1–2004 (American Society of Mechanical Engineers, 2005ac), respectively. Both ASME codes are accepted in the nuclear industry and provide reasonable engineering design guidelines on overload protection, redundant braking systems, overtravel switches, and protective devices to make the likelihood of a load drop by a crane extremely small. In addition, the DOE design of the overhead and bridge cranes is based on site-specific ground motions (SAR p. 1.2.2-21). Therefore, the design methodology is reasonable.

Design and Design Analysis

The main design features of cranes are load path redundancy; conservative design factors, such as allowable stresses being well below the material yield strength and hoist speeds being inversely proportional to rated load; overload protection; redundant braking systems; and overtravel limit switches to ensure that a load drop is mitigated. DOE stated that it will follow the guidelines in ASME NOG–1–2004 Sections 4200, 5200, and 6200 (American Society of Mechanical Engineers, 2005aa) for the crane material selection.

In SAR Section 1.2.2.2.9.2.1, DOE defined the loads for which the cranes are designed. DOE considered the following design loads for the cranes: dead loads, live loads, dynamic loads, seismic loads, environmental loads, and event sequence loads. In SAR Section 1.2.2.2.9.2.2, DOE defined the design loads for jib cranes. DOE considered the same design loads for the jib crane.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP and the codes and standards [ASME NOG–1–2004 and ASME NUM–1–2004 (American Society of Mechanical Engineers, 2005aa,ac)]. Both ASME NOG–1–2004 and ASME NUM–1–2004 (American Society of Mechanical Engineers, 2005aa,ac) are accepted codes for the design of conservative, fail-safe crane systems that are currently in operation in various NRC-licensed nuclear facilities. The NRC staff verified that the design is consistent with the cited codes. The load combinations used in the design analysis were based on ASME NOG–1–2004 Sections 4140 and 5310 (American Society of Mechanical Engineers, 2005aa). Therefore, the loads used in the design analysis are reasonable for crane design. The NRC staff also notes that DOE reasonably considered seismic design loads for DBGM–2 levels, which are above and beyond the design load specifications of ASME NOG–1–2004 and ASME NUM–1–2004 (American Society of Mechanical Engineers, 2005aa,ac).

2.1.1.7.3.4.2 Special Lifting Devices

DOE provided design information for special lifting devices in SAR Section 1.2.2.2.1. This includes yokes, grapples, and adapters that are typically located at the end of mechanical handling equipment 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 have mechanical jaw actuation mechanisms with safety release features.

Design Bases and Design Criteria

DOE presented the design bases and design criteria for specialized lifting devices such as yokes and grapples 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, DOE provided specific design criteria to meet each of the design bases.

To protect against a cask drop or load drop onto a cask/canister, DOE's design criterion is based on ANSI N14.6–1993 [American National Standards Institute, 1993aa; as modified by NUREG–0612, Section 5.1.1(4) (NRC, 1980aa)]. In addition, DOE has special safeguards in the lifting device design to prevent a load drop. For example, the naval waste package inner lid grapple uses three lifting jaws, equally spaced, to engage the lid of the waste package.

Further, raising or lowering the hoist is possible only if the grapple is fully engaged with the load. The grapple with a suspended waste package inner lid is mechanically prevented from unintentional disengagement.

To protect against a load drop during a seismic event, DOE required that the special lifting devices be designed for loads and accelerations associated with a DBGM–2 seismic event.

NRC Staff Evaluation: The NRC staff reviewed DOE’s design bases and design criteria using the guidance in the YMRP. The NRC staff also verified that the design criteria for the special lifting devices are based on ANSI N14.6–1993 (American National Standards Institute, 1993aa). The use of this code is reasonable for designing the special lifting devices because it specifically applies to lifting devices for radioactive containers weighing more than 4,500 kg [10,000 lb]. Therefore, the information DOE provided on design bases and design criteria for special lifting devices is reasonable, and DOE reasonably defined the relationship between design bases and design criteria.

Design Methodologies

DOE’s design methodology for special lifting devices follows the recommendations of ANSI N14.6–1993 (American National Standards Institute, 1993aa), as modified by NUREG–0612 Section 5.1.1(4) (NRC, 1980aa). NUREG–0612 Section 5.1.1 (4) (NRC, 1980aa) modifies the calculation of the stress design factor on the basis of the combined maximum static and dynamic loads instead of only the static weight as recommended in ANSI N14.6–1993 (American National Standards Institute, 1993aa).

In addition, DOE considered seismic safety by requiring the design to account for loads and accelerations associated with a DBGM–2 seismic event.

NRC Staff Evaluation: The NRC staff reviewed DOE’s design methodology using the guidance in the YMRP and design guidelines of ANSI N14.6–1993 (American National Standards Institute, 1993aa). In some cases, DOE’s proposed safety design features are over and above the design guidelines of ANSI N14.6–1993 (American National Standards Institute, 1993aa). For example, the mechanical jaw actuation mechanisms of the grapples’ safety release features and special lifting devices have interlocks to prevent accidental device actuation if the special lifting device is not properly connected to the adaptor. Therefore, DOE’s design methodology for the special lifting devices is reasonable.

Design and Design Analysis

The main design features of the special lifting devices are conservative design factors based on the design guidelines of ANSI N14.6–1993 (American National Standards Institute, 1993aa). In SAR Section 1.2.2.2.7, DOE stated that the materials used for the special lifting devices are consistent with ANSI N14.6–1993 Section 4 (American National Standards Institute, 1993aa). In addition to the design guidelines of ANSI N14.6–1993 (American National Standards Institute, 1993aa), DOE stated in SAR Section 1.2.4.2.1.1.3.1 that the design will include 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 that prevent grapple disengagement when a load is suspended from the grapple. All these design features are used to mitigate a load drop from the special lifting device.

In SAR Section 1.2.2.2.9.2.3, DOE defined the following specific design loads for the design and analysis of the special lifting devices: (i) loads related to normal operation, (ii) loads due to a seismic event, and (iii) loads due to a collision. In the case of loads related to normal operation, DOE followed the design requirements of ASME NOG–1–2004 Section 4140 and 5310 (American Society of Mechanical Engineers, 2005aa). In the case of loads due to a seismic event, DOE considered dead loads, live loads, and seismic loads of DBGM–2 levels. For a collision, DOE considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed DOE's design and design analyses using the guidance in the YMRP. The NRC staff reviewed DOE's information using the design recommendations of ANSI N14.6–1993 (American National Standards Institute, 1993aa) and ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) and verified that the design is consistent with the cited codes. The design loads were in accordance with ASME NOG–1–2004 Sections 4130 and 4140 (American Society of Mechanical Engineers, 2005aa). The materials used for the special lifting devices were consistent with ANSI N14.6–1993 Section 4 (American National Standards Institute, 1993aa). Because both ASME NOG–1–2004 and ANSI N14.6–1993 (American Society of Mechanical Engineers, 2005aa; American National Standards Institute, 1993aa) are widely accepted codes and standards in the nuclear industry, the design and the loads used in the design analysis of special lifting devices are reasonable.

2.1.1.7.3.4.3 Other Mechanical Structures

DOE described other mechanical structures in GROA surface facilities. In this category of other mechanical structures, four subcategories were reviewed: shield and confinement doors, rails, platforms, and racks.

The principal function of the systems, which were grouped into the shield and confinement doors (SAR Section 1.2.4.2.1.1.3.1) and sliding gates, is to protect facility personnel from direct radiation.

The rails support the WPTT and TEV. Rails also support large gantry cranes such as the bridge crane of the CTM. For the TEV, the rails also provide the electrical power for the traction motors.

The platforms (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) include multilevel steel structures that provide personnel and tool access to the top of aging overpacks or transportation casks. These steel structures also provide a single operating platform to access the top of the shielded enclosure of the WPTT for maintenance purposes.

The racks (SAR Section 1.2.5.2.2.1.3) stage SNF assemblies and TAD canisters to blend fuel assemblies for thermal management and to allow for loading and unloading flexibility. The TAD canister staging racks are steel structures that hold TAD canisters for staging purposes. The staging racks provide seismic support for the canisters and support canisters at an elevation that minimizes potential drop height.

<u>Design Bases and Design Criteria</u>

DOE 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 are presented in SAR Table 1.2.4-4 (CRCF).

To protect against direct personnel exposure and mitigate radionuclide consequences, the shield doors and slide gates consist 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 have obstruction sensors that prevent the door from operating if any object is on its travel path. Additionally, interlocks prevent the shield doors from opening if other doors are open or if dedicated radiation monitors are triggered. The motors that operate the door cannot produce the torque required to breach a canister.

To protect against TEV derailment during waste package loading, the rails are designed for a DBGM–2 seismic event so that a derailment is prevented.

To prevent the platforms from a seismic-induced collapse or a waste container breach due to seismically induced impact, DOE stated that it will use the design methods and practices provided in American Institute of Steel Construction, Inc. (1997aa). Additional structural capacity, over and beyond the safety recommendations of the aforementioned code and standard, is provided to preclude platform collapse. Finally, the platform design includes energy-absorbing features to limit impact forces on the waste container.

In the case of the racks, to protect against seismically induced SNF canister tipover or canister impact, the rack design criterion includes seismic supports. In addition, a protective wall adjacent to the SNF staging rack ensures that large objects cannot collide with the rack, preventing damage to either the rack or SNF assemblies or both. To protect against fire-induced canister breach, the design criterion of staging racks includes several safety design features such as (i) fixed neutron absorbers for criticality control in accordance with ANSI/ANS 8.21–1995 (American Nuclear Society, 1995aa) and ANSI/ANS 8.14–2004 (American Nuclear Society, 2004aa), (ii) fuel assembly spacing to prevent criticality, (iii) a thermal barrier that encloses the bottom and sides of the canisters to control canister temperatures during certain fire scenarios, and (iv) ventilation of staging areas to remove decay heat.

NRC Staff Evaluation: The NRC staff reviewed DOE’s design bases and design criteria using the guidance in the YMRP. The NRC staff notes that the design bases and the design criteria for the shield doors, slide gates, platforms, rails, and racks are reasonable because the design bases and design criteria addressed the relevant safety functions to protect against direct exposure of personnel, limit damage to the waste container due to collapse or closing of doors or gates, limit collapse of the platforms, limit the potential for waste container damage from derailment or tipover during a seismic event, and limit damage to the waste canister in the staging rack from either collapse or fire. Additionally, DOE has used industry-accepted standards (American Institute of Steel Construction, Inc., 1997aa) for design methods and practices. The design of platforms and the design of fixed neutron absorbers for criticality control will be in accordance with ANSI/ANS 8.21–1995 (American Nuclear Society, 1995aa) and ANSI/ANS 8.14–2004 (American Nuclear Society, 2004aa).

Design Methodologies

In SAR Section 1.2.4.2.1.6, DOE stated that the design methodology for shield and confinement doors is based on ANSI/AISC N690–1994 Section Q1.2 (American Institute of Steel Construction, 1994aa). In SAR Section 1.2.4.1.6, DOE stated that the design methodology for TEV and WPTT rails is based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). DOE’s design methodologies for steel platforms and racks are based on the ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa) code and standard.

NRC Staff Evaluation: The NRC staff reviewed DOE's design methodologies using the guidance in the YMRP and design guidelines of ANSI/AISC N690–1994 and ASME NOG–1–2004 (American Institute of Steel Construction, 1994aa; American Society of Mechanical Engineers, 2005aa), and notes that the methodologies are in accordance with the cited codes and standards. The shield doors, confinement doors, platforms, and racks are within the scope of ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). Further, ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa) has been referenced in past nuclear facilities NRC licensed and is commonly used for the design of steel safety-related structures for nuclear facilities. The NRC staff also notes that ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) is applicable for rails because this code addresses the use of rails in conjunction with overhead bridge cranes and trolleys. For example, ASME NOG–1–2004 Sections 4160 and 4460 (American Society of Mechanical Engineers, 2005aa) specify tolerances and requirements, respectively, for runway rails of gantry cranes. Therefore, DOE's design methodologies are reasonable because they are consistent with the design guidelines in accepted codes and standards.

Design and Design Analysis

DOE summarized the design features of the shield and confinement doors, such as (i) the shield doors are interlocked so that they will 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 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 feature of platforms and racks is to provide personnel safety and seismic protection for canisters that are staged on various racks.

DOE defined the load combinations for the shield and confinement doors and platforms as per ANSI/AISC N690–1994 Table Q1.5.7.1 (American Institute of Steel Construction, 1994aa). The material used for this category of mechanical systems is consistent with ANSI/AISC N690–1994 Section Q1.4 (American Institute of Steel Construction, 1994aa). In SAR Section 1.2.4.1.7, DOE stated that the materials of construction and design loads for the TEV and WPTT rails will be in accordance with ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). In SAR Section 1.2.4.2.2.9, DOE specified the load combinations for racks in accordance with ANSI/AISC N690–1994 Table Q1.5.7.1 (American Institute of Steel Construction, 1994aa). DOE proposed to follow the load combinations in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) for the design of TAD canister staging racks and truck cask handling frames. For other types of ITS racks or platforms and frames, DOE plans to use only ANSI/AISC N690–1994 Table Q1.5.7.1 (American Institute of Steel Construction, 1994aa) for load combinations.

NRC Staff Evaluation: The NRC staff reviewed DOE's design and design analyses using the guidance in the YMRP. The NRC staff reviewed DOE's information on the shield doors, rails, platforms, and racks using the design recommendations of ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa) and ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The NRC staff verified that the design is consistent with the cited codes. DOE used the load combinations and stress limits of ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa) to conform to appropriate safe engineering practices when designing safety-related steel structures for nuclear facilities. Because both ASME NOG–1–2004 and ANSI/AISC N690–1994 (American Society of Mechanical Engineers, 2005aa; American Institute of Steel Construction, 1994aa) are referenced in the nuclear industry, the NRC staff notes that the design and the loads used in the design analysis of the shield doors, rails, platforms, and racks are reasonable.

2.1.1.7.3.5 Transportation Systems

DOE provided design information for the ITS transportation systems used at the GROA. This information was provided in SAR Sections 1.2.8.4, 1.3.2, 1.3.3, and 1.3.4. The four ITS transportation systems are (i) TEV, (ii) Site Transporter, (iii) Cask Tractor and Cask Transfer Trailer (CTCTT), and (iv) Site Prime Mover. The NRC staff's review focused on the design bases and design criteria, design methodology, and design and design analysis.

The NRC staff review on the description of the surface facilities are provided in TER Section 2.1.1.2, and the ability of transportation system to perform its intended safety function is provided in TER Section 2.1.1.6.

2.1.1.7.3.5.1 Transport and Emplacement Vehicle

DOE plans to use the Transport and Emplacement Vehicle (TEV) to transport the loaded waste packages from the surface facilities (CRCF and IHF) to the designated locations in the emplacement drifts. The entire TEV operation consists of (i) handling the waste packages by accepting, lifting, and securing the waste packages inside a protective structure of the TEV for transport; (ii) shielding personnel from the waste packages in unrestricted areas; (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) safely returning to the surface facility. DOE also plans to use the TEV for waste package retrieval operations. Under normal conditions, the retrieval operations consist of performing the reverse sequence of steps that defines the emplacement operations. Under off-normal conditions, the TEV will perform the retrieval operations after the off-normal condition is restored to normal. The NRC staff's evaluation of the recovery and retrieval processes is described in the TER Section 2.1.2.

The TEV is a crane rail-based transporter {5-m [16-ft]-diameter operating envelope} with a shielded enclosure (SAR Figure 1.3.4-20) and propelled by eight electric motors that are powered by an electrified third rail. The TEV is a manually operated or computer controlled, fully instrumented handling equipment with sensors and communication networks. It contains a battery backup system with sufficient capacity to only power the sensors and maintain communication with the Central Control Command in the event of a power failure. It contains a restraint system, redundant braking systems, and a shielded enclosure that surrounds the waste package.

DOE presented the TEV design features in SAR Sections 1.3.2, 1.3.3, and 1.3.4. DOE also provided a complete mechanical handling and design report on the TEV that included its detailed design. In addition, DOE provided mechanical envelope calculations (DOE, 2009ez) and I&C system diagrams for the TEV. The NRC staff's review of the TEV description is provided in TER Section 2.1.1.2, and the ability of the TEV to perform its intended safety functions is provided in TER Section 2.1.1.6.

Design Bases and Design Criteria

DOE provided the nuclear safety design bases and their relationship to the design criteria for the TEV in SAR Table 1.3.3-5. Specific design criteria for each design basis were provided, along with controlling parameters and bounding values.

To protect against tipover during a DBGM–2 seismic event, the design criterion is to minimize the TEV center of gravity. The main feature the TEV design specified was a wide base {e.g., 3.4 m [11 ft]}.

To protect against runaway during operations, the design criterion specified was to employ special drive mechanisms and braking systems. More specifically, DOE specified that wheel size, gearbox configuration, and disk brakes be designed into the TEV to achieve runaway prevention requirements.

DOE also considered protection against derailment using ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as the design criteria. In addition, the TEV and its interface with the rails at the loadout station are required to restrain the TEV during a DBGM–2 seismic event.

Protection of personnel from direct exposure to radioactivity is specified with the criterion that requires the use of interlocks and shielding materials on the TEV. For example, DOE specified interlocks that prevent opening the TEV doors in unrestricted areas between the surface handling facility and the emplacement drift turnouts.

Finally, to protect the waste packages from ejection during a spectrum of seismic events, the design criterion listed in the SAR provides locks to the TEV shield doors. DOE indicated it would use electromechanical locks that mechanically prevent unintentional motion of the shield doors.

NRC Staff Evaluation: The NRC staff reviewed the design bases and their relationship to the design criteria for the TEV using the guidance in the YMRP. The NRC staff notes that the information DOE provided on the design bases and design criteria is reasonable because the information addresses the design bases and criteria of the TEV relevant to safety functions to protect against derailment of TEV, tipover of the TEV, ejection of the waste package from the TEV, and direct radiation exposure of workers. Additionally, DOE's use of ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) as a design criterion for protection against derailment of the TEV is reasonable due to its use in the nuclear industry. Therefore, DOE's design criteria for the design bases are reasonable.

Design Methodologies

In designing the TEV, DOE first considered specific characteristics of the GROA site, such as (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 2.7×10^5 kg [300 tons], maximum length of 6,299 mm [248 in], and maximum height of 2,349 mm [92.49 in]}. Due to the unique nature of the GROA site and the specialized nature of the TEV operations, DOE utilized a methodology for designing the TEV that included several studies. In one study (BSC, 2008ck), DOE identified appropriate codes and standards [primarily, the ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) standard for Type I cranes along with six additional supporting standards]. In a second study (BSC, 2008cl), DOE (i) evaluated the applicability of the identified standards to the TEV application, (ii) identified inadequacies ("gaps") in the standards, and (iii) defined additional requirements to supplement the existing codes and compensate for the deficiencies. In the third and final study (BSC, 2008ac,be), DOE outlined a design development plan such that the TEV can be relied upon to perform its intended safety functions. On the basis of these studies, DOE stated that it will (i) perform further reliability analyses and (ii) generate 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 the TEV's safety functions (SAR Table 1.3.3-7).

NRC Staff Evaluation: The NRC staff reviewed the design methodologies of the TEV using guidance in the YMRP. The NRC staff reviewed the SAR and the three design studies listed previously and notes that DOE described a design methodology that considered the relevant safety components of the TEV (e.g., drive motors, locks, hardwired circuitry, restraint features, and brake system components) needed to satisfy the design criteria. DOE stated that consensus codes and standards may not be fully applicable due to the specialized nature of the TEV (SAR p. 1.3.3-45). The TEV is a rail-based, shielded unit used to move waste packages that operates over a crane rail track; therefore, DOE's application of codes and standards for cranes and rail designs is reasonable to the extent the codes and standards represent similar or analogous situations. In particular, DOE's selection of the ASME NOG–1–2004 standard (American Society of Mechanical Engineers, 2005aa) as the primary guidance in the design is reasonable due to its consideration for dynamic seismic qualifications, materials controls, harsh/radiation environmental-condition requirements, single-failure proof requirements, and testing requirements and its use by the nuclear industry. Additionally, DOE stated that extended factory acceptance testing of key TEV equipment systems will be performed, in an environment that simulates actual operating conditions as closely as possible, to determine whether the performance and reliability of the TEV SSCs meet the design criteria (SAR p. 1.3.3-45). DOE's design methodology for the TEV is reasonable because (i) the design methodology is consistent with the DOE-cited codes and standards, (ii) the DOE design development plan calls for additional reliability analyses as part of the design process, and (iii) extended factory acceptance testing at full scale is planned when the TEV is completed.

Design and Design Analysis

DOE used the ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) Type I Crane industry standard to define the material specifications, load combinations, and methodologies for the design of the TEV lifting and propulsion functions. DOE performed design calculations related to horsepower requirements on a grade, which is not addressed in ASME NOG–1–2004, by applying the guidelines in Cummins and Given (1973aa). DOE also specified the industry standard Doman (1988aa) for the design and future construction of the shielded enclosure, front shield doors, door drives, hinges, and locks. DOE relied on this document as guidance for anticipated load combinations, design considerations for the hinges, door drive systems, and safety devices associated with the shield doors. Each of the nuclear safety design bases and the design criteria in the TEV design is detailed next.

Protection of TEV Against Tipover

DOE addressed preventing TEV tipover during a DBGM–2 seismic event by designing the TEV with a wide vehicle base {3.4 m [11 ft]} and a low center of gravity. The TEV design ensures that the waste package positioning is low in the vertical direction {i.e., less than 356 mm [14 in] from top of the rail to bottom of the emplacement pallet}. In addition, DOE specified ASME NOG–1–2004 Section 4457 (American Society of Mechanical Engineers, 2005aa) to define a requirement for gantry stability during extreme environmental or abnormal conditions, as described in BSC Section 6.12 (2008ck). 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 includes a seismic restraint between the TEV chassis and the rails (SAR Figure 1.3.3-41) and the TEV operating speed is limited to 2.7 km/hour [1.7 mph] (SAR Section 1.3.1.2.2).

NRC Staff Evaluation: The NRC staff reviewed DOE's information on preventing TEV tipover using the guidance in the YMRP. For the TEV design, the NRC staff notes that the TEV design includes reasonable measures for limiting the frequency of tipover of the TEV because the TEV has a wide base, has a low operating speed, uses double-flanged wheels designed to prevent wheel climb when the TEV travels around a curve, uses a seismic restraint between the TEV chassis and the rails, and uses ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) safety factors.

Protection of TEV Against Runaway

To protect the TEV against runaway during operations, DOE incorporated five design and control elements: (i) high-torque drive motors, (ii) integral disk brakes in the drive motors, (iii) high-ratio gearboxes, (iv) drive components to mechanically limit the speed of the TEV to 2.7 km/h [1.7 mph], and (v) rail brakes (SAR p. 1.3.3-43).

DOE determined the TEV speed limit using ASME NOG–1–2004 Table 5333.1-1 (American Society of Mechanical Engineers, 2005aa) and the largest TEV payload of 80,829 kg [89.1 ton] that is consistent with waste package envelope information in other analyses. DOE chose the fastest appropriate TEV speed of 2.7 km/h [1.7 mph] (SAR Section 1.3.3.1.2.2) with a 10 percent variation. To achieve the desired speed limit, DOE selected a 1,750-rpm motor coupled to a 914-mm [36-in] wheel.

DOE considered the interaction of the TEV with other SSCs that could potentially affect the speed control, such as electrical power failure. DOE included redundancy to mitigate the effect of the power loss by incorporating eight integrated double disc brakes, one pair for each drive motor.

To supplement the disc brakes, DOE added a second level of runaway-prevention redundancy by adding eight high gear-ratio (100.75:1) gearboxes, one for each drive motor. The final TEV design will use noncoasting gearboxes currently available through commercial vendors (DOE, 2009ez).

DOE also introduced an additional braking system for parked or off-normal conditions. DOE selected rail brakes (or "thrusters") that directly couple the TEV to the rail in a wedgelike braking action.

NRC Staff Evaluation: The NRC staff reviewed DOE's information on protection of the TEV against runaway using the guidance in the YMRP. The NRC staff notes that DOE's design approach in defining the TEV maximum speed is reasonable because the value selected is consistent with the ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) and less than the largest impact speed of 6.1 m/sec [22 km/hour] [20 ft/sec (13.6 mph)], equivalent to a drop height of 2 m [6.5 ft], that may cause a waste package breach, as detailed in BSC Section 3.2.1.10 (2008bz).

DOE's design is reasonable because the engagement of the disc brakes is not only redundant but also independent of a human operator's reaction time, sensor response time, or software code performance. The actuation is both automatic and fail-safe because the brakes are normally mechanically engaged through mechanical springs and released only on command. This is consistent with rail vehicle design in ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) and on-road heavy-duty vehicle practices for safety.

In addition, DOE's rail breaks approach is reasonable because it utilizes the weight of the TEV to produce the frictional normal force rather than relying on an external actuation system that may fail. DOE stated that rail brakes (thrusters) are commercially available units and are designed to railroad standards.

Prevention of TEV From Derailment

DOE addressed TEV derailment prevention at the loadout station during a seismic event. DOE described a custom seismic restraint system as shown in SAR Figure 1.3.3-41. This passive restraint system consists of L-shaped structures located on the underside of the TEV chassis and extending under the railhead.

DOE also addressed derailment due to wheel climb on the tightest curve {radius of 61 m [200 ft]}. DOE performed a geometric assessment to ensure that the wheel climb would not occur. If a derailment did occur, the TEV design, which limits the height of wheel drop, would reduce the impact on the waste package, thereby minimizing potential breach. By specifying that the bottom faces of the TEV chassis and the base plate are at the same height from the rails, the design ensures that a potential drop would only be 76 mm [3 in].

NRC Staff Evaluation: The NRC staff reviewed DOE's information on TEV derailment using the guidance in the YMRP. The NRC staff notes that design of the L-shaped seismic restraint, which is located at the front and back and left and right sides of the TEV, represents a passive and simple design with the potential to prevent derailment by limiting the vertical motion of the TEV during a seismic event (SAR Figure 1.3.3-41). DOE indicated that the TEV will be designed based on the requirements of ASME NOG–1–2004 to mitigate derailment due to seismic events (SAR Table 1.3.3-5). ASME NOG–1–2004 Section 7000 also includes structural and weld testing for seismic restraint systems. Therefore, DOE provided reasonable information supporting the functionality and effectiveness of the seismic restraint system to prevent derailment.

In addition, outside the loadout station (i.e., during transport), DOE reduced the potential for derailment by specifying double-flanged wheels for the TEV. The NRC staff notes this is a reasonable design for reducing the potential for derailment because (i) considerable energy would be required to lift the wheels beyond the flanges and (ii) the guidelines related to material selection, loading, clearances, and flange width and height for these types of flanged wheels are provided in ASME NOG–1–2004.

DOE performed structural analyses indicating that the waste package would satisfy DOE's acceptance criteria in the event of a 508-mm [20-in] drop (SAR Table 1.5.2-9). The NRC staff notes that the TEV mitigation for derailment in regard to a 76-mm [3-in] drop height is reasonable because the DOE structural analysis determined that a 508-mm [20-in] drop would result in the outer corrosion barrier of the waste package receiving only 25 percent of the energy necessary for the waste package to breach (SAR Table 1.5.2-9).

Therefore, the DOE design for the TEV and the structural analysis that considered a drop of the waste package provide reasonable support for the DOE approach with respect to the prevention of the TEV from derailment.

Protection Against Waste Package Ejection and Against Inadvertent Door Opening

The TEV incorporates doorlock systems that are electrically activated. Because the shield door system consists of two, outward-swinging doors, one of the doors houses the lock solenoids while the second shield door contains structurally featured holes in which the steel shot bolts penetrate to prevent door motion. The design features protect against waste package ejection resulting from a seismic event, collision, derailment, normal transport, or tipover. DOE indicated that the cross-sectional area and material strength will be selected to withstand loads resulting from a DBGM–2 seismic event. The ejection control and the incorporation of a non-ITS collision avoidance system (SAR Section 1.3.2.1) provide redundancy in reducing the probability of collisions.

DOE also considered the prevention of inadvertent opening of the TEV's front and rear doors in unrestricted areas. DOE incorporated a series of interlocks and a hardwired ITS switch mounted on the TEV, designed per ASME NOG–1–2004 and Institute of Electrical and Electronics Engineers (IEEE) IEEE 384–1992, 323–2003, 383–2003, 344–2004, and 336–2005 (Institute of Electrical and Electronics Engineers, 2006aa; 2005aa; 2004aa,ab; 1998aa).

Externally mounted mechanical arms (permanently located only in areas that are safe for door opening) physically engage and actuate the ITS switch on the TEV. This ITS switch is interlocked with the door solenoid circuitry that prevents opening of the doors when the ITS switch is electrically determined to be in the incorrect position. DOE referenced Doman (1988aa) as the industry guidance for the design of these components. In addition, DOE indicated that operators in the Control Center will confirm the proper position of the ITS switch before the TEV is allowed to proceed.

NRC Staff Evaluation: The NRC staff reviewed DOE's information on protection against waste package ejection and inadvertent door opening using the guidance in the YMRP. The TEV provides reasonable protection against waste package ejection and inadvertent door opening because the design incorporates redundancy (such as collision mitigation systems) and is based on industry standards that are accepted in the nuclear industry (BSC, 2008ck), which will ensure that the design specifications and testing of the hardwired interlocks meet their intended safety function. A more detailed evaluation of interlocks and ITS controls is provided in TER Section 2.1.1.7.3.7.

2.1.1.7.3.5.2 Site Transporter

DOE 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, aging facility, and low-level radioactive waste facility. DOE 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. DOE also indicated that the site transporter is designed to withstand the natural phenomena included in SAR Table 1.2.2-1 as well as horizontal and vertical ground motion shown in SAR Figures 1.2.2-8 to 1.2.2-13.

Design Bases and Design Criteria

DOE presented the nuclear safety design bases for the site transporter and their relationship to the design criteria in SAR Table 1.2.8-2. DOE provided specific design criteria to meet each of the nuclear safety design bases.

To protect against spurious movement, DOE defined procedural controls for site transporter motion to prevent spurious movement. Complementing this requirement, DOE defined a second design basis related to motion. This would protect against runaway, which was controlled by the design criterion specifying the control of the site transporter speed to 4 km/hour [2.5 mph].

DOE also defined a design basis to preclude fuel tank explosion. DOE indicated that the site transporter will be equipped with a limited supply of fuel and an onboard fire suppression system. In addition, DOE specified operational procedures to prevent the use of the internal combustion engines inside the surface facilities.

Furthermore, three additional design bases with corresponding design criteria were delineated for the site transporter: (i) a design specification that limits the aging overpack lift height to 0.91 m [3 ft] thus reducing the severity of a drop; (ii) a design criteria specifying a wide base resulting in an inherent vertical stability that protects against tipover; and (iii) a design requirement for reasonable clearance and energy-absorbing features to minimize sliding impact and inducing stresses on the waste container.

NRC Staff Evaluation: The NRC staff reviewed the design bases and design criteria using the guidance in the YMRP. The DOE design bases for the site transporter are reasonable because the design bases consider protection against spurious movement (including a limit for the speed of the site transporter), fuel tank explosion, tipover, drop of the waste package, and sliding impact and inducing stress on the waste package. Further, the NRC staff notes that the design criteria DOE provided are reasonable because DOE provided specific design criteria for all of the safety functions assigned to the site transporter in the design bases (SAR Table 1.2.8-2).

Design Methodologies

DOE based the design of the site transporter on the requirements of the ASME NOG–1–2004 Type I Crane industry standard (American Society of Mechanical Engineers, 2005aa). In addition, DOE indicated that other codes ASME NOG–1–2004 recommended will also be utilized: (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, which are 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 (National Electrical Manufacturers Association, 2006aa) for motor size selection.

NRC Staff Evaluation: The NRC staff reviewed the design methodologies using the guidance in the YMRP. The NRC staff reviewed the design methodology for the site transporter and notes that DOE provided reasonable guidelines for material specifications, load combinations, and methodologies for the design of the structural support, lifting, propulsion, and braking functions of the site transporter because the design guidelines are consistent with ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa), an industry-accepted standard, and other codes that are recommended for use in ASME NOG–1–2004.

Design and Design Analysis

The design features for satisfying each of the nuclear bases and criteria follow.

Protection Against Runaway

DOE specified a safety design requirement to limit the site transporter to 4 km/hour [2.5 mph (220 ft/min)] through proper sizing of the electric motors and gearboxes that constrains maximum rotational speed. DOE acknowledged a deviation of the site transporter speed specification from ASME NOG–1–2004 Table 5333.1-1 (American Society of Mechanical Engineers, 2005aa). The code recommends a maximum Type I crane speed of 3.2 km/hour [1.98 mph] (closest value to the site transporter speed) corresponding to transport loads weighing between 0 and 44,452 kg [0 and 49 ton]. Higher loads like a typical 2.3×10^5-kg [250-ton] vertical aging overpack expected to be handled by the site transporter would require even lower speeds. To assess the impact of this deviation, DOE quantified the robustness of transportation casks that were required to survive a 1,016-mm [40-in] horizontal drop onto an unyielding object. The analysis indicated that the impact energy of a 4 km/hour [2.5 mph] site transporter speed is a factor of 90 less than the impact energy of the 1,016-mm [40-in] drop, as described in DOE Number 8 (2009ez). DOE concluded that no breach would occur from a site transporter collision at its design speed limit.

NRC Staff Evaluation: The NRC staff reviewed the protection against runaway of the site transporter using the guidance in the YMRP. DOE acknowledged that the safety design speed limit for the site transporter of 4 km/hour [2.5 mph (220 ft/min)] is larger than the speed recommended in ASME NOG–1–2004 Table 5333.1-1 (American Society of Mechanical Engineers, 2005aa). However, DOE conducted structural analysis to support the DOE design speed limit that indicated the speed of 4.0 km/h [2.5 mph] was a factor of 90 less than the impact energy the cask is designed to survive from a drop. The NRC staff notes that the 4.0 km/h [2.5 mph] speed limit is reasonable because the DOE safety margin is a factor of 90 against a cask breach and the DOE probability of a collision at 4.0 km/hour [2.5 mph] is less than 1×10^{-8}.

Protection Against Fuel Tank Explosion

DOE stated that the site transporter will be equipped with a fire suppression system and carries a maximum of 378 L [100 gal] of fuel (SAR p. 1.2.8-32). DOE also stated that the TAD canister is designed to withstand a fully engulfing fire without failure of its containment function and the aging overpack is designed to withstand the same fully engulfing fire without failure of its shielding function. The burning period for the fully engulfing fire is determined based on a pool fire of all site transporter hydrocarbon fuel and other combustible lubricating and hydraulic fluids plus other combustible and flammable materials on the site transporter (SAR p. 1.4.3-3).

NRC Staff Evaluation: The NRC staff reviewed the protection against fuel tank explosion of the site transporter using the guidance in the YMRP. The NRC staff notes that the DOE design of the site transporter for the protection against fuel tank explosion is reasonable because the site transporter has an onboard fire-suppression system and the TAD canister and aging overpack are designed to withstand a fully engulfing fire (SAR Section 1.4.3.1.2).

Protection Against Spurious Movement

DOE credited the site transporter with the safety function of protection against spurious movement while performing lifting/lowering maneuvers. While a loaded canister is being placed into or removed from the aging overpack, DOE required a Procedural Safety Control (PSC) to disconnect the electrical power to the site transporter and setting of the brakes (SAR p. 1.2.8-33). DOE also required independent verification of the deactivation of the electrical power.

NRC Staff Evaluation: The NRC staff reviewed the protection against spurious movement of the site transporter using the guidance in the YMRP. The NRC staff notes that the DOE site transporter design for protection against spurious movement is reasonable because DOE specified a PSC that includes both deactivation of power to the site transporter and the setting of the brakes.

Reduction in the Severity of a Drop

DOE credited the site transporter with reducing the severity of a drop by designing the site transporter to limit lifting height of an aging overpack to 0.3 m [1 ft]. In addition, DOE 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 will be conducted to provide assurance against premature failure of the lifting members. DOE also referenced an industry standard—ASME NQA–1–2000 Subpart 2.15 (American Society of Mechanical Engineers, 2000aa)—that will be followed and that addresses hoisting, rigging, and transporting of items. In addition, DOE indicated in BSC Section 4.8.1.2.7 (2007bi) its adoption of NUREG–0612 (NRC, 1980aa) and ANSI N14.6–1993 (American National Standards Institute, 1993aa). The former is a standard for the control of heavy loads and the latter for the design of special lifting devices for shipping containers weighing 4,536 kg [10,000 lbs] or more.

NRC Staff Evaluation: The NRC staff reviewed reduction in the severity of a drop using the guidance in the YMRP. The DOE site transporter design for reducing the severity of a drop is reasonable because DOE has included industry-accepted standards for the site transporter's lifting system design; designed the site transporter lift height to a limit of 0.3 m [1 ft], which is below the DOE design criteria of 0.91 m [3 ft]; and stated that the equipment qualification program will be used to ensure the SSCs ITS have the ability to perform their safety function (SAR Section 1.13).

Protection Against Sliding Impacts

DOE credited the site transporter by limiting the frequency of sliding impact of the site transporter into a wall and inducing stresses on the waste package due to seismic events. DOE defined operating clearances and energy-absorbing features (SAR Table 1.2.8-2). In addition, DOE designed the site transporter with a drive system consisting of tracks to provide significant resistance to sliding.

NRC Staff Evaluation: The NRC staff reviewed protection against sliding impacts using the guidance in the YMRP. The NRC staff notes that the DOE design approach for the site transporter to provide protection against sliding impacts is reasonable because the design provides for operating clearance and energy-absorbing features to ensure that the waste container is not breached from sliding impacts. DOE stated that it will perform equipment qualification testing to ensure that the SSCs ITS have the ability to perform their safety function (SAR Section 1.13).

Protection Against Tipover

DOE credited the site transporter with the safety function of protecting against tipover. DOE 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 also synchronized to share the load, reduce the probability of both a drop from overloading and a tipover. DOE stated that the passive cask restraint system provides stabilization during cask movement. The restraint system, which contacts the cask after it has been raised to the correct height, was illustrated in SAR Figure 1.2.8-49. Pins in each of the arms lock the restraint in place in case of an assembly failure.

NRC Staff Evaluation: The NRC staff reviewed protection against tipover using the guidance in the YMRP. The DOE design approach for the site transporter to provide protection against tipover is reasonable because the design provides for a wide base including a restraint system incorporated into the site transporter that provides for three-point stability. DOE stated that it will perform equipment qualification testing to ensure that the SSCs ITS have the ability to perform their safety function (SAR Section 1.13).

2.1.1.7.3.5.3 Cask Tractor and Cask Transfer Trailers

The cask tractor will be used in intrasite operations to pull cask transfer trailers carrying a transportation cask containing a horizontal DPC from the RF to the aging pad. It will also be used to pull cask transfer trailers carrying a horizontal shielded transfer cask containing a horizontal DPC from the aging pad to the WHF. The two 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. These systems are designed to raise, level, and stabilize the cask transfer trailer while transferring the DPC at the horizontal aging module. The CTCTT performs functions at the GROA that are similar to functions performed by similar vehicles (commercially available equipment) utilized in the nuclear facilities such as independent spent fuel storage sites.

DOE described the tractor as a vehicle driven by a human operator. The seat is equipped with sensors and interlocks that shut 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 design description and operational processes for the CTCTT were described in SAR Section 1.2.8.4.2. DOE also provided a mechanical equipment envelope in SAR Figure 1.2.8-50 and indicated that the CTCTT will be designed to withstand the natural phenomena loading parameters provided in SAR Table 1.2.2-1, as applicable.

Design Bases and Design Criteria

DOE presented the nuclear safety design bases for the CTCTT and their relationship to the design criteria in SAR Table 1.2.8-2. DOE provided specific design criteria to meet each of the nuclear safety design bases.

DOE defined a design criterion that limits the maximum speed of the CTCTT, which is intended to protect against runaway. In addition, DOE specified Procedural Safety Control 2 (PSC–2) to the CTCTT (SAR p. 1.2.8-36) that states the CTCTT is to be deactivated and the brakes set during waste handling operation. DOE also specified that a fire suppression system onboard the vehicle precludes fuel tank explosions. Furthermore, DOE specified a criterion for a maximum trailer height for accepting the cask that reduces the severity of a drop. Finally, DOE 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 Evaluation: The NRC staff reviewed the design bases and design criteria using the guidance in the YMRP and notes that the design bases and criteria are reasonable because they address the relevant safety functions for protection from runaway, fuel tank explosions, drops, and cask punctures by considering speed, height, impact forces on the cask, and temperature rise due to fires.

Design Methodologies

DOE based the cask tractor design on industry-accepted standards for the design of the CTCTT. These standards provide guidance for the design of safety systems for (i) personnel and burden carriers [ANSI/ITSDF B56.8 (Industrial Truck Standards Development Foundation, 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 (American National Standards Institute, 1992aa)].

NRC Staff Evaluation: The NRC staff reviewed the design methodologies using the guidance in the YMRP. The NRC staff reviewed the applicability of the standards to the CTCTT design and determined that the standards cover the safety aspects of the equipment necessary to satisfy the nuclear safety design criteria. These standards are appropriate because they apply to operator-controlled equipment. These standards are also applied to the design of commercially available trailers that carry nuclear payloads traveling at interstate highway speeds. These speeds are higher than the speed limit specified for the GROA site. The application of the standards is therefore conservative, and the design methodology is reasonable.

Design and Design Analysis

The design features with respect to the cask tractor and cask transfer trailers DOE used to address each of its nuclear safety design bases and design criteria follow.

Protection Against Runaway

DOE specified a safety design requirement for the cask tractor to limit travel speed to 4 km/hour [2.5 mph]. The tractor is equipped with a dual-brake system and an alarm that notifies the operator of a system failure. DOE also designed the trailer with a braking system that is independent of the tractor and engages automatically when the trailer is disconnected. This brake system is designed to hold the trailer on a 5 percent grade with a 2 percent cross slope (SAR p. 1.2.8-35). DOE also indicated that it will use ANSI/ITSDF B56.9–2006 Section 7.13 (DOE, 2009ez) to design the physical speed controls of the cask tractor.

NRC Staff Evaluation: The NRC staff reviewed protection against runaway using the guidance in the YMRP. The DOE design approach of the CTCTT to provide protection against runaway is reasonable because the aforementioned standard used for the design of the speed controls is consistent with equipment commonly used in nuclear facilities that is designed to provide motion and handling controls of tow tractors with a sit-down rider and is powered by an internal combustion engine. The NRC staff notes that the DOE approach is also reasonable because the cask tractor and cask transfer trailer are equipped with braking systems that operate in tandem when connected and are 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 DOE Number 9 (2009ez).

<u>Preclusion of Fuel Tank Explosion</u>

The tractor design is also credited with the safety function of precluding fuel tank explosion. DOE did not provide any specific design detail on the fuel tank, but indicated that the tractor will be designed in accordance with the ANSI/ITSDF B56.9–2006 (Industrial Truck Standards Development Foundation, 2006ab). DOE described the fuel tank explosion in its responses to NRC staff RAIs (DOE, 2009ez,fa).

NRC Staff Evaluation: The NRC staff reviewed the preclusion of fuel tank explosion using the guidance in the YMRP. The NRC staff considers ANSI/ITSDF B56.9–2006 (Industrial Truck Standards Development Foundation, 2006ab) applicable to the fuel tank design of the cask tractor because this standard addresses the potential fuel tank explosion condition and requires the tow tractor to comply with UL 558 (Underwriters Laboratories, 1996aa). UL 558 (Underwriters Laboratories, 1996aa) is an industry standard that addresses fire safety aspects of diesel-fueled industrial tow tractors. The DOE design of the fuel tank of the cask tractor to preclude fuel tank explosion is reasonable because the design is consistent with the industry-accepted standard ANSI/ITSDF B56.9–2006 (Industrial Truck Standards Development Foundation, 2006ab).

<u>Reduction in Severity of a Drop</u>

DOE designed the cask tractor trailer to minimize the potential cask drop height (SAR p. 1.2.8-35) and credited the cask tractor trailer with preventing a cask drop from a height of more than 1.8 m [6 ft] (SAR Table 1.2.8-2).

NRC Staff Evaluation: The NRC staff reviewed reduction in severity of a drop using the guidance in the YMRP. The DOE cask tractor trailer design for reducing the severity of a drop is reasonable because the potential cask drop height of 1.8 m [6 ft] limits the drop height well below that for other potential drops {e.g., two-block drop height for the cask handling crane is 9.1 m [30 ft]} and DOE stated that the equipment qualification program will be used to ensure the SSCs ITS design criteria are met consistent with the safety functions [SAR Section 1.13; DOE Number 9 (2009ez)].

Cask Puncture Prevention

The cask transfer trailer precludes transportation cask puncture due to a collision or to hydraulic ram operation. DOE indicated that the transportation cask with a thick steel lid is constructed of 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. DOE further indicated that the inherent toughness of the casks provides the necessary puncture resistance. In addition, DOE included a relief valve in the hydraulic ram design to prevent actuator overpressure (DOE, 2009ez).

NRC Staff Evaluation: The NRC staff reviewed cask puncture prevention using the guidance in the YMRP. Because utilizing a pressure relief valve to limit the maximum push and pull force of the hydraulic ram is a standard practice in the industry and the cask is resistant to puncture by design, DOE's approach of limiting damage to the DPCs inside the cask is reasonable.

2.1.1.7.3.5.4 Site Prime Movers

DOE plans to deploy three types of site prime movers in the GROA whose primary function will be to move cask cars and trailers loaded with casks between the buffer area and the handling facilities. These prime movers are rubber-tired tractors, steel-wheeled locomotives, or a hybrid prime mover that consists of both rubber and steel wheels. The truck tractor pulls trailers carrying loaded truck casks, whereas the steel-wheeled, rail-based switcher locomotive moves rail cask cars. These towing functions at the repository are similar to commercially available vehicles used in commercial and military industrial facilities. These prime movers will be operator-controlled or driven and will be equipped with a 378-L [100-gal] diesel fuel tank, speed control features, and air-based braking systems with onboard air compressors.

DOE provided design information including operational processes, PSCs, design criteria and design bases and their interrelationships, design methodologies, codes and standards, and design load combinations associated with the site prime mover in SAR Sections 1.2.8.4.3. DOE provided additional design envelope information on the site prime movers (DOE, 2009ez). DOE 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 rail carrier is 3,251 mm [128 in] wide, and the railcar outside length is 27.4 m [90 ft]. The railcar will accommodate the maximum combined naval transportation cask and rail carrier weight of 3.6×10^5 kg [395 tons (789,000 lb)] with the loaded naval transportation cask alone having a weight of 2.7×10^5 kg [295 tons (590,000 lb)].

Design Bases and Design Criteria

DOE presented the nuclear safety design bases for the prime mover and their relationships to the design criteria in SAR Table 1.2.8-2. DOE identified two design criteria to address the safety design bases.

First, to protect against runaway, DOE specified that the site prime mover would include equipment to limit the speed of the site prime mover to 14.5 km/hour [9 mph] within the GROA and 4.4 km/hour [2.75 mph] while approaching handling facilities (SAR p. 1.2.8-37). Secondly, to preclude fuel tank explosion, DOE 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 DOE Number 13 (2009fa). Also, DOE identified (i) Procedural Safety Control-2 (PSC-2) to limit spurious movement of the rail-based site prime mover during handling operations with loaded waste containers by specifying that the site prime mover is to be detached prior to loaded waste containers being placed on or taken off the railcar (SAR p. 1.2.8-38) and (ii) for truck-based site prime movers, the site prime mover operating procedure will specify that the site prime mover be detached or deactivated with its brakes set prior to waste handling operations (SAR p. 1.2.8.39).

NRC Staff Evaluation: The NRC staff reviewed the design bases and design criteria using the guidance in the YMRP. DOE's design bases and design criteria are reasonable because the safety functions are relevant to the site prime mover (i.e., precluding a fuel tank explosion and maintaining a speed limit significantly below the speeds considered for transportation to the site). The site prime mover is attached to the cask transportation vehicle (railcar or truck trailer) once the cask and its contents have been accepted into the repository Cask Receipt Security Station (SAR p. 1.2.1-9).

Design Methodology

DOE stated that the site prime mover design is based on industry-accepted standards for prime movers. These standards included 49 CFR 571.121 and 49 CFR 571.108, American Association of State Highway and Transportation Officials (2004aa), and American Railway Engineering and Maintenance-of-Way Association (2007aa). DOE stated that it will apply the sections of the 49 CFR 571, which are related to the lamps, reflective devices, and associated equipment used on the vehicle, to ensure proper motion signaling. DOE will also utilize other applicable regulations, such as 49 CFR 571.106, which provides guidelines for the design of the brake hoses, and 49 CFR 571.301, which provides guidance for the design of fuel system integrity (DOE, 2009ez).

NRC Staff Evaluation: The NRC staff reviewed the design methodology using the guidance in the YMRP. DOE's design methodology for the site prime movers is reasonable because the design methodology is based on industry-accepted standards for site prime mover design used for rail-based activities and the trucking industry, and the methodology addresses the relevant safety functions (i.e., limits speed and precludes fuel tank explosion).

Design and Design Analyses

Runaway Prevention

DOE 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 4.42 km/hour [2.75 mph] when approaching the handling facilities. The speed is controlled by a governor on the engine and a transmission constraint that ensures speed limits. DOE indicated that the site prime movers and the cask conveyances are also equipped with braking systems that operate in tandem when these systems are connected. These braking systems are designed such that the brakes are automatically applied when the 14.5 km/hour [9-mph] design limit is exceeded. For rail-based cars, DOE indicated that it will use Association of American Railroads Section M (2004aa) to address the braking and speed limit control features of the rail-based site prime movers (DOE, 2009ez).

NRC Staff Evaluation: The NRC staff reviewed runaway prevention using the guidance in the YMRP. The DOE speed control approach for runaway prevention is reasonable because the use of engine governors and transmission constraints is a common practice observed across many fields and applications, the braking demands for these vehicles are significantly lower than the speeds considered for the brake design for transportation to the site on roads and rails {e.g., 97 km/h [60 mph]}, and the design is based on industry-accepted standards for the railroad industry and the trucking industry.

Protection Against Fire Explosion

DOE credited the site prime movers with the safety function of precluding fuel tank explosion. DOE 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). DOE indicated that fuel tanks that preclude explosions, which could be used if deemed necessary in the final design analysis, are commercially available from TSS International and Rodgard/Hutchinson Worldwide (DOE, 2009fa). 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 satisfy the explosion-proof requirement for the site prime movers. DOE further indicated that application of U.S. Department of Defense guidelines (2006aa) is conservative because it applies to environments more severe than those expected at the Yucca Mountain repository.

DOE stated that it will perform further analyses on the final design of the site prime movers. It will verify the performance of the fuel tank materials selected, the braking system design, and the vehicle speed control in complying with the credited safety functions.

NRC Staff Evaluation: The NRC staff reviewed protection against fire explosion using the guidance in the YMRP. DOE's design and design analysis for the protection of the site prime movers against fire explosion is reasonable because (i) DOE stated its intent to rely on the use of commercially available designs to preclude fuel tank explosions and (ii) the rail-based site prime movers do not enter the handling facilities, thus preventing any fire or explosion inside the building from the diesel fuel tank (SAR p. 1.2.8-38).

2.1.1.7.3.6 Electrical Power Systems

DOE provided design information in SAR Sections 1.4.1.2 and 1.4.1.3 for the proposed ITS Electrical Power System to be used at the GROA. The ITS electrical power system receives electric power from the non-ITS normal electrical power system and provides power to ITS systems and equipment that require electrical power to perform a safety function. If the normal electrical power system is lost, the ITS electrical power system receives electric power from the ITS diesel generators.

The ITS electrical power system consists of three subsystems: (i) ITS Alternating Current (AC) subsystem, (ii) ITS Direct Current (DC) subsystem, and (iii) ITS Uninterruptible Power Supply (UPS) subsystem. The ITS AC subsystem provides power to ITS equipment and other ITS and non-ITS systems, while the ITS DC subsystem primarily provides power for the control actions of the ITS electrical power system switchgear. The ITS UPS subsystem provides power to ITS instruments and controls that must be continuously powered to perform their safety functions. The ITS AC, DC, and UPS subsystems are described in TER Section 2.1.1.2, and the performance of these subsystems is evaluated in TER Section 2.1.1.6.

The nuclear safety design bases and design criteria for the ITS electrical power system were described in SAR Tables 1.9-3 (CRCF) and 1.9-4 (WHF). ITS electrical power system design bases and design criteria were presented in SAR Table 1.4.1-1 and described in SAR Sections 1.4.1.2.5 and 1.4.1.3.5. The design criteria of redundancy and reliability attributes of the ITS electrical power system were discussed 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 considers whether the description and discussion of the proposed ITS electrical power system design for both surface and subsurface operations reasonably describe the relationship between DOE's proposed design criteria and performance requirements for the ITS electrical power system, and the relationship between the design bases and the design criteria.

Design Bases and Design Criteria

DOE 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. DOE provided specific design criteria to address the nuclear safety design bases.

DOE provided several design criteria for the safety design bases to (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, in part, on the IEEE standards for electrical power systems for nuclear facilities as listed by DOE, apply when offsite commercial electrical power is available and during LOSP events.

To provide reliable ITS electrical power during an LOSP, the following design criteria were provided: (i) two independent ITS diesel generators must be included in the ITS electrical power system design, (ii) support systems for each ITS diesel generator must be electrically and physically independent of the support systems for the other ITS diesel generator, and (iii) fuel oil storage for each ITS diesel generator must be sized for 14 days of continuous operation and must be capable of being refilled while the ITS diesel generators are operating.

To assure reliable and continuous availability of the ITS electrical power system distribution system, DOE included design criteria such that (i) ITS electrical distribution equipment and associated raceways must be electrically independent and physically separated; (ii) upon occurrence of an LOSP, the ITS diesel generator switchgear must be isolated from the normal electrical power system; and (iii) ITS loads must be automatically sequenced onto the ITS diesel generators.

DOE identified battery-powered DC electrical power SSCs within the EDGF as subject to ITS electrical power system distribution system design criteria. The ITS electrical power system must 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.

Certain ITS SSCs require continuous AC power to support or perform a safety function when AC power becomes unavailable. The ITS electrical power system distribution system is subject to related design criteria as identified by DOE. Battery-operated UPS SSCs are required within the CRCF, WHF, and EDGF to provide continuous AC power for these SSCs.

Reasonable environmental conditions must be maintained in ITS electrical equipment and battery rooms for reliable and continuous operation. A design criterion was provided to ensure that ITS electrical equipment and battery rooms in the EDGF would be cooled. The HVAC system in the EDGF must include an ITS HVAC subsystem that provides an independent HVAC train 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, DOE indicated that the reliability for the ITS electrical power system SSCs supplying power to ITS HVAC systems in the EDGF is the same as the reliability for similar electrical power systems of the CRCF and WHF.

For each of the design criteria DOE identified, the nuclear safety design bases for the ITS electrical power system design have been determined on the basis of PCSA results. The PCSA identified specific safety functions that must be performed so that the facility safety objectives are met. From this analysis, the controlling parameters for each of these functions are identified. Finally, a set of design criteria is applied to the ITS electrical power system design to ensure the availability of safety systems. DOE selected these design criteria to ensure the ability of SSCs to perform their intended safety functions.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using the guidance in the YMRP. The NRC staff notes that design bases have been identified for the ITS electrical power system supporting the operation of ITS SSCs, and that these design bases have been derived from the site characteristics and consequence analyses which form a part of the PCSA. The design criteria DOE provided are reasonable because the criteria address both when offsite commercial electrical power is available and when loss of offsite power (LOSP) events occur, the criteria are consistent with IEEE standards for electrical power systems for nuclear facilities, and the design includes two independent diesel generators and battery power to ensure reliable and continuous availability of ITS electrical power.

Design Methodologies

DOE stated that it will design the ITS electrical power system in accordance 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 (Institute of Electrical and Electronics Engineers, 2006aa–ac; 2003aa,ab; 2001aa; 1998aa; 1997aa; 1986aa). Additionally, DOE stated that it will design the ITS AC and DC electrical power system in accordance with Regulatory Guide 1.41 (NRC, 1973ad) and, in response to an NRC staff RAI (DOE, 2009fc), following Regulatory Guide 1.9 (NRC, 2007ag).

DOE further indicated that it will design the ITS diesel generators in accordance with additional industry codes and standards, including NFPA 70 and 110 (National Fire Protection Association, 2005ab,ac) and IEEE 387–1995 and 446–1995 (Institute of Electrical and Electronics Engineers, 1996aa,ab). In addition to the codes and standards listed previously, DOE plans to design the ITS 125-V DC supply in accordance with IEEE 485–1997 and 946–2004 (Institute of Electrical and Electronics Engineers, 2005ab, 1997ab). DOE also stated that it will design the ITS UPS SSCs in accordance with ANSI/IEEE 944–1986 and IEEE 1184–1994 (Institute of Electrical and Electronics Engineers, 1995aa, 1986ab).

However, DOE's design methodology incorporates selected design criteria from the cited codes and standards. DOE applied the selected criteria contained in the cited codes and standards in conjunction with and subject to the results of the PCSA to arrive at the proposed ITS electrical power system design. In response to an NRC staff RAI (DOE, 2009dl), DOE described its application of specific criteria used in the identified industry codes and standards, the safety function of the ITS for electrical power systems, the applicability of the principal codes and standards to each ITS component, and the rationale or comment where an alternative approach to the standard was used. In response to an NRC staff RAI (DOE, 2009dl), DOE proposed alternatives (exceptions) to portions of the cited industry codes and standards affecting system reliability. DOE's proposed alternatives (exceptions) to portions of the cited industry codes and standards (and the resulting proposed design of the ITS electrical power system) were derived from the PCSA.

NRC Staff Evaluation: The NRC staff reviewed DOE's design methodology using the guidance in the YMRP to assess whether those design criteria are consistent with the PCSA results and the codes and standards to be used in the design and construction of facility ITS electrical power systems have been identified. The DOE design methodologies reflect design bases derived from the site characteristics and consequence analyses of the PCSA. The NRC staff notes that DOE has (i) reasonably described proposed design methodologies and their relationships to design criteria for the ITS electrical power system and (ii) provided information relative to the codes and standards that DOE proposes to apply in conjunction with the PCSA to the design and construction of the ITS electrical power system aspects of the GROA. The NRC staff notes that DOE-proposed alternatives (exceptions) to portions of the cited industry codes and standards used for the proposed design of the ITS electrical power system are based, in part, on assumptions and component reliability data used in the PCSA. DOE should confirm that the assumptions and component reliability data DOE used in the PCSA are consistent with the final design of the ITS electric power system and with reliabilities for the types and manufacturing specifications of ITS electrical power system equipment procured for use in the GROA.

Design and Design Analyses

DOE described the power supply specifications and proposed design criteria for power supply feeder sizing and margin requirements to be used in the design of the system distributing ITS power to ITS electrical power system loads in the EDGF, CRCF, and WHF. The significant features of the proposed ITS electrical power system design include independent, redundant, separate trains of ITS electrical power that provide power to designated independent, redundant, separate trains of ITS loads, such as ITS HVAC systems in fuel handling facilities. DOE 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 redundant ITS electrical power system design reflects an independent Train A and B configuration for the combined ITS electrical power system and ITS HVAC systems within the proposed GROA. For example, Train A ITS electrical power system can power only Train A ITS HVAC SSCs in all CRCFs, WHF, EDGF, and (non-ITS) RF. An identical relationship exists between Train B ITS electrical power system and ITS HVAC in the same facilities.

The design of ITS power distribution/connection beyond main ITS switchgear, ITS load centers, and ITS motor control centers was not described in the SAR. In response to an NRC staff RAI (DOE, 2009fc), DOE stated that specific ITS power flow and control components will be identified as part of the detailed design process and that, for the ITS electrical power system including power flow and control components for connection and control of ITS power to the end users within each facility, "unless an exception applies as a result of the PCSA, this design will include the characteristics of spatial diversity, independence, isolation, redundancy, and single-failure protection."

NRC Staff Evaluation: The NRC staff reviewed DOE's design and design analyses using the guidance in the YMRP. DOE's design and the design analyses, at the current stage of design, are reasonable because redundancy of ITS electrical power is addressed, the major functional and architectural attributes of the ITS electrical power system and major ITS electrical power system SSCs are described, and the ITS electrical power system and HVAC systems architecture is consistent with accepted engineering practice.

DOE should confirm that the final design of the ITS electric power system (including ITS electrical power SSCs beyond main ITS switchgear, ITS load centers, and ITS motor control centers) and the reliabilities for the types and manufacturing specifications of ITS electrical power system equipment, procured for use in the GROA, are consistent with the PCSA.

2.1.1.7.3.7 Instrumentation and Controls

DOE 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 discuss the proper operation of repository processes and enable facility operators to continuously monitor the status of all packaging and emplacement functions. The ITS I&C equipment is designed to sense conditions indicative of the onset of an analyzed event sequence and to initiate actions to prevent or mitigate those event sequences. The NRC staff focused on the design bases, design criteria, design methodology, and design analysis in support of this intended safety function.

Design Bases and Design Criteria

In the SAR, DOE provided general control philosophy for the GROA, stating that repetitive operations will utilize automation to support the facility operators. To facilitate this, DOE proposed using various non-ITS local digital control systems. Facility operators stationed in the Central Control Center (CCC) Facility and in the operations rooms of the various surface facilities will operate these systems using human–machine interface (HMI) consoles. The local digital control systems can be monitored and controlled through a GROA-wide Digital Control and Management Information System 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. If the CCC becomes uninhabitable, the surface facilities can continue operations using the local HMI consoles. 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 controlled take priority. Operators in the facility operations room or in the CCC can stop an activity that is locally controlled, but cannot override a command input.

DOE stated in the SAR that human actions and digital controllers are 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. When programmable logic controllers are used, their operation is constrained by the ITS controls associated with the system being controlled. ITS functions will be implemented using mechanical, electromechanical, or electrical devices with known, high reliability. Facility operators using the Digital Control and Management Information System cannot override safety functions.

DOE 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, DOE indicated that all ITS controls will consist of individual hardwired devices, instead of being driven by software or programmable devices. The use of programmable components is limited to normal operating functions, and the hardwired ITS controls will be integrated into the design of the ITS SSCs in a way that prevents the ability of other normal use, non-ITS controls from overriding any of the ITS control functions. 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. 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.

DOE proposed that the ITS I&C equipment will function as part of the ITS SSCs to accomplish ITS functions. These SSC functions are summarized at a high level in tables in SAR Section 1.9. For example, ITS I&C equipment that is 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. However, a few ITS SSCs provide interlock functions to ensure that the interactions between SSCs do not result in conditions adverse to safety.

DOE described the ITS I&C equipment functions needed to ensure the SCCs achieve the safety functions and described the design bases and design analyses 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. These SAR sections described the design concepts and the intended normal operations and safety functions for proposed ITS SSCs (i.e., ITS mechanical handling systems, ITS HVAC systems, standby, ITS emergency diesel generators and their support systems, and the GROA communications and monitoring systems). In general, these systems function within the IHF, CRCF, WHF, RF, and EDGF or on transport systems that can travel among these facilities or between this set of facilities and the subsurface emplacement facilities.

DOE presented the nuclear safety design bases for the ITS I&C equipment and their relationship with the design criteria in SAR Tables 1.9-1 through 1.9-7. DOE provided specific design criteria 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 SSCs that are implemented through the use of 29 key groups of ITS controls. SAR Table 1.9-1 presented the results of the preclosure safety classification of SSCs within the GROA, while SAR Tables 1.9-2 to 1.9-7 identified the safety functions and controlling parameters and values of the nuclear safety design bases for the ITS SSCs. Conceptual process and instrumentation diagrams, conceptual control logic diagrams, and conceptual digital control diagrams for various ITS controls were provided. The various design considerations and design criteria were described in SAR Sections 1.9.1.1 through 1.9.1.13.

The SAR did not explicitly describe where facility operating and nonoperating personnel will be when conducting operations locally. However, it was implied that personnel are not physically prohibited from being present in any area of the CRCF, WHF, IHF, RF, or EDGF while operations are being conducted. In those instances where there is a potential danger to facility personnel from inadvertent exposure to high radioactivity canisters and packages, equipment and personnel access shield doors will limit such accidental exposure, and in some instances administrative procedural controls have been identified to further limit such potential exposure. The controls for these doors have safety interlocks to prevent inadvertent opening when highly radioactive materials are present.

ITS controls have required safety functions; a single component or several components working together may perform these safety actions. These components include limit switches, load sensors, and interlocks for ASD controllers, among other devices, configured to perform safety functions needed to accomplish event sequence mitigation or prevention. Most of the ITS controls DOE identified serve in applications within the surface facilities. However, some ITS controls are incorporated within the TEV, which travels between the surface and subsurface facilities.

The nuclear safety design bases for the ITS control design have been determined on the basis of PCSA results. The PCSA identified specific safety functions that must be performed to meet facility safety objectives. From this analysis, the controlling parameters for each of these functions are identified. Finally, a set of design criteria is applied to ITS control design to ensure the availability of safety systems. These design criteria are selected to ensure the ability of SSCs to perform their intended safety functions. 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 a drop of a canister containing radioactive materials, leading to an inadvertent breach of the canister and subsequent release of these materials.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using the guidance in the YMRP. To support the evaluation of these nuclear safety design bases, the NRC staff performed a confirmatory evaluation that examined the design and the intended CRCF operations 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. Specifically, the NRC staff reviewed CRCF event sequences ESD18–DSTD–SEQ2 and ESD18–TAD–SEQ2 to evaluate ITS I&C equipment design with respect to potential interactions with material handling equipment associated with equipment shield door and cask port slide gate operations and determined they were consistent with the design bases and design criteria. The NRC staff examined ESD09–HLW–SEQ3–RF and ESD09–TAD–SEQ3–RRF in conjunction with the review of ITS controls needed to support continued ITS electrical power system operation. The NRC staff determined that the design bases and design criteria were consistent with the safety function of HVAC confinement and electrical equipment cooling capability for the CRCF following a radionuclide release. The NRC staff reviewed operational sequences, as described in the SAR, for the CTM and associated automated shielding SSCs to gain an understanding of potential challenges to the ITS I&C functions if they are needed to perform during a temporary loss of power condition. The NRC staff determined that the design bases and design are consistent with the safety functions relied on during a temporary loss of power. The NRC staff notes that the design bases and design criteria for the ITS I&C supporting the operation of ITS SSCs are reasonable because they (i) have been derived from the site characteristics and consequence analyses which form a part of the PCSA and (ii) address the event sequences that could affect the safety function of ITS I&C.

<u>Design Methodologies</u>

During both the functional (conceptual) and the detailed phases of the design, the controlling parameters identified in the PCSA and the criteria contained within the applicable industry codes and standards are required to be applied to the design. This ensures that the ITS controls will be available to perform their required actions, assuming that the conditions associated with the occurrence of event sequences are present when the safety action is needed. In SAR Sections 1.2.4, 1.4.1, and 1.13, DOE identified that certain industry design codes and standards will be applied to the ITS control design. These criteria include guidelines in ASME NOG–1–2004 (overhead crane controls) (American Society Mechanical Engineers, 2005aa); IEEE 344 (seismic qualification), 323 (environmental qualification), and 603 (criteria for safety systems—applicable to ITS electrical power system controls and to ITS HVAC controls) (Institute of Electrical and Electronics Engineers, 2005aa, 2004aa, 1998ab); Regulatory Guides 1.100 (seismic qualification) and 1.89 (environmental qualification) (NRC, 1988aa, 1984aa); and NFPA 70 (national electrical code) (National Fire Protection Association, 2005ab). The stage of the design presented in the application is conceptual. DOE identified the principal industry design codes and standards it intends to use when completing the detailed design of the ITS controls. However, DOE took several exceptions to the application of certain design criteria in nuclear safety industry codes and standards. DOE described that the basis for these exceptions was the use of a risk-informed PCSA process to identify where criteria, such as redundancy, diversity, and single-failure proof design, are appropriate.

In responses to the NRC staff RAI (DOE, 2009dl,do) related to ITS controls, DOE additionally stated that IEEE–308, 379, 384, and 603 (Institute of Electrical and Electronics Engineers, 2001aa,ab; 1998ab; 1992aa) and ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa) will be used as applicable principal codes and standards for addressing the criteria of spatial separation, independence, isolation, and redundancy used in ITS control design. DOE stated that these standards have been either adopted in whole or in part on the basis of whether they are determined applicable or appropriately adapted for use in the Yucca Mountain repository design. In these responses, DOE provided tables showing which sections of these principal codes and standards applied to ITS controls associated with SSCs and listed 31 SSCs and associated safety functions implemented through ITS controls to which these principal codes and standards will apply. Within these tables, DOE indicated that exceptions to portions of the cited codes and standards that provide added reliability (e.g., redundancy, independence, single-failure criteria) would be based on reliabilities determined using the PCSA. DOE stated that specific items which rely on the application of redundancy and/or diversity to achieve the reliability specified in the nuclear safety design bases will be further clarified as detailed design of ITS I&C and interlocks progresses.

In responses to the NRC staff RAI regarding design criteria exceptions (DOE, 2009dl,do), DOE cited the results of calculations of the likelihood of event sequence occurrence within the PCSA to justify these exceptions. Using event tree analyses as input to the event categorization process, DOE determined that the bounding probability of failure on demand of 2.75×10^{-5} [derived from multiple data sources listed in NPRD-95 (Denson, et al., 1994aa) on the basis of failure on demand performance of a type of solid-state relay] applies to a majority of ITS controls evaluated in the event sequence analyses because DOE believes it to be an industry standard data source. DOE stated that this value is representative of historical operational experience identified from military and/or industrial applications, and that it has been used to perform event sequence analyses associated with all or nearly all ITS interlocks. The SAR and supplemental materials, however, describe some ITS interlock functions that are likely to be subsystems comprising multiple components, typically including, at a minimum, sensor(s) of various types and complexity, an amplifier or signal conditioner, an interposing relay, interconnections, and in some cases an electrical power source. The results of the iterative design process influence the outcome of the classification of SSCs as ITS and associated design criteria and maintenance/surveillance planning. DOE stated that the iterative design process could result in application of additional design criteria associated with alternatives (exceptions) to portions of cited codes and standards, such as the ability of ITS interlocks to perform intended safety functions in the event a single component is not operating within specifications and the use of redundancy to enhance reliability, independence, isolation, and diversity in final designs of ITS I&C and interlock SSCs. At the current stage of ITS I&C system design, the ITS I&C component reliability data DOE used in support of the PCSA could differ from what might be applicable for the types or manufacturing specifications of ITS I&C system equipment that may be procured for use in the GROA.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review DOE's proposed methodology for determining appropriate design criteria to be applied to the design of the ITS controls design described in its SAR, as well as for implementing specific criteria of nuclear industry safety design codes and standards to be used for completing the detailed GROA design. DOE's design methodology for ITS I&C is reasonable because the design methodology was derived from the site characteristics and consequence analyses which form a part of the PCSA, and the methodology provides the criteria relevant to the safety functions (e.g., SAR Table 1.2.3-3 states equipment and personnel shield doors shall have a mean probability of inadvertent opening of less than or equal to 1×10^{-6} per transfer). The NRC staff notes DOE's proposed alternatives (exceptions) to portions of cited codes and standards for the proposed design of the ITS I&C and interlocks are based, in part, on assumptions and component reliability data used in the PCSA. DOE should confirm that the assumptions and component reliability data DOE used in the PCSA are consistent with the final design of the ITS I&C and interlocks and with reliabilities for the types and manufacturing specifications of ITS I&C and interlock equipment procured for use in the GROA.

Design and Design Analyses

DOE provided 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 NRC staff's evaluation and due to similarity in design methods used, the NRC staff broadly categorizes these 29 groups in this TER into three types of ITS control and interlock applications. Examples of each of the three types are discussed next.

Doors, Materials Handling Cranes, and WPTT

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). A mechanical outline drawing, piping and instrumentation diagrams, and a conceptual logic diagram for the port slide gate was provided in SAR Figures 1.2.4-20, 1.2.4-51, 1.2.4-58, and 1.2.4-61. 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. Both rely on prohibiting the opening of the slide gate unless the CTM shield skirt is in place (DOE, 2009dk).

The CTM (materials handling crane) transfers HLW from different types of canisters into waste packages in the IHF, CRCF, WHF, and RF. The CTM was described 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.

The four safety functions the ITS controls perform are as follows:

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 misspool" switch trips.

2. Limit drop height by preventing hoist raising/lowering without safety permissives, limit lift heights, and require a "grapple engaged" signal to allow the load to be lifted further.

3. Protect against spurious movement using the hoist holding brake that will 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.

4. Protect against inadvertent exposure of personnel to radiation by requiring 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.

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 safety function for ITS controls implemented for the WPTT is to protect against spurious movement while the CTM is lowering the canister.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review information DOE provided on the design and design analyses of I&C for doors, cranes, and the WPTT. DOE’s design and design analyses of I&C for doors, cranes, and the WPTT are reasonable because (i) DOE has clearly described the major design and functional attributes of these ITS SSCs (i.e., protection against a drop and radiation exposure), (ii) DOE’s design criteria for controls and safety interlocks governing safety actions of the specific mechanical handling equipment functions follow established safety-related design criteria in ASME NOG–1–2004, and (iii) DOE stated that exceptions to codes and standards that are identified in the final design will be justified on the basis of the PCSA.

DOE should confirm that the assumptions and component reliability data DOE used in the PCSA are consistent with the final designs of ITS I&C and ITS interlock SSCs and with the reliabilities for the types and manufacturing specifications of ITS I&C and ITS interlock equipment procured for use in the GROA.

HVAC (CRCF, WHF, and EDGF)

The CRCF HVAC systems provide temperature control, flow control, and filtration during normal CRCF operation. ITS portions of the HVAC system for the CRCF ensure reliable confinement and filtration of radiological releases from event sequences involving breach of waste containers or damaged SNF assemblies and provide appropriate environmental conditions for ITS electrical and mechanical equipment that support the filtration function. The safety functions for ITS controls for CRCF HVAC are evaluated next.

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 DOE identified is that the failure of one train (A/B) initiates 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 ITS confinement areas are normally maintained at a negative pressure relative to the atmosphere. ITS exhaust fans exhaust the air through two stages of HEPA filters before discharging it to the atmosphere. The ITS ASDs vary fan speed as required to maintain proper differential pressure relative to the atmosphere. 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.

DOE 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 will 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.

<u>Support the ITS Electrical Function by Providing Cooling</u>

SAR Figure 1.2.4-104 depicted the CRCF 1 Composite Ventilation Flow Diagram Tertiary Confinement ITS HVAC Systems, Electrical, and Battery Rooms. SAR Figures 1.2.4-105 through 1.2.4-108 depicted the CRCF 1 Confinement ITS Electrical Room and Battery Room HVAC System—Train A and B Ventilation and Instrumentation Diagrams. SAR Figures 1.2.4-109 through 1.2.4-111 depicted logic diagrams related to the ITS fan coil unit and battery room exhaust fan related to CRCF (Trains A and B).

SAR Section 1.2.4.4.1 stated redundant sets of HVAC supply and exhaust equipment serve each group of ITS electrical rooms and battery rooms (Train A and Train B).

NRC Staff Evaluation: The NRC staff reviewed DOE’s information on the design and design analyses of I&C for the CRCF ITS HVAC SSCs using the guidance in the YMRP. DOE’s design and design analyses of I&C for the CRCF ITS HVAC SSCs are reasonable because the applicable criteria for the design of safety functions are contained within the industry-accepted IEEE standards DOE cited and DOE stated that exceptions to codes and standards that are identified in the final design will be consistent with the PCSA. DOE should confirm that the assumptions and component reliability data DOE used in the PCSA are consistent with the final designs of ITS I&C and ITS interlock SSCs and with the reliabilities for the types and manufacturing specifications of ITS I&C and ITS interlock SSCs equipment, procured for use in the GROA.

<u>Diesel Generator</u>

The SAR stated that two independent ITS diesel generators (Train A and Train B) and supporting ITS mechanical systems are provided in the EDGF. The safety function of the ITS diesel generator and associated mechanical supporting system are evaluated next from an ITS control perspective.

- ITS electrical power must be provided.

SAR Section 1.4.1.2.1 and Table 1.4.1-1 described the ITS diesel generators and their associated safety functions. ITS electrical power is provided to ITS surface nuclear confinement

HVAC systems in the CRCF, WHF, and EDGF and the non-ITS HVAC and electrical power systems in the RF. Design criteria follow.

- Two independent diesel generators are required.

DOE proposed two independent, 100 percent load diesel generators. DOE responded to the NRC staff RAI (DOE, 2009dv) that the controls used in conjunction with the ITS diesel generators conform to IEEE 387–1995 and IEEE 741–1997 for circuit breaker interlocks in addition to the design codes and standards included in the response to the NRC staff RAI (DOE, 2009do). Circuit breaker electrical interlocks are provided to prevent automatic closing of an ITS diesel generator circuit breaker to an energized or faulted bus. DOE also described (SAR Section 1.4.1.2.1) a solid state type undervoltage device for sensing a loss of normal 13.8 kv power into the EDGF feeder bus. DOE further stated that logic is provided so the incoming breaker can trip on undervoltage logic signals.

- Each ITS diesel generator design has ITS support systems, including related ITS I&C and ITS interlock equipment, that are electrically and physically independent from the support systems for the other ITS diesel generator. Each ITS diesel generator fuel oil storage tank is to be sized for 14 days of continuous operation and be capable of online refueling.

NRC Staff Evaluation: The NRC staff reviewed DOE's information on the design and design analyses of the EDGF ITS I&C and interlock systems using the guidance in the YMRP. DOE's design and design analyses of the EDGF ITS I&C and interlock systems are reasonable because the applicable criteria for the design of safety functions are contained within the industry-accepted IEEE standards DOE cited; DOE stated that exceptions to codes and standards that are identified in the final design will be consistent with the PCSA. DOE should confirm that the assumptions and component reliability data DOE used in the PCSA are consistent with the final designs of ITS I&C and ITS interlock SSCs and with the reliabilities for the types and manufacturing specifications of ITS I&C and ITS interlock SSCs equipment procured for use in the GROA.

2.1.1.7.3.8 Fire Protection Systems

DOE provided design information for the ITS fire protection systems used at the GROA in SAR Section 1.4.3.2.1 and Table 1.9-1. In addition, DOE provided information on the design bases and design criteria for the fire protection systems in the waste handling buildings in SAR Table 1.4.3-2. DOE identified several double-interlock preaction (DIPA) sprinkler systems as ITS. These ITS DIPA systems are used in moderator-controlled areas of the CRCF and WHF to reduce the likelihood of inadvertent water discharge, where the breach of a loaded canister and water intrusion may lead to a criticality event (DOE, 2008ab). The NRC staff focused on the design bases, design criteria, design methodology, and design analysis in support of this intended safety function.

The NRC staff's review on the description of the fire protection systems is provided in TER Section 2.1.1.2, and the ability of the fire protection system to perform its intended safety functions is reviewed in TER Section 2.1.1.6.

Design Bases and Design Criteria

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. An ITS DIPA sprinkler system achieves this goal by requiring independent heat detection (individual sprinkler head response) and supplemental fire detection (e.g., smoke, flame, or other form of fire detection) before the interlocks are achieved and water is delivered. The DIPA sprinkler system is a variation of a traditional wet pipe system and commonly used in spaces where the inadvertent introduction of water is undesirable.

DOE provided the nuclear safety design bases and design criteria for the ITS DIPA sprinkler systems in SAR Table 1.4.3-2. The nuclear safety design bases are specified as the mean probability of inadvertent introduction of fire suppression water into a canister. This mean probability must be less than or equal to 1×10^{-6} over a 720-hour period following a radionuclide release in the CRCF and less than 6×10^{-7} over a 720-hour period following a radionuclide release in the WHF.

DOE indicated in its response to an NRC staff RAI, as outlined in DOE Enclosure 1 (2009fg), that preventing boron pool dilution resulting from a spurious sprinkler system activation in the WHF was not considered a nuclear design basis. DOE stated that the boron content of the pool in the WHF is still sufficient to control criticality, even if suppression system water were to hypothetically drain into the pool.

SAR Section 1.4.3.2.1.2 and Table 1.4.3-2 indicated that the ITS DIPA sprinkler system will be designed to meet NFPA 13 (National Fire Protection Association, 2007ab) and NFPA 72 (National Fire Protection Association, 2006aa) requirements.

NRC Staff Evaluation: The NRC staff reviewed the design bases and design criteria using the guidance in the YMRP. The safety function of the ITS DIPA system is to prevent the inadvertent introduction of water into a breached canister. There are no other ITS functions designated for this system (e.g., no ITS fire detection, alarm, or suppression functions).

DOE’s design bases and design criteria for the ITS DIPA systems are reasonable to prevent inadvertent introduction of water into a breached canister because the ITS DIPA sprinkler system includes independent heat detection and supplemental fire detection and the design codes and standards used to develop the ITS DIPA sprinkler systems are consistent with the standard industry practice and follow Regulatory Guide 1.189 (NRC, 2009ac).

Design Methodologies

DOE indicated in SAR Section 1.4.3.2.1.2 and Table 1.4.3-2 that the ITS DIPA system design will be performed in accordance with nationally recognized sprinkler and fire alarm codes NFPA 13 and NFPA 72 (National Fire Protection Association, 2007ab, 2006aa), respectively.

NRC Staff Evaluation: The NRC staff reviewed the ITS fire protection systems using the guidance in the YMRP and notes that DOE used nationally recognized design codes and standards for the ITS DIPA system, which represents an appropriate design methodology.

Design and Design Analysis

DOE provided ITS DIPA design information in SAR Sections 1.4.3.2.1 and 1.4.3.2.1.2 and Table 1.4.3-2. In addition, DOE provided responses to the NRC staff RAI in DOE Enclosures 1 and 2 (2009fg) to supplement the design information presented in the SAR.

DOE noted that DIPA system equipment will be specifically listed and labeled for its suitability for fire protection service. These components and systems will be tested for reliability and suitability prior to use in fire protection systems and will provide the high component reliability expected in the PCSA.

NRC Staff Evaluation: The NRC staff reviewed the design and design analysis using the guidance in the YMRP. DOE's design of the ITS DIPA sprinkler system is reasonable because the design bases provided in SAR Table 1.4.3-2 are achievable using standard components, the design is consistent with nationally recognized design standards such as NFPA 13 and NFPA 72 (National Fire Protection Association, 2007ab, 2006aa), and the DIPA system equipment are specifically listed and labeled for suitability for fire protection service.

2.1.1.7.3.9 Canisters and Overpacks

DOE provided design information for ITS overpacks and canisters used at the GROA. The ITS overpacks and canisters reviewed in this section are categorized as (i) waste package, (ii) TAD canister, and (iii) other canisters, overpacks, and casks. The NRC staff's review focused on the design bases and design criteria, design methodology, and design and design analysis. The description of these ITS canisters and overpacks is evaluated in TER Section 2.1.1.2, and their performance is evaluated in TER Section 2.1.1.4.

2.1.1.7.3.9.1 Waste Package

DOE 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 ITS waste package is an engineered barrier for disposal of CSNF, HLW, and DOE and naval SNF. These ITS waste packages are also important to waste isolation. The waste packages are designed to accommodate six different loading configurations, depending on the waste form. 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 TER Section 2.1.1.2.3.5.1 for more information on waste package configurations). 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, different waste package configurations have multiple internal structures and different external dimensions to accommodate various waste forms.

Design Bases and Design Criteria

DOE presented the nuclear safety design bases for the waste packages and their relationship with the design criteria in SAR Table 1.5.2-6. DOE also provided the specific design criteria for each of the design bases, along with controlling parameters and bounding values.

DOE 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 the spectrum of fire and (ii) protect against breach of the waste package from a rock or vibratory ground motion impacts. These design criteria, based on 2001 ASME Boiler and Pressure Vessel Code, Sections II and III (American Society of Mechanical Engineers, 2001aa), are ultimate tensile stress limits and limits based on material energy absorption capabilities. Also, the controlling parameters imposed on each design basis are a mean conditional probability or a mean frequency of breach of the waste package.

In addition, for the waste packages, the design criteria to address design bases for the important to waste isolation include (i) a minimum thickness of 25 mm [1 in] for the outer corrosion barrier for codisposal, TAD-bearing, and naval waste packages; (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, including effects of a high temperature and fuel rod gas release.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design bases and design criteria for the waste package and its components using the guidance in the YMRP. DOE's design criteria and design bases for the waste package and its components are reasonable because they are derived from the specific site characteristics and consequence analyses, and the design criteria reasonably consider PCSA results. DOE's design bases and design criteria are reasonable because the design bases and design criteria address the relevant events (e.g., 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 the design criteria rely on the industry-accepted 2001 ASME Boiler and Pressure Vessel Code, Sections II and III.

<u>Design Methodologies</u>

The waste package is designed on the basis of the structural and thermal design methodologies presented in the waste package component design methodology report (BSC, 2007bi): (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 analytical methods and computational tools, and (vi) demonstrating that design requirements are satisfied. DOE presented the structural design methodologies, including the analyses performed for various load combinations (normal loads and event sequences loads), and DOE's acceptance criteria in SAR Tables 1.5.2-8 and 1.5.2-9. The thermal design methodologies for the waste package were addressed by performing parametric studies to analyze waste package response to accidental fires.

DOE performed a finite element analysis 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. DOE stated that, for the normal loads, its acceptance criterion was for generated stresses to remain in the elastic range and below the threshold for stress corrosion cracking of Alloy 22. The threshold values for stress corrosion cracking of the waste package outer corrosion barrier are discussed and evaluated in TER Section 2.2.1.3.1.3.3, where the NRC staff determined that DOE's methodologies to establish the value of the threshold are reasonable because they would not overestimate the value of this parameter.

DOE used elastic-plastic finite element analyses and analytical methods to evaluate waste package performance for event sequence loads. DOE calculated 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) the waste package while horizontal inside the WPTT on the waste package transfer carriage subjected to the dynamic loads imposed by vibratory ground motion, (v) the waste package subjected to loads produced by general drift collapse in the lithophysal portions of the repository caused by vibratory ground motion, and (vi) the waste package subjected to loads produced by rockfall in the nonlithophysal portions of the repository. DOE stated that, for event sequence loads, the DOE acceptance criteria allowed stresses to be generated beyond elastic range and invoked the tiered screening criteria method (SAR Table 1.5.2-10). The tiered screening criteria method is a deterministic approach based on elastic-plastic analysis methods provided in ASME 2001, Section III, Appendix F (American Society of Mechanical Engineers, 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.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding structural and thermal design methodologies for the waste package and its components using guidance in the YMRP. The NRC staff notes that the proposed design methodologies are supported by reasonable technical bases because they are consistent with established industry practice and applicable codes and standards. Therefore, DOE's proposed design methodologies are reasonable.

<u>Design and Design Analyses</u>

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 waste package inner vessel is designed in accordance with 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (American Society of Mechanical Engineers, 2001aa) for Class 2 components, will be stamped with an N symbol, and therefore will be identified as a pressure vessel. The outer corrosion barrier is designed with applicable technical requirements of 2001 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (American Society of Mechanical Engineers, 2001aa) for Class 2 components; however, it will not be stamped with an N symbol and therefore will not be identified as a pressure vessel.

The materials, design, fabrication, testing, and examination of the waste package will meet the requirements of the following codes and standards in American Society of Mechanical Engineers (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”

DOE 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, DOE described the methods for the closure welds in SAR Section 1.2.4.2.3.

The waste package inner vessel is made of ASME SA–240 (UNS S31600) with additional controls on nitrogen and carbon, referred to as Stainless Steel 316. The waste package outer corrosion barrier is made of ASME SB–575 (UNS N06022) with limited constituents of chromium, molybdenum, tungsten, and iron, referred to as Alloy 22 (BSC, 2007bi). The inner vessel is designed as a load-bearing component (i.e., a pressure vessel) for internal pressure and deadweight loads. The outer corrosion barrier is included as a corrosion-resistant component to address postclosure requirements and is not a pressure vessel (BSC, 2007bi).

Using these design methods, DOE analyzed three waste package configurations (TAD canister, DOE short codisposal canister, and naval long canister). In response to an NRC staff RAI (DOE, 2009er), DOE stated that the remaining three waste package configurations (5-DHLW/DOE long codisposal, 2-MCO/2-DHLW codisposal, and naval canistered SNF short waste package) will be subject to similar analyses and the final qualifications of the waste package design will be included in proposed administration controls.

In SAR Section 1.5.2.7.3, DOE stated it will perform a nondestructive examination, in accordance with the 2001 ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NC–5000 (American Society of Mechanical Engineers, 2001aa), on all major fabrication welds of the waste package after final machining, surfacing, and heat treatment except liquid dye penetrant testing. Radiographic examination, liquid dye penetrant testing, and ultrasonic examination would be used to examine the outer corrosion barrier longitudinal weld, circumferential weld, bottom lid weld, and upper sleeve to outer corrosion barrier weld. The liquid dye penetrant testing method would only be 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. Radiographic examination and liquid dye penetrant testing methods would be used to examine the inner vessel longitudinal weld, inner vessel circumferential weld, and inner vessel bottom lid weld. The outer closure welds would be inspected using visual, eddy current, and ultrasonic inspection techniques.

DOE provided representative samples of structural and thermal finite element analyses of the waste package performance under normal and event sequence load combinations (DOE, 2009er). These include the following analyses: (i) naval canistered SNF long waste package oblique impact inside the TEV (BSC, 2007cn), (ii) nonlithophysal rockfall on waste packages (BSC, 2007co), (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), (v) waste package capability analysis for nonlithophysal rock impacts (BSC, 2007cr), and (vi) thermal responses of TAD and 5-DHLW/DOE SNF waste packages to a hypothetical fire accident (BSC, 2007cs).

NRC Staff Evaluation: The NRC staff reviewed DOE's information regarding codes and standards for the materials, design, fabrication, testing, and examination of the waste package and its components using the guidance in the YMRP. The NRC staff notes that the cited codes and standards conform to standard engineering practice and accepted industry technology. While DOE used the 2001 ASME Boiler and Pressure Vessel Code rather than the 2003 ASME Boiler and Pressure Vessel Code, the NRC staff determined that the 2001 version is reasonable because DOE reasonably demonstrated that the changes applied to the ASME 2003 Code version would not affect the code requirements imposed onto the waste package design.

The NRC staff also notes 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 reasonable because these materials are consistent with the design methodologies used, are consistent with standard engineering practice, and are based on accepted industry technology.

The NRC staff also reviewed the information provided regarding proposed fabrication materials, fabrication processes, and closure methods for the waste package. The processes proposed for the waste package fabrication, assembly, and closure are consistent with the design methodologies used, are in compliance with applicable sections of 2001 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa), and are based on accepted industry technology.

The NRC staff reviewed the information provided regarding proposed nondestructive examination methods for fabricated waste packages to detect and evaluate fabrication and any other defects. DOE's proposed nondestructive examination methods are consistent with the design methodologies used, are based on accepted industry technology, and are consistent with applicable sections of 2001 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa).

The NRC staff reviewed representative samples of structural and thermal finite element analyses to evaluate the waste package performance under normal and event sequence load combinations. DOE only evaluated waste package configurations for the 21–PWR/44–BWR TAD canister bearing, 5-DHLW/DOE short codisposal, and naval canistered SNF long waste packages. For these analyses, the calculated stresses in the waste package outer corrosion barrier satisfied the DOE tiered screening criteria used to evaluate material failure for mechanical loading (SAR Table 1.5.2-10) and the calculated temperature inside of the waste package stayed below the temperature limit for accidental conditions. The NRC staff notes that (i) the design analyses are reasonable and conform to established practices for mechanical/structural performance assessment using finite element methods (Bathe, 1996aa), (ii) the waste package components are designed to sustain loads from normal operations and Category 1 and 2 event sequences, and (iii) the waste package thermal controls are such that the fuel cladding temperature will be sufficiently low to prevent cladding failure.

2.1.1.7.3.9.2 Transportation, Aging, and Disposal Canister

DOE plans to use the TAD canister to dispose of CSNF. 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. The TAD canister will be used in surface facilities including the CRCF, RF, and WHF, and in the subsurface facility where it will be inside a waste package.

DOE provided the design features of the TAD canister in SAR Section 1.5.1.1.1 and a mechanical envelope diagram in SAR Figure 1.5.1-5. DOE 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 Bases and Design Criteria

DOE identified the TAD canister as ITS, because it is relied upon in the PCSA to prevent or mitigate the consequences of an event sequence (SAR Section 1.9). DOE also identified the TAD canister as important to waste isolation, because it prevents or substantially reduces the release rate and rate of movement of radionuclides to the accessible environment (SAR Section 1.9.2). In SAR Table 1.5.1-7 DOE provided the nuclear safety design bases for the TAD canister and their relationship to TAD canister structural characteristics. Specifically, the TAD canister is required to provide containment to radioactive materials when subject 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 is also required to provide containment when subject to thermal challenges over a spectrum of fires while contained within a cask, waste package, aging overpack, or the CTM shield bell.

In the SAR Section 1.5.1.1.1.2.5, DOE identified the TAD canister design criteria and design bases. DOE stated that the TAD canister will provide moderator control to ensure subcriticality during all possible event sequences for handling operations that are important to criticality. In addition, DOE stated that the TAD canister shall have thermal characteristics such that the cladding temperature is not to 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. The NRC staff's evaluation pertaining to criticality and cladding temperature limits is discussed in the subsection to follow on design and design analyses.

NRC Staff Evaluation: The NRC staff has reviewed the relationship between the design bases and design criteria of the TAD canister using the guidance in the YMRP. The NRC staff notes that the design criteria and design bases DOE used are reasonably derived from the PCSA results and are consistent with the canister's intended safety function to provide containment from structural or thermal challenges at the site. Additionally, the thermal design criteria and bases are consistent with regulatory guidance of NUREG–1536 (NRC, 1997ae). Therefore, the information DOE provided on the design bases and design criteria is reasonable and provided a clear relationship between design bases and design criteria.

Design Methodologies

In SAR Section 1.5.1.1.1.2.6, DOE presented performance specifications and methodologies that should be used to design the TAD canister such that it will meet the performance specifications. DOE focused on two parameters: leakage rate and fuel cladding temperature. DOE specified that the TAD canister maintain a normal condition maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second for 2,000- and 10,000-year return periods for seismic events. During these events, the TAD canister will be 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 must also maintain the maximum off-normal condition leakage rate of 9.3×10^{-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). DOE specified 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). Similarly, the TAD canister, while contained in an aging overpack subjected to impact from an aircraft crash event (i.e., a F-15 military aircraft) must maintain a maximum leakage rate of 9.3×10^{-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 Evaluation: The NRC staff reviewed the information provided regarding design specifications and methodology for the TAD canister using the guidance in the YMRP. DOE's design methodologies used for the TAD canister are reasonable because a range of normal and off-normal conditions, including aircraft crash, were considered for the effect on leakage rates and cladding temperature. In particular, the normal and off-normal cladding temperature limits are consistent with the guidelines of NUREG–1536 (NRC, 1997ae), which specifies cladding temperature limits of 400 °C [752 °F] (normal) and 570 °C [1,058 °F] (off-normal).

Design and Design Analyses

The information presented in SAR Section 1.5.1.1.1.2.6.1.2 regarding the TAD canister fuel handling processes, which includes drying and inerting of the canister, specifies that NUREG–1536 (NRC, 1997ae) will be used as the basis for water draining and drying procedures. Further, helium will be used to inert the TAD canister to prevent oxidation of the spent fuel cladding.

DOE specified in SAR Section 1.5.1.1.1.2.6.1.2 that the fabrication of the TAD canister will be based on the 2004 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (American Society of Mechanical Engineers, 2004aa). In addition, the TAD canister (except thermal shunts and criticality control materials) is specified to be fabricated using 300-series stainless steel for the canister shell and structural internals, as per ASTM A276–06 (ASTM International, 2006ab). With respect to criticality, SAR Section 1.5.1.1.1.2.2.2 described the characteristics and materials of the neutron absorbers. In SAR Section 1.14.2.3.1.3, DOE listed the TAD canister components that are designed to prevent and control criticality. Of primary importance is the shell of the canister, which prevents a moderator from being introduced into the SNF.

The TAD canister containment characteristics were described in SAR Section 1.5.1.1.1.2.6.1.2. The TAD canister must sustain a 0.3-m [1-ft] vertical flat bottom drop such that a specified maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second is maintained while the cladding temperature remains below 400 °C [752 °F]. DOE specified that the TAD canister closure welds must conform to the requirements set forth in SFPO–ISG–18 (NRC, 2008ae); alternatively, closure weld helium leak testing must conform to the testing procedures in ANSI N14.5–97 (American National Standards Institute, 1998aa). SFPO–ISG–18 (NRC, 2008ae) was developed to address the qualification of final closure welds on austenitic stainless steel canisters. DOE did not provide any information specific to the type of welding procedure to be used or the type of nondestructive evaluation of the welds other than stating that the guidance of SFPO–ISG–18 (NRC, 2008ae) will be followed. However, SFPO–ISG–18 (NRC, 2008ae) 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.

DOE established in SAR Section 1.5.1.1.1.2.7 that the material for the TAD canister and structural internals shall be constructed of a 300-series stainless steel as per ASTM A276–06 (ASTM International, 2006ab) and will be designed to be compatible with either borated or unborated water environments as defined in DOE Table 3.1-4 (2008ag). DOE stated that the selection of stainless steel is based on the assumption that the canister degradation from corrosion must have minimal impact on the pH of the aqueous solution(s) that would contact the TAD canister and waste form after TAD canister breach (DOE, 2007ac). In addition, DOE identified a list of prohibited or restricted materials that cannot be used to construct the TAD canister, including the cleanliness specifications that shall be followed as defined in ASME NQA–1–2000, Subpart 2.1, Classification C (American Society of Mechanical Engineers, 2000aa).

DOE 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 requirements of the following codes and standards:

- ANSI N14.5–97, American Standard for Radioactive Materials—Leakage Test on Packages for Shipment (American National Standards Institute, 1998aa)

- 2004 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2004aa)

- ASCE 7–98, Minimum Design Loads for Buildings and Other Structures (American Society of Civil Engineers, 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 (American Society of Civil Engineers, 2003aa)

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design and design analysis for the TAD canister using the guidance in the YMRP. The NRC staff also reviewed the applicability of the guidance provided by SFPO–ISG–18 (NRC, 2008ae) for the TAD canister confinement characteristics and notes that it is reasonable for repository applications with respect to the existence of welding flaws of sufficient size that could impair the weld structural strength or confinement capability because welding flaws of this nature are specifically addressed in SFPO–ISG–18.

The NRC staff reviewed the characteristics and design specifications of the criticality-significant components of the TAD canister. The NRC staff notes that the canister shell reasonably prevents and controls criticality by preventing a moderator from entering the canister. In addition, the fixed neutron absorbers specifications for controlling criticality are consistent with standard engineering practices. The information DOE provided is consistent with the criticality standards in Regulatory Guide 3.71 (NRC, 2005ac).

The NRC staff reviewed the specification DOE identified for drying and inerting of the TAD canister. The NRC staff notes that this specification follows standard industry practices used in commercial canisters and is therefore reasonable.

The NRC staff reviewed the material specifications and restrictions for TAD canister construction and notes that the specifications and restrictions, including standards used, are consistent with standard engineering practices and are therefore reasonable.

The NRC staff also reviewed the design codes and standards to be used for the TAD canister design and construction. The NRC staff notes that the design codes and standards are reasonable because they are in conformance with standard engineering practices.

2.1.1.7.3.9.3 Other Canisters, Overpacks, and Casks

The NRC staff organized its review and evaluation into the following topics: (i) DOE's standardized canisters for SNF, (ii) HLW canisters, (iii) DPCs, (iv) naval canisters for U.S. Navy SNF; (v) aging overpack, and (vi) transportation cask.

2.1.1.7.3.9.3.1 U.S. Department of Energy Standardized Canister

DOE provided the design information for the DOE standardized canister in SAR Section 1.5.1.3.1.2.1.1 and a mechanical envelope diagram of a small-diameter standardized canister in SAR Figure 1.5.1-9. 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 or 4.6 m [10 or 15 ft] long. These standardized canisters are fabricated from Stainless Steel Type 316L. The large diameter canister weighs between 4,077 kg (3.1-m length) and 4,536 kg (4.6-m length) [9,000 lb (10-ft length) and 10,000 lb (15-ft length)]. The weight of the small diameter canister is between 2,265 kg (3.1-m length) and 2,722 kg (4.6-m length) [5,000 lb (10-ft length) and 6,000 lb (15-ft length)].

<u>Design Bases and Design Criteria</u>

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 DOE 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.

On the basis of the PCSA, the canister must provide 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, drop of one DOE standardized canister onto another DOE standardized canister, and low-speed collisions with structures during transport. For all of these events, the design criterion is given in terms of the maximum effective plastic strain that results from a structural challenge, and it must be determined whether the maximum effective plastic strain meets the required reliability when compared to the DOE standardized canister capacity curve.

The canisters must also provide containment when they are subjected to thermal challenges (i.e., a spectrum of fires). The inclusion of the fire in the design bases is derived from the PCSA results. Fire was identified as a possible internal initiating event (SAR Section 1.6.3) that may 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 canisters must be able to withstand the thermal challenges while contained within an overpack or a cask. In addition, the DOE standardized canister also must be able to withstand a spectrum of fires while placed on a staging rack.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria of the DOE standardized canister using the guidance in the YMRP. The NRC staff notes, on the basis of standard engineering practice, that maximum effective plastic strain is a reasonable structural failure criterion for evaluating whether loss of containment or breach of a canister has occurred. The design criteria and design bases DOE used are appropriately derived from the PCSA results and are consistent with the canister's intended safety function.

Design Methodologies

In SAR Section 1.5.1.3.1.2.6.1, DOE provided the overall methodology used for DOE standardized canister design. DOE stated that the structural integrity of the DOE standardized canister will be relied on to maintain containment for accidental events, such as drops and low-speed collisions during waste handling operations. Although the DOE standardized canisters are designed in accordance with 1998 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998aa), DOE stated that, because the code does not specifically address drop conditions, alternative methods such as drop tests and finite element analyses are used to evaluate the structural behavior of the canister when subject to a drop.

The NRC staff has reviewed SAR Section 1.5.1.3.1.2.6.1, which references 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 Sandia National Laboratory for the relevant structural challenges, as identified in SAR Table 1.5.1-26. DOE used these full-scale test results to validate the 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 actual full-scale tests for three drop events of a 457-mm [18-in] diameter standardized canister.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design specifications and methodology for the DOE standardized canister using the guidance in the YMRP. The design methodology used for the DOE standardized canister is reasonable because the canister design is consistent with the industry-accepted 1998 ASME Boiler and Pressure Vessel Code and the behavior of the canister when subject to a drop was based on full-scale tests and finite element analysis. Further, the NRC staff determined that the deformation patterns the DOE finite element analyses predicted are qualitatively consistent with those of the tested canisters, and therefore the DOE finite element methodology is reasonable for estimating the canister deformation behavior.

Design and Design Analyses

Dimensions of the small diameter DOE standardized canister were shown in SAR Figure 1.5.1-9. Stainless Steel Type 316L, ASME SA–312 (UNS S31603) is specified for the canister shell. The DOE standardized canister design has a design feature that is a skirt along the circumferential edge on each end of the canister. DOE stated that this feature is important because it can absorb energy when subjected to an end drop. Dished heads are located at each end of the canister and are to be fabricated from Stainless Steel Type 316L, ASME SA–240 (UNS S31600) that has been annealed. DOE stated that stainless steel was selected due to its resistance to degradation (e.g., chemical and galvanic).

In SAR Section 1.5.1.3.1.2.8.1 DOE stated that the following code requirements in American Society of Mechanical Engineers (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 DOE presented information on the peak equivalent plastic strains occurring within the containment boundary for specific drop scenarios for the standardized 457- and 610-mm [18- and 24-in] diameter canisters. For the DOE standardized canisters, DOE used a through-wall strain limit of 48 percent as the failure criteria. DOE showed 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 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 all drop events. On the basis of the finite element analysis results, DOE concluded that the containment boundary for the 457- and 610-mm [18- and 24-in]-diameter DOE standardized canisters remains intact for the drop events.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design and design analysis for the DOE standardized canister using the guidance in the YMRP. The choice of stainless steel for the DOE standardized canister is reasonable because it provides resistance to environmental degradation and its high ductility is necessary for the DOE standardized canister to withstand the demand imposed when subjected to a structural challenge. The NRC staff also notes that the fabrication methodology based upon the ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1997ab) follows standard industry practice and is reasonable.

The NRC staff also reviewed DOE's approach for evaluating the DOE canisters' capacity to withstand possible repository structural challenges. On the basis of the results of the full-scale tests and the corresponding finite element analyses, the NRC staff notes that the DOE canisters will perform their intended safety functions when they are subjected to the drop events discussed previously.

2.1.1.7.3.9.3.2 High-Level Radiological Waste Canisters

The proposed repository will receive HLW from four sources: (i) the Hanford Waste Treatment and Immobilization Plant, (ii) the Defense Waste Processing Facility at the Savannah River Site, (iii) the Idaho National Laboratory, and (iv) the West Valley Demonstration Project. The HLW canisters 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 structural design data for the HLW canisters. 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]. 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]. 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]. All canisters are fabricated from Stainless Steel Type 304L (UNS S30400). SAR Section 1.5.1.2.5 provided the design criteria and design bases. The HLW canisters are filled with a molten mixture of HLW and other constituents (e.g., silica sand), which are poured into the HLW canisters, and the canister is sealed once the waste solidifies.

Design Bases and Design Criteria

The safety function of the HLW canister is to provide containment of radioactive materials. On the basis of the PCSA, the HLW canister was classified as ITS. In SAR Section 1.5.1.2.1.5 DOE 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.

On the basis of the PCSA results, the canister must provide containment when it is subjected to structural and thermal challenges. DOE considered the potential structural challenges for the canister design, 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, a drop of a DOE standardized canister onto a HLW canister, and low-speed collisions with structures during transport. For all of these events, the design criterion is given in terms of the maximum effective plastic strain that results from a structural challenge, and it must be determined whether the maximum effective plastic strain meets the required reliability when compared with the capacity curve of the HLW canister.

The HLW canister may also be subjected to thermal challenges (i.e., a spectrum of fires). The inclusion of the fire in the design bases is derived from the PCSA results. The HLW canister must be able to withstand the thermal challenges while contained within an overpack or a cask, within a cask or waste package, or within the CTM shield bell.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria of the HLW canister using guidance in the YMRP. The NRC staff notes, on the basis of standard engineering practice, that maximum effective plastic strain is a reasonable structural failure criterion for evaluating whether loss of containment or breach of canister has occurred.

The design criteria and design bases DOE used are reasonably derived from the PCSA results and are consistent with the canister’s intended safety function.

Design Methodologies

DOE referred to SAR Section 1.7.2.3.1 for the HLW canister design methodology and the analysis basis for loss of containment. In SAR Section 1.7.2.3.1 DOE 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. DOE stated that for all tests, the HLW canister did not breach.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design methodology for the HLW canister using the guidance in the YMRP. The NRC staff notes that the results of the actual full-scale tests showing no breach of the HLW canister demonstrate that the design is reasonable and, therefore, the design methodology used is also reasonable.

Design and Design Analyses

In SAR Table 1.5.1-16, DOE provided geometric details of the four HLW canisters from Hanford, Idaho National Laboratory, Savannah River, and West Valley. The HLW canisters will have a length of 300–450 cm [118–177 in], a diameter of 61 cm [24 in], and a shell thickness ranging from 0.34–0.95 cm [0.13–0.37 in]. The four HLW canisters will be constructed of an austenitic stainless steel. The HLW canisters will be designed to the design codes and standards listed in SAR Table 1.5.1-18. The canister welding and nondestructive weld evaluation will be performed under the guidance of the 2001 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa). DOE stated that the canister welding procedures will follow the industry-accepted standards set forth by the 2001 ASME Boiler and Pressure Vessel Code, Section IX (American Society of Mechanical Engineers 2001aa). All full penetration butt welds from the Hanford, Idaho National Laboratory, and Savannah River Site canisters will have a nondestructive evaluation radiographic examination per 2001 ASME Boiler and Pressure Vessel Code, Section V (American Society of Mechanical Engineers 2001aa). The West Valley canister will use dye penetration of all fabrication welds as per 2001 ASME Boiler and Pressure Vessel Code, Section V (American Society of Mechanical Engineers, 2001aa).

NRC Staff Evaluation: The NRC staff reviewed DOE's information on design of HLW canisters using the guidance in the YMRP. The HLW canister fabrication design is reasonable because the canister welding, welding procedures, and nondestructive weld evaluations are consistent with the industry accepted 2001 ASME Boiler and Pressure Vessel Code.

2.1.1.7.3.9.3.3 Dual-Purpose Canister

DOE designed the dual-purpose canister (DPC) to store CSNF at the utility sites and to ship the SNF. DOE has not made a decision whether to ship SNF in DPCs from the utility sites or repackage SNF into TADs prior to shipping from the utility sites. DOE stated that the DPC can be placed within a properly designed overpack for aging; however, DOE also stated that the current DPC design has not been shown to be suitable for disposal. Therefore, SNF in DPCs would need to be repackaged into a TAD canister for disposal in a waste package. In SAR Section 1.5.1.1.1.2.1.2, DOE briefly discussed the DPC.

Design Bases and Design Criteria

The DPC safety function provides containment of radioactive materials. On the basis of the PCSA, the canister was classified as ITS. SAR Table 1.5.1-9 presented the nuclear safety design bases for the DPC.

On the basis of the PCSA, DOE considered the following potential structural challenges for the DPC design: drop of the canister or a load onto the canister, side impact or collision, and low-speed collisions with structures during transport. The design criterion is given in terms of the maximum effective plastic strain that results from a structural challenge and whether the maximum effective plastic strain meets the required reliability when compared to the canister's capacity curve. The DPC must be able to withstand the thermal challenges while contained within an overpack or a cask and within the CTM shield bell.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria of the DPC using guidance in the YMRP. The NRC staff notes, based on standard engineering practice, that maximum effective plastic strain is a reasonable structural failure criterion for evaluating whether loss of containment has occurred. The design criteria and design bases DOE used are derived from the PCSA results and are consistent with the canister's intended safety function.

Design Methodology

In SAR Section 1.5.1.1.1.2.1.2, DOE stated that analyses (e.g., structural, thermal, criticality) will be required to show compliance with the PCSA design bases before any DPC system (along with the necessary overpacks) is used at the repository. However, DOE stated that similar structural analyses have been performed on generic canisters (BSC, 2008cp). DOE derived generic canister geometrical and material properties on the basis of typical DPC and naval canisters. These structural analyses focused on various canister drop scenarios at differing drop heights and orientations. DOE used the results in quantifying an estimate of the passive reliability for a generic canister. DOE stated that the finite-element analyses used to model structural challenges to representative containers within a class of containers encompasses TAD canisters, naval SNF canisters, and a variety of DPCs (SAR p. 1.7-28).

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design methodology for the DPC using the guidance in the YMRP. The DOE design methodology used for the DPC is reasonable because the structural analyses DOE performed for a generic canister were based on geometrical and material properties of typical DPC and naval canisters. Also, DOE's approach for evaluating the generic canister capacity to withstand possible repository structural challenges is consistent with the approach used for the Transportation, Aging, and Disposal (TAD) canister, which the NRC staff determined was reasonable in TER Section 2.1.1.7.3.9.2.

Design and Design Analyses

Currently, DPC systems are licensed for storage at utility sites under 10 CFR Part 72 and for transportation under 10 CFR Part 71. DOE stated in SAR Section 1.5.1.1.1.2.1.2 that if the selected DPC falls within the design envelope of the generic canister, then the structural analyses based on the generic canister (BSC, 2008cp) will be used to evaluate the structural performance of the DPC.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design and design analyses for the DPC using the guidance in the YMRP. The NRC staff notes that the selected design approach is reasonable because the approach is consistent with the design approach used for the TAD, which the NRC determined was reasonable in TER Section 2.1.1.7.3.9.2. The decision to ship SNF in DPCs from the utility sites or repackage SNF into TADs prior to shipping from the utility sites has not been made. DOE stated that it would perform additional structural and criticality analyses once receipt of a specific DPC type is planned to confirm consistency with the design basis (SAR Section 1.5.1.1.1.2.1.2).

2.1.1.7.3.9.3.4 Naval Canister

Naval SNF will be 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. The naval SNF canister can be described as a circular cylinder with a bottom plate and a top shield plug. The bottom plate is 8.9 cm thick [3.5 in], the top shield plug is 38.1 cm thick [15 in], and the canister walls are 2.5 cm thick [1 in]. 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 maximum outer diameter of the naval SNF canister is 167 cm [66.5 in]. DOE 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 naval SNF canister is fabricated from a stainless steel that is similar to Stainless Steel Types 316 and 316L.

Design Bases and Design Criteria

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 DOE provided the design criteria and design bases for the naval canister. The nuclear safety design bases and the design criteria for the naval canister were given in SAR Table 1.5.1-30. The nuclear safety design bases in SAR Table 1.5.1-30 stated that the naval canister is analyzed as a representative canister.

DOE considered a drop of the canister, a drop of a load onto the canister, and a side impact or collision as potential structural challenges for the canister design. For these events, the design criterion is given in terms of the maximum effective plastic strain that results from the structural challenge and whether the maximum effective plastic strain meets the required reliability when compared to the canister’s capacity curve. The naval canister must be able to withstand the thermal challenges while contained within a cask, within a waste package, and within the CTM shield bell.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria of the naval canister using guidance in the YMRP. The NRC staff notes, on the basis of standard engineering practice, that maximum effective plastic strain is a reasonable structural failure criterion for evaluating whether loss of containment has occurred. The design criteria and design bases DOE used are derived from the PCSA results and are consistent with the canister’s intended safety function.

Design Methodology

In the PCSA, the naval SNF canister structural reliability is determined by using a representative canister which is selected such that it encompasses TAD canisters, DPCs, and naval canisters. The probability of a representative canister breach is evaluated for structural and thermal challenges including fire, loss of cooling inside a surface facility, seismic events, a flat bottom drop, collision with an object or structure, and the drop of an object on the canister. DOE stated that finite element programs, such as ANSYS, ABAQUS/Explicit®, and LS-DYNA™, have been used to simulate structural and thermal challenges.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design methodology for the naval canister using the guidance in the YMRP. On the basis of engineering judgment and staff experience, the NRC staff notes that software programs for structural reliability assessment, such as ANSYS, ABAQUS, and LS-DYNA, are appropriate because they are well-established commercial finite element software programs that are applicable for the types of analyses DOE performed. Further, the structural and the thermal models were verified using hand calculations, independent models, previous analyses, and thermal tests.

Design and Design Analyses

In SAR Section 1.5.1.4.1.2.8 DOE provided the codes and standards that the naval canister system must satisfy. The naval canister will be primarily 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), for normal and accident conditions of storage and transportation. With respect to the lifting features of the naval SNF canister, ANSI N14.6–93 (American National Standards Institute, 1993aa) will be followed 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 N14.5–97 (American National Standards Institute, 1998aa).

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design methodology for the naval canister using the guidance in the YMRP. The NRC staff notes that the cited codes and standards to be used for the design and fabrication of the naval SNF canister are in conformance with standard engineering practices and are therefore reasonable.

2.1.1.7.3.9.3.5 Aging Overpack

In SAR Section 1.2.7.1, DOE stated that two types of aging overpacks will be used: (i) vertical overpacks for a TAD canister and a DPC and (ii) a horizontal aging module for a DPC. 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 concrete sidewall and the top of the vertical overpack shield and protect the canister against natural environmental phenomena, such as tornadoes, airborne missiles, ambient-temperature extremes, and earthquakes. The aging overpacks are specified to have a maximum fully loaded weight of 227 metric tons [250 tons], a maximum overpack diameter of 3.66 m [12 ft], and a maximum overpack height of 6.71 m [22 ft]. A conceptual 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 wall thickness is approximately 0.91 m [3 ft], which provides shielding. The horizontal aging module has a maximum height of 6.40 m [21 ft], a maximum width of 2.59 m [8.5 ft], and a minimum length of 7.11 m [23.3 ft]. 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.

Design Bases and Design Criteria

DOE proposes to use an aging overpack system (i.e., either a vertical aging overpack or a horizontal aging module as appropriate for the canister) to protect the CSNF within TAD canisters and DPCs while in the aging facility. As DOE described in SAR Section 1.2.7.1.3.2, the aging overpack is a missile barrier and a radiation shield for the DPCs and TAD canisters within the aging facility.

DOE provided the design bases and design criteria for the aging overpack system in SAR Section 1.2.7.5. The aging overpack should (i) provide stability (i.e., prevent tipover during a seismic event with a PGA of 3 g), passive cooling, and cushioning of the canister for a drop or collision and (ii) protect the TAD canisters or DPCs from natural phenomena, so that they can maintain containment of radioactive materials. DOE stated that all of these aging overpack functions prevent canister stress and leakage limits and waste form and material temperature limits from being exceeded.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria for the aging overpack using guidance in the YMRP. The NRC staff notes that the design criteria and design bases DOE used are derived from the PCSA results and are consistent with the intended safety functions of the overpack.

Design Methodology

In SAR Section 1.2.7.6.2, DOE stated that before an aging overpack is used at the GROA, the aging overpack system will be evaluated for normal handling, dead, and thermal loads and loads that result from event sequences. The allowable stress and leakage rates will be compared to these loads to determine the acceptability of the aging overpack system. For example, DOE expresses the leakage rate limits in terms of the TAD canister leakage rate limit specifications. In addition, cladding temperature limits for both normal and off-normal are given in terms of the TAD canister specifications (SAR Section 1.5.1.1.1.2.6.1.1). DOE stated that, for DPCs contained in an aging overpack, the maximum canister leakage rate is equal to its design value.

DOE also stated in SAR Section 1.2.7 that the aging overpack system's structural design will be evaluated through the use of fragility assessments as described in SAR Section 1.7. A corresponding structural analysis of an aging overpack containing an SNF canister is presented in BSC (2008cp). The structural analysis focused on impact events including drop onto unyielding ground and slapdown from an upright position. DOE used these structural analyses results to provide an estimate of the failure probability for each of these impact events.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding the design methodology for the aging overpack using the guidance in the YMRP. The NRC staff notes that, for the TAD aging overpack, the leakage rate and cladding temperature limits are specified in terms of those of the internal TAD canister. These limits are consistent with the performance specifications for the vertical TAD aging overpack (DOE, 2008ag).

The NRC staff also reviewed DOE's approach for evaluating the aging overpack's structural capacity as given in BSC (2008cp). DOE used nonlinear finite element analysis for modeling the drop (impact) analyses. DOE's approach is reasonable because it is commonly used in industry for performing the highly nonlinear, transient analysis characteristic of impact.

On the basis of this review, DOE's structural analyses methodology is reasonable. In addition, in SAR Section 1.2.7, DOE stated that prior to use, any specific aging overpack system will be evaluated for normal handling, dead, and thermal loads and loads that result from event sequences.

Design and Design Analyses

In SAR Section 1.2.7.7 DOE provided general material specifications for the fabrication of the aging overpack system. DOE stated that the vertical aging overpack will be constructed of a metal liner, which is surrounded by reinforced concrete sidewalls and a top. DOE also stated that the concrete for the aging overpack will be in conformance with the requirements of ACI 349–01/349R–01 (American Concrete Institute, 2001aa). The reinforcing steel will comply with ASTM A706/A706M–06a (ASTM International, 2006ac) or ASTM A615/A615M–06a (ASTM International, 2006ad).

DOE identified the design codes and standards applicable to the aging overpack system design in SAR Section 1.2.7.8. Because the aging overpack system is classified as ITS, DOE stated in SAR Section 1.2.7.9 that the overpack system will be evaluated for normal handling loads, dead loads, thermal loads, and event sequence loads. It will also withstand the natural phenomena parameters listed in SAR Table 1.2.2-1.

In SAR Section 1.2.7.8, DOE stated that the design will follow these requirements:

- ASCE/SEI 43–05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities (American Society of Civil Engineers, 2005aa)
- ACI 349–01/349R–01, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary (American Concrete Institute, 2001aa)
- ANSI/ANS–6.4–1997, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, Appendix A (American National Standards Institute, 1997ab)

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design and design analyses for the aging overpack using the guidance in the YMRP. The NRC staff reviewed the cited design codes and standards to be used for the design and construction of the aging overpack system. The NRC staff notes that the cited codes and standards are in conformance with standard engineering practices and are therefore reasonable.

2.1.1.7.3.9.3.6 Transportation Cask

DOE proposes to use transportation casks to transport the different categories of waste forms to the repository. In SAR Section 1.5.1.1.1.2.1.3 DOE identified the TAD transportation cask as an ITS component of the TAD canister system. DOE listed preclosure nuclear safety design bases and criteria for the TAD transportation cask in SAR Table 1.9-2. DOE provided performance specifications for the TAD transportation cask and the vertical aging overpack for a TAD canister (DOE, 2008ag).

Design Bases and Design Criteria

DOE 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 transportation cask serves two safety functions: to provide confinement and to protect personnel from direct exposure (i.e., shielding). The transportation cask is required to provide confinement when the cask is subject to a drop or a low speed impact and collision, where the cask confinement is evaluated on the basis of canister capacity. The cask must also maintain shielding when subject to a drop or a low speed impact and collision. DOE stated that the transportation casks used for shipping SNF to the repository are casks certified by the U.S. Nuclear Regulatory Commission under 10 CFR Part 71 (SAR p. 1.2.8-41).

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria for the transportation cask using guidance in the YMRP. DOE’s design criteria and design bases are reasonable because the design bases and design criteria address the relevant safety functions of the transportation cask (i.e., confinement of radioactive material from drops, impacts, and collisions; protection of personnel from direct radiation exposure).

Design Methodology

In SAR Section 1.2.8.4.5.6, DOE provided the design methodologies used in the design of the transportation casks including the codes and standards. As part of the design methodology, DOE presented details in SAR Section 1.7.2.3.1 on the methodology used to estimate the transportation cask containment capacity to withstand repository structural challenges. In BSC (2008cp) structural analyses were presented for a transportation cask containing a representative SNF canister. The structural analyses focused on different drop/impact conditions. DOE used these structural analyses results to provide an estimate of the failure probability with respect to loss of containment.

NRC Staff Evaluation: The NRC staff reviewed the design methodology for the transportation cask using guidance in the YMRP. The NRC staff notes that the design methodology is reasonable because it is based upon an NRC-approved methodology.

The NRC staff reviewed DOE's approach for evaluating the transportation cask's structural capacity as given in BSC (2008cp). The NRC staff notes that DOE used nonlinear finite element analysis for modeling the drop (impact) analyses. DOE's approach is reasonable because it is commonly used in industry for performing the highly nonlinear, transient analysis characteristic of impact. On the basis of this review, the NRC staff notes that DOE's structural analyses methodology is reasonable.

Design and Design Analyses

DOE has specified that the transportation cask design will be based on the codes and standards, materials of construction, and design load combinations used for NRC certification of transportation cask designs.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design and design analyses for the transportation cask using guidance in the YMRP. The NRC staff notes that the design of the transportation cask is reasonable because it is based on the codes and standards, materials of construction, and design load combinations used by NRC to evaluate applications for transportation cask certification.

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. DOE provided this information in SAR Sections 1.14, 1.2.1 to 1.2.8, 1.9, and 1.10.3. The NRC staff's review focused on the design bases and design criteria, design methodology, and design analysis.

2.1.1.7.3.10.1 Criticality Prevention

DOE provided design information for the ITS features for prevention and control of nuclear criticality. The objective of the review is to verify the design of criticality prevention and control features.

In SAR Section 1.14, DOE described how its criticality safety program prevents and controls criticality during the preclosure period. The criticality safety program includes the analysis and design of SSCs, which was 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}, will not exceed the design basis value of the Upper Subcritical Limit (USL). In SAR Section 1.7.5, DOE stated that no Category 1 or 2 event sequences important to criticality were identified. The design features and PSCs relied upon to prevent criticality were listed in SAR Section 1.9.

The criticality safety analysis process was described in SAR Section 1.14.2.2. DOE's analysis of preclosure criticality considered how the canister designs, facility designs, and characteristics, as well as operations, affect the criticality control parameters. The parameters considered important to criticality are waste form characteristics, moderation, neutron absorbers, geometry, interaction, and reflection. DOE's criticality analyses evaluated changes to these parameters to provide input to the PCSA.

DOE's technical program included criticality safety requirements, analysis process, and evaluation results. The evaluations were based upon the current facility design and expected operations. Updated evaluations will be performed, as shown in SAR Table 5.10-3, to demonstrate that actual designs and fuel comply with criticality safety requirements.

DOE 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). DOE stated that it screened out Category 1 and 2 event sequences that could result in criticality. DOE relied on the use of passive design features (physical barriers against introduction of moderation), engineered design features, and procedural controls to screen out criticality.

DOE provided seven criticality control parameters it plans to use for each major waste type listed in SAR Table 1.14-2 for dry and wet handling conditions. Each parameter was classified as either conditionally controlled or as the primary parameter used to prevent criticality (SAR Section 1.14.2.2). DOE performed sensitivity calculations to understand the effect of varying these parameters on k_{eff}, which were discussed in BSC (2008ba) and whose results were summarized in SAR Section 1.14.2.3.

Design Bases and Design Criteria

Each ITS SSC has design bases and design criteria to meet those bases. Some ITS SSCs with design bases and design criteria related to criticality are briefly discussed next.

All the canisters to be handled at the GROA were classified as ITS. The criticality-related design basis provides containment, which prevents the introduction of moderator into the canisters. The NRC staff evaluates the canisters used in the GROA in TER Section 2.1.1.7.3.9.

Cranes and other lifting devices also have design bases and design criteria that are used to prevent canister breach, as discussed in TER Sections 2.1.1.7.3.4.1 and 2.1.1.7.3.4.2. Some cranes are also ITS because of design bases that help to control moderator, such as requiring the mean probability of inadvertent introduction of an oil moderator into a canister to be less than or equal to 9×10^{-5} over a 720-hour period following a radioactive release. To meet this design basis, DOE used design criteria where cranes have double retention capability on the areas of the crane where leaked oil could enter a breached canister (SAR Table 1.2.4-4).

If a canister is breached, water from the fire protection system is one of the main sources that could introduce moderator into the canister. Thus DOE had design bases and design criteria in place to limit the probability of water being introduced from the fire protection system. The NRC staff's evaluation of the fire protection system is discussed in TER Section 2.1.1.7.3.8.

To prevent a canister from being crushed by a closing slide gate or equipment shield door, DOE used a design criterion requiring the force of the closing slide gates to be power limited so they are not able to breach a canister or sever the hoisting ropes and cause a drop. Interlocks and obstruction sensors are also used. The NRC staff evaluates ITS interlocks in TER Section 2.1.1.7.3.7. The design bases and criteria were given in SAR Table 1.2.4-4 for the slide gates in the CRCF, and other slide gates have similar design bases. The equipment shield doors are further evaluated in TER Section 2.1.1.7.3.4.3.

Staging racks have design bases and design criteria to prevent criticality as discussed in TER Section 2.1.1.7.3.4.3 and in the design analysis section, next.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to review the information regarding the design bases and design criteria for criticality prevention. The NRC staff notes that the design bases and design criteria are reasonable because (i) the design bases and design criteria address the relevant safety functions for moderator exclusion that is relied on for criticality prevention (i.e., damage to the canisters that could allow moderator to enter the canister and the means to limit the presence of the moderator's water and oil if a canister were breached); (ii) the method of preventing the introduction of oil is consistent with ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac), which NRC endorses in Regulatory Guide 3.71 (NRC, 2005ac); and (iii) limiting the power of the gates and using interlocks for preventing canister breach, which prevents the introduction of moderator, is consistent with ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac).

Design Methodologies

DOE described its methodology for screening potential criticality events beyond Category 2 through controlling criticality parameters in SAR Sections 1.14.2.3.2.1 and 1.14.2.3.2.3. DOE stated that criticality is prevented through a combination of ITS SSCs and PSCs. DOE did not analyze potential criticality dose consequences, because criticality was screened out as beyond Category 2.

For dry handling in the surface and subsurface facilities, DOE relied primarily on moderator control to prevent criticality. The moderator of most concern was water. DOE controls moderator by keeping moderator out of areas where canisters are handled and by ensuring that the canisters are not breached in a drop or other accident. Without moderators present, low-enriched fissile systems such as CSNF and most DOE SNF cannot go critical. As part of its approach, DOE identified several SSCs that are ITS because they were relied on to exclude moderator or prevent a canister breach. The NRC staff considers DOE's method of preventing criticality by maintaining moderator control to be reasonable and in accordance with ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac).

In the WHF pool, where water cannot be avoided, DOE relied primarily on neutron absorbers to control criticality. For wet handling operations, DOE relied on the presence of 2,500 mg/L [2,500 ppm] of soluble boron enriched to 90 wt% B-10 as the primary criticality control parameter. DOE selected the chemical form of the neutron absorber to be orthoboric acid (H_3BO_3), which was to be injected into the water in the pool and in the transportation cask and

DPC fill water. To ensure the presence of enough enriched boron, DOE developed PSC-9, which requires operators to check the boron concentration.

NRC Staff Evaluation: The NRC staff reviewed DOE's design methodologies for ITS SSC used to prevent criticality as discussed in the SAR and responses to the NRC staff RAIs using the guidance in the YMRP. The NRC staff compared DOE's design methodology for dry handling with ANSI/ANS–8.21–1995 and ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac, 1995aa), which NRC endorses. The NRC staff notes that DOE's design methodology for dry handling is appropriate because it is consistent with the accepted standards. The NRC staff also notes that DOE's use of soluble neutron absorbers for the WHF pool is appropriate because it is consistent with ANSI/ANS–8.14–2004 (American Nuclear Society, 2004aa).

Design and Design Analysis

Code Validation

DOE used MCNP code models to determine the SNF k_{eff}. The ability of the models to accurately calculate k_{eff} was validated in BSC (2008ce,cf; 2003ai; 2002ac). DOE used the methodology of ANSI/ANS-8.1–1998 (American Nuclear Society, 2007aa) using the benchmark experiment results for model validation and model bias and uncertainties determination.

For CSNF, DOE specified the range of applicability (ROA) for six parameters represented in the benchmark experiments against which the model has been checked in BSC Table 34 (2008cf). In this table, DOE also provided the values and ranges for the CSNF models. On the basis of the MCNP results, DOE determined the critical limit for a CSNF is 0.988, which was rounded down to 0.98.

For the DOE SNF, the analysis of benchmark experiments was documented in BSC (2008ce, 2003ai, 2002ac). BSC (2003ai) recorded how DOE calculated critical limits for the different groups of the DOE SNF. BSC Table 6-43 (2003ai) summarized the calculated critical limit values and equations. The critical limits were calculated using the k_{eff}s of the benchmarks that applied to the fuel type. In BSC (2002ac), DOE described the benchmark experiments and provided tables containing the 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 supplied in BSC (2002ac), which determined the benchmarks to apply to the different configurations of the DOE SNF. BSC (2008ce) updated the material compositions used. BSC Table 7-2 (2008ce) listed the updated bias and bias uncertainty for the DOE SNF groups. On the basis of MCNP results, DOE determined the critical limit for all DOE SNF is 0.948, which was rounded down to 0.94, as described in BSC Section 2.3.10 (2008ba).

In SAR Section 1.14.2.3.4, DOE subtracted an administrative margin (Δk_m) of 0.05 from the critical limits to get a USL of 0.93 for CSNF and 0.89 for the DOE SNF.

NRC Staff Evaluation: The NRC staff reviewed the MCNP software validation information DOE provided using the guidance in the YMRP and Regulatory Guide 3.71 (NRC, 2005ac). The NRC staff considers that this software is reasonable for criticality analysis because MCNP is a commonly used computer code in the nuclear industry. The NRC staff also considers DOE's use of the ENDF V and VI neutron cross section libraries is reasonable because they are commonly used data libraries. Furthermore, DOE's use of the methodology for MCNP model validation specified in ANSI/ANS–8.1–1998 (American Nuclear Society, 2007aa) and endorsed in Regulatory Guide 3.71 (NRC, 2005ac) is reasonable because DOE detailed the validation that showed the guidance of the standard was appropriately followed. The NRC staff also notes that DOE's treatment of the bias and ROA of the benchmarks is reasonable because it is in accordance with ANSI/ANS–8.1–1998 (American Nuclear Society, 2007aa).

While most of the parameters for CSNF are within the ROA of the benchmarks, the NRC staff notes some discrepancies exist between the CSNF models and the benchmarks. The discrepancies include different energy spectrums and more soluble boron in the CSNF models than in any of the benchmarks. The NRC staff evaluated these differences and notes, on the basis of its knowledge of neutron physics, that the larger amount of B–10 in the CSNF models absorbs more of the thermal neutrons contributing to the model's energy spectrum because the models have relatively more intermediate and fast neutrons than the benchmark experiments. Therefore, the benchmarks provide reasonable validation of DOE's CSNF models because the differences in the energy spectrum are insignificant compared to the decrease in reactivity the larger amount of B–10 affects the energy spectrum and the overall similarity between the CSNF models and the benchmarks.

The NRC staff notes that the use of a Δk_m of 0.05 is reasonable and consistent with standard industry practice and the USLs DOE determined are reasonable for preclosure given the small change in the critical limit compared to the margin of safety incorporated into the USL.

Dry Handling

DOE described the general characteristics of the canisters used in the GROA in SAR Section 1.5.1. Outside the WHF pool, DOE relied on moderator and interaction control to prevent criticality. DOE used MNCP models to determine whether the k_{eff} of a configuration exceeded the USL. Configurations with a k_{eff}s that exceeds the USL were considered critical.

DOE relied on the TAD's 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, DOE discussed the criticality potential of the DOE standardized SNF canisters. DOE described the eight combinations of canister, basket, and representative fuel that are relevant to criticality in BSC Section 2.3.1.1.2 (2008ba). The DOE standardized SNF canister's 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 Figure 61 (2008ba) presented the results of an analysis that shows 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. On the basis of these results, DOE concluded that the minimum canister spacing that would ensure subcriticality is 30 cm [12 in]. DOE used this spacing in the DOE SNF staging rack design, discussed next. DOE's screening arguments for criticality-initiating events, including interaction, are evaluated in TER Section 2.1.1.3.3.2.1.

The HLW containers are used for the vitrified (glass) waste. DOE stated that individual HLW canisters (i.e., canisters holding glass made from highly radioactive liquid solutions) are subcritical as per ANSI/ANS–8.1–1998 Table 1 (American Nuclear Society, 2007aa) due to their low concentrations of fissile isotopes, as detailed in BSC Section 2.3.1.1.3 (2008ba).

DOE stated that the naval SNF canisters during the preclosure period criticality are considered to be controlled because the probability of a naval canister being breached is beyond Category 2 (SAR Section 1.5.1.4.1.2.6.3). Criticality resulting from the interaction of multiple naval canisters is prevented by the IHF design, as discussed in TER

Section 2.1.1.3.3.2.1. TER Section 2.1.1.7.3.9.3.4 documents NRC staff's review of the naval canisters. The DOE 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). DOE's model of interaction between DOE SNF canisters was discussed in BSC Sections 2.3.1.3.4 and 2.3.2.3.4 (2008ba).

NRC Staff Evaluation: The NRC staff reviewed the general characteristics of the canisters for moderator and interaction control to prevent criticality using the guidance in the YMRP. The NRC staff notes that designing the canisters with a low probability of breach is a reasonable method of maintaining moderator control and is consistent with the standards of ANSI/ANS–8.22–1997 (American Nuclear Society, 1997ac). The NRC staff evaluated DOE's modeling of a closely packed array of four DOE SNF canisters and notes that it is appropriate because the analysis used assumptions that result in an increase in the calculated k_{eff} and the canisters remain subcritical even with an unrealistically conservative reflector (lead) and the most reactive type of DOE SNF. For the DOE canisters, DOE's design provides reasonable control to limit interaction between the DOE canisters in a staging rack because the spacing between canisters would exceed 60 cm [24 in], as outlined in BSC Section 2.3.1.3.4 (2008ba), that is greater than the minimum canister spacing {30 cm [12 in]} that would prevent criticality.

The NRC staff notes that the individual HLW canister design is subcritical because the fissile isotope concentrations listed in SAR Table 1.14-1 are below the limit from ANSI/ANS–8.1–1998 Table 1 (American Nuclear Society, 2007aa), which provided a margin of safety.

Wet Handling

In the WHF pool, another type of ITS staging rack is used. This is a submerged SNF staging rack used to hold PWR and BWR assemblies. DOE claimed to have designed the SNF staging racks in accordance with ANSI/ANS–8.21–1995 (American Nuclear Society, 1995aa) (SAR Section 1.2.5.2.2.1.3). DOE detailed the staging racks in BSC Sections 2.3.1.3.1 and 2.3.1.3.2 (2008ba). The fixed neutron absorber used in the staging racks is Boral (BSC, 2008ba).

DOE modeled the result of 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 flux traps as collapsed. DOE determined that it required 30 percent of the soluble boron concentration to prevent criticality (SAR Section 1.14.2.3.2.2.4).

DOE considered potential event sequences that result in the interaction of a single assembly with the staging racks or shielded transfer casks containing TAD canisters or DPCs to remain subcritical while crediting no more than 15 percent of the minimum required soluble boron concentration (SAR Section 1.14.2.3.2.2.4). DOE considered event sequences concerning drops and earthquakes during transfer operations into or out of the WHF pool that could modify the system geometry. DOE stated that criticality could be prevented while the canister baskets and the fixed neutron absorber are omitted (SAR Section 1.14.2.3.2.2.4).

The criticality-related PSCs were listed in SAR Table 1.9-10, along with the basis for each of the PSCs. In the WHF, PSC-6 and PSC-9 were relied upon to prevent criticality. PSC–9 is used 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, which is pumped into the pool to maintain the boron concentration. The water in the pool is manually sampled and analyzed on a regular basis to monitor boron concentration (SAR Sections 1.2.5.3.2.1.3.3 and 1.2.5.3.2.2). PSC–6, on the other hand, is relied upon to control interaction by preventing assemblies from falling out of a cask that tipped over into the pool. Other PSCs were relied upon to prevent a canister breach that might potentially be followed by an introduction of moderator into a canister.

NRC Staff Evaluation: The NRC staff reviewed the design and design analysis for the wet handling facility to prevent criticality using the guidance in the YMRP. The NRC staff notes that the design and design analysis for the prevention of criticality are reasonable because Boral is a commonly used absorber, the design of the staging rack is consistent with the industry-accepted standard ANSI/ANS–8.21–1995 (American Nuclear Society, 1995aa), and the design analysis uses the industry-accepted code MCNP and 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 [2,500 ppm] 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 DOE modeled Boral as replaced with steel and the flux-traps are modeled as collapsed. The NRC staff calculations show that the model remains subcritical when fresh water replaces the borated water, by crediting the flux traps and Boral. 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 modeled both a Westinghouse 17 × 17 assembly and B&W 15 × 15 assembly submerged in borated water with 2,500 mg/L [2,500 ppm] of boron enriched to 90 wt% B-10. The modeling results show that both models were subcritical with the B&W assembly being more reactive.

The DOE approach for preventing criticality during Category 1 and 2 event sequences is reasonable because the approach relies on the control of boron concentration and enrichment through the boron makeup system and PSC-9, the large amount of boron in the WHF pool, the administrative margin, and the fresh fuel assumption.

2.1.1.7.3.10.2 Shielding Systems

DOE provided design information on the shielding features used at the GROA. This information was provided in SAR Sections 1.2.1 to 1.2.8, 1.9, and 1.10.3. The ITS shielding SSCs are those features that are credited in the PCSA for reducing the mean frequency of inadvertent exposure of personnel to below the Category 1 events sequence mean frequency. The ITS shielding components 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 objective of the NRC staff's review is to assess the design of ITS shielding features.

<u>Design Bases and Design Criteria</u>

DOE provided design bases 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 provide the bases for the shielding evaluation of the various facility areas and the radiation zones established for each.

DOE stated that it will use ANSI/ANS–6.4–2006 Table 5.2 (American Nuclear Society, 2006aa) and ACI–349–01/349R–01 (American Concrete Institute, 2001aa) for the concrete shielding design.

To ensure they perform their safety function, the shield doors will be interlocked to prevent inadvertent opening when complementary shielding is not closed, and the doors will be interlocked to radiation monitors. 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.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria using guidance in the YMRP with regard to DOE's presentation of shielding design objectives. On the basis of the review, the NRC staff notes that the design information DOE provided for design bases and design criteria is reasonable because the design criteria DOE provided are comprehensive enough to provide design bounding limits for the ITS shielding design, and the relationship between design bases and design criteria is clear.

<u>Design Methodologies</u>

DOE stated that the primary material used for shielding will be 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, Table 5.2 (American Nuclear Society, 2006aa). The 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 National Standards Institute, 1997ab).

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. To perform the shielding analysis, DOE used commonly accepted methods and computer codes such as MCNP and SCALE. This demonstrates 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 results were presented for the various areas and components in SAR Tables 1.10-35 through 1.10-46. Because the overall shielding design methodology is the same for both ITS and non-ITS shielding, the NRC staff evaluates it in TER Section 2.1.1.8.3.3.

NRC Staff Evaluation: The NRC staff reviewed DOE's methodology using the guidance in the YMRP and the design recommendations of Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff notes that DOE specified the use of applicable guidance and standards to develop the design methodology for shielding ITS components because the proposed design methodologies are supported by reasonable technical bases and are consistent with established industry practice. The codes and standards used for shielding design and construction are consistent with industry practice, and the computer codes used to perform shielding calculations are applicable for the radiation types and sources expected at the GROA.

Design and Design Analysis

The main safety function of ITS shielding is to protect personnel from direct radiation exposure. DOE 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 shielding analyses for both ITS and non-ITS use the same data and assumptions. The final step in the evaluation of the ITS shielding component design is to ensure the ITS SSCs design criteria are carried forward into the repository construction. Both steps were discussed in SAR Section 5.10. DOE proposed a two-step approach, using the proposed conditions listed in SAR Table 5.10-1 that include limiting conditions for operations. The specific, proposed conditions of the ITS SSCs evaluated in this TER section involve 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.

The second approach is for ITS SSCs that do not meet the criteria for limiting conditions for operations. DOE will detail the controls necessary to assure reasonable design and functionality of these ITS SSCs in the Technical Requirements Manual described in SAR Section 5.10.2.4.2. This manual is intended to provide a central location for the maintenance of operational and design restrictions not specifically contained in the limiting conditions. DOE's commitment to the design bases of ITS components in the Technical Requirements Manual was provided in SAR Table 5.10-3.

NRC Staff Evaluation: The NRC staff reviewed the design methodology using the guidance in the YMRP. The NRC staff notes that the design and design analysis method DOE used is reasonable because (i) this approach is based on industry-accepted approaches for radiation shielding and operational controls for limiting exposures and (ii) the DOE shielding analyses were evaluated and determined to be reasonable in TER Section 2.1.1.8.

2.1.1.7.4 NRC Staff Conclusions

The NRC staff notes that DOE's information relevant to design of important to safety (ITS) structures, systems, and components (SSCs) and safety controls (SCs) for the geologic repository operations area (GROA) is consistent with the guidance in the YMRP. The NRC staff also notes that DOE provided reasonable information on design of ITS SSCs for preclosure operations as discussed in this chapter.

As part of the detailed design process, DOE stated it would conduct additional analyses that will provide further information on and evaluation of design parameters and assumptions. The NRC staff notes that this information could be used to confirm that more refined soil properties and detailed designs are consistent with DOE's currently estimated demand-to-capacity ratios for the structural integrity of surface structures (DOE, 2009ev) (TER Section 2.1.1.7.3.1.1). As part of the detailed design process, DOE should (i) evaluate the effect of soil–structure interaction on the response of the aging pad prior to excavation to confirm the demand-to-capacity ratio estimated for the aging pad (TER Section 2.1.1.7.3.1.2); (ii) confirm the coefficient of friction between concrete pad and aging cask, and between concrete pad and horizontal aging module (TER Section 2.1.1.7.3.1.2); and (iii) confirm that the reliabilities for the types and manufacturing specifications of the ITS electrical power system, ITS I&C, and ITS interlock equipment procured for use in the GROA are consistent with the PCSA and final designs (TER Sections 2.1.1.7.3.6 and 2.1.1.7.3.7).

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CHAPTER 8

2.1.1.8 As Low As Is Reasonably Achievable for Category 1 Sequences

2.1.1.8.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of Safety Analysis Report (SAR) Section 1.10 (DOE, 2008ab) and the Operational Radiation Protection Program (RPP) described in SAR Section 5.11. The objective of this review is to verify that the U.S. Department of Energy's (DOE'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 activity is undertaken.

DOE has described the policy, design, and operational work practices of the repository relied upon to reduce doses to members of the public and occupational doses to workers to ALARA. DOE'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. DOE also described the facility shielding design used to meet the ALARA requirements for normal operations and Category 1 event sequences. DOE's implementation of 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 Evaluation Criteria

The regulatory requirements applicable to this section are the 10 CFR Part 20 ALARA requirements for normal operations and Category 1 event sequences, as required by 10 CFR 63.111(a)(1). 10 CFR 63.21(c)(6) requires DOE to submit a description of its program for control and monitoring of radioactive effluents and occupational radiological exposures. According to 10 CFR Part 20, DOE is required to develop, document, and implement an RPP commensurate with the scope and extent of the planned activities. In particular, 10 CFR Part 20 mandates (i) the use, to the extent practical, of procedures and engineering controls based upon sound radiation protection principles to achieve doses to members of the public and occupational doses that are ALARA; (ii) establishment of a constraint on air emissions of radioactive material, excluding Rn-222 and its daughters, to the environment; and (iii) control of the annual occupational dose to individual adults.

The NRC staff reviewed DOE's ALARA section using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). In addition, the NRC staff used HLWRS–ISG–03 (NRC, 2007ac), which supplements the YMRP. The relevant acceptance criteria follow:

- 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 reasonably incorporate ALARA principles.

- The RPP is described.

The following NRC guidance documents were also used in this review:

- NRC 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 (NRC, 1978ab)

- NRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," Rev. 1-R (NRC, 1997ac)

- NUREG–0800, Chapter 12.5, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NRC, 1987aa)

- NUREG–1567, Chapter 11, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (NRC, 2000ab)

2.1.1.8.3 Technical Evaluation

DOE stated that the objective of its ALARA program is to keep doses to repository workers and the public ALARA and that ALARA will be incorporated into the design, operations, maintenance, decommissioning, and dismantling activities. DOE submitted a description of its RPP for control and monitoring of radioactive effluents and occupational radiological exposures. The Operational RPP was described in SAR Section 5.11, and DOE stated that the Operational RPP would be made available prior to the receipt of radioactive waste at the site.

The NRC staff's review of DOE's program description for implementing ALARA principles, including its RPP, is discussed next.

2.1.1.8.3.1 DOE's Management Commitment To Maintain Doses As Low As Is Reasonably Achievable

In SAR Section 1.10.1, DOE described its management commitment to maintain doses ALARA. As a part of its management commitment, DOE stated that it will control worker doses and releases of radioactive materials to the environment. DOE 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 DOE's management commitment are the development of an ALARA program, implementation of an Operational RPP, and personnel training.

In the SAR, DOE stated that, during design and construction, it will conduct ALARA-specific reviews to ensure ALARA principles are incorporated in the design. DOE also stated that it will conduct and document audits consistent with the recommendations of NRC Regulatory Guide 8.8 (NRC, 1978ab). DOE stated that it will estimate occupational doses in accordance 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, DOE's management commitment is to implement an operational ALARA program at the repository in accordance with NRC Regulatory Guide 8.8, Position C.1 (NRC, 1978ab). DOE 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), DOE 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. DOE stated that it will develop procedures for implementing ALARA principles in decommissioning and dismantlement activities. Technical Evaluation Report (TER) Section 2.1.3 provides the NRC staff's evaluation of DOE's plans for permanent closure and decontamination or decontamination and dismantlement of the Yucca Mountain surface facilities.

NRC Staff Evaluation: The NRC staff evaluated DOE's management commitment to maintain doses ALARA, using the guidance in the YMRP. DOE's management commitment to implement radiation controls into its work activities is reasonable because DOE stated that it will incorporate ALARA principles during design and construction, operations, and decontamination and decommissioning. The controls are reasonable because DOE stated that it 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, as applicable. DOE also stated that it will implement an operational RPP that will follow the guidance in NRC Regulatory Guide 8.8, Position C.1 (NRC, 1978ab), as applicable.

According to SAR Section 1.10.1, personnel will be made aware of DOE's management commitment to ALARA through policy and instruction. DOE indicated that personnel will be instructed on their individual responsibilities related to ALARA implementation to ensure that the ALARA criteria are followed. DOE also stated that supervisors will be instructed to integrate appropriate radiation protection controls into work activities. DOE's approach for making personnel aware of DOE's management commitment to ALARA principles is reasonable because it is consistent with NRC Regulatory Guide 8.10 (NRC, 1997ac).

DOE's management commitment to provide appropriate radiation training is reasonable because DOE 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], and the training will be commensurate with the duties and responsibilities of training recipients. DOE also indicated that workers will be periodically retrained in radiation protection procedures and techniques on the basis of job responsibility.

DOE described its RPP commensurate with the scope and extent of the planned activities to provide protection for workers, members of the public, and the environment. DOE stated that it will provide operational radiation protection requirements through written policies and procedures. In SAR Section 5.11, DOE described the Operational RPP. DOE stated in SAR Section 5.11 that it will use program policies and procedures to limit radiation exposures to workers and members of the public during operations and to maintain exposures ALARA. The NRC staff's evaluation of the RPP is provided in TER Section 2.1.1.8.3.5.

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, DOE discussed the application of the ALARA principle into the design. DOE 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 consistent application of ALARA principles in the design process. Formal design criteria are used to implement ALARA design considerations. Reviews during the design process considered potential radiation exposure and contamination from normal operations and any Category 1 event sequences. DOE'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 for an individual radiation worker. ALARA principles were applied to both collective and individual doses for radiation workers. The ALARA program considered estimated worker doses 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, consideration was given to reducing doses for workers conducting recovery actions from potential event sequences.

DOE described design objectives, considerations, and features for the facility layout and equipment design. ALARA aspects were discussed for specific equipment, such as shield doors, shielded viewing windows, ventilation confinement, and radiation and airborne radioactivity monitoring. Access controls would be applied to high and very high radiation areas as well as restricted areas. Radiation zone designations were used to identify the need for design features to maintain doses ALARA. Where contamination could occur, DOE incorporated design features to control the spread of contamination and facilitate maintenance and decommissioning.

NRC Staff Evaluation: The NRC staff evaluated DOE's consideration of ALARA principles in design and modifications by using the guidance in the YMRP and notes that DOE's design of the geological operations area considered the ALARA philosophy. The NRC staff notes that DOE considered ALARA objectives and principles 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 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

DOE stated (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. DOE also stated that the facility design would be sufficient to limit the exposure to onsite members of the public in unrestricted areas to within DOE's proposed dose requirements specified in SAR Tables 1.8-28 and 1.8-36. TER Section 2.1.1.5 provides additional details on the NRC review of DOE's assessment of worker and public exposure during waste handling operations. DOE stated that it will control access to the restricted area and apply access controls to high and very high radiation areas using the guidance in NRC Regulatory Guide 8.38 (NRC, 2006ac). Therefore, DOE factored the ALARA principle into facility design (including facility location).

DOE stated that it will conduct ALARA design reviews using multidisciplinary teams with experience in radiological safety, operations, and engineering. DOE's approach for ALARA design reviews is reasonable 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 DOE decides to make potential modifications or improvements. This approach assures that modifications would not adversely influence other components of the design. The NRC staff also compared DOE's dose estimates for radiation workers during normal operations to DOE's ALARA goal and determined that estimated doses exceeded the annual ALARA dose goal at several GROA facilities (BSC, 2008al). DOE acknowledged situations when estimated doses did not meet the ALARA design goal and identified options for dose reduction (BSC, 2008bw). DOE 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, DOE presented average worker doses (BSC, 2008al) that were below the annual ALARA dose goal of 5 mSv [0.5 rem]. Therefore, DOE factored the ALARA principle into its assessment of radiological consequences for radiation workers.

2.1.1.8.3.3 Facility Shielding Design

In SAR Section 1.10.3, DOE discussed the facility surface and subsurface shielding design objectives, criteria, and evaluation used to implement ALARA criteria for normal operations and Category 1 event sequences. DOE stated that the objective is to design facility shielding to reduce dose rates from radiation sources such that worker doses are limited and are ALARA when combined with the program to control personnel access and occupancy of restricted areas. DOE performed shielding evaluations to ensure that reasonable 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. DOE 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, DOE provided its shielding design descriptions for individual facilities used in the shielding evaluation. The shielding design 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 include worker occupancy time, external radiation sources, radiation effects on components, and bounding source terms. 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, Table 1 (American Nuclear Society, 2006aa). 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.

DOE 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. TER Section 2.1.1.5.3.1 evaluates the use of this standard as well as the updated 1991 version of the standard. TER Section 2.1.1.5.3.1 notes that DOE's use of the 1977 standard is reasonable because it is based on conservative assumptions and results in an overestimate 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 to 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. The shielding evaluation includes factors such as the radiation source, distance from the source to the shielding, shielding thickness, shielding material, and the radiation zones of each facility provided in SAR Table 1.10-1. The radiation zones have been designated R1 through R5, which include unlimited occupancy through limited or no occupancy areas, respectively, with the dose rate range or limit of each zone. DOE implemented ALARA principles through the combination of facility shielding design and the RPP.

NRC Staff Evaluation: The NRC staff evaluated DOE's considerations of ALARA principles in the facility shielding design using NRC Regulatory Guide 8.8, Position C.2 (NRC, 1978ab). DOE's evaluation is based upon the shielding design objectives (SAR Section 1.10.3.1) and shielding design considerations (SAR Section 1.10.3.1.1), which the NRC staff notes are reasonable because they are based on the design recommendations of NRC Regulatory Guide 8.8 (NRC, 1978ab). In accordance with NRC Regulatory Guide 8.8 (NRC, 1978ab), DOE's design objectives include providing shielding that will ensure that (i) personnel radiation doses are ALARA, (ii) worker access and occupancy times allow for normal operations, and (iii) minimum radiation damage occurs to equipment not intended 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 notes that the shielding design considerations, as discussed in SAR Section 1.10.3.1.1, address reducing direct and scattered radiation. The source terms used in the shielding evaluation, as described in SAR Section 1.8, are reasonable to establish the shielding design criteria as well as the radiation zoning areas because DOE used bounding source terms. TER Section 2.1.1.5 provides the NRC staff's evaluation of DOE's source term calculations, direct exposure calculations, and radiation dose calculations to workers and members of public from airborne radionuclides, which the NRC staff notes are reasonable.

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, DOE 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 for restricted areas; reducing or preventing radioactive contamination; and monitoring and reducing radioactive waste production. DOE 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. DOE 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. DOE stated that, during operations, it will apply preplanning for significant worker doses and dry-run training for jobs associated with significant collective doses. DOE indicated that localized areas with higher radiation levels will be identified and factored into work planning. Work planning will include surveys of radiation levels, contamination, and airborne material concentrations; consideration of remotely operated equipment use; 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. DOE stated that radiation areas will be designated and posted within restricted areas, and access to high and very high radiation areas will be controlled. DOE indicated that ALARA reviews will be conducted before design changes and administrative control changes are approved. DOE 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, DOE did consider reduction of worker doses for recovery from potential event sequences that DOE described as off-normal events. TER Section 2.1.1.4 describes the NRC staff's evaluation of DOE's categorization of event sequences.

NRC Staff Evaluation: The NRC staff evaluated DOE's description of how DOE would incorporate the ALARA principles into operations, using the guidance in the YMRP and using HLWRS–ISG–03 (NRC, 2007ac). Because the ALARA principle applies during operations, TER Section 2.5.6 provides the NRC staff's evaluation of DOE's plans for conducting normal activities, including maintenance, surveillance, and testing of structures, systems, and components. In TER Section 2.5.6, the NRC staff notes that DOE described the plans for conduct of normal activities, including maintenance, surveillance, and periodic testing that would be implemented before DOE receives, possesses, processes, stores, or disposes high-level radioactive waste.

The NRC staff notes that DOE stated that it will incorporate ALARA guidance from NRC Regulatory Guides 8.8 and 8.10 (NRC, 1978ab; 1997ac) into repository processes and procedures. DOE stated it will apply ALARA principles to both individual and collective doses ALARA, which reconciles potential situations when reductions in collective dose could lead to significant increases to the dose for an individual. Because DOE's ALARA approach weighs associated drawbacks against potential benefits, such as dose increases from the installation and removal of temporary shielding versus the potential dose reductions when the shielding is in place, DOE's approach for assessing the usage of temporary shielding is reasonable. The NRC staff notes that DOE's approach of using operational administrative controls, such as radiological work permits for sampling, inspection, maintenance, and calibration procedures, is a standard industry practice. According to the SAR, preplanning and dry-run training will be required for significant worker doses and for jobs associated with significant collective doses. Because ALARA aspects would be considered as part of the review and approval process for issuing radiological work permits, DOE's implementation of the ALARA principles is tied to operational activities with higher levels of expected radiological exposure, which thereby increases the effectiveness of DOE's ALARA program. DOE stated that previous experience and data, as well as potential off-normal occurrences and contingency planning, will be factored into task planning and preparation, so that modifications to the proposed operations will be reviewed to assure that they do not adversely influence other aspects of area operations. The NRC staff notes DOE has reasonably described how it would incorporate the ALARA principle into proposed operations and that this approach provides additional confidence that DOE would execute an effective ALARA program.

In accordance with the ALARA criteria, DOE 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 DOE'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. The NRC staff notes that this approach is consistent with HLWRS–ISG–03 (NRC, 2007ac) and is reasonable. Nevertheless, DOE acknowledged that ALARA principles would be factored into the review of any proposed recovery actions so that dose reduction measures would be included.

2.1.1.8.3.5 Radiation Protection Program

DOE 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, which will be documented in a detailed RPP. As indicated in the introduction of SAR Section 5.11, DOE stated that a detailed RPP would be available prior to receiving and possessing spent nuclear fuel and high-level radioactive waste. Accordingly, the NRC conducted its review on the basis of the information available at this time. The purpose of the RPP is to establish policies and procedures to provide control of radioactive material; minimize the potential for contamination; minimize generation of low-level radioactive waste and effluents; and provide reasonable facilities, equipment, qualified staff, and radiation protection training. 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.

2.1.1.8.3.5.1 Administrative Organization

DOE described the RPP organization in SAR Section 5.11.1. DOE stated that it will have the radiation protection and criticality safety (RPCS) program organization under the RPCS manager. DOE indicated that the RPP organization will work independently of the operations and maintenance organizations. According to SAR Section 5.3.1.2 and DOE's response to a staff request for additional information (DOE, 2009az), the RPCS manager will report directly to the site operations manager and chief nuclear officer. DOE 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. DOE 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 Evaluation: The NRC staff reviewed DOE's description of the administrative organization of its RPP using the guidance in the YMRP, HLWRS–ISG–03 (NRC, 2007ac), NRC Regulatory Guide 1.8 (NRC, 2000ae), and NRC Regulatory Guide 8.8 (NRC, 1978ab). The NRC staff notes that the RPP organization description has defined the responsibilities of the RPCS manager. DOE's description of the RPCS manager duties and authority, discussed previously, is 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. DOE indicated 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). 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, DOE 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 experience should be at a level requiring policy planning and decisionmaking related to the programmatic aspects of RPP as a whole. The description for supervisory or management experience specified in ANSI/ANS–3.1–1993, Section 6.3 (American Nuclear Society, 1993aa) includes policy planning and decisionmaking. Therefore, these qualification requirements are consistent with NRC Regulatory Guide 1.8 (NRC, 2000ae). DOE also stated that it will include radiological response personnel to support emergency response functions. These statements provide additional confidence that there would be reasonable resources to maintain ALARA goals and objectives. DOE also stated that it will review and assess the adequacy of the radiation protection program and the content at least annually.

2.1.1.8.3.5.2 Equipment, Instrumentation, and Facilities

As a part of the RPP, DOE 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 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, DOE 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. DOE stated that its instrumentation will be periodically calibrated to the National Institute of Standards and Technology standards and routinely tested for operability. According to the SAR, DOE 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. DOE also stated that it will provide a radiation protection organization with appropriate facilities to effectively implement its RPP.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of the radiation protection equipment, instrumentation, and facilities using the guidance in the YMRP and HLWRS–ISG–03 (NRC, 2007ac). The NRC staff notes that DOE provided a high-level description of the type of protective equipment that DOE will include in its RPP, as described previously. DOE stated that it will describe the radiation protection equipment in the RPP. DOE's statement is reasonable at the current level of design because it is consistent with the guidance provided in NUREG–1567, Section 11.4.4.2 (NRC, 2000ab), as applicable. DOE stated that it will provide a detailed RPP.

For the instrumentation, DOE's description of its approach for selecting radiation protection instruments is reasonable because DOE stated it will (i) consider the radiation types, levels, and energies and (ii) consider the environmental conditions. DOE also stated it will be 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. DOE's approach for selection and calibration of radiation protection instrumentation is reasonable because it is consistent with the guidance in NUREG–0800, Section 12.5 (NRC, 1987aa), as applicable. DOE 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. For area monitoring, DOE 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. DOE stated that this system will provide both historical and real-time information and will operate on a continuous basis.

The NRC staff reviewed DOE'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. This is consistent with the general 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, Section 11.4.4.2 (NRC, 2000ab) and NUREG–0800, Section 12.5 (NRC, 1987aa), as applicable.

2.1.1.8.3.5.3 Policies and Procedures

DOE described the policies and procedures to be used to implement the RPP in SAR Section 5.11.3. DOE stated that it will develop the following policies and procedures:

- Radiation Surveys and Radiological Postings—DOE stated that radiation survey policies and procedures will be developed in accordance with 10 CFR 20.1501, 20.1502, 20.1703, 20.1906, and 20.2101, and radiological postings will be in accordance with 10 CFR 20.1901 through 20.1903.

- Radiological Access Control and Onsite Dose—DOE stated that the access control system will be developed to comply with 10 CFR 20.1601 and 20.1602. DOE also stated that it will follow NRC Regulatory Guide 8.38 (NRC, 2006ac). According to the SAR, the onsite dose limits in 10 CFR 20.1201 through 20.1208 and 10 CFR 20.1301 will be met by the RPP identifying occupational dose monitoring practices and a methodology to monitor dose limits for members of the public. DOE also stated that it will follow NRC Regulatory Guide 8.35 (NRC, 1992ac) for planned special exposures.

- Control of Radiological Material and Contamination—DOE 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 20.1905. DOE stated that it will meet the criteria of NRC Regulatory Guide 1.86, Table 1 (NRC, 1974aa) for determining whether materials and equipment can be released outside of restricted areas.

- Monitoring of External and Internal Dose—DOE stated that procedures and policies will be developed following 10 CFR 20.1501(c), 20.1502, and 20.1204. DOE also indicated that it will follow NRC Regulatory Guide 8.34 (NRC, 1992ab) for monitoring methods and criteria for occupational doses. DOE stated that it will follow NRC Regulatory Guide 8.9 (NRC, 1993aa) for internal dose monitoring. DOE also stated that it will select dosimeters on the basis of NRC Regulatory Guide 8.4, Paragraphs C and C.1 (NRC, 1973aa) and ANSI N322–1997 (American Nuclear Society, 1997ab). According to the SAR, DOE 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—DOE stated that procedures will be developed to meet the 10 CFR Part 20 requirements for surveys and measurements and that it will follow the guidance in NRC Regulatory Guide 8.25 (NRC, 1992aa).

- Respiratory Protection—DOE stated that, to be consistent with 10 CFR 20.1701 through 20.1705, it will follow the guidance in NRC Regulatory Guide 8.15 (NRC, 1999ac).

- Radiation Protection Training—DOE stated that its training will be consistent with guidance in NRC Regulatory Guide 8.8, Section C.2 (NRC, 1978ab); 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—DOE stated that it will post notices in accordance with 10 CFR 19.11 and 63.9(e)(1).

- Protection of the Pregnant Worker and Embryo/Fetus—DOE 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, Section C (NRC, 1999ab) and NRC Regulatory Guide 8.36 (NRC, 1992ad).

- Radiation Protection Records and Reports—According to the SAR, DOE's program will address the applicable requirements in 10 CFR 20.2101 through 20.2110, 10 CFR 20.2201 through 20.2206, and 10 CFR 19.13. DOE also stated that it will incorporate guidance in NRC Regulatory Guide 8.7 (NRC, 2005ab) and ANSI/HPS N13.6–1999 (American Nuclear Society, 1999aa).

- Environmental Radiological Monitoring—DOE indicated that its environmental monitoring program will be consistent with 10 CFR 20.1101(d), 20.1301, 20.1302, 20.1501, and 20.2001, as well as NRC Regulatory Guide 1.21 (NRC, 2009aa).

NRC Staff Evaluation: Using the guidance in NUREG–1567 (NRC, 2000ab) and NUREG–0800 (NRC, 1987aa), the NRC staff confirmed that DOE's description of the policies and procedures was consistent with commonly accepted programs and practices for radiation protection. DOE described the major program elements and implementation.

The NRC staff determined 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) and that the RPP 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 determined that the description of the RPP is consistent with the assumptions used in the PCSA consequence estimates, as reviewed in TER Section 2.1.1.5; the means to limit dose, as reviewed in TER Section 2.1.1.6; and the ALARA considerations, as reviewed in TER Section 2.1.1.8. Therefore, DOE's policies and procedures in its RPP are reasonable and commensurate with the scope of normal activities for the GROA.

2.1.1.8.4 NRC Staff Conclusions

The NRC staff notes that DOE's description of its proposed Operational Radiation Protection Program (RPP) is consistent with the guidance in the Yucca Mountain Review Plan (YMRP). The NRC staff also notes that DOE's RPP reflects as low as is reasonably achievable (ALARA) considerations as discussed in this chapter.

DOE stated that it would provide a detailed RPP for control and monitoring of radioactive effluents and occupational radiological exposures when it becomes available and prior to the receipt of radioactive waste at the site (TER Section 2.1.1.8.3.5.2).

2.1.1.8.5 References

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CHAPTER 9

2.1.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes

2.1.2.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of Safety Analysis Report (SAR) Section 1.11 (DOE, 2008ab) as supplemented by the 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 DOE'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, DOE described its plans for retrieval and alternate storage of radioactive wastes. DOE's description of its alternate storage plan identified a proposed alternate storage facility, including the location, size, and storage operations. DOE also provided a schedule for retrieval operations, should retrieval become necessary.

2.1.2.2 Evaluation Criteria

The regulatory requirements applicable to this section are in 10 CFR 63.21(c)(7), which requires DOE to describe its plans for retrieval and alternate storage, and 10 CFR 63.111(e), which defines how the preclosure performance objective related to retrievability of waste can be met.

- 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) requires that the geologic repository operations area be designed to preserve the option of waste retrieval throughout the preclosure period.

- 10 CFR 63.111(e) also mandates that the geologic repository operations area be designed so that any or all emplaced waste could be retrieved on a reasonable schedule.

Finally, the retrieval operations are to be conducted in a manner consistent with the criteria for the safety analysis of preclosure operations, including maintaining doses as low as is reasonably achievable (ALARA).

The NRC staff reviewed DOE's information using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The relevant acceptance criteria are (i) plans for retrieval of waste packages are provided and can be implemented on a reasonable schedule, if necessary; (ii) the retrieval plan incorporates ALARA considerations; (iii) the proposed alternate storage of retrieved radioactive wastes is reasonable; and (iv) a schedule for potential retrieval operations is provided.

2.1.2.3 Technical Evaluation

DOE's retrieval plan consists of maintaining access to waste packages in emplacement drifts through the preclosure period, such that waste packages could be retrieved, if necessary, by reversing the operational procedure used for waste emplacement. DOE 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. DOE also stated that if off-normal events occurred, such as collapse of an emplacement drift section, specialized procedures and equipment could be developed to restore access to waste packages. DOE also identified an alternate storage facility location. DOE did not propose the option of backfilling of emplacement drifts.

The NRC staff reviewed DOE's description of its retrieval operations provided in SAR Section 1.11. Specifically, the NRC staff reviewed DOE'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) DOE identified a reasonable range of potential problems (off-normal scenarios) during retrieval, and (iii) DOE described a feasible approach 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 DOE's retrieval operations schedule and description of alternate waste storage plans. The NRC staff review of these areas follows.

2.1.2.3.1 Waste Retrieval Plan

<u>Retrieval Under Normal Operations</u>

DOE described its waste retrieval plan in SAR Section 1.11.1. In the plan, DOE described the structures, systems, and components (SSCs) used for retrieval. DOE 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. DOE's plan includes maintaining access to the emplacement drifts and keeping the SSCs available throughout the preclosure period (DOE, 2009ba, 2009bb).

DOE described a monitoring and maintenance plan for the ground support system to keep the subsurface facility openings sufficiently stable to permit access to the SSCs and waste packages. DOE'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). DOE 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). DOE 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. DOE 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 Evaluation: The NRC staff used the guidance in the YMRP to determine whether DOE 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 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 could 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 evaluation of the TEV design in TER Section 2.1.1.7.3.1.3.1 notes that the TEV could support waste package transportation and its drive system is designed to negotiate a 2.5 percent grade fully loaded, regardless of direction. As discussed in TER 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 and the monitoring and maintenance programs for these would ensure accessibility to waste packages throughout the preclosure period.

The NRC staff also evaluated whether DOE's plan to inspect the emplacement drifts and turnouts using remotely operated equipment provides maintenance that would support retrieval if necessary. In response to NRC staff's RAI, DOE stated that it will inspect the entire length of every emplacement drift and turnout annually. After reviewing DOE's RAI response (DOE, 2009bb), the NRC staff notes that DOE has provided spatial and temporal coverage of observations necessary to assess performance of the ground support systems. The NRC staff also notes that in DOE (2009bb), DOE stated that it might change its inspection frequency if information gathered to that point in time supports such a change. DOE stated that the basis for changes in the inspection frequency of ground support would be properly documented and supported. The NRC staff notes that DOE could adjust the temporal frequency of annual inspections as conditions change, provided the inspection is frequent enough to permit an assessment of the rate of any change in ground support conditions. The NRC staff notes that DOE's plan to inspect the emplacement drifts and turnouts once a year initially and modify the inspection frequency as necessary provides temporal coverage of observations necessary to assess performance of the ground support systems.

DOE Retrieval Scenarios Under Off-Normal Conditions

DOE evaluated recovery strategies after an off-normal event and identified two off-normal occurrences that could hinder access to waste packages during the preclosure period (BSC, 2007bw): derailment of a TEV and rockfall resulting in rubble accumulation. DOE used these two scenarios to encompass the range of potential SSC 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 that are related to rockfall occurrences were 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.

DOE described the conceptual design of a maintenance and repair vehicle (MRV) for recovering from potential off-normal occurrences. According to DOE's description, the MRV design will be based on the TEV. The NRC staff reviews the TEV in TER Section 2.1.1.7. DOE 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. DOE's plan relied on this concept of a multipurpose vehicle for restoring a derailed TEV or a collapsed emplacement drift to normal conditions.

DOE identified three derailment conditions within emplacement drifts that encompass several severity levels of TEV failures. The minor failure case considered a derailment where the TEV retains full functionality. DOE's recovery plan consisted of placing commercially available "rerailers" along the rail and using the MRV to drag the TEV back onto the rail. A more severe case considered a derailment with 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. DOE proposed to use winches, rail clamps, and rerailers to remove the base plate and pull the TEV onto the rails. The most critical scenario considered involved the repair of damaged rails near emplaced waste packages. In DOE Enclosure 3 (DOE, 2009ba), DOE 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.

DOE identified (BSC, 2008bt) three waste package failure modes that could result from overheating of a waste package buried in rubble: (i) loss of impact properties for the outer corrosion barrier (OCB) due to exposure to a temperature of 538 ˚C [1000.4 ˚F] or higher; (ii) creep rupture of minimum-strength weldment material due to an OCB temperature of 501 ˚C [933.8 ˚F] or higher; and (iii) pressure-induced rupture of the bottom lid of the OCB for minimum-strength material due to exposure to a temperature of 400 ˚C [1000.4 ˚F] or higher near the center of the waste package lid. DOE analyzed the thermal effects on a waste package buried in rubble using the industry-accepted software ANSYS and considering representative heat transfer parameters for the drift. DOE evaluated 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 determined no Category 1 or 2 event sequences would interfere with retrieval. DOE also determined, on the basis of its calculations, that low-probability, beyond design bases conditions would not likely interfere with retrieval operations.

DOE 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. DOE described the design and development of an MRV that would be needed for rubble removal. According to DOE's plan, it would take approximately 8 years to initiate and complete the decision process, which includes design, development, and building of the MRV to initiate a recovery process.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to determine whether DOE identified retrieval scenarios under degraded drift conditions and methodologies were established for identifying and analyzing potential problems for the various retrieval operation scenarios. For an off-normal condition involving rockfall, the NRC staff reviewed DOE's evaluation of a scenario involving waste package burial in BSC (2008bt) and its response to staff's RAI in DOE Enclosure 4 (2009ba). The NRC staff reviewed the DOE assumptions for representing a rockfall condition and the analytical approach for determining the thermal effects of rock rubble and notes that they are reasonable. The NRC staff reviewed DOE's calculation of the temperature profile of a waste package partially or completely buried in rubble and analyses of failure modes of concern and the associated temperature limits. The NRC staff review considered any reliance on subsurface ventilation for cooling the buried waste package and potential impacts on the retrieval schedule under off-normal conditions.

The NRC staff reviewed the thermal limits provided in DOE (2009ba) and noted that the impact and creep rupture limits were based on ASME codes and the pressure-induced rupture limit was based on a calculation using the ANSYS code. The NRC staff considers these thermal limits for the failure modes (impact, creep rupture, and failure of the bottom lid due to over pressurization) reasonable because they are based on standard codes.

The NRC staff reviewed the calculations, and their bases, that DOE used to estimate the temperature of the waste package due to drift collapse and loss of ventilation. DOE determined 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 DOE stated in SAR Section 1.3.1.2.4. The NRC staff notes that the DOE calculations are reasonable.

Concerning the installation of a temporary shield wall during recovery from an off-normal condition, the NRC staff identified a potential limitation of DOE's plan. Depending on the length of the damaged section of rail or invert, the telescoping boom might not be able to span across the entire damaged section to build the shield wall. This limitation could significantly slow down the installation of the temporary shield wall DOE described. However, the NRC staff notes that retrieval would not be precluded as a result of this potential slowdown.

The NRC staff notes that although the occurrence of off-normal events could complicate the retrieval operations, the potential effects of such events on waste retrievability should be considered in light of the low likelihood of occurrence of the off-normal conditions. The NRC staff recognizes that the seismic ground motions strong enough to significantly damage an emplacement drift have a low likelihood of occurring during a 100-year preclosure period. In addition, the NRC staff considered that DOE has plans to inspect, monitor, and maintain the emplacement drifts and invert structure during the preclosure period, as mentioned earlier. The NRC staff notes that DOE's retrieval plan under off-normal conditions would be feasible and could be implemented within the proposed repository design concepts.

2.1.2.3.2 Preclosure Safety During Retrieval

In SAR Section 1.11.1.3.1, DOE discussed its approach to 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, DOE discussed, in general terms, how as low as is reasonably achievable (ALARA) concepts would be implemented. DOE did not develop occupational dose limits for retrieval. However, DOE stated that whatever radiation exposure considerations are applicable to emplacement operations would also apply to retrieval scenarios.

NRC Staff Evaluation: According to SAR Section 1.11.1.3.1, DOE did not identify any new event sequences for retrieval scenarios. DOE'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 DOE's preclosure safety analyses for waste emplacement operations and notes that the analyses are reasonable, as documented in TER Section 2.1.1.4 in general and TER Section 2.1.1.5 in particular. As stated earlier, the results of the preclosure safety analyses conducted for the waste emplacement operations are applicable to retrieval operations carried out in the reverse order. In addition, DOE stated in SAR Section 1.11.3 that it will submit additional details on its retrieval plan, as needed.

The NRC staff's review of the Operational Radiation Protection Program (RPP) is documented in TER Section 2.1.1.8, which notes that DOE's RPP, including implementation of ALARA principles, is reasonable. On the basis of the description of the RPP program, including ALARA implementation, for the preclosure operations and DOE's acknowledgment that similar radiation exposure considerations are applicable to retrieval, the NRC staff notes that DOE's ALARA program would also be reasonable for retrieval operations. In addition, the NRC staff notes that, if retrieval is required, DOE plans to implement radiation protection, including the ALARA program, consistent with the radiation protection guidelines current at the time of retrieval. DOE's retrieval plan is reasonable and consistent with the ALARA principles.

2.1.2.3.3 Proposed Alternate Storage Plans

DOE indicated in SAR Section 1.11.2 that facilities for handling and storing 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). DOE estimated that the alternate storage location can be developed to accommodate waste packages containing 7×10^7 kg [70,000 metric tons] of heavy metal. As DOE 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 DOE identified has been characterized for surface waste handling buildings and aging pads, as reviewed in TER Section 2.1.1.1.

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP. The NRC staff notes that the description of surface facilities for handling and storing retrieved waste packages is reasonable because the alternate storage site could be sited at the locations described in SAR Figure 1.11-1. The site capacity can accommodate all the waste, and DOE owns the site. Also, the NRC staff understands that the actual facility design need only be provided following a decision to retrieve, at which time the amount of waste to be retrieved, the nature of the shielded storage containers, and the storage configuration could be determined.

2.1.2.3.4 Retrieval Operations Schedule

DOE provided a conceptual retrieval timeline in SAR Figure 1.11-2 and BSC Section 4.2 (2008ad).

NRC Staff Evaluation: The NRC staff reviewed DOE's information using the guidance in the YMRP. The NRC staff notes that other conditions could potentially affect the schedule if DOE relied on the MRV for recovery. For instance, the retrieval planning timeline shown in SAR Figure 1.11-2 indicates that DOE expects that retrieval operations could initiate as late as 8.5 years after a decision to retrieve. However, no details are provided in the SAR or the references the NRC staff reviewed regarding the completion of design, fabrication, and delivery of the MRV for recovery operations. Given the potential for the waste package surface temperature to rise within a short period of time depending on the ventilation status [about 162 days according to SAR Section 1.3.5.3.2.1 and even earlier according to BSC (2008bt)], the NRC staff notes that it is unlikely that DOE could design, construct, and procure an MRV and execute the recovery in such a short period from the assumed condition of a waste package buried under rubble that also blocks ventilation.

However, any rubble accumulation is unlikely to be extensive. DOE used the term “off-normal conditions” to depict occurrences or conditions outside the bounds of routine operations but within the range of analyzed conditions for SSCs (SAR Section 1.3.1.2.1.7). In addition, 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. Even if rubble accumulation was extensive, it is likely to be a local phenomenon affecting only a few waste packages.

The retrieval schedules presented in the SAR are reasonable for normal conditions. The NRC staff notes that a severe impact on the overall schedule is not expected if DOE decided to retrieve the entire inventory even under “off-normal” conditions that DOE discussed. Therefore, on the basis of the reasonableness of the assumptions and the conceptual details provided in the SAR and supporting documents, the NRC staff notes that the overall schedule is achievable under the assumption of Category 2 events and those events beyond Category 2 design bases that did not exceed the thermal limits per DOE’s calculations, as stated in DOE Enclosure 4 (2009ba).

2.1.2.4 NRC Staff Conclusions

The NRC staff notes that DOE’s description of its plans for retrieval and alternate storage of radioactive wastes is consistent with the guidance in the YMRP. The NRC staff also notes that DOE’s plans for retrieval and alternate storage of radioactive wastes are reasonable as discussed in this chapter.

2.1.2.5 References

BSC. 2008ad. “Concepts for Waste Retrieval and Alternate Storage of Radioactive Waste.” 800–30R–HER0–00100–000. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2008bt. “Evaluation of an Event Sequence for Waste Package Burial.” 800–K0C–WIS0–00600–000–00A. 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. 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: DOE, 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: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

CHAPTER 10

2.1.3 Permanent Closure and Decontamination

2.1.3.1 Introduction

This chapter contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the proposed plans for permanent closure and decontamination or decontamination and dismantlement (PCDDD) of the Yucca Mountain surface facilities. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.12 (DOE, 2008ab) and the information DOE provided in response to the NRC staff's request for additional information (RAI) (DOE, 2009ao).

DOE included the design considerations in the SAR to facilitate PCDDD and a planning timeline for decontamination and dismantlement.

2.1.3.2 Evaluation Criteria

The regulatory requirements applicable to this section are 10 CFR 63.21(c)(8) and 10 CFR 63.21(c)(22)(vi), which require the SAR to describe the design considerations that are intended to facilitate PCDDD of surface facilities and to include information on plans for PCDDD of surface facilities. 10 CFR 63.21(a) specifies that the SAR must be as complete as possible in light of information that is reasonably available at the time of docketing.

The NRC staff reviewed the DOE information using the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The relevant acceptance criteria are (i) describe and provide bases for the features of the geologic repository operation area (GROA) design that will facilitate PCDDD and (ii) preliminary plans for PCDDD of surface facilities are adequate.

In addition, the NRC staff used other applicable guidance in NUREGs and regulatory guides to support the NRC staff's review. These additional guidance documents are discussed in the relevant sections that follow.

2.1.3.3 Technical Evaluation

DOE provided its key considerations for PCDDD and stated that it will develop and implement a plan that follows the guidance of NUREG–1575 (NRC, 2000ac) and NUREG–1757 (NRC, 2006aa). DOE stated that it will provide the plan to NRC in accordance with the planning timeline shown in SAR Figure 1.12-1. DOE stated that it will submit an application to amend the SAR before permanent closure of a geologic repository at the Yucca Mountain site, and this submission must consist of an update of the SAR. NRC would review this submission.

The NRC staff review of the design considerations that will facilitate PCDDD and the PCDDD plan follows.

2.1.3.3.1 Design Considerations That Will Facilitate Permanent Closure and Decontamination or Decontamination and Dismantlement

DOE described the criteria that it will use to ensure that the design will facilitate and support PCDDD in SAR Section 1.12.1. DOE also specified that the criteria applied as the design evolves will ensure maintaining radiation doses to workers and the public as low as is reasonably achievable (ALARA). The SAR indicated that some of the following considerations will be used to facilitate the PCDDD: selection of materials and processes to minimize waste production, use of a stainless-steel-lined wet handling pool with a leak-detection drainage system to minimize the contamination of concrete around the pool, and incorporation of features to contain leaks and spills. DOE also provided examples of design and operational considerations, described in SAR Section 1.12.3, that it will use to prevent contamination, such as minimizing the handling of uncanistered radioactive waste.

In its response to the NRC staff's RAI, DOE identified requirements and criteria that will ensure that design considerations to facilitate PCDDD will be evaluated as the design evolves. Specifically, DOE indicated that it will manage the bases for design in accordance with its Quality Assurance Requirement and Description (DOE, 2008af) document. DOE also noted that it will use its ALARA review process to ensure the design features that will facilitate decontamination and dismantlement are considered and evaluated. DOE stated that information necessary at the time of PCDDD would be available through the DOE record management and document control program.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to determine whether DOE described and provided bases for the GROA design considerations that will facilitate PCDDD. The NRC staff also evaluated the information DOE provided in its response to the NRC staff's RAI (DOE, 2009ao). The NRC staff notes that the type of design considerations described by DOE is typically used in nuclear facilities because these design considerations facilitate the decontamination process. For example, DOE's use of a stainless-steel-lined wet handling pool and features, such as berms, to contain leaks and spills will minimize site contamination. Operational considerations, such as minimizing the number of canisters to be opened, would also help prevent the spread of contamination. The NRC staff notes that these considerations are consistent with the guidance in the YMRP and conform to standard industry practice. Therefore, DOE reasonably described design features and their functions as they relate to PCDDD. On the basis of the NRC staff evaluation of DOE's response (DOE, 2009ao), the NRC staff notes that the use of the Quality Assurance Requirement and Description and ALARA processes would ensure that the design considerations to facilitate PCDDD would be considered as the design evolves and that DOE described and provided the basis for the GROA design considerations that it would use.

2.1.3.3.2 Plans for Permanent Closure and Decontamination or Decontamination and Dismantlement

DOE described in SAR Section 1.12.3 the information that it would collect on PCDDD and how it will maintain that information during the facility lifetime. SAR Section 1.12 stated that DOE will submit the final plans for the decontamination and dismantlement of the repository facilities in the GROA before permanent closure for NRC review. The NRC staff's review and evaluation of DOE's information are described next.

2.1.3.3.2.1 Facility History

DOE stated that the following facility history information will be available: (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 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 (SAR Section 1.12.3.1).

NRC Staff Evaluation: Using the guidance in the YMRP, the NRC staff reviewed the description of the information that DOE will collect. Because DOE indicated that the information will include past activities and all proposed operational activities that involve radioactive material, DOE considered the type of facility information needed to support PCDDD. The NRC staff notes that the scope of activities that DOE considered, as previously described, is comprehensive and consistent with the guidance in the YMRP and NUREG–1757 (NRC, 2006aa). The NRC staff also notes that the facility history information would be available at the time of permanent closure and decommissioning because DOE stated that it will create and maintain this information in accordance with its records and management and document control processes, as described in SAR Section 5.2.1.

2.1.3.3.2.2 Facility Description and Dose Modeling

DOE described information related to the GROA and its environment that will be used to estimate doses (SAR Section 1.12.3.2). DOE also described the objective of dose models in SAR Section 1.12.3.4, which is to show that the total effective dose equivalent to a critical group of individuals near the preclosure controlled area is consistent with ALARA principles. DOE also discussed the type of information that it would use for dose modeling, such as source term information and physical features used to model exposure pathways.

NRC Staff Evaluation: Using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa), the NRC staff reviewed DOE’s description of the information to be used for its facility dose modeling evaluations at the time of permanent closure and decommissioning. The DOE information, including facility site information and populations, source term information, exposure time estimates, and dose rate for PCDDD activities, is consistent with the information identified in the guidance in the YMRP and NUREG–1757 (NRC, 2006aa). Therefore, this information is reasonable for determining doses in support of PCDDD. DOE also indicated that it will create and maintain information related to its dose modeling programs in accordance with its records and management and document control processes, described in SAR Section 5.2.1. The NRC staff notes that this process of maintaining information would ensure the availability of this information at the time of permanent closure and decommissioning.

2.1.3.3.2.3 Facility Radiological Status

DOE described how it would evaluate the radiological status of the facility and determine the anticipated magnitude of decontamination activities or of decontamination and dismantlement activities (SAR Section 1.12.3.3). DOE indicated in its SAR that the information for these evaluations will be based on facilities’ 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, DOE listed the information that will be available during PCDDD to evaluate the radiological status, 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 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, DOE stated that the following information will be available during PCDDD to evaluate the radiological status of structures and buildings: (i) a summary of the soil 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, DOE discussed its plans for addressing potential water contamination from process operations. On the basis of the site characterization, DOE determined that there are no natural surface water bodies at the site (SAR Section 1.1.1.2). DOE indicated that storm water drainage diversion channels will protect the GROA from runoff from slopes above the facilities and keep storm water from becoming contaminated. According to the SAR, DOE will provide two storm water detention impoundments and analyze the radiation of the water in these impoundments. DOE stated that 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.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of information pertaining to the site radiological status using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa). The NRC staff notes that this information provided a reasonable basis for estimating the scope of PCDDD activities because the information to be collected is based on the site characterization information and proposed operational activities, as previously described. DOE provided details on how it will address radiological issues due to potential water contamination from process operations. For example, DOE indicated that storm water drainage diversion channels will protect the GROA from runoff from slopes above the facilities and that DOE will provide two storm water detention impoundments and examine the radiation of the water in these impoundments. For structures and buildings, systems and components, and contaminated soil, DOE identified and described the process to collect the information needed to address those radiological issues. The scope of activities that DOE considered is consistent with the guidance provided in NUREG–1757 (NRC, 2006aa). The NRC staff also notes that DOE showed that the radiological status information would be available at the time of permanent closure and decommissioning because DOE stated that it will create and maintain this information in accordance with its records and management and document control processes, described in SAR Section 5.2.1.

2.1.3.3.2.4 Alternatives for Decommissioning

DOE stated that it will evaluate alternative decontamination and dismantlement activities (SAR Section 1.12.3.5). DOE indicated that the strategy was to evaluate the alternatives consistent with ALARA principles while minimizing the generation of low-level waste. Information used includes (i) the determination of the anticipated physical condition of the facilities, components, and structures over time and (ii) the determination of appropriate methods of low-level radioactive waste disposal.

NRC Staff Evaluation: Using the guidance in the YMRP, the NRC staff reviewed DOE's description of its strategy for evaluating alternatives for decommissioning and notes that this description is reasonable because DOE stated that it will consider ALARA principles while minimizing radiological waste and environmental impacts as previously described. DOE explained that the information would be available at the time of permanent closure and decommissioning because DOE stated that it will create and maintain this information in accordance with its records and management and document control processes, described in SAR Section 5.2.1.

2.1.3.3.2.5 As Low As Is Reasonably Achievable Analysis

DOE described the scope of its ALARA assessment and the information that would be provided to support the ALARA analyses to facilitate PCDDD (SAR Section 1.12.3.6). DOE stated that the plan will be provided to NRC in accordance with the planning timeline shown in SAR Figure 1.12-1. As noted in the SAR, examples of the information that DOE 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 are consistent with ALARA principles.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of its ALARA analyses to support PCDDD using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa). The NRC staff notes that the type of information that DOE proposed to use is consistent with the guidance provided in NUREG–1757, Volume 2, Chapter 6 (NRC, 2006aa) because DOE would perform the assessment based upon the planned PCDDD activities and would describe ALARA goals and the implementation of the ALARA program. DOE explained that the information would be available at the time of permanent closure and decommissioning because DOE stated that it will create and maintain this information in accordance with its records and management and document control processes, described in SAR Section 5.2.1.

2.1.3.3.2.6 Planned Decommissioning Activities

DOE described the information that it would create and maintain to support its planned decommissioning activities (SAR Section 1.12.3.7). DOE described this information for contaminated structures, contaminated systems and components, and contaminated soil. DOE provided a basis for assuming that there are no natural surface water bodies at the site (SAR Section 1.1.1.2) and, therefore, no information for surface water is expected to be included in the plan. DOE proposed to build artificial water retention impoundments at the GROA for the purpose of water collection and evaporation. DOE indicated that the impoundment will be decommissioned after facility operations support is no longer required and the water remaining in the impoundment will be appropriately processed.

DOE also described the scope of the schedule. DOE stated that it will provide its plan in accordance with the planning timeline shown in SAR Figure 1.12-1.

NRC Staff Evaluation: The NRC staff reviewed DOE's description of information that DOE would use for its decommissioning activities using the guidance in the YMRP. The NRC staff notes that the type of information that DOE described would provide a reasonable basis for estimating the scope of PCDDD activities because it addresses PCDDD activities related to the site as well as the surface and subsurface facilities and structures, systems, and components. The information will include schedules and descriptions of the PCDDD tasks and techniques used, personnel protection methods, and the use of written procedures, consistent with standard industry practice. DOE explained that the information would be available at the time of permanent closure and decommissioning because DOE stated that it will create and maintain this information in accordance with its records and management and document control processes, described in SAR Section 5.2.1.

2.1.3.3.2.7 Project Management and Organization

DOE described its plan for developing a management organization that is responsible for task management for PCDDD (SAR Section 1.12.3.8). Specifically, DOE described the information to be included concerning management organization, including a description of the reporting hierarchy and project units' responsibilities and authority. DOE also described the information that will be included concerning task management; management positions and qualifications; training, such as descriptions of the radiation safety training that DOE intends to provide to each employee; and documentation that will be maintained to show that training needs are met.

NRC Staff Evaluation: Using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa), the NRC staff reviewed DOE's description of information to be used to establish DOE's PCDDD project management and organization. As previously described, DOE addressed the management structure and responsibilities and task management information needed for managing PCDDD activities, because DOE's information is consistent with the guidance in the YMRP and NUREG–1757 (NRC, 2006aa).

2.1.3.3.2.8 Health and Safety Program During Permanent Closure and Decontamination or Decontamination and Dismantlement

DOE stated in SAR Section 1.12.3.9 that it will modify the preclosure operational RPP to address PCDDD activities and listed the features (e.g., 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) that will be included in this modification.

NRC Staff Evaluation: Using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa), the NRC staff reviewed DOE's description of information to be used to establish DOE's PCDDD radiological health and safety program. On the basis of the evaluations in TER Section 2.1.1.8, the NRC staff notes that DOE's plan to modify its preclosure operational RPP to address PCDDD is reasonable because this approach provides integration with (i) the preclosure program and (ii) DOE's preclosure operational RPP, as documented in TER Section 2.1.1.8.

The NRC staff also notes that the activities that DOE proposed to include in the PCDDD RPP, such as air sampling and contamination control, showed that it considered the health and safety issues during PCDDD activities and is consistent with the guidance in the YMRP and NUREG–1757 (NRC, 2006aa).

2.1.3.3.2.9 Environmental Monitoring and Control Program During Permanent Closure and Decontamination or Decontamination Based on DOE's Statement on Developing Details in Detailed Design and Dismantlement

DOE stated in SAR Section 1.12.3.10 that the environmental monitoring and control information necessary at the time of PCDDD will include 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. DOE 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. According to the SAR, the Environmental Radiological Monitoring Program records will be available to evaluate and use for closure planning.

NRC Staff Evaluation: The NRC staff reviewed the information regarding DOE's environmental monitoring program using the guidance in the YMRP and NUREG-1757 (NRC, 2006aa) and notes that the environmental monitoring program information that DOE provided is consistent with NUREG–1757 and standard industry practice for engineering and process controls. DOE stated that it will evaluate and revise its environmental monitoring and control program, as necessary, to measure impacts to the site environment during PCDDD. DOE also indicated that it will ensure that the necessary records will be available through the DOE record management and document control program. Therefore, the environmental monitoring program information that DOE provided is reasonable because it is consistent with the previously cited guidance and industry practice.

2.1.3.3.2.10 Radioactive Waste Management Program

DOE stated that the information necessary at the time of PCDDD with respect to the management of low-level waste will be available through the record management and document control program (SAR Section 1.12.3.11). SAR Section 1.12.3.11.1 provided an estimated low-level radioactive waste quantity of 3,500 m^3 [123,620 ft^3] after treatment, which is expected to be generated during the closure phase of the repository. In the former section, DOE also listed the information to be developed for the PCDDD activities. Examples of the information DOE cited are (i) an estimated volume of each solid low-level radioactive waste type and (ii) the radionuclides with their estimated activity in each solid low-level radioactive waste type. DOE stated that it will update the estimated types and quantities of low-level waste generated during the development of the final plans for PCDDD activities. DOE does not expect to generate mixed radioactive waste as part of its routine operations, as stated in SAR Section 1.4.5.1.1.4.

SAR Section 1.12.3.11.2 listed the information to be included in the plans for minimizing the quantities of low-level radioactive waste and for disposing of the low-level waste. Examples of the information DOE cited are (i) a description of the waste volume reduction techniques to be used to minimize the amount of waste requiring burial and (ii) a description of the methods intended to be used to package and transport each waste type to its designated disposal facility.

NRC Staff Evaluation: The NRC staff reviewed DOE's radioactive waste management information using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa). The NRC staff notes that DOE described and provided the basis for a radioactive waste management program because it provided preliminary estimates of waste quantities and plans for managing radioactive waste generated during the preclosure period in SAR Section 1.12.3.11, and the information is consistent with NUREG–1757. In addition, DOE is expected to update the estimated types and quantities of low-level waste generated during the development of the final plans for PCDDD activities. Considering the previous information, the NRC staff notes that the radioactive waste management program information necessary at the time of PCDDD would be available through DOE's record management and document control program.

2.1.3.3.2.11 Radiation Surveys

DOE stated in SAR Section 1.12.3.13 that the radiological information necessary at the time of PCDDD 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. DOE indicated that this radiological information will be created and maintained in accordance with the records management and document control processes.

NRC Staff Evaluation: The NRC staff reviewed the radiological survey information that DOE provided using the guidance in the YMRP and NUREG–1757 (NRC, 2006aa) and notes that it is consistent with NUREG–1757. The NRC staff notes that DOE described and provided a basis for radiological surveys, because DOE stated that it will maintain the records associated with radiological surveys through DOE's record management and document control program.

2.1.3.3.2.12 Quality Assurance Program

In SAR Section 1.12.3.12, DOE stated that information required to facilitate PCDDD with respect to quality assurance (QA) will be integrated with the preclosure QA program. The information includes 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 Evaluation: The NRC staff used the guidance in the YMRP to determine whether DOE described the QA information (i.e., the type of information that will be required to facilitate decommissioning, with respect to QA) that would be used for PCDDD. The NRC staff notes that DOE's statement that it will integrate the information previously described with its QA program provides integration with the preclosure program. On the basis of the NRC staff evaluation and DOE statement, the NRC staff notes that DOE is cognizant of the QA program requirements that are to be integrated with PCDDD activities and that the information DOE provided is reasonable for preliminary PCDDD QA plans.

2.1.3.4 NRC Staff Conclusions

The NRC staff notes that DOE's proposed plans for permanent closure and decontamination or decontamination and dismantlement (PCDDD) of the Yucca Mountain surface facilities are consistent with the guidance in the YMRP. The NRC staff also notes that DOE's proposed plans for permanent closure and decontamination or decontamination and dismantlement are reasonable as discussed in this chapter.

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 and ML090690439. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

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

NRC. 2000ac. NUREG–1575, "Multi-Agency Radiation Survey and Site Investigation Manual." Washington, DC: NRC.

CHAPTER 11

Conclusions

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the Safety Analysis Report (SAR) and the other information submitted by the U.S. Department of Energy (DOE), and the NRC staff notes that, consistent the YMRP guidance, (i) the preclosure safety analysis (PCSA), which includes consideration of the design of the proposed geologic repository operations area (GROA) and activities associated with the period of operations, is reasonable; (ii) the identification of structures, systems, and components (SSCs) important to safety (ITS) is reasonable; (iii) the design of the GROA maintains the option of retrievability; and (iv) plans for permanent closure and plans for the decontamination or decontamination and dismantlement (PCDDD) of the surface facilities are reasonable. In particular, the NRC staff notes the following:

Preclosure Safety Analysis

- Site descriptions are reasonable for identification of natural and human-induced hazards that might affect the GROA design and conduct of the preclosure safety analysis (PCSA).
- Descriptions of structures, systems, and components (SSCs) and operational activities are reasonable to support PCSA.
- Identification of hazards, including the probability of hazards, is reasonable.
- Event sequences are reasonably developed, quantified, and categorized.
- Consequence calculations in supporting PCSA are reasonable.
- Important to safety structures, systems, and components and procedural safety controls are reasonably identified through PCSA.
- Nuclear safety design bases, including safety functions and controlling parameters for the important to safety structures, systems, and components, are based on the PCSA results.
- As low as is reasonably achievable (ALARA) principles are reasonably considered in the preclosure design and activities.

Retrieval and Alternate Storage

- Plans for retrieval and proposed alternate storage areas preserve the option to retrieve any or all of the emplaced waste on a reasonable schedule.

Permanent Closure and Decontamination

- Plans for PCDDD are reasonably described.
- Design features to facilitate PCDDD are reasonably considered.

DOE stated it would evaluate additional design details and conduct analyses to confirm the safety functions of structures, systems, and components (SSCs) important to safety (ITS) are consistent with what was presented in the SAR. DOE should confirm the (i) alluvium properties with respect to the design of foundations of surface facilities (TER Section 2.1.1.1.3.5.4) and (ii) mechanical properties of lithophysal and nonlithophysal rock in the repository block as part of a performance confirmation program (TER Section 2.1.1.1.3.5.4).

As part of the detailed design process, DOE should

1. Confirm the shear strength properties of alluvium, including uncertainty (TER Section 2.1.1.1.3.5.4)

2. Confirm the allowable maximum bearing pressure for mat foundations design on the basis of settlement criterion and during a design basis seismic event (TER Section 2.1.1.1.3.5.4)

3. Confirm the stability of slopes under applicable seismic loading conditions using an approach that accounts for uncertainties in the shear strength of alluvium (TER Section 2.1.1.1.3.5.4)

4. Confirm that its human reliability analyses (e.g., task analyses) identified potential vulnerabilities for the repository facilities and associated activities (TER Section 2.1.1.3.3.2.2.2)

5. Confirm that elastic spring constants to model soil at the BDBGM seismic level of 0.91 g for evaluation of $C_{1\%}$ are reasonable (TER Section 2.1.1.4.3.3.1.2)

6. Conduct seismic structural and foundation analyses to confirm the adequacy of $C_{1\%}$, which defines the fragility curves as shown in BSC Table 6.2-1 (2008bg) (TER Section 2.1.1.4.3.3.1.2)

7. Confirm that the identification of ITS components and the associated nuclear safety design bases are consistent with the design (TER Sections 2.1.1.4.3.3.2.1 and 2.1.1.6.3.1)

8. Confirm that the fault tree modeling specifies the components used to quantify its basic events (TER Section 2.1.1.4.3.3.2.1)

9. Confirm that the exposure time of containers is consistent with exposure time used in the PSCA for event sequence quantification and categorization (TER Section 2.1.1.4.3.4.2)

10. Confirm that the safety functions identified in the PCSA for passive and active systems that are credited to screen out initiating events are consistent with the design (TER Section 2.1.1.6.3.1)

11. Evaluate the effect of soil–structure interaction on the response of the aging pad to confirm the demand-to-capacity ratio estimated for the aging pad (TER Section 2.1.1.7.3.1.2)

12. Confirm the coefficient of friction between the concrete pad and the aging cask, and between the concrete pad and the horizontal aging module (TER Section 2.1.1.7.3.1.2)

13. Confirm that the reliabilities for the types and manufacturing specifications of the ITS electrical power system, ITS I&C, and ITS interlock equipment procured for use in the GROA are consistent with the PCSA and final designs (TER Sections 2.1.1.7.3.6 and 2.1.1.7.3.7)

CHAPTER 12

Glossary

This glossary is provided for information and is not exhaustive. Terms shown in *italics* are included in this glossary.

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*.

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.

***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.

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 waste package design.

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.

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*.

aquifer: A saturated underground geologic formation of sufficient permeability to transmit *groundwater* and yield water of sufficient quality and quantity to a well or spring for an intended beneficial use.

ash: Fragments of volcanic rock that are broken during an explosive volcanic eruption to less than 2 mm [0.08 inches] in diameter. See also *tephra* and *pyroclastic*.

basalt: A common type of *igneous* rock that forms black, rubbly-to-smooth-surfaced lavas and black-to-red *tephra* deposits (frequently used as "lava rock" for barbecues).

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.

***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) The process of comparing the conditions, *processes*, and *parameter* values used in a *model* against actual data points or interpolations (e.g., contour maps) from measurements at or close to the site to ensure that the *model* is compatible with reality, to the extent feasible. (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. (3) In operations, the process to ensure accuracy of instruments and any set points for automation actuations of items important to safety.

canister: An unshielded cylindrical metal receptacle that facilitates handling, transportation, storage, and/or disposal of high-level *radioactive* waste. It may serve as (1) a pour mold and container for vitrified high-level *radioactive* waste; (2) a container for loose or damaged fuel rods, nonfuel components and assemblies, and other debris containing *radionuclides*; or (3) 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 packages.

carbon steel: A steel made of 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.

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 around a volcanic vent. Synonymous with *scoria cone*.

cladding: The metal outer sheath of a fuel rod generally made of a zirconium alloy, and in the early nuclear power reactors of *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. Often referred to as "clad."

climate: Weather conditions, including temperature, wind velocity, precipitation, and other factors, that prevail in a region.

code (computer): The set of commands used to implement a mathematical *model* on a computer.

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 flared 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.

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 release of energy as a result of accidental production of a self-sustaining or divergent neutron chain reaction.

design concept: An idea to design and operate the aboveground and belowground portions of a proposed repository.

dike: A tabular, generally vertical body of *igneous* rock that cuts across the *structure* of adjacent rocks. Dikes *transport* molten rock from depth to an erupting volcanic.

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 (substance dissolved in *groundwater*) 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*.

disposal canister: A cylindrical metal receptacle designed to contain *spent nuclear fuel* and high-level *radioactive* waste as an integral part of the waste package.

dissolution: Dissolving a substance in a solvent.

drift: From mining terminology, a horizontal underground passage. In the proposed Yucca Mountain repository design, drifts include excavations for emplacement (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* for storing (in a storage facility) and shipping (in a transportation *cask*) *commercial spent nuclear fuel* assemblies.

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. Also 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*.

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*. (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.

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 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. 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.

finite element analysis: A commonly used numerical method for solving mathematical equations in a variety of areas (e.g., hydrology, mechanical deformation). A technique in which algebraic equations are used to approximate the partial differential equations that comprise mathematical *models* to produce a form of the problem that can be solved on a computer. For this type of approximation, the area being modeled is formed into a grid with irregularly shaped blocks. This method provides an advantage in handling irregularly shaped boundaries (e.g., internal features such as faults) and surfaces of engineered materials. Values for *parameters* are frequently calculated at nodes for convenience, but are defined everywhere in the blocks by means of interpolation functions.

fissure: In relation to *igneous* activity, a fissure is an elongate 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.

fluvial: *Processes* related to the downslope movement of water on the Earth's surface.

fracture: A planar discontinuity in rock along which loss of cohesion has occurred. It is often caused by the 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. Fractures may act as fast paths for *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.

galvanic corrosion: Accelerated corrosion of a metal resulting from electrical contact with a more noble metal or nonmetallic conductor in a corrosive electrolyte.

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): The 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 contained in pores or *fractures* in either the unsaturated or saturated zones below ground level.

human failure event: An event that would be modeled as a basic event in the logic *models* of a safety assessment, and that is the result of one or more unsafe actions.

human reliability analysis: Human reliability analysis evaluates the potential for, and mechanisms of, human errors that may affect the safety of the proposed geologic repository operation area operations, 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) A type of rock that has formed from a molten, or partially molten, material. (2) A type of activity related to the formation and movement of molten rock, either in the subsurface (*intrusive*) or on the surface (*extrusive*).

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 an eruption. 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.

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.

matrix: Rock material and its pore space exclusive of *fractures*. As applied to Yucca Mountain *tuff*, the ground mass of an *igneous* rock that contains larger crystals.

median: A value such that one-half of the observations are less than that value and one-half are greater than the value.

meteorology: The study of climatic conditions, such as precipitation, wind, temperature, and relative humidity.

migration: *Radionuclide* movement from one location to another within the engineered barrier system or the environment.

model: A depiction of a system, phenomenon, or *process*, including any hypotheses required to describe the system or explain the phenomenon or *process*.

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.

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 uniform *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 pathways.

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.

pit: 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* activity, this describes fragments or fragmental rocks and deposits produced by explosive volcanic activity, 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 2.6 million years ago to the present day.

radiation worker**:** A proposed geologic repository operation area worker within the controlled area boundary, with assigned duties that involve exposure to radiation or radioactive material and who receives an *occupational dose*.

radiation protection program**:** A program for controlling and monitoring radioactive effluents and occupational radiological exposures to maintain such effluents and exposures.

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) or 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: Radioactive type of atom with an unstable nucleus that spontaneously decays, usually emitting radiation in the process. Radioactive elements are characterized by their atomic mass and atomic number.

range (statistics): The numerical difference between the highest and lowest value in any set.

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.

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 based on 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.

rock fall: The release of *fracture*-bounded blocks of rock from the *drift* wall, usually in response to an earthquake.

rock matrix: See *matrix*.

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*).

scoria cone**:** See *cinder cone*.

seepage: The inflow of *groundwater* moving in *fractures* or *matrix* pores of permeable rock to an open space in the rock. For the proposed Yucca Mountain repository, seepage refers to water dripping into a *drift*.

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.

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.

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: Nuclear reactor fuel that has been used to the extent that it can no longer effectively sustain a chain reaction and that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. This fuel is more *radioactive* than it was before irradiation and releases significant amounts of heat from the decay of its fission product *radionuclides*.

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*:** A layer of rock or soil with geologic characteristics that differ from layers above or below it.

stress corrosion cracking: A cracking *process* that requires the simultaneous action of a corrodent substance and sustained (residual or applied) tensile stress. Stress corrosion cracking excludes both the fracture of already corroded sections and the localized corrosion *processes* that can disintegrate an alloy without the action of residual or applied stress.

structure: In geology, the arrangement of the parts of geologic features or areas of interest such as folds or faults. This includes features such as *fractures* created by faulting, and joints caused by 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.

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: Transparency is achieved when it is easy to detect and understand the process by which a study was completed, the assumptions driving the results, the way in which the assumptions were reached, the methodology used, the rigor of the analyses, and the outcome of the study.

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: A *canister* for transportation from a remote location, *aging* at a centralized site, and disposal at the proposed repository.

tuff: A general term for volcanic rocks that formed from rock fragments and *magma* 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 site.

uncertainty: How much a calculated or measured value varies from the unknown true value. See also *aleatory uncertainty* and *epistemic uncertainty*.

variable: A nonunique property or attribute.

variability (statistical): A measure of how a quantity varies over time or space.

volcanism: Pertaining to *extrusive igneous* activity.

wash: In a relation to landforms, a streambed, dry or running, usually in an arid environment.

watershed: The area drained by a river system including the adjacent ridges and hillslopes.

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11. ABSTRACT (200 words or less)

This “Technical Evaluation Report on the Content of the U.S. Department of Energy’s Yucca Mountain Repository License Application—Preclosure Volume: Repository Safety Before Permanent Closure” presents the U.S. Nuclear Regulatory Commission (NRC) staff’s review of the preclosure design and operations that the U.S. Department of Energy (DOE) described and provided in its Safety Analysis Report (SAR), dated June 3, 2008, as updated on February 19, 2009. The NRC staff also reviewed information DOE provided in response to NRC staff’s requests for additional information and other information that DOE provided related to the SAR. In particular, this report provides information on the NRC staff’s evaluation of (i) Site Description as it Pertains to Preclosure Safety Analysis; (ii) Description of Structures, Systems, Components, Equipment, and Operational Process Activities; (iii) Identification of Hazards and Initiating Events; (iv) Identification of Event Sequences; (v) Consequence Analyses; (vi) Identification of Structures, Systems, and Components Important to Safety, Safety Controls, and Measures to Ensure Availability of the Safety Systems; (vii) Design of Structures, Systems, and Components Important to Safety and Safety Controls; (viii) As Low As Reasonably Achievable for Category 1 Sequences; (ix) Plans for Retrieval and Alternate Storage of Radioactive Wastes; and (x) Permanent Closure and Decontamination, or Decontamination and Dismantlement (PCDDD) of Surface Facilities.

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