NOTE TO READER: This volume is one of five volumes shown in the table below that comprise the Safety Evaluation Report (SER). Each volume is to be published separately as it is completed; however, the volume number may not be published in sequence (e.g., Volume 3 is anticipated to be published before Volume 2). The SER volume number and section number of chapters within a volume, are based on the Yucca Mountain Review Plan (YMRP)[1] that the U.S. Nuclear Regulatory Commission (NRC) staff used to guide its review of the license application. Use of SER section numbers that correspond to the YMRP section numbers facilitated the NRC staff's writing of the SER, allowing the interested reader to easily find the applicable review methods and acceptance criteria within the YMRP. The following table provides the topics and SER sections for each volume. The table will help the reader locate SER section cross-references in each volume.

Chapter	*SER Section*	*Title*
Volume 1 General Information		
1	1.1	General Description
2	1.2	Proposed Schedules for Construction, Receipt, and Emplacement of Waste
3	1.3	Physical Protection Plan
4	1.4	Material Control and Accounting Program
5	1.5	Description of Site Characterization Work
Volume 2 Repository Safety Before Permanent Closure		
1	2.1.1.1	Site Description as it Pertains to Preclosure Safety Analysis
2	2.1.1.2	Description of Structures, Systems, Components, Equipment, and Operational Process Activities
3	2.1.1.3	Identification of Hazards and Initiating Events
4	2.1.1.4	Identification of Event Sequences
5	2.1.1.5	Consequence Analyses
6	2.1.1.6	Identification of Structures, Systems, and Components Important to Safety; and Measures to Ensure Availability of the Safety Systems
7	2.1.1.7	Design of Structures, Systems, and Components Important to Safety and Safety Controls
8	2.1.1.8	Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences
9	2.1.2	Plans for Retrieval and Alternate Storage of Radioactive Wastes
10	2.1.3	Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities
Volume 3 Repository Safety After Permanent Closure		
1	2.2.1.1	System Description and Demonstration of Multiple Barriers
2	2.2.1.2.1	Scenario Analysis
3	2.2.1.2.2	Identification of Events with Probabilities Greater Than 10^{-8} Per Year
4	2.2.1.3.1	Degradation of Engineered Barriers
5	2.2.1.3.2	Mechanical Disruption of Engineered Barriers
6	2.2.1.3.3	Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms

[1]NRC. 2003. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

Chapter	SER Section	Title
Volume 3 Repository Safety After Permanent Closure (continued)		
7	2.2.1.3.4	Radionuclide Release Rates and Solubility Limits
8	2.2.1.3.5	Climate and Infiltration
9	2.2.1.3.6	Unsaturated Zone Flow
10	2.2.1.3.7	Radionuclide Transport in the Unsaturated Zone
11	2.2.1.3.8	Flow Paths in the Saturated Zone
12	2.2.1.3.9	Radionuclide Transport in the Saturated Zone
13	2.2.1.3.10	Igneous Disruption of Waste Packages
14	2.2.1.3.12	Concentration of Radionuclides in Groundwater
15	2.2.1.3.13	Airborne Transportation and Redistribution of Radionuclides*
16	2.2.1.3.14	Biosphere Characteristics
17	2.2.1.4.1	Demonstration of Compliance with the Postclosure Individual Protection Standard
18	2.2.1.4.2	Demonstration of Compliance with the Human Intrusion Standard
19	2.2.1.4.3	Demonstration of Compliance with the Separate Groundwater Protection Standards
20	2.5.4	Expert Elicitation
Volume 4 Administrative and Programmatic Requirements		
1	2.3	Research and Development Program to Resolve Safety Questions
2	2.4	Performance Confirmation Program
3	2.5.1	Quality Assurance Program
4	2.5.2	Records, Reports, Tests, and Inspections
5	2.5.3.1	U.S. Department of Energy Organizational Structure as it Pertains to Construction and Operation of Geologic Repository Operations Area
6	2.5.3.2	Key Positions Assigned Responsibility for Safety and Operations of Geologic Repository Operations Area
7	2.5.3.3	Personnel Qualifications and Training Requirements
8	2.5.5	Plans for Startup Activities and Testing
9	2.5.6	Plans for Conduct of Normal Activities, Including Maintenance, Surveillance, and Periodic Testing
10	2.5.7	Emergency Planning
11	2.5.8	Controls to Restrict Access and Regulate Land Uses
12	2.5.9	Uses of Geologic Repository Operations Area for Purposes Other Than Disposal of Radioactive Wastes
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*This SER Section combines the review of information addressed in the YMRP Sections 2.2.1.3.11 and 2.2.1.3.13.

ABSTRACT

This is the second volume of the U.S. Nuclear Regulatory Commission (NRC) staff's "Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada." This document, the NRC staff's Safety Evaluation Report (SER), Volume 2, summarizes the results of the NRC staff's review of the Preclosure operations that the U.S. Department of Energy (DOE) described and provided in its June 3, 2008, license application, as updated on February 19, 2009, seeking an authorization to begin construction of a repository at Yucca Mountain. In particular, this SER Volume 2 documents the results of NRC staff's evaluation to determine whether the design of the proposed geologic repository operations area for Yucca Mountain will comply with the performance objectives and requirements that apply before the repository is permanently closed. A prior volume of the SER, Volume 1, documents General Information. In subsequent volumes of the SER, Volumes 3–5, NRC staff plans to present the remainder of its review and evaluation of the Safety Analysis Report included in DOE's license application.

Consistent with NRC's requirements and performance objectives, the NRC staff reviewed the following: (i) Site Description as it Pertains to the 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) Meeting the 10 CFR Part 20 As Low As Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences, (ix) Plans for Retrieval and Alternate Storage of Radioactive Wastes, (x) Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities.

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Abbreviations and Acronyms

AC	alternating current
AF	aging facility
AFE	annual frequency of exceedence
ALARA	as low as is reasonably achievable
ANSI/ANS	American National Standards Institute/American Nuclear Society
AO	aging overpack
APE	annual probability of exceedence
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
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
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 exceedence
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
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
RAI	request for additional information
RF	receipt facility
RHH	repository host horizon
RMS	radiation/radiological monitoring systems
ROA	range of applicability
ROVs	remotely operated vehicle
RPCS	radiation protection and criticality safety
RPP	Radiation Protection Program
RVT	random vibration theory
SAR	Safety Analysis Report
SASW	spectral analysis of the surface wave
SCs	safety controls
SER	Safety Evaluation Report
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
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

EXECUTIVE SUMMARY

Background

On June 3, 2008, the U.S. Department of Energy (DOE or the applicant) submitted a license application to the U.S. Nuclear Regulatory Commission (NRC) seeking authorization to begin construction of a geologic repository for high-level radioactive waste disposal at Yucca Mountain, Nevada.[1] The license application consists of general information and a Safety Analysis Report (SAR). This document, the NRC staff's Safety Evaluation Report (SER), Volume 2, presents certain results of staff's review of the SAR DOE provided in its June 3, 2008, license application and as updated on February 19, 2009.[2] The NRC staff also reviewed (1) information DOE provided in response to NRC staff's requests for additional information and (2) other documents that DOE has provided to NRC, which are not part of the License Application. For example, DOE provided 196 separate technical documents to the NRC on June 3, 2008 when DOE submitted the License Application. Where appropriate, the each volume of the SER provides specific citations to these additional sources in the context of the NRC staff's safety review. SER Volume 2, in particular, documents the results of the staff's evaluation to determine whether the proposed repository design for Yucca Mountain will comply with the technical criteria and public health and environmental standards that apply before the repository is permanently closed. These criteria and standards can be found in NRC's regulations at 10 CFR Part 63, Subparts E and K. In conducting its review of the license application, the NRC staff was guided by the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan.[3]

Other portions of the NRC staff's safety review have been, or will be, documented elsewhere. SER Volume 1, NUREG–XXXX (NRC, 2010XX) contains the staff's evaluation of DOE's general information. SER Volume 3, NUREG- XXXY (NRC, 2010XY) contains the staff's evaluation of DOE's compliance with postclosure safety objectives and requirements; SER Volume 4 will document the staff's evaluation of DOE's demonstration of compliance with administrative and programmatic requirements; and SER Volume 5 will document the staff's evaluation of the DOE's proposed probable subjects of license specification and will document the applicant's commitments, licensing specifications, and conditions. NRC will be able to determine if DOE has satisfied the requirements for a construction authorization for a geologic repository for high-level waste disposal at Yucca Mountain, Nevada, only after the NRC staff has completed all volumes of the SER. A decision to authorize construction will be effective only after a full public hearing has been conducted and the Commission has completed its review under 10 CFR 2.1023.

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[1]DOE. 2008. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.
[2]DOE. 2009. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.
[3]NRC. 2003. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev 2. Washington, DC: NRC

Performance Objectives for the Geologic Repository Operations Area through Permanent Closure

To decide whether to authorize construction of a geologic repository at the Yucca Mountain site, NRC must determine whether there is reasonable assurance that the types and amounts of radioactive material described in the license application can be received and possessed in a geologic repository operations area of the design proposed by DOE without unreasonable risk to health and safety of the public. In arriving at that determination, NRC must consider, among other things, whether the site and design comply with the performance objectives and requirements contained in NRC's regulations at 10 CFR Part 63, Subparts E and K.

NRC expects DOE to demonstrate compliance with five pre closure performance objectives:

(1)Operation of a geologic repository must provide protection against radiation exposures and releases of radioactive materials. First, and foremost, NRC must be satisfied that DOE's proposed geologic repository operations area will comply with the Preclosure Public Health and Environmental Standards set forth in Subpart K. DOE must also show that the geologic repository operations area will meet the same radiation protection requirements applicable to other NRC-licensed facilities, as specified at 10 CFR Part 20.

(2) Second, the geologic repository operations area must be designed to ensure that these dose limits will be met during normal operations and during any event expected to occur at least once before permanent closure. For less likely events (those having at least one chance in ten thousand of occurring before permanent closure)the design must show compliance with higher, specified limits.

(3) Third, DOE must show compliance with these quantitative limits using a particular type of quantitative analysis called a pre closure safety analysis or PCSA. This systematic set of projections and calculations must meet specific requirements laid out in NRC's regulations.

(4) Fourth, the geologic repository operations area must be designed to permit implementation of a performance confirmation program that complies with specific NRC requirements for monitoring, laboratory and field testing and on-site experiments. This program must provide data that indicates whether actual conditions during construction differ from limits assumed in the licensing review; and whether natural and engineered systems and components required for repository operation are, indeed, functioning as intended.

(5) Finally, the geologic repository operations area must be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and thereafter until the completion of the performance confirmation program and Commission review of information obtained from it.

Input to Executive Summary

Note: The key aspects of review in Volume 2 are focused on preclosure safety analysis, retrievability, and decommissioning. The preclosure safety analysis includes site and facility information; hazard and initiating event identification; event sequence identification and categorization; consequence analysis; identification of structures, systems, and components important to safety; design information on structures, systems, and components important to safety; and ALARA considerations. The relevant regulatory requirements are 10 CFR 63.111 for performance objectives and 10 CFR 63.112(a)–(f) for preclosure safety analysis.

Site Description as it Pertains to Preclosure Safety Analysis

NRC requires that the site descriptions as they pertains to preclosure safety analysis (PCSA) and the geologic repository operations area (GROA) design identify GROA location. The site descriptions must include 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. The data is to be pertaining to the Yucca Mountain site and surrounding region, to the extent necessary, to identify naturally occurring and human-induced hazards at the GROA. The applicant provided descriptions of the following aspects of the site to conduct the PCSA and design the GROA: (i) site stratigraphy; (ii) regional demography; (iii) local meteorology and regional climatology; (iv) regional and local surface and ground-water hydrology; (v) site geology and seismology, including geoengineering properties that are relevant to design of surface and subsurface facilities; (vi) igneous activity; (vii) site geomorphology; (viii) site geochemistry; and (ix) land use, structures and facilities, and residual radioactivity within the entire land withdrawal area.

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Description of Structures, Systems, Components, and Operational Activities as it pertains to Preclosure Safety Analysis

NRC regulations, as stated in 10 CFR 63.112(a), stipulate that the applicant must provide in its PCSA a general description of the structures, systems, components (SSCs), equipment, and process activities at the GROA. The applicant provided GROA information required to perform PCSA and design of the GROA. This information includes (1) description of location of both surface and subsurface facilities and their functions; (2) description and discussion of the design of SSCs and safety controls (SCs); (3) civil and structural systems; (4) mechanical systems; (5)

electrical power systems; (6) heating, ventilation, and air conditioning (HVAC) systems; (7) radiation and radiological monitoring systems; (8) types of radioactive waste; (9) waste containers; (10) instrumentation and communication systems; and (11) facility operational processes.

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Identification of Hazards and Initiating Events

10 CFR 63.112(b) specifies that the applicant must identify and systematically analyze naturally occurring and human-induced hazards at the GROA. In addition, 10 CFR 63.112(d) specifies that the applicant must provide technical basis for either inclusion or exclusion of a specific hazard.

Based on the specific nature of external (both naturally occurring and human-induced) and internal (operational) hazards, the applicant used different approaches to identify the initiating events that can potentially lead to event sequences. The applicant developed a list of 89 external hazards based on past licensing experience with different nuclear facilities and also conducted a study to identify hazards specific to the subsurface facilities at the repository. The applicant evaluated these hazards using criteria regarding applicability, frequency and rate of occurrence, and associated consequences at the repository. Based on the screening analysis, the applicant concluded that seismic events and loss of power have the potential to initiate event sequences at the repository during the preclosure period.

For operation-related hazards (internal initiating events), the applicant used Master-Logic Diagram and Hazard and Operability techniques to identify them from equipment failures and human actions. The applicant screened the internal initiating events from further analysis after grouping them and placing them in event sequence diagrams using engineering analysis to determine that either (i) the event could not occur, (ii) the initiating event bounded by another event, or, (iii) the probability of occurrence is too low to become a credible hazard. The applicant quantified the initiating events by direct probability assignments and fault tree analysis. As a part of fault tree quantification, the applicant developed the failure probability of individual components or individual pieces of equipment and included equipment and component failures as basic events. The applicant also developed human failure events and included these failures as basic events in fault trees.

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(b)(5) In screening individual internal initiating events involving drops and collisions, the applicant included system information related to specific equipment operating modes (b)(5) (b)(5) and considered the operating environment including maintenance and testing. The applicant employed the probabilistic fire risk assessment methodology, originally developed for facilities handling hazardous materials, to identify fire-related initiating events. The applicant used data for ignition frequency from historical information at either radioactive material handling facilities or other industrial facilities. (b)(5)

The applicant screened out internal flooding because of design of the surface facilities such that important to safety (ITS) equipment would not be wetted or submerged. Additionally, a waste form container exposed to water will not lose its structural integrity or shielding capability. The applicant screened out all criticality-related initiating events during the preclosure period by conducting sensitivity analyses of seven important to criticality parameters. For each of these parameters, the applicant determined the impact of their variations on reactivity. These deterministic screening evaluations were performed for commercial spent nuclear fuel (SNF) and a representative fuel from each of the DOE SNF groups to cover the possible operational conditions to which the waste form may be exposed at the GROA.

The applicant used the fault tree approach for assessing initiating events at the subsurface facilities (b)(5) The applicant used the system information and the information on the operating environment to construct the fault tree model to screen out initiating events at the subsurface facilities. The applicant also used the fault tree analysis for direct exposure-related initiating events. The applicant used a nine-step approach for conducting human failure analysis (b)(5) It quantified the human failure events (b)(5) (b)(5)

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(b)(5)

Identification of Event Sequences

10CFR Part 63.112(b) and 63.111(c) require the applicant to perform a PCSA and identify event sequences. The applicant developed event sequences based on the initiating events it

identified and considered failure of associated combination of SSCs that can lead to radiological dose to public or workers. The applicant 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 propagates through the facility. For quantification of event sequences, the applicant used event tree analysis modeling sequence of responses of SSCs providing containment, shielding, confinement, and criticality control functions as the pivotal events and relied on the active and passive SSCs for prevention or mitigation of event sequences. In the applicant's methodology, event sequences are quantified by aggregation of event sequences based on the types of waste form and operations.

The applicant determined passive reliability of canisters and waste packages for containment and cask systems for shielding under structural and thermal challenges while reliability of active systems such as HVAC systems for confinement and moderator intrusion control systems for criticality control were determined using fault tree analyses. For seismically initiated event sequences, the applicant determined the probability of failure of structural systems and equipment by convolving the fragility curves of the SSCs and site specific hazard curve.

The event sequences resulted in end states that consisted of exposure to workers either direct with no mitigation, or because of loss or degradation of shielding, filtered and unfiltered radionuclide release, and releases from criticality consequences. The applicant relied on reliability of SSCs, interlock and control systems, and procedural safety controls (PSCs) and determined there are no Category 1 event sequences precluding any worker dose evaluation. Limited Category 2 event sequences involved filtered and unfiltered releases from breach of canisters, SNF assemblies, and transportation cask with uncanistered SNF resulted mostly from operational event sequences and some from seismic event sequences. Category 1 event sequences include SSC 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 applicant determined most event sequences as beyond Category 2 requiring no dose consequence evaluation.

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Consequence Analysis

The regulatory requirements of 10 CFR 63.111(a)–(c) specify the dose objectives for normal operations and Category 1 and Category 2 event sequences. 10 CFR 63.204 requires that, during the normal operation, the public in the general environment as defined in 10 CFR 63.202

will not receive an annual dose exceeding 0.15 mSv [15 mrem] from combined management and storage (defined in 40 CFR 191.2) and storage (defined in 10 CFR 63.202) of radioactive material inside the Yucca Mountain repository.

The applicant 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, the applicant 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. Individuals located on the south and west boundaries are considered members of the public in the general environment. The general environment is the area that allows public access and is subject to the 0.15 mSv/year [15 mrem/year] limit in 10 CFR 63.111(a)(2) for normal operations and 0.05 Sv [5 rem] exposure for any single Category 2 event sequence per 10 CFR 63.111(b)(2). For the areas north and east of the site boundary, which are areas controlled by the Nevada Test Site and the Nevada Test and Training Range, access by general members of the public is restricted. These areas are referred to as "offsite but not in the general environment." Individuals working in these areas are subject to the 1 mSv/year [100 mrem/year] public dose limit in 10 CFR 20.1301 for normal operations and 0.05 Sv [5 rem] for any single Category 2 event sequence as required in 10 CFR 63.111(b)(2).

The applicant performed dose calculations for radiation workers and the onsite and offsite members of the public. Because the applicant did not identified Category 1 event sequences, potential radiation worker dose calculation included only those related to normal operations. The applicant used computer codes MCNP and SCALE for dose calculations. The applicant used the GENII Gaussian statistical model, which implemented dosimetry models recommended by the International Commission on Radiological Protection to assess internal exposures. The applicant provided description of the kind, amount, and specifications of the radioactive material to be received and possessed at the GROA. In addition, the applicant discussed the source term release inputs, the material at risk, the damage ratio for fuel release, the release and respirable fractions, and leak path factors for normal operations and Category 1 and Category 2 event sequences. The applicant used the methodology and input data discussed above for consequence analyses.

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

NRC regulations stipulate that the applicant must perform PCSA and the PCSA must include an analysis to identify SSCs ITS and measures taken to ensure the availability of safety systems. SSCs are identified as ITS if they are relied on to provide assurance that HLW operations will not exceed the performance requirements of 10 CFR 63.111(b). The applicant used four criteria

to identify ITS SSCs. A SSC is ITS if it is required to satisfy one or more of the following criteria: (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, or (iv) perform dose mitigation or criticality safety control functions.

In performing the analysis to identify ITS SSCs, the NRC, in 10 CFR 63.112(e)(1) through 63.112(e)(13), requires the applicant to consider 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 applicant presented the ITS SSCs it identified through PCSA using the four criteria discussed above. Along with these ITS SSCs, the applicant also included nuclear safety design bases for the ITS SSCs. For each ITS SSC, the applicant also presented a representative event sequence that requires this ITS SSC to either reduce its frequencies or perform dose mitigation function.

The applicant 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. The applicant does not rely on radioactive contamination disposal monitoring and control systems to reduce event sequence frequency or mitigate dose consequence and, therefore, these systems are not ITS. The goal of the radiation monitoring and alarm system is to monitor the radiation releases in surface and subsurface areas, and to provide alarms and radiation level indication to personnel. The applicant determined that the radiation monitoring and alarm system is not 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 applicant 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 and it is a part of the as low as is reasonably achievable (ALARA) program. Furthermore, the applicant identified administrative procedures and interlocks and other controls on shield doors to provide access control to 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. The applicant 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 criticality concern, the applicant conducted a separate criticality initiating events and event sequence analysis for the preclosure operations. The applicant evaluated seven parameters to determine whether they need to be controlled to prevent criticality during the preclosure period. Though this analysis, the applicant identified the SSCs relied on to maintain subcriticality by preventing moderator from contacting the fissile materials as ITS. Additionally, to provide moderator control, the applicant identified the double-interlock preaction sprinkler systems as ITS to protect moderator-controlled areas where spurious water delivery is not desirable.

The applicant estimated the reliability of the ITS SSCs and preclosure PSCs to perform their intended safety functions to prevent and mitigate potential event sequences on the basis of reliability assessments as part of PCSA. To provide additional assurance that the ITS SSCs will perform their safety functions to an appropriate level of reliability, the applicant plans to qualify these ITS SSCs for the range of environmental conditions anticipated at the time of functional demand. The applicant also commits to develop reliability-centered maintenance, inspection, and testing programs for the ITS SSCs, as necessary, to ensure their continued functioning and readiness. These programs will be reviewed by the NRC staff prior to the applicant's receipt and possession of radioactive waste.

The applicant considered redundancy in the design of ITS SSCs, as appropriate, to meet design basis reliability demands by either following codes and standards that include redundant design features or including diversity for achieving the same safety functions. To ensure the reliable and timely emergency power supply to ITS SSCs that require electrical power to support the intended safety functions, the applicant plans to use standby and ITS emergency diesel generators. The applicant identified the ITS emergency diesel generators and feeds, up to and including ITS loads in the ITS power subsystem as ITS. The applicant evaluated radioactive waste and effluents control in its PCSA. The applicant determined that the releases of radioactive waste and effluents generated during operations will not cause exposures exceeding the regulatory limits specified in 10 CFR 63.111(b).

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(b)(5)

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

10 CFR 63.21(c)(2) requires the applicant to provide information for codes and standards that the applicant proposes to apply to the GROA design and construction. 10 CFR 63.21(c)(3) requires the applicant to describe and discuss the design, including (i) dimensions, materials properties, specifications, and analytical and design methods along with any applicable codes and standards. 10 CFR 63.21(c)(3) and 63.112(f) require a description and discussion of the GROA design, including (i) the relationship between design criteria and the preclosure

performance objectives specified at 10 CFR 63.111(a) and 63.111(b) and (ii) the design bases and their relation to the design criteria.

For the surface structural and civil facilities, the applicant provided design information on the ITS surface waste-handling facilities, the aging facility, and flood control features. The applicant determined that seismic loading bounds the design of the ITS surface buildings. The applicant performed structural analyses of the superstructure and the foundations of the surface structural facilities to protect ITS SSCs inside the buildings from loads induced by external hazards. The applicant 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 protection against aging overpack tipover and sliding. The applicant performed probable maximum flood (PMF) and flood inundation analyses for the proposed flood control features and provided design information for the proposed flood control features for preventing inundation of the surface facilities from a PMF at the GROA. The applicant provided design and design analyses information on the ITS mechanical systems, HVAC systems, transportation systems used to move HLW, electrical power systems, instrumentation and control systems, fire protection systems, waste packages and canister systems to be used in the GROA.

(b)(5)

As Low As Is Reasonably Achievable For Category 1 Sequences

The NRC staff reviewed the applicant's description of its proposed Radiation Protection Program (RPP) to verify that the RPP reflects as low as is reasonably achievable (ALARA) considerations of maintaining the occupational doses to workers, and doses to members of the public, to as far below regulatory limits as is practical, consistent with the purpose for which the licensed activity is undertaken. The applicant's description included the following: (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 area, 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.

(b)(5)

Plans for Retrieval and Alternate Storage of Radioactive Wastes

NRC regulations 10 CFR 63.21(c)(7) stipulate that the applicant must provide a description of plans for retrieval and alternate storage of the radioactive wastes should retrieval be necessary. NRC regulations 10 CFR 63.111(e) also require that the applicant must design its GROA to preserve the option of waste retrieval. The applicant described its plans for retrieval and alternate storage of the radioactive wastes, including a discussion of compliance with the preclosure performance objectives. The applicant, in its description of its alternate storage plan, identified a proposed alternate storage site for a facility, including the location, size, and storage operations. The applicant also provided a schedule for retrieval operations, as required by the regulation. The applicant 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.

(b)(5)

Permanent Closure and Decontamination

NRC requires the applicant to describe the design considerations that are intended to facilitate decontamination and dismantlement (PCDDD) of surface facilities and include information on plans for PCDDD of surface facilities. The applicant provided design considerations to facilitate PCDDD and a planning timeline for decontamination and dismantlement. (b)(5)

(b)(5)

VOL 2 - INTRODUCTION

On June 3, 2008, the U.S. Department of Energy (DOE or the applicant) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for authorization to begin construction of a geologic repository for high-level radioactive waste disposal at Yucca Mountain, Nevada. The license application consists of general information and a Safety Analysis Report (SAR). This document, the NRC staff's Safety Evaluation Report (SER), Volume 2, summarizes the results of the staff's review of the SAR the applicant provided in its June 3, 2008, license application and as updated on February 19, 2009. In particular, this SER Volume 2 documents the results of staff's evaluation to determine whether the design of the proposed geologic repository operations area (GROA) for Yucca Mountain will comply with the performance objectives and requirements that apply before the repository is permanently closed. These performance objectives and requirements can be found in NRC's regulations at 10 CFR Part 63, Subparts E and K.

Other portions of the NRC staff's safety review have been, or will be, documented elsewhere. SER Volume 1, NUREG–1949 (NRC, 2010XX) contains the results of the staff's review of the applicant's General Information. SER Volume 3 contains the results of the staff's review and evaluation of the applicant's compliance with postclosure performance objectives and requirements; SER Volume 4 will document, the results of the staff's review and evaluation of applicant's demonstration of compliance with administrative and programmatic requirements; and SER Volume 5 will document commitments by the applicant and potential licensing specifications and conditions. In conducting its review of the entire license application, the NRC staff was guided by the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan (YMRP)[1].

Preclosure Performance Objectives and Requirements

To decide whether to authorize construction of surface and subsurface facilities of a geologic repository at the Yucca Mountain site, the NRC must determine, among other things, whether there is reasonable assurance that the types, amounts, and specifications of radioactive material described in the license application can be received, possessed, and disposed at the facility without unreasonable risk to the health and safety of workers and the public. In arriving at that determination, the NRC staff must consider, among other things, whether the site and design comply with the performance objectives and requirements contained in 10 CFR Part 63, Subparts E and K, of NRC's regulations. As per the preclosure performance objectives, the applicant must perform a preclosure safety analysis (PCSA) to identify and categorize event sequences and identify structures, systems, and components (SSCs) important to safety (ITS) and associated design bases and criteria. An event sequence is a series of actions and/or occurrences within the natural and engineered components of the facility that could potentially lead to exposure of individuals to radiation. The applicant's PCSA must demonstrate that the repository as proposed to be designed, constructed, and operated will meet the specified exposure limits throughout the preclosure period. The applicant must also demonstrate that the design of GROA will not preclude retrievability of the wastes, in whole or in part, from the underground facility where these wastes will be emplaced for permanent disposal.

Risk-Informed, Performance-Based Review

[1]NRC. 2003. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

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The PCSA quantifies performance of GROA as a means of demonstrating compliance with the preclosure performance objectives at 10 CFR 63.112. PCSA is a systematic analysis that must answer three basic questions used to define risk: (i) what can go wrong?; (ii) how likely is it?; and, (iii) what are the consequences? The applicant's PSCA includes a number of evaluations such as: (i) identification of hazards and initiating event sequences;(ii) development and categorization of event sequences; (iii) failure mode and reliability assessments of structures, systems, and components; and (iv) their fragility assessments. Because the PCSA encompasses a broad range of issues, the NRC staff used risk information throughout the review process to ensure that the review focused on significant items that affect preclosure performance. Section 2.1.1 of the YMRP provides guidance as to how the NRC staff must apply risk information throughout the review of the PCSA.

Applicant's Preclosure Safety Analysis

In order to answer the three questions related to risk, the applicant conduced a PCSA to systematically examine the site, the design, and the potential hazards, initiating events, and event sequences and their radiological safety consequences. The analysis identified structures, systems and components important to safety. The applicant included the PCSA as an integral part of its repository design process. The NRC staff focused its review to determine whether or not the applicant's PCSA contains sufficient information to satisfy the regulatory requirements and a clear link is established between design bases and design criteria. The focus of review of various chapters of this volume of the SER is summarized below.

Site Description as it Pertains to Preclosure Safety Analysis (SER Section 2.1.1.1)

This SER section provides the NRC staff's evaluation of the site description information that the applicant used to identify natural and human-induced hazards with the emphasis on those features, events, and processes that might affect the design of the GROA and conduct of the PCSA. Features such as 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 man-made sources of radiation can influence design of GROA and may be important parameters for PCSA consideration.

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

This SER section provides the NRC staff's evaluation of the applicant's description of SSCs, safety controls, equipment, and operational process activities in surface and subsurface facilities. More specifically, the NRC staff evaluates (i) civil and structural systems; (ii) mechanical systems; (iii) electrical power systems; (iv) heating, ventilation, and air conditioning systems; (v) radiation/radiological monitoring systems; (vi) types of radioactive waste; (vii) waste containers; and (viii) operation of the facilities. In the evaluation, the NRC staff gives more emphasis on the ITS SSCs and risk-significant operation processes involving radioactive waste handling.

Identification of Hazards and Initiating Events (SER Section 2.1.1.3)

This SER section provides the NRC staff's evaluation of 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 SER section assesses the probability of the potential hazards taking into account the 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 (SER Section 2.1.1.4)

This SER section provides the NRC staff's evaluation of the information on identification of event sequences to determine compliance with the regulatory requirements for 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 SER section also assesses the methodology for event sequence development and categorization and reliability of SSCs.

Consequence Analysis (SER Section 2.1.1.5)

This SER section provides the NRC staff's evaluation of the consequence analysis the applicant conducted to support its PCSA. The NRC staff focuses its review on the (i) dose calculation methodology, (ii) atmospheric dispersion determination, (iii) assumptions and input parameter, (iv) source terms, (v) methodology for the worker and public dose determination, and (vi) demonstration compliance with 10 CFR 63.111(a)–(c) and 63.204.

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

This SER section provides the NRC staff's evaluation of identification of SSCs ITS and procedural safety controls for reducing event sequences or mitigating dose consequences. In addition, the NRC staff evaluates the applicant's consideration of the following aspects in its PCSA: (i) means to limit the radioactive material concentration in air; (ii) the means to limit the time required to perform work near the radioactive materials; (iii) shielding protection; (iv) monitoring and controlling dispersal of radioactive contamination; (v) access control to high radiation or airborne radioactivity areas; (vi) criticality control and prevention; (vii) radiation alarm; (viii) ability of SSCs ITS to perform their intended safety functions; (ix) fire detection and suppression; (x) radioactive waste and effluent controls; (xi) the means to provide timely and reliable emergency power; (xii) redundant systems; and (xiii) SSCs ITS inspection, testing, and maintenance. This SER section assesses the criteria the applicant developed for identification of SSCs ITS and procedural safety controls. The NRC staff evaluates the nuclear safety design basis requirements the applicant developed for the SSCs ITS from the PCSA event sequence analyses.

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

This SER section provides the NRC staff's evaluation of the information for the SSCs ITS design, construction, and operation. This evaluation includes (i) information relative to the codes and standards for design and construction of the GROA, (ii) design methodologies, and (iii) design analysis. The NRC staff also evaluates the relationship between the proposed

design criteria and the GROA performance requirements, and the relationship between the design bases and the design criteria.

As Low As Is Reasonably Achievable For Category 1 Sequences (SER Section 2.1.1.8)

This SER section provides the NRC staff's evaluation of the Radiation Protection Program (RPP) the applicant proposed to verify that the RPP reflects ALARA considerations of maintaining the occupational doses to workers, and doses to members of the public, to as far below regulatory limits as is practical, consistent with the purpose for which the licensed activity is undertaken. The 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 area, 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.

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

NRC regulations require that the applicant must provide a description of plans for retrieval and alternate storage of the radioactive wastes should retrieval be necessary and design its GROA to preserve the option of waste retrieval. This SER section provides the NRC staff's evaluation of applicant's retrieval plan 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. The applicant's description of 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 and Decontamination (SER Section 2.1.3)

NRC regulations require the applicant to provide information concerning plans for permanent closure and plans for the decontamination or decontamination and dismantlement (PCDDD) of the surface facilities. NRC regulations further require the applicant to describe the design considerations that are intended to facilitate PCDDD of surface facilities. In the SER section, the NRC staff evaluates (i) the design conditions to facilitate PCDDD and (ii) 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.

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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 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 sufficient to identify 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 the applicant 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. The applicant 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 Regulatory Requirements and Regulatory Guidance

The regulatory requirements for site description as they pertain to the preclosure period are defined in 10 CFR 63.21(c)(1)(i–iii) and 10 CFR 63.112(c). Specifically, the SAR must 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.

The information 10 CFR 63.21 requires includes the GROA location, geology, hydrology (both surface and groundwater), geochemistry, geomechanical properties and conditions of soil and rock, and meteorology and climatic conditions. The same site information is used to demonstrate compliance with 10 CFR 63.112(c), which requires the applicant 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.

The NRC staff reviewed the applicant's site information using the guidelines identified in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.1 (NRC, 2003aa). NRC staff 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 Staff Review and Analysis

The applicant stated that to demonstrate compliance with the regulatory requirements of 10 CFR 63.21(c)(1)(i–iii) and 10 CFR 63.112(c), it used the regulatory acceptance criteria in YMRP Section 2.1.1.1. NRC staff organized its evaluation of the applicant's site description generally following the YMRP outline. NRC staff focused on verifying that the site information is sufficient 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, the applicant 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. 10 CFR 63.21(c)(1)(i) requires the applicant to provide the location of the GROA with respect to the site boundary. The applicant also provided site geographic information used to identify naturally occurring and human-induced hazards, as 10 CFR 63.112(c) requires. The applicant'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 Safety Evaluation Report (SER) 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 the licensee will control. The applicant 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, the applicant 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), the applicant committed to 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) in a future license application update, because the approximately 0.8 km^2 [200 acres] are private property and are 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. The applicant also committed to 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 the applicant's description of the repository boundaries in SAR Section 1.1.1.1 and in the RAI responses using the guidance in the YMRP.

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(b)(5) SAR Figures 1.1-1, 1.1-2, and 1.1-3 clearly delineate the boundaries and are of sufficient detail and scale to permit review of the site boundary and the preclosure controlled area, access points, the general environment, and a detailed representation of the surface GROA, including its phased development.

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Natural Features

In SAR Section 1.1.1.2, the applicant 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), the applicant 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 the applicant's descriptions of the natural features in SAR Section 1.1.1.2 using the guidance in the YMRP. (b)(5)

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Man-Made Features

In SAR Section 1.1.1.3, the applicant 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 the applicant'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), the applicant 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. (b)(5)

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NRC Staff Conclusion

On the basis of NRC staff review of SAR Section 1.1.1 described previously, NRC staff finds that the applicant adequately described information regarding Yucca Mountain site geography, (including site boundaries, location of natural and man-made features, and relevant facilities outside of the GROA) and provided a basis for PCSA and GROA design. NRC staff finds, with reasonable assurance, that the applicant's information regarding site geography is acceptable for use in other SAR sections to develop design bases and to perform PCSA, and complies with the regulatory requirements of 10 CFR 63.21(c)(1)(i) and 10 CFR 63.112(c).

2.1.1.1.3.2 Regional Demography

To demonstrate compliance with 10 CFR 63.21(c)(1), the applicant described the regional demography in SAR Section 1.1.2. This information is used to determine the location of the real members of the public and to support a PCSA and design of the GROA. The applicant provided the population distribution in the demographic study area it established based upon NRC Regulatory Guide 4.2 (NRC, 1976aa). The applicant also described the population locations, regional population centers, and population projections for the preclosure operations period (2017–2067) under consideration.

Demographic Study Area

The applicant 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.

The applicant 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 the applicant estimated the 2003 resident population located in each study area grid cell and presented these estimates in SAR Table 1.1-2 and Figure 1.1-11. This information is the baseline population distribution within the 84-km [52-mi] grid and that the applicant 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 the applicant's demographic data and its methodology to establish the demographic study area using the guidance in the YMRP. (b)(5)

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Population Centers

In SAR Section 1.1.2.2, the applicant listed the nearby Nevada population centers: Boulder City, Henderson, Las Vegas, Mesquite, and North Las Vegas in Clark County; Caliente, Alamo, Panaca, and Pioche in Lincoln County; Beatty, Gabbs, Manhattan, Pahrump, Round Mountain, Tonopah, and the town of Amargosa Valley in Nye County; and Goldfield and Silver Peak in Esmeralda County. The nearby California population centers are Bishop and Death Valley National Park in Inyo County. The closest large population center to the GROA 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 the applicant's information pertaining to population centers near the Yucca Mountain Repository. (b)(5)

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Population Projections

The applicant 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 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 the Center for Business and Economic Research made for the period of 2003–2035; an assumed constant average growth rate of 1.08 percent was used from 2035–2067 on the basis of constant growth rate between 2032 and 2035 (BSC, 2007bz).

The applicant based the annual rate of change for Inyo County in California on Inyo County population projections from 2000–2050 the Demographic Research Unit of the California State Department of Finance made (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 a constant average decline rate of 0.6 percent was used from 2050–2060; and an assumed no-change was applied from 2060–2067 ~~based on~~on the basis of the assumption that no decline in population is expected beyond 2060.

The applicant also estimated projected populations in Nye and Clark Counties due to construction and operation of the proposed repository and 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). (b)(5)

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(b)(5) 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. The applicant 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, no projection estimates were performed.

NRC Staff Evaluation: The NRC staff reviewed the applicant's data, assumptions, and methodology used for the population distribution projections within 84 km [52 mi] of the GROA

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NRC Staff Conclusion

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2.1.1.1.3.3 Local Meteorology and Regional Climatology

To address the requirements of 10 CFR 63.21(c)(1)(iii) and 10 CFR 63.112(c) with respect to meteorology and climatology, the applicant described local meteorology and regional climatology conditions that could pose hazards to GROA facilities or repository safety during the preclosure period. This information, presented in SAR Section 1.1.3, is used to develop design bases for structures, systems, and components (SSCs) at the site. Atmospheric conditions, such as atmospheric stability categories, average wind speeds, and prevailing wind direction, are also described in SAR Section 1.1.3. The applicant used this information to evaluate the consequences of airborne radionuclide transport in hypothetical preclosure release scenarios.

Data Summaries and Collection Techniques

The applicant set up 12 meteorological monitoring stations to characterize the site meteorological conditions. The applicant 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. The applicant 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 wind speed and direction, temperature, humidity, and precipitation. The information collected from 1994–2006 was provided in the applicant's report on local meteorology of Yucca Mountain (BSC, 2007bs) and summarized in the SAR. The summaries included mean monthly values as well as observed precipitation and temperature extremes. The applicant 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). The applicant also described the data reduction techniques used to calculate atmospheric stability and classify wind speed 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 the applicant's data summaries and collection techniques using the guidance in YMRP Section 2.1.1.1.3 (b)(5)

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Annual and Probable Maximum Precipitation

The applicant 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. The applicant 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. The applicant provided the maximum 24-hour precipitation totals for September 21 through 22, 2007, which the applicant described as the largest precipitation event reported at the site, in SAR Table 1.1-23; the largest reported 24-hour precipitation total among the 12 stations was 87.1 mm [3.4 in].

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, 2007ab), the applicant 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, the applicant estimated values of the probable maximum precipitation to be 335 mm [13.2 in] for a 6-hour storm event for the basins encompassing the North Portal pad and 328 mm [12.9 in] for the basins encompassing the South Portal pad. For comparison, these 6-hour totals are approximately 3.8 times larger than the largest reported 24-hour precipitation total observed at any Yucca Mountain precipitation monitoring station.

To characterize snowfall at the site, the applicant 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. The applicant 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]. The applicant 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).

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NRC Staff Evaluation: The NRC staff reviewed the applicant's precipitation information using the guidance in the YMRP/(b)(5)

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Severe Weather

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

Tornadoes: The applicant 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).

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The applicant followed procedures described in NRC Regulatory Guide 1.76 (NRC, 1976ab) to develop design basis tornado characteristics, except that, as allowed by NRC Regulatory Guide 1.76, the applicant used more extreme tornado characteristics to describe the design basis tornado. The applicant used information in NUREG/CR–4461 (Ramsdell and Andrews, 1986aa) to develop the alternative design basis tornado characteristics. The applicant established design basis tornado parameters including a wind speed of 304 km/hr [189 mph], a pressure drop of 5.6 kPa [0.81 psi], and a rate of pressure drop of 2.1 kPa/s [0.3 psi/s] (SAR Section 1. 1.3.6.1). For comparison, NRC Regulatory Guide 1.76 Table 1 (NRC, 1976ab) recommends using a wind speed 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] as appropriate 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 the applicant's information using the guidance in the YMRP. (b)(5)
(b)(The applicant used appropriate site data to determine that tornadoes have been observed within Nye County. (b)(5)
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Extreme Winds: The applicant considered hazards arising from extreme winds by using a probabilistic approach applied to measured wind speeds to estimate a maximum 3-second gust

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straight wind of 193 km/hr [120 mph] at a million-year recurrence interval. The applicant used wind speeds 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). The applicant 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. The applicant concluded that the design basis tornado {including a wind speed 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 the applicant's assessment of hazards related to extreme winds in SER Section 2.1.1.3.3.1.2.3 using the guidance in the YMRP/(b)(5)

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Lightning: The applicant considered lightning strikes in evaluating external hazards that might initiate an event sequence at the repository. The applicant 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. The applicant indicated that these observations are generally consistent with other estimates for southern Nevada. On the basis of this information, the applicant determined that a direct lightning strike is an initiating event at the repository (SAR Section 1.6.3.4.6) and the applicant provided lightning protection in the design of facilities in the GROA.

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NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidance in the YMRP/(b)(5)

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Hurricanes: In addition to extreme winds from hurricanes as discussed previously, the applicant considered external flooding as a result of hurricanes in evaluating external hazards. The applicant 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.

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NRC Staff Evaluation: (b)(5)

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Other Severe Weather: The applicant also considered sandstorms and dust storms at the site. The applicant 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, the applicant 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.

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NRC Staff Evaluation: (b)(5)

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NRC Staff Conclusion

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2.1.1.1.3.4 Regional and Local Surface and Groundwater Hydrology

To demonstrate compliance with 10 CFR 63.21(c)(1)(iii) and 10 CFR 63.112(c), the applicant 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, the applicant estimated the probable maximum flood resulting from the probable maximum precipitation (see SER Section 2.1.1.1.3.3) to determine the design bases for the flood protection structures the applicant plans to construct to protect the GROA from runoff and debris flows. NRC staff evaluated the applicant's hydrological information presented in SAR Section 1.1.4.

Surface and Groundwater Hydrologic Features

The applicant provided information pertaining to the regional and local surface and groundwater hydrology. The applicant charactenzed 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. The applicant summarized the flooding history in the Yucca Mountain area on the basis of both literature reviews and actual stream gauging records, as described in BSC, Section 3.4.3 (2004bj). The applicant characterized the regional groundwater flow as occurring in an asymmetric radial-flow pattern, flowing from recharge areas in mountains and other highlands toward Death Valley (SAR Section 1.1.4.2).

The 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), the applicant 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). The applicant 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 the applicant's description of surface and groundwater hydrologic features using the guidance in the YMRP/ (b)(5)

(b)(5)

Probable Maximum Flood

In SAR Section 1.1.4.3, the applicant provided a flood inundation analysis. The applicant analyzed probable maximum flooding to assess the performance of preventive measures such as dikes and channels around critical facilities to control run on. The applicant's analyses considered a surface facility design with and without a flood-inundation mitigation measure. The applicant 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). The applicant 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, the applicant used HEC-RAS to determine the magnitude and duration of runoff that would occur during the HEC-1-predicted probable maximum flood event.

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

Subarea Hydraulic Properties: The applicant used a unit hydrograph method to develop a runoff hydrograph. Determination of a unit hydrograph requires subarea size and time of concentration. The applicant 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. The applicant assumed a uniform infiltration rate of 38.1 mm/hr [1.5 in/hr], which is lower than the lowest infiltration values obtained from 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).

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(b)(5)

Comment: (b)(5)
(b)(5)

The applicant used a bulking factor to account for increased flow depths caused by the presence of entrained air, debris, and sediment load. The applicant used a bulking factor of 10 percent. In other words, the applicant increased the peak discharges by 10 percent in its probable maximum flood analyses. A literature review by the applicant 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 (b)(5)

(b)(5)

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), the applicant considered the range of Manning's coefficient for three flow conditions: clear water flow, high sediment transport, and mudflow. The applicant used a Manning's coefficient of 0.035 for the clear water channel flow condition, 0.09 for high sediment transport flow, and 0.16 for the mudflow. The values were selected on the basis of calibration studies the applicant provided in DOE (2009bf).

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For probable maximum flood analysis, the applicant considered the amount of clear water runoff that would be large enough so that mudflow condition is unlikely to develop. Results of the applicant's sensitivity study showed that increasing the Manning's coefficient from 0.035 to 0.09 resulted in a 2.4-m [8-ft] increase of predicted water surface elevation near the North Portal pad; increasing Manning's coefficient further from 0.09 to 0.16, however, only resulted in an insignificant increase of 0.15 m [0.5 ft] (DOE, 2009bf). Therefore, the applicant used a Manning's coefficient of 0.09 in its probable maximum flood analysis, corresponding to the high sediment transport flow condition.

NRC Staff Evaluation: The NRC staff reviewed the procedure the applicant used to determine the Manning's coefficient of 0.09 for the probable maximum flood analysis. (b)(5)
(b)(5)

Probable Maximum Flood Rate: The maximum probable maximum flood peak flow rate resulting from the applicant's HEC-1 model is 1,564 m^3/s [55,240 cfs], as shown in BSC Table 7-1 (2007br). (b)(5)
(b)(5)

(b)(5) NRC staff reviewed the application of the analysis of the probable maximum flood from a flood hazard perspective in SER Chapter 2.1.1.7.

NRC Staff Evaluation: (b)(5)
(b)(5)

Comment (b)(5)
(b)(5)

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Application of Probable Maximum Flood Analyses: The applicant 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, the applicant assumed that the planned facilities upstream were not constructed and the flood control measures were not implemented. In SAR Figure 1.1-57, the applicant showed that in the no-mitigation case, the runoff from the probable maximum flood event would inundate the North Portal pad and important to safety (ITS) facilities in the vicinity of the North Portal.

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The applicant'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, in SAR Figure 1.2.2-7 the applicant showed 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: (b)(5)
(b)(5)

(b)(5)

(b)(5) The NRC staff's evaluation of the effects of proposed changes to natural drainage features and flood barriers is provided in SER Chapter 2.1.1.7.

NRC Staff Conclusion

(b)(5)

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

To demonstrate compliance with 10 CFR 63.21(c)(1)(ii), the applicant 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 as required by 10 CFR 63.112(c). 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, the applicant provided the geologic information from its site characterization investigations that was used to support PCSAs and GROA designs. In SAR Section 1.1.5.1, the applicant described the geologic site conditions of the rocks and alluvial deposits (sediments deposited by streams in valleys) on which the proposed surface GROA facilities are proposed to be built and into which waste packages will be placed in the underground (subsurface) GROA. The applicant also identified and described geologic structures, including faults, fractures, and the inclined layering of rocks, and characteristics of the rocks such as the degree of fusion of the rock matrix and relative abundance of lithophysae (voids in the rocks formed by volcanic-gas bubbles), 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.

Stratigraphy of the Subsurface GROA

In SAR Section 1.1.5.1, the applicant 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 will 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), the applicant proposed that the two lithophysal zones in the RHH comprise approximately 85 percent of the waste emplacement area.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of stratigraphy of the subsurface GROA, including the applicant's stratigraphical studies conducted in the Yucca Mountain region (BSC, 2004bi; Sawyer, et al., 1994aa). (b)(5)

(b)(5)

Structural Geology of the Subsurface GROA

In SAR Section 1.1.5.1, the applicant provided information on site structural geology and tectonics to understand past geologic hazards and potential future hazards caused by faulting, seismicity, rockfall, and volcanism. The applicant also provided information on the structural geologic studies it conducted in the Yucca Mountain region (BSC, 2004bi; Day, et al., 1998aa).

(b)(5)

Faulting at the Subsurface GROA

The applicant located and characterized hundreds of faults within a 100-km [62-mi] radius of Yucca Mountain (BSC, 2004bi; Day, et al., 1998aa). The applicant 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. The applicant 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. The applicant determined that displacement of such block-bounding faults could generate the largest displacement and vibratory ground motions (i.e., earthquakes) at the site.

The applicant 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. The applicant 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.

The applicant also described intrablock faults, which are defined as less continuous (i.e., shorter length) and having smaller displacement than the block-bounding faults. The applicant classified the Sundance and Ghost Dance faults, located within the subsurface GROA, as intrablock faults. The applicant observed additional small-displacement faults and shear fractures in the ESF and ECRB. (b)(5)

(b)(5)

The applicant explained 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. The applicant also observed greater extension in the southern portion of the site than in the northern portion. The applicant explained 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, the applicant 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. The applicant explained that the 60-m [197-ft] setback distance it established as design control parameter 01-05 was applied to these two faults as a postclosure criterion. The applicant 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 completion ~~of design of~~ the subsurface GROA design. In particular, the location of the westernmost end-points of emplacement drifts (and therefore, the location and length of emplacement drifts), is dependent upon the location of the west access main, which the applicant proposes to also set back from the Solitario Canyon fault (DOE, 2009bf). The applicant proposed to estimate the setback distance for subsurface

openings on the basis of the locations, strikes, and dips of known faults. This commitment will be confirmed during excavation of the openings.

However, the applicant 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 ~~exceedence~~exceedance probability of 10^{-5} (BSC, 2003aj). In addition to the hazard of direct fault displacement, the applicant determined that faults and their damage zones can disturb drift stability (SAR Section 1.3.4.2.2) and increase rockfall hazard (DOE, 2009bf). The applicant asserted that the 60-m [197-ft] standoff of emplacement drifts from the Solitario Canyon fault is sufficient to mitigate this increased hazard.

The applicant determined that the subsurface GROA will only approach the Solitario Canyon fault (not the Bow Ridge fault) closely enough for this standoff to apply (SAR Figure 2.2-12). Because the proposed subsurface GROA will have its western boundary delimited by the subsurface trace of the Solitario Canyon fault at the level of the subsurface GROA, the component of the GROA that will be closest to the Solitario Canyon fault will be the perimeter access main (DOE, 2009bf). 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].

The applicant expects to encounter faults during drift construction ~~of drifts~~ 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, the applicant 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).

The applicant determined that, for preclosure safety considerations, fault shear displacements of more than 3 m [10 ft] during a 100-year preclosure period have annual ~~exceedence~~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 ~~exceedence~~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 SER Section 2.1.1.1.3.5.5.

Comment: (b)(5)
(b)(5)

NRC Staff Evaluation: NRC staff reviewed the applicant'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. (b)(5)
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(b)(5) 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.

(b)(5)

(b)(5)

The applicant committed to confirm actual fault locations and characteristics (including fault width and displacement), for faults such as the Solitario Canyon fault [Postclosure Design Control 01-05; DOE Section 1.3.1 (2009bf); SAR Section 1.3.4.2.2] and for intrablock faults that are unnamed or undiscovered, 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 ~~Procedural Safety Control (PSC) 25.~~ Table 1.9-10, Procedural Safety Control (PSC)-25]. The applicant also specified a setback distance measurement process for aging pads from the Bow Ridge fault, as outlined in DOE Section 1.1.3 (2009bf). The applicant would then use this new information to confirm the design of the subsurface GROA structures as it pertains to setback distances from faults. (b)(5)

Comment (b)(5)

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(b)(5) Also, the applicant recognized the characterization of intrablock faults as "probably needing administrative control" (SAR Table 5.10-3).

(b)(5)

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 the applicant, ~~When evaluating~~any known or unknown faults encountered during construction (as the applicant proposes in Design Control 01-05 and PSC-25)~~, they~~ 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). (b)(5)

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Fracture Characteristics at the Subsurface GROA

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

The applicant considered rockfall (spallation of tunnel wall rock blocks) and drift degradation (major tunnel collapse) to be "fracture hazards", controlled by aspects of the fracture networks it measured in different RHH zones. The applicant 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 SER Sections 2.1.1.1.3 and 2.1.1.1.4.

The applicant'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.

The applicant described the fractures in the upper lithophysal zone as having a predominantly north- and northwest-striking tectonic orientation, with spacing that ranges from 0.5 to 3 m [1.6 to 9.8 ft] and lengths of less than 3 m [10 ft]. The lower lithophysal zone—the rock layer proposed to contain most of the waste packages—has a few long fractures but many small fractures less than 1 m [3 ft] long, which are steeply dipping and have a spacing of a few centimeters [few inches]. The applicant characterized the middle nonlithophysal zone as a network of long, relatively closely spaced fractures that the applicant separated into four sets on the basis of orientation: two sets are subvertical with northwest 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, the applicant identified three steeply dipping sets, with the most prominent striking northwest. The applicant also identified a northwest-striking, shallowly dipping set among the lower nonlithophysal zone fractures. The applicant indicated high fracture frequencies {19 to 24 fractures per each 3-m [10-ft] interval} in the lower nonlithophysal zone, similar to the intensities in the middle nonlithophysal zone.

The applicant described the zone of influence around faults (DOE, 2009bf). This zone is defined as the region near a fault where fracture intensity is increased or orientation changes.

According to the applicant, the intensity of long fractures {greater than 1 m [3.3 ft]} correlates to rock type, but not to proximity to faults. However, for shorter fractures, the applicant made four general observations on the zone of influence. First, the width of the zone of influence adjacent to a fault ranges from 1 to 7 m [3.3 to 23 ft] 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, the applicant 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 the applicant's information on fractures using the guidance in the YMRP (b)(5)

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(b)(5) The applicant's characterization of the fracture data from the ~~repository host horizon~~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. The applicant committed to confirm that the fracture parameters input to design and performance reflect the actual field observations made during construction of underground openings [10 CFR 63.132(a) and 63.44].

Commitments

- "Emplacement drifts shall be located a minimum of 60 m from a Quaternary fault with potential for significant displacement" ~~[Design Control Parameter~~(SAR Table 1.9-9, DCP 01-05 ~~(DCP) (SAR Table 1.9-9)]~~). "During construction activities (underground) in the vicinity of the Solitario Canyon fault, the location of the fault will be confirmed, and

the condition of the rock near the fault will be examined" ([DOE, 2009bf, Enclosure 1 (2009bf)]). "A construction standoff will then be evaluated on observational data to confirm the design basis" (SAR Section 1.3.4.2.2).

- "The rock condition is to be observed as emplacement drift boring is accomplished. Observed faults are to be specifically evaluated to ensure that conditions cannot credibly lead to a breach of a waste package during preclosure period or a standoff distance from the fault is to be established [PSC-25 (SAR Table 1.9-10), DOE, 2009 as, Enclosure 3]."

- Assess tThe significance of displacement of faults it the applicant expects to encounter will be assessed during drift construction using a variety of methods principally to establish a basis for setback distance [PSC-25; DCP 01-05; Administrative Control for Waste Package Loading, Handling, and Emplacement (SAR Table 5.10-3); DOE Sections 1.2.3 and, 1.3.2 (2009as); and DOE Section 1.6 (2009bg)].

- Observe rRock conditions will be observed as emplacement drift boring is accomplished, including fracture characteristics such as orientation, spacing, length, intensity, and connectivity (PSC-25), in part to ensure emplacement drifts are constructed nominally parallel with the design azimuth (70–80°) (SAR Table 1.9-9, DCP 01-08 and DCP 01-14, both at inSAR Table 1.9-9).

Comment [(b)(5)
(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.5.1.2 Geology of the Surface Geologic Repository Operations Area

In SAR Section 1.1.5.1.4, the applicant provided information on the stratigraphy and structural geology of the surface GROA. The surface GROA facilities would be constructed mainly in Midway Valley on alluvium. However, the applicant 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 The applicant used this information for analyses of potential hazards to the facilities such as earthquakes, surface faulting, landslides, and erosion of and deposition on the surface GROA.

Stratigraphy of the Surface GROA

In SAR Section 1.1.5.1.4, the applicant characterized the near-surface stratigraphy using geologic mapping, boreholes, test pits, trenches, and geophysical investigations (BSC, 2002aa;

SNL, 2008af). The applicant determined that the surface GROA is underlain 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: (b)(5)

(b)(5)

Structural Geology of the Surface GROA

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. The applicant described 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, the applicant 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.

The applicant 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, the applicant indicated that the fault location is consistent with the locations of previously identified faults.

NRC Staff Evaluation: NRC staff reviewed the applicant's faulting information (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5) Using the displacement approach and the earthquake approach (methods described in SER Section 2.1.1.1.3.5.5), the applicant stated that the Bow Ridge Fault would have no impact on ground motion at the surface GROA structures (DOE, 2009as). The applicant explained 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. (b)(5)

(b)(5)

(b)(5)

(b)(5) To further address these faulting uncertainties, the applicant indicated 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). (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

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, the applicant provided its description of site seismology. The applicant described its analysis of potential seismic hazards in SAR Section 1.1.5.2.4, the overall approach to developing a seismic hazard assessment for Yucca Mountain in SAR Section 2.2.2.1, and the conditioning (adaption or modification) of the ground motion hazard for seismic design at Yucca Mountain in SAR Section 1.1.5.2.5.1. Additional information ~~was provided in the~~ applicant'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.

The applicant 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. Conducting an expert elicitation in the late 1990s to develop a PSHA for Yucca Mountain. This assessment included a PFDHA that is discussed in SER 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. Conditioning PSHA ground motion results to constrain the large low-probability ground motions to ground motion levels that, according to the applicant, are more consistent with observed geologic and seismic conditions at Yucca Mountain, as provided in BSC ACN02 (2005aj).

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

The applicant applied these three steps for seismic hazard assessment 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 the applicant'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 SER. The third step involving site response modeling is evaluated in SER Section 2.1.1.1.3.5.3.

PSHA—Methodology

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

Project (DOE, 1997aa). The applicant stated that its PSHA methodology followed the guidance 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, the applicant 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, the applicant convened two panels of experts as described in SAR Section 2.2.2.1.1.1. The first expert panel consisted of six 3-member teams of geologists and geophysicists (seismic source teams) who developed probabilistic distributions to characterize 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 the applicant, the U.S. Geological Survey, and Risk Engineering Inc. (1998aa), ~~who~~ 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 the applicant'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. (b)(5)

(b)(5)

PSHA—Input Data and Interpretations

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

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

NRC Staff Evaluation: The NRC staff reviewed the applicant's PSHA input data and interpretations as described in SAR Sections 1.1.5.2 and 2.2.2.1.1. (b)(5)

(b)(5)

(b)(5)

Conditioning of Ground Motion Hazard

In SAR Section 1.1.5.2.5.1, the applicant 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 (Coraddini, 2003aa), and the applicant itself (BSC, 2004bj) questioned whether the very large ground motions the PSHA predicted at low annual ~~exceedence~~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 the applicant'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 ~~exceedence~~exceedance probability to 10^{-4} or 10^{-5} (typical annual ~~exceedence~~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 ~~exceedence~~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, the applicant 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, the applicant chose to treat the issue as part of the ground response analysis. Accordingly, the applicant'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

The applicant 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 annual probability of exceedance (APE). As the applicant 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 $\sigma_{\ln}$ 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), the applicant conducted conditioning using, in series, the shear-strain-threshold and extreme-stress-drop methods.

NRC Staff Evaluation: The NRC staff reviewed the applicant's methods for conditioning of PSHA results in SAR Section 1.1.5.2.5.1 and the applicant responses to NRC RAIs (DOE, 2009aq) (b)(5)

(b)(5)

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). The applicant 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 empincal 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), the applicant 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 the applicant also clarified and updated the formulations for the two conditioning methods, as outlined in BSC Appendix A (2008bl).

NRC Staff Evaluation: (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.5.3 Seismic Site Response Modeling

The applicant provided information in SAR Section 1.1.5.2.5.2 on how the surface and subsurface GROA might behave if the site was subjected to seismic loads. Seismic site response modeling is the last step in the development of seismic inputs for preclosure seismic design and PCSA. The applicant 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, the applicant provided information in the following areas: (i) site response modeling methodology; (ii) geophysical information to develop compression waves (V_p), 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 the applicant'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, the applicant discussed how the various types and thicknesses of rocks, alluvium, and soils that comprise the GROA and the site would likely respond to earthquake ground motions. The results of site-response modeling included 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). The applicant used the site-specific ground motion curves that are consistent with the conditioned PSHA ground motion hazard curves.

The applicant 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 annual probability of ~~exceedence~~exceedance (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. The applicant 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 the applicant's overall approach to site-response modeling using the guidance of NUREG/CR–6728 and YMRP Section 2.1.1.3.3, Acceptance Criterion 1. (b)(5)

(b)(5)

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

The applicant relied on a 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 the applicant'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 the applicant 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

The applicant 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.

1. Point Source vs. Finite Source: As part of the model validation, the applicant demonstrated the acceptability of a point source in the site-response modeling by showing that the point-source models produce ground motion response spectra (5 percent damped pseudo-absolute response spectra) that are in reasonable agreement with observed data, as outlined in BSC Section 7 (2008bl). The applicant

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also included the evaluation of modeling variability and model bias. The applicant'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 the applicant's justification for using a point-source model. (b)(5)

(b)(5)

2. Equivalent Linear vs. Nonlinear: The applicant 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.

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In the validation study for the one-dimensional equivalent-linear model described in EPRI Appendix 6.B~~, conducted by the EPRI~~ (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: (b)(5)

(b)(5)

3. One-Dimensional vs. Two-~~Dimensional~~ or Three-Dimensional Modeling: The applicant used a one-dimensional site response model to account for two-~~dimensional~~ and three-dimensional effects. Therefore, two-~~dimensional~~ and three-dimensional effects, such as topographic amplification and lateral variability in dynamic material properties, are not explicitly addressed in the site response model. The applicant stated that two-~~dimensional~~ 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).

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To improve the one-dimensional model, the applicant randomized soil properties over the site as a way to accommodate lateral variations in soil properties. Secondly, the applicant developed several one-dimensional profiles with multiple alluvium thicknesses

For example, the applicant 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).

The applicant noted in BSC Section 7.4 (2004aj) that although the two-dimensional effects are not explicitly included in the one-dimensional model, the two-~~dimensional~~ 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-~~dimensional~~ and three-dimensional ground motion effects.

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NRC Staff Evaluation: (b)(5)

(b)(5)

4. 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 ~~for~~ of this 1995 Japanese earthquake ~~is~~ 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).

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NRC Staff Evaluation: (b)(5)

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5. Comparison of Combined Source, Path, and Site-Response Modeling to Observed Data: The applicant performed combined model validation to confirm that the model is acceptable 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.

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NRC Staff Evaluation: (b)(5)

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6. 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 M 5.7 Little Skull Mountain earthquake. The applicant 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. The applicant also noted potential two-dimensional effects associated with the dipping alluvial-rock interface, which are not explicitly included in the one-dimensional model.

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The applicant summarized the shear strain from ~~the~~ EPRI's (1993aa) validation study and compared it with the applicant'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 ~~the~~ 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: (b)(5)

(b)(5)

(b)(5) The simplification and approximation of the model the applicant made included choosing point source over finite source, stochastic over deterministic for the source modeling, and one-dimensional over two-dimensional or three-dimensional equivalent-linear over nonlinear for the site-response modeling.

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NRC Staff Conclusion

(b)(5)

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

Comment (b)(5)

(b)(5)

Comment [LIM8R7]:

As part of site characterization activities, the applicant 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. The applicant's information included the following: (i) depth to the alluvium-tuff contact; (ii) subsurface configuration of volcanic strata and subsurface location of faults; (iii) V_S and V_P 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, the applicant 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, the applicant collected data from 89 exploratory boreholes and surface wave survey lines across the site. The applicant 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 the applicant'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 the applicant's one-dimensional RASCALS site-response models. NRC staff review of the dynamic material property information is provided in SER Section 2.1.1.1.3.5.3.3.

Alluvium Thickness Calculations

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

NRC Staff Evaluation: (b)(5)

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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 the applicant 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, the applicant relied on the borehole and SASW methods to develop profiles for site-response models because the borehole-based techniques provided reliable information on velocities in the immediate vicinity of the borehole SASW surveys complemented the borehole-based measurements and provided information on the average V_s over a larger volume of the subsurface.

NRC Staff Evaluation: NRC staff evaluated the applicant'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 (b)(5)

(b)(5)

Compression Waves

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

NRC Staff Evaluation: NRC staff evaluated the applicant's information in the SAR and supporting documentation (b)(5)

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Density

The bulk density is important because it influences the site-response modeling, especially damping. The applicant 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: (b)(5)

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Seismic Velocity Profiles for Surface GROA

The development of seismic velocity profiles as input to the seismic site-response model for the surface GROA is described in SAR Section 1.1.5.2.7.2 with additional detailed information in BSC (2008bl), SNL (2008af), and in the applicant's response to staff's RAI (DOE, 2009aq). On the basis of the available velocity data and site geology, the applicant developed 13 basecase velocity profiles for the surface GROA to fully capture the variability and uncertainty of the site. To capture the randomness of the site response, the applicant used each basecase 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 ~~exceedence~~exceedance probabilities in the PSHA to develop representative surface hazard curves. To capture the spatial variability of the site, including differences across the Exile Hill fault splay or variability in the stiffness of the underlying tuff, the applicant enveloped the site-specific hazard results to develop a single hazard curve for the entire surface GROA.

NRC Staff Evaluation: (b)(5)

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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. The applicant determined that these variations coincided with lateral changes in rock conditions, such as variations in lithology, stratal contacts, or the degree of fracturing in the tuffs. As a result, the applicant developed four separate velocity profiles to represent a central "stiff" zone and three relatively "softer" zones. Akin to the methodology for the surface GROA, the applicant developed a suite of site-response models that were combined to produce representative hazard curves for the repository block.

NRC Staff Evaluation: (b)(5)

(b)(5)

NRC Staff Conclusion

The NRC staff reviewed the applicant's description of geophysical information to develop V_p, V_s, and density profiles (b)(5)

(b)(5)

2.1.1.1.3.5.3.3 Geotechnical Information Used ~~t~~To Develop Dynamic Material Properties

The applicant 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. The applicant derived these values from experiments conducted over the past two decades. The applicant detailed descriptions of the data acquisition activities in BSC (2002aa). Both resonant column and torsional shear tests were performed in a sequential series on the same specimen over a shear strain range from about 10^{-4} percent to 10^{-1} percent (BSC, 2002aa, 2004aj; SNL, 2008af).

Normalized Shear Modulus and Damping

The applicant provided normalized shear modulus and material damping values used to assess the ground response at the surface from a controlled ground motion at the rock outcrop level [SAR Section 1.1.5.3.2.6.3 and BSC Section 6.4.4 (2008bl)]. Additional information needed for reduced shear modulus and damping values for each rock layer present at the site ~~are~~ 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.

Comment (b)(5)

NRC Staff Evaluation: The NRC staff reviewed the information provided in the SAR and relevant documents using YMRP Section 2.1.1.1.3. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5) The applicant used two sets of mean normalized shear modulus and damping ratio curves developed for both tuff and alluvium to bound the uncertainty. (b)(5)

(b)(5) The applicant 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. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.5.3.4 Development of Seismic Design Inputs

The applicant's development of the seismic design inputs was provided in SAR Sections 1.1.5.2.5.3, 1.1.5.2.5.4, 1.1.5.2.5.5, and 1.1.5.2.5.6. The applicant 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, the applicant used NUREG/CR–6728 to develop these ground motion inputs.

The Design Ground Motions

The applicant 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 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 adequacy of 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. (b)(5)

(b)(5)

(b)(5) The applicant developed vertical hazard curves by applying distributions of vertical-to-horizontal 5 percent damped response spectral ratios to the site-specific horizontal hazard curves. (b)(5)

(b)(5)

Ground Motion Inputs for Surface GROA and Subsurface GROA

The applicant 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 basecase velocity profiles (south and northeast of the Exile Hill Fault splay), two basecase 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.

Site-Specific Hazard Curves: The seven combinations of 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).

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UHS and Design Response Spectra (5 Percent Damped): 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}, 5×10^{-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).

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(b)(5)

Earthquake Time Histories: 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).

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Strain-Compatible Soil Properties: 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}

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NRC Staff Evaluation: (b)(5)

(b)(5)

(b)(5) To develop the hazard curves for the surface-area facility and repository block, the applicant used 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). (b)(5)

(b)(5)

(b)(5) The strain-compatible soil properties are the products of the site-response analysis described previously.

NRC Staff Conclusion

(b)(5)

Comment (b)(5)
(b)(5)

2.1.1.1.3.5.4 Site Geotechnical Conditions and Stability of Subsurface Materials

To demonstrate compliance with 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c), the applicant described, in SAR Section 1.1.5.3, the geomechanical properties and conditions of the repository site for design and PCSA of the GROA. The applicant 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 applicant'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

The applicant provided information pertaining to the nature and geometrical configuration of subsurface materials at the surface facility site. The applicant conducted geological and geophysical studies at the site, including geologic mapping of outcrops, characterization of cuttings from geophysical testing boreholes, observations in test pits and trenches, and surface- and borehole-based geophysical testing. The applicant concluded that the surface facility site is underlain by Quaternary alluvium and colluvium of up to 61 m [200 ft] thick, which overlies a sequence of volcanic tuff, as shown in SNL Table 6.2-1 (2008af). The applicant 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. The applicant determined the variation of the alluvium thickness through geologic analysis of site-specific borehole data as described in SNL (2008af). The applicant 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. The applicant did not provide information regarding the special relationship between the cemented and uncemented 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.

Comment (b)(5)
(b)(5)

NRC Staff Evaluation: The NRC staff reviewed the applicant's information pertaining to the types and geometrical configuration of subsurface materials at the surface facility site using staff guidance in ~~the YMRP Section~~YMRP Section 2.1.1.1 (b)(5)

(b)(5)

Stability of Subsurface Materials at the Surface GROA

Shear Strength of the Alluvium: The applicant 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. The applicant's assessment of the performance of surface facility structures assumed that the subsurface materials supporting the foundations will be stable during the preclosure period and undergo only elastic deformations when subjected to static and seismic loading (BSC, 2007ba). To support this assumption, the applicant provided information pertaining to the bearing capacity of the alluvium that is the allowable foundation bearing pressure on alluvium, which the applicant calculated using estimates of the shear strength of the alluvium.

To estimate the shear strength of the alluvium, the applicant 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, the applicant 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, the applicant 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. The applicant used these density indices to calculate relative density of the alluvium. The applicant 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).

The applicant 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. The applicant 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. The applicant determined a mean relative density of 68 and standard deviation of 21 for the alluvium by averaging the measurements. As described in the applicant's soils engineering report, BSC Appendix I (2002ab), the applicant 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°. The applicant 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.

The applicant'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 the applicant 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, the applicant indicated the alluvium is variably cemented, but did not provide information regarding spatial relationships between the cemented and uncemented alluvium

In response to an NRC staff RAI, the applicant explained 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 the applicant 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 the applicant's assumption that uniformity of the derived soil classification and gradation of the alluvium would imply uniformity of the alluvium shear strength, the applicant 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, the applicant 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.

The applicant'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 the applicant described the shear strength of the alluvium using a mean shear strength value of 39°. The applicant 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 Section~~YMRP Section 2.1.1.1 (b)(5)

(b)(5)

(b)(5)

(b)(5)

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.

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In response to NRC staff RAIs, the applicant 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: (b)(5)

(b)(5)

Allowable Bearing Pressure and Settlement of Foundations of Surface GROA Facilities: The applicant provided information pertaining to allowable bearing pressure for the foundations of the surface facility structures in its soils engineering reports in BSC Appendix B (2007bq). The applicant determined the allowable bearing pressure for three conditions using shear strength of alluvium on the basis of an internal friction angle of 39°: (i) square and strip footings with no limit on settlement, as described in BSC 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). The applicant's results for condition (i) determined potential limits on foundation loading without causing a generalized shear failure of the subsurface materials (i.e., rotational failure of the foundation and underlying materials). 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.

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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].

The applicant stated in BSC Table 2-2 and Figures 7-2 and 7-3 (2007qq) 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. However, the applicant did not provide an adequate basis for recommending a maximum allowable bearing pressure of 2,394 kPa [50 ksf] for the extreme loading condition. In response to an NRC staff RAI, the applicant stated in DOE Enclosure 1 (2009aq), DOE Enclosure 1 (2009eh), and DOE Enclosure 1 (2009ei) that its design approach for mat foundations at the surface facility is based on (i) using a finite element (FE) modeling in which the subsurface material is represented by soil springs and the deformation of the soil springs from FE modeling is used to calculate foundation pressures and settlements and (ii) checking the calculated pressures and settlements against the applicant's predetermined limits. The applicant performed an FE analysis to design mat foundation for the design basis seismic load condition and cited a maximum allowable bearing pressure of 2,394 kPa [50 ksf] under a seismic loading condition, which NRC staff evaluated in SER Section 2.1.1.7.3.1.1. In BSC (2007bq), the applicant evaluated elastic, or short-term, settlement of a 91.4 by 122.9-m [300 by 400-ft] mat on a 36.6-m [120-ft]-thick alluvium, subjected to normal loading conditions (dead load plus live load) of 144, 239, and 335 kPa [3, 5, and 7 ksf]. BSC Table B7-1 (2007qq) presented the calculated settlements, ranging from 5 to 76 mm [0.2 to 3.0 in] for the range of load [refer to BSC Table 7-1 (2007qq) for details]. In response to an NRC staff RAI (DOE, 2009ei), the applicant provided: (i) Table 1, presenting new results of total and differential settlements for the static loading conditions (normal ~~L~~load) for all ITS structures; (ii) Figure 1, showing calculated allowable bearing capacity for foundations ~~of width~~ up to 92.6 m [300 ft] wide, which is similar to BSC Figure B6-2 ~~in BSC~~ (2007qq) for rotational shear failure of foundation material discussed ~~above~~previously; and (iii) Figure 2, showing allowable bearing pressure for foundations ~~of width~~ up to 92.6 m [300 ft] wide, which would limit the settlement to 50 mm [2 in]. ~~Based on~~On the basis of DOE Figure 2 (2009ei), limiting the settlement to 50 mm [2 in] criteria for large mat foundations, the applicant recommended an allowable bearing pressure of 479 kPa [10 ksf] for normal loading conditions. The applicant provided the rationale

for ~~limiting the~~a settlement limit not exceeding~~up to~~ 50 mm [2 in] for large mat foundations of ITS structures. For extreme loading the applicant recommended an allowable bearing capacity of 2,394 kPa [50 ksf]~~,~~ from author Figure 1 year ~~based on~~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 (~~DOE~~ 2009ei) lists the average foundation pressure for various ITS structures under normal loading to range from 81 ~~kPa [1.7 ksf]~~ 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 the applicant's information on allowable bearing pressure for surface GROA facility foundations using guidance in ~~the YMRP Section~~YMRP Section 2.1.1.1. (b)(5)

(b)(5)

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). The uncertainty associated with the shear strength and relative density data is discussed in the previous paragraph and needs to be considered in using this information for footings up to 9.1 m [30 ft] wide. ~~It is noted~~Note that all the proposed ITS structures' foundations are much larger than 9.1 metric units missing [30 ft].

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. The applicant 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 (2007qq) presented calculated settlements at the center and corners of the mat foundation. (b)(5)

(b)(5)

In ~~T~~the applicant's response to an NRC staff RAI, DOE Figure 2 (~~DOE~~, 2009ei). ~~Figure 2~~ presented a revised evaluation of allowable bearing pressure ~~based on~~on the basis of limiting the settlement to 50 mm [2 in]. (b)(5)

(b)(5)

To use a value of bearing capacity that does not include a limit on foundation settlement, the applicant 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.

(b)(5)

The applicant 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 FE modeling analysis used to design mat foundation represents the alluvium as a spring (soil spring using shear modulus calculated from shear wave velocity). (b)(5)

(b)(5)

(b)(5)

(b)(5) As NRC staff ~~described by the NRC staff in~~describes in SER Section 2.1.1.7.3.1.1, the applicant's FE analysis is based on the assumption that the mat foundation will respond linearly under vertical loading. (b)(5)

(b)(5)

(b)(5)

Comment (b)(5)
(b)(5)

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, the applicant committed to evaluate the stability of the slopes as part of the detailed design [DOE Enclosure 2 (2009aq) and DOE Enclosure 2 (2009eh)]. The applicant also stated that aging pads will be built on terraces to minimize the amount of cut and fill, and any cut-and-fill slopes will not be steeper than 2:1 (ratio of horizontal to vertical dimensions). The applicant 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 the applicant's information on stability of the cut-and-fill slopes using staff guidance in ~~the YMRP Section~~YMRP Section 2.1.1.1.(b)(5)

(b)(5)

(b)(5)

Geotechnical Conditions at the Subsurface GROA

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

RHH Materials: In SAR Section 1.1.5.3.1.1, the applicant 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). The applicant explained that the repository host rock includes lithophysal and nonlithophysal subunits; the nonlithophysal subunits comprise approximately 15 percent of the emplacement area and the lithophysal subunits approximately 85 percent, with approximately 80 percent within the Lower Lithophysal subunit (Tptpll) (SAR Figure 2.3.4-22). The applicant explained in SAR Section 1.1.5.3.1.1 that the lithophysal and nonlithophysal rock types are compositionally similar but have different physical, thermal, and mechanical properties because of the differences in their internal geologic structures.

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

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on RHH materials using guidance in the ~~YMRP Section~~YMRP Section 2.1.1.1. (b)(5)
(b)(5)
(b)(5) On the basis of the geologic studies, the applicant described the expected locations of stratigraphic contacts and estimated the percentage occurrence of each rock type within the repository horizon.
(b)(5)

Mechanical Properties of Nonlithophysal Rocks: To characterize the mechanical properties of nonlithophysal rock, the applicant (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.

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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 the applicant 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. The applicant's information provided an engineering characterization of the nonlithophysal rock units that the applicant encountered in the ESF tunnel. (b)(5)
(b)(5)

NRC Staff Evaluation: NRC staff reviewed the information on the mechanical properties of the nonlithophysal rocks using guidance in ~~the YMRP Section~~YMRP Section 2.1.1.1. (b)(5)
(b)(5)

(b)(5)

Mechanical Properties of Lithophysal Rocks: To characterize the mechanical properties of lithophysal rock, the applicant (i) tested 29 large-diameter specimens from the lithophysal rock units and used the results to group the rock mass into 5 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.

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The applicant 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, the applicant 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 the applicant's information regarding the mechanical properties of lithophysal rocks using staff guidance in ~~the YMRP Section~~YMRP Section 2.1.1.1. (b)(5)

(b)(5)

Comment (b)(5)
(b)(5)

(b)(5) The applicant 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), the applicant stated this is suggested by the behavior of existing tunnels at the subsurface facility site. The applicant'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. (b)(5)

(b)(5)

(b)(5)

(b)(5)

Other Geotechnical Properties at the Subsurface GROA: The applicant 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).

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NRC Staff Evaluation: The applicant used standard techniques for the laboratory and field tests and provided information to define potential uncertainties in the test results. (b)(5)

(b)(5)

Commitment

Confirm stability of slopes in vicinity of the Aging Facility under applicable seismic loading conditions accounting for uncertainties in shear strength of alluvium, ~~prior to~~before excavation of Aging Facility foundation (DOE, 2009aq).

Condition of Construction Authorization

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.1.3.5.5 Fault Displacement Hazard Assessment

In SAR Section 1.1.5.2.4.1, the applicant 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 the applicant'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

The applicant conducted its PFDHA at the same time as its PSHA using the same procedures as discussed in SER Section 2.1.1.1.3.5.2. The applicant 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, the applicant used the PFDHA frequency and magnitude information and estimates as input to the seismic ground motion hazard assessments its seismic experts made.

The applicant 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. The applicant used experts from different areas of expertise closely related to seismic hazard to represent differences in experience, model selection, and analytical approach. The applicant used the logic tree approach to ensure consistent and quantifiable results were obtained.

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

The process the applicant followed in the PFDHA included specific characteristics and uncertainties that the applicant 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).

The applicant 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: (b)(5)

(b)(5)

(b)(5)

(b)(5)

Probabilistic Fault Displacement Hazard Assessment (PFDHA)—Input Data and Interpretations

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. The applicant analyzed 100 earthquakes in the Basin and Range region to determine the relationships among the amounts and patterns of both principal and distributed fault displacements, the minimum magnitude at which an earthquake may produce surface faulting, and the maximum magnitude at which an earthquake does not displace the surface.

For the largest mapped faults at Yucca Mountain, the 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).

The applicant 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 9 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 the applicant's information regarding input to the PFDHA in the SAR and supporting documents. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.6 Site Igneous Activity

To demonstrate compliance with 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c), the applicant provided information in SAR Sections 1.1.6, 2.2.1, and 2.3.11 on the known intrusive and extrusive (volcanic) igneous activity in the Yucca Mountain region as they pertain to PCSA and the GROA design. The applicant also provided information on the probabilistic volcanic hazard analysis (PVHA) conducted. The applicant indicated that volcanic activity has occurred in the tectonically active Yucca Mountain region and could continue into the future. On the basis of this, igneous activity may possibly affect GROA design and preclosure repository performance.

In SAR Section 1.1.6, the applicant assessed the location and magma types of past volcanism in the Yucca Mountain region, described the characteristics of basaltic volcanism in the region, and presented evidence for simultaneous seismic activity and volcanic eruption. The applicant 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, pursuant to 10 CFR Part 63. Aspects of the hazard and risk that igneous activity in the postclosure period pose to repository performance are evaluated in SER Section 2.2.1.3.10.1.

In this SER chapter, NRC staff's evaluation of the information presented in SAR Section 1.1.6 is coordinated with that of SER 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; on relevant applicant-provided reports; and on publicly available information. This review concentrates on volcanic (extrusive) surface activity as it is more likely to affect the repository surface facilities and workings in the active operation period than an intrusive event, even

though the applicant showed the probability of a future eruption within the preclosure period to be extremely low.

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

Rhyolitic Igneous Activity: The applicant assessed the types of igneous activity and location of past volcanism in the Yucca Mountain region in SAR Section 1.1.6.1. The applicant gave the age range of the major volcanic flare-up that formed the rhyolitic ash-flow tuffs of Yucca Mountain, the host rocks for the repository, as 13 to 10 million years ago (SAR Section 2.3.11.1). The applicant 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, the applicant 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).

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NRC Staff Evaluation: NRC staff reviewed the information the applicant provided using the guidance in the YMRP. (b)(5)

(b)(5)

Basaltic Igneous Activity: With regard to smaller scale basaltic igneous activity, the applicant presented evidence that basaltic eruptions and intrusions in the Yucca Mountain region have fallen into 2 major time periods or phases: (i) from 11 million to 8 million years ago and (ii) beginning about 4.6 million years ago and continuing to the latest eruption 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).

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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, the applicant concluded that their presence does not significantly increase the future probability of an eruption at the repository site on the basis of the number of post-Pliocene igneous events (BSC, 2004af).

The applicant focused on the young (post-Pliocene) basaltic volcanic deposits, lavas, and intrusions because they can be used to determine the type and style of volcanism that has occurred most recently and that may recur in the 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. The applicant also determined that the volumes of basaltic magma erupted in the Yucca Mountain region are very small (on a comparative global scale), that the youngest phase of igneous activity has featured the smallest eruptions, and that activity has generally decreased in volume over time.

NRC Staff Evaluation: (b)(5)

(b)(5)

Relationship Between Seismic and Igneous Activity

In SAR Section 1.1.6.1.2, the applicant identified that rising magma could cause seismic activity in the form of small earthquakes that in turn could trigger other larger earthquakes. The applicant also described the occurrence of patches of basaltic ash particles showing signs of minimal abrasion in some alluvium horizons and ground cracks (fissures) exposed in trenches the applicant excavated to investigate faults. The applicant concluded that the cracks were caused by faulting. The applicant 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 the applicant provided using the guidance in the YMRP. (b)(5)

(b)(5)

Probabilistic Igneous Hazard Analysis

In SAR Section 1.1.6.2, the applicant assessed the likelihood of future basaltic igneous activity in the repository area, together with an estimate of the uncertainty, by relying upon the result of a PVHA (CRWMS M&O, 1996aa; BSC, 2004bi). Further relevant information is found in the applicant'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 the applicant's PVHA, the applicant determined (i) the mean annual frequency of the likelihood of a basaltic dike intruding the underground repository as 1.7×10^{-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). (b)(5)

(b)(5)

These values indicated the general range of probabilities the applicant determined for an igneous intrusion into, and a volcanic eruption within, the subsurface GROA. Actual probability values applicable to the preclosure period were not stated in the SAR, but were described in the applicant's external events hazard screening analysis in BSC Section 6.3 (2008ai) as lower than 10^{-8} per year. An applicant-conducted PVHA update (SNL, 2008ah) made similar conclusions about the annual probability of future intrusive and volcanic activity at the repository site.

In general, the probability of a dike intruding the repository, as the applicant 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 the applicant's information using the guidance in the YMRP (b)(5)

(b)(5)

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

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

The applicant 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, the applicant 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.

Potential Ash Fall From Distant (Caldera) Volcanoes: The applicant considered the ash-fall hazard posed by extremely rare distal explosive eruptions, such as large caldera-forming events at Yellowstone (Wyoming) and Long Valley (California) that occurred within the past 1 million years. In the past, such eruptions have deposited ash falls up to a few tens of centimeters [~10–20 in] in the Yucca Mountain area, as described in Perry and Crowe, 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).

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Potential Ash Fall From Nearby (Basaltic) Volcanoes: The applicant contended 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. The applicant 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 the applicant-conducted PVHA (CRWMS M&O, 1996aa), concluded that it was less than 10^{-6} per year, as described in SAR Section 1.6.3.4.3 and BSC Section 6.3 (2008ai).

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The applicant 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 the applicant provided using the guidance in the YMRP (b)(5)

(b)(5)

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 the applicant'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.

The applicant 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.

The applicant 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, the applicant 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, the applicant claimed 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).

The applicant 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], the applicant indicated that there was sufficient space of approximately 40 cm [16 in] below ventilation system intakes 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]}. The applicant claimed 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: (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.7 Site Geomorphology

To demonstrate compliance with 10 CFR 63.21(c)(ii) and 10 CFR 63.112(c), in SAR Section 1.1.7 the applicant 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. The applicant assessed the site's landscape response to climate change and erosional and depositional processes. These assessments are the applicant's bases for evaluating whether a geomorphic hazard could affect site structures or operations. The applicant 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.

Geomorphic Information and Tectonic Activity

Erosion, Erosion Rates, and Deposition: The applicant conducted geomorphic studies in the Yucca Mountain region to characterize the site, as described in BSC 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. The applicant'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. The applicant 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. The applicant concluded that these geomorphic observations support a slow rate of erosion for the region.

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Additionally, the applicant presented geomorphic information related to volcanism in the Yucca Mountain region that it used to determine erosion rates. The applicant examined cinder cones (also known as scoria cones) and their associated basaltic lava flows in Crater Flat. The degree of cinder cone erosion 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.

The applicant 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. (b)(5)

(b)(5)

NRC Staff Evaluation: NRC staff reviewed the applicant's description of erosional and depositional processes and landforms in SAR Section 1.1.7 using guidance in the YMRP. NRC staff reviewed the applicant's description of the hazards posed by geomorphic processes at and near the repository site relevant to the preclosure period. (b)(5)

(b)(5)

(b)(5)

(b)(5)

Variability of Quaternary Processes

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

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The applicant concluded that over the next 10,000 years, under climatic conditions similar to the present, the local rate of sediment accumulation around Yucca Mountain should prevail. 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, the applicant 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.

The applicant 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) the applicant 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: (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.8 Site Geochemistry

To demonstrate compliance with 10 CFR 63.21(c)(1)(ii) and 10 CFR 63.112(c), the applicant described Yucca Mountain site geochemistry in SAR Section 1.1.8 and references therein. In particular, the applicant 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. The applicant 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, the applicant focused on characteristics of the near-field environment (i.e., the excavated repository drifts and adjacent host rock) and how preclosure activities would affect near-field geochemical conditions. In SAR Section 1.1.8, the applicant identified four factors associated with the preclosure period that would modify present-day geochemical conditions in the near-field environment: elevated temperatures, gamma radiation, underground construction activities, and underground ventilation processes.

The applicant 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, the applicant'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).

Elevated Temperature and Ventilation Effects

The applicant 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).

The applicant 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, the applicant reasoned 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). In particular, the applicant reasoned 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, the applicant cited field observations of wall rock dewatering due to forced ventilation in the ESF under ambient conditions (SAR Section 2.3.3) and thermal-hydrologic-chemical and seepage evaporation modeling analyses, as described in BSC 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 the applicant'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. (b)(5)

(b)(5)

Gamma Radiation and Ventilation Effects

In describing the geochemistry of the Yucca Mountain site for preclosure conditions, the applicant considered how gamma radiation from the emplaced waste packages might affect water–rock interactions in the repository near field. The applicant conducted irradiation experiments to investigate radiation effects on repository host rock and observed that even at much higher doses than anticipated in the repository, gamma radiation damage was limited to small changes in mechanical properties of minerals due to the radiolysis of water in the rock samples, which resulted in some fracturing and alteration. The applicant concluded that radiation was not important in terms of preclosure site geochemistry, because (i) even at maximum field strength, the gamma radiation would penetrate no more than a few centimeters [inches] into the repository host rock and (ii) the scarcity of water in the rocks 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 the applicant related the potential effects of gamma radiation to the geochemistry of the repository near-field environment. (b)(5)

(b)(5)

(b)(5) The applicant cited field observations and coupled heat transfer modeling analyses to support the assumption that forced ventilation and elevated temperatures during the preclosure period would limit the availability of water for radiolysis in the repository near field. (b)(5)

(b)(5)

Construction Activities and Ventilation Effects

The applicant 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). The applicant 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.

The applicant reasoned 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 the applicant provided. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.1.3.9 Land Use, Structures and Facilities, and Residual Radioactivity

To demonstrate compliance with 10 CFR 63.112(c), the applicant investigated the following to determine potential human-induced hazards at the site: (i) previous and ongoing 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. The applicant presented this information in SAR Sections 1.1.9, 5.8.1, and 5.8.2 and in its response to NRC staff RAIs on land use (DOE, 2009au).

The applicant stated in SAR Section 5.8.1 that it intends to pursue legislation permanently withdrawing approximately 59,500 ha [147,000 acres] to construct and operate a repository at Yucca Mountain. The analyzed withdrawal area is the same as the preclosure controlled area (SAR Figure 5.8-1). The applicant stated that it will permanently control the land to restrict access and reduce potential human-induced hazards during the preclosure period.

Previous and Ongoing Land Use

The applicant summarized previous and ongoing 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. The applicant 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.

The applicant detailed ongoing land-use interests and existing rights-of-way in SAR Sections 5.8.1 and 5.8.2. The description included maps with locations of existing public land withdrawals and repository surface facilities with respect to preclosure and postclosure controlled area boundaries. At present, NTS lands (Area 25) used for Yucca Mountain project activities are managed under a memorandum of agreement between the DOE National Nuclear Security Administration and the DOE Office of Civilian Radioactive Waste Management. For the purposes of site characterization of Yucca Mountain, the U.S. Bureau of Land Management has granted the applicant two rights-of-way (N-48602 and N-47748) and issued two public land orders (PLO-7653 and PLO-7534) to withdraw the land from appropriation under other laws such as mining or leasing.

The applicant identified existing mining claims located just outside and just within the southern boundary of the proposed preclosure controlled area. The applicant also identified and described both patented and unpatented mining claims located about 15 km [9 mi] south of the proposed GROA (SAR Section 2.1.1.1.3.1). In addition to the mining claims, the applicant identified other encumbrances such as the Nye County Early Warning Drilling Program and free-use permits the U.S. Bureau of Land Management issued to DOE for two borrow pits located within the proposed withdrawal area.

The applicant described that the legislation proposed in the Nuclear Fuel Management Act (Bodman, 2007aa) would require DOE to prepare and submit a management plan to Congress and the State of Nevada that would establish priority for repository-related activities within the preclosure controlled area. If the proposed legislation is enacted, this management plan would govern potential land use conflicts and establish a process for other activities to be conducted under DOE-established terms and condition.

NRC Staff Evaluation: NRC staff used the guidance in the YMRP to review the applicant's information on land use. (b)(5)

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(b)(5) The NRC staff evaluation of the plans for land control is contained in SER Chapter 2.5.8.

Existing Structures and Facilities

The applicant 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. The applicant noted that these existing structures and facilities are subject to being replaced during construction activities at the GROA, in accordance with planned repository structures and facilities described in SAR Section 1.2.

NRC Staff Evaluation: NRC staff used the guidance in the YMRP (b)(5)
(b)(5)

Potential Exposure to Residual Radioactivity

Radiological Surveys: The applicant 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.

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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).

Emissions from the NTS: The applicant 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

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to tritium, resuspension of plutonium and americium from soil contaminated by past nuclear testing continues to contribute to radioactive emissions. The applicant 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). The applicant 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 Monitoring of Releases. 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. The applicant 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, the applicant determined that the concentrations in 2004 and 2005 were less than 1 percent of the compliance levels for the national emission standards for hazardous air pollutants.

Residual Radioactivity from Previous Land Uses. As a result of the surveys described previously, the applicant 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 the applicant, any residual radioactivity within the proposed land withdrawal area will make a negligible contribution to worker and public radiation exposure.

NRC Staff Evaluation: (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.1.1.4 Evaluation Findings

NRC staff reviewed the applicant's SAR and other information submitted in support of the license application, relevant to a general description of work conducted to characterize the Yucca Mountain site. (b)(5)

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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 the applicant'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 the applicant's information adequately 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 the applicant's responses to the NRC staff requests for additional information (RAIs) (DOE, 2009dl–dn,dp,dq,ds,dt–dv,dx,ea–ee).

The 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 Regulatory Requirements and Regulatory Guidance

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 should satisfy the general information input for the PCSA as required in 10 CFR 63.112(a).

The NRC staff evaluated this identified information using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.2 (NRC, 2003aa). 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.

2.1.1.2.3 Staff Review and Analysis

The structure of this Safety Evaluation Report (SER) chapter follows the review guidance provided in YMRP Section 2.1.1.2. SER Section 2.1.1.2.3.1 discusses the location and functions of surface facilities. SER Section 2.1.1.2.3.2 covers the SSCs, SCs, equipment, and utility systems for the surface facilities; the main surface facilities include the receipt facility (RF), initial handling facility (IHF), canister receipt and closure facility (CRCF), wet handling facility (WHF), and the aging facility (AF). SER Section 2.1.1.2.3.3 details the SSCs, equipment, and

utility systems for the subsurface facilities. SER Section 2.1.1.2.3.4 describes high-level radioactive waste (HLW) characteristics. SER Section 2.1.1.2.3.5 covers the engineered barrier system (EBS) components (e.g., drip shield, waste package), spent nuclear fuel (SNF) waste canisters, and overpacks. SER Section 2.1.1.2.3.6 covers the operational processes associated with the GROA and reviews the communication; instrumentation, and control systems for both surface and subsurface facilities. Although the applicant classifies the subsurface facility as non-ITS, SER Section 2.1.1.2.3.7 presents a review of the description of design and operations of the subsurface facility and focused review of the emplacement drift, which is a key component of the GROA subsurface operations. The applicant 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 approach, which includes a review of (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 functions of the SSCs and equipment in the context of operations and any interaction with other SSCs are reviewed. The NRC staff also determined 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

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant provided an overview of the surface facilities and their associated operations (SAR Section 1.2.1). Information provided in the Yucca Mountain Repository License Application General Information Volume, Chapter 1, Section 1.1.2 ~~of Yucca Mountain Repository License Application~~ 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 IHF, CRCF, WHF, RF, and 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). The applicant 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 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 ~~discussion on~~ function of each of the waste handling facilities ~~follows~~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 the applicant 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. The applicant 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 the review using the guidance provided in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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

This section presents the NRC staff's evaluation of the applicant's information on description and discussion of design of the surface facility SSCs, equipment, and utility systems. Surface facilities are evaluated 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

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described the structural design of the building facilities in SAR Sections 1.2.3 through 1.2.7. The applicant used this information in PCSA~~,~~ and in the design and performance evaluation of the building facilities. On the basis of the PCSA, the applicant designated the CRCF, IHF, WHF, RF, AF, and 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 SER section.

ITS Structures

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. The applicant 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 IHF cask handling process area main structure is a braced-frame steel structure approximately 52 m [170 ft] wide, 57 m [187 ft] long, and 32 m [105 ft] high. The interior reinforced concrete structure consists of 1.2-m [4-ft]-thick walls and roof that comprise ~~of~~ 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. The applicant 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 7 pads for about 1,250 vertical aging overpacks. The aging pad area 17R (SAR Figure 1.2.7-4) has 8 pads with space for about 1,150 vertical aging overpacks, and 2 pads with space for 100 horizontal DPCs in horizontal aging modules, with 50 modules on each pad. Vertical aging overpacks 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). The applicant 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 the applicant, 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. The applicant considered flood drainage channels to carry away water from a probable maximum flood (PMF) surrounding the aging

pads. The distance from the aging pads to upslope hillsides and the location of the drainage channel precludes soil from the slope sliding onto the concrete aging pads and contacting the aging overpacks (see SER 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, the applicant described ITS flood control features credited with preventing inundation of the surface facilities from a PMF at the site. The applicant'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 related to the description of the structural design of the ITS facilities using the guidance in YMRP Section 2.1.1.2.2. The NRC staff reviewed the list of codes and standards, drawings, materials, and loads associated with each facility. (b)(5)

(b)(5)

(b)(5)

Non-ITS Structures

SAR Section 1.2.8 provided the applicant's description of the non-ITS facilities. The location of the non-ITS facilities relative to other surface facilities was shown in SAR Figure 1.2.1-2 SAR Table 1.2.8-1 presented a list of all the non-ITS facilities and a general description of structural systems used in their design and construction. SAR Section 1.2.8 1.3 listed the codes and standards that the applicant proposed to use for the design of the non-ITS facilities. Additional information related to the structural design of the non-ITS facilities can be found 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 the applicant's structural description of the EDGF. The EDGF has an overall footprint approximately 53 m [174 ft] wide by 30 m [98 ft] long. The applicant provided the general arrangement plans for the floor and roof in SAR Figures 1.2.8-1 through 1.2.8-3. Cross-sectional views of the facility were shown in SAR Figures 1.2.8-4 through 1.2.8-7. The 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 the applicant, 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 the SAR structural design description of the EDGF and LLWF buildings using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.2 Layout of Mechanical Handling Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3)(i), and 63.112(a), the applicant 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 SER 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 and evaluated by the NRC staff in SER Section 2.1.1.7.3.1.2.

NRC Staff Evaluation: The NRC staff evaluated the applicant's description of the cask handling systems using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

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 and evaluated by the NRC staff in SER Section 2.1.1.7.3.1.2.1.

NRC Staff Evaluation: The NRC staff evaluated the applicant's description of the canister transfer systems using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

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 the applicant's description of the waste package closure system in the CRCF and IHF facilities using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

Waste Package Load-Out System

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 and evaluated by the NRC staff in SER Section 2.1.1.7.3.1.2.1.

NRC Staff Evaluation: The NRC staff evaluated the applicant's description of the waste package load-out systems using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

SNF Assembly Transfer System

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 and evaluated by the NRC staff in SER Section 2.1.1.7.3.1.2.1.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of the SNF assembly transfer system using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

Dual Purpose Canister Cutting System

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 and are evaluated by the NRC staff in SER Section 2.1.1.2.3.2.5.

NRC Staff Evaluation: The NRC staff evaluated the applicant's description of the DPC cutting system using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

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 the 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 the applicant's description of the TAD canister closure system using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.3 Geologic Repository Operations Area Electric Power Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3)(i), and 63.112(a), the applicant 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 the applicant 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. The applicant did not identify any ITS SSCs in the subsurface facilities. The applicant provides separate subsurface normal electrical power system feeds that facilitate simultaneous operations and construction/expansion activities. During operations, some areas of the subsurface facility will be inaccessible to workers as waste emplacement proceeds.

The electrical power system includes normal electrical power system and ITS electrical power system subsystems, 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. The applicant 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 equipment qualification program requirements, 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). The applicant stated that the electrified third rail design will be based on applicable codes and standards and accepted industry practices (SAR Section 1.3.3.4.1) and that the materials used to construct the electrified third rail conductor are contingent on the subsurface transportation equipment design (SAR Section 1.3.2.4.6.4). The applicant stated that commercially available materials will be used for the electrified third rail (SAR Section 1.3.4.5.7) 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). (b)(5)

(b)(5)

Comment (b)(5)

(b)(5)

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). Inspection and potential maintenance operations for these areas will be performed using one or more types of ROVs. The applicant also described an ROV that may be battery powered with tracks or rubber

tires (BSC, 2008ca). The applicant also presented concepts whereby alternating current mains or a tethered insulated cable may be used to provide power to an ROV.

NRC Staff Evaluation: The NRC staff evaluated the design descriptions of the proposed normal electrical power system using the guidance in YMRP Section 2.1.1.2.2. To evaluate the description of the normal electrical power system, the NRC staff focused on the adequacy of 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 the applicant. 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) compliance with design codes and standards, and (iii) provisions for adequate preventive and corrective maintenance operations. The evaluation focused on whether the design can adequately perform the stated functions as defined by the applicant and whether the design would interfere with deployment of alternative designs, if needed, to support operations during the preclosure period. In addition, the NRC staff also evaluated the adequacy of the design described in the SAR and supplemental materials to support potential operations and/or waste retrieval during the preclosure period, subject to 10 CFR 63.111(e).

(b)(5)

(b)(5)

(b)(5) 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 vaguely describe multiple, additional planned ROVs and related power provisions that are intended to enable operations in nonacccessible areas. (b)(5)

Comment (b)(5)
(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

ITS Electric Power System

The applicant provided a high-level conceptual and functional description of the ITS electrical power system design for designated surface facilities in SAR Sections 1.4.1.2 and 1.4.1.3. The proposed ITS electrical power system consists of ITS switchgear, ITS diesel generators and associated ITS diesel generator mechanical support systems, ITS 13.8-kV breaker automatic load sequencers, ITS 13.8-kV to 480-V transformers, ITS 480 V load centers and ITS motor control centers, ITS 125-V DC battery power supplies, and ITS UPS SSCs. The applicant listed and discussed applicable principal codes and standards for design methods and practices for the ITS electrical power system in SAR Section 1.4.1.2.8. The applicant provided supplemental information regarding the applicability of cited principal codes and standards (DOE, 2009dl) and discussed the proposed application of specific sections of principal codes and industry standards that the applicant intends to apply to the final design of the ITS electrical power system (DOE, 2009do). The applicant proposed to use IEEE 308, 379, 384, and 603 (Institute of Electrical and Electronics Engineers, 2001aa,ab; 1998aa,ab) as the principal codes and standards for the ITS electrical power system design. These standards describe the need to incorporate design criteria such as redundancy, spatial separation, independence between redundant channels, and isolation between safety and nonsafety circuits The applicant further described how it 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 YMRP Section 2.1.1.2.2. The NRC staff focused on the adequacy of 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 the applicant.

(b)(5)

(b)(5) A safety evaluation of the design of the ITS electrical power systems is included in SER Sections 2.1.1.6.3.2.8.4 and 2.1.1.7.3.6. (b)(5)

(b)(5)

Commitment

The applicant commits to provide in the final design a complete design description of electrical power SSCs located in nonaccessible areas, including design, construction, connections, configurable switching (i.e., at turnouts), and plans for inspection, observation, and maintenance and repair if needed, during the preclosure period for all ROVs ~~which~~ that are planned to operate in nonaccessible areas and require external power provisions (BSC, 2008ca).

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.4 Heating, Ventilation, and Air Conditioning and Filtration Systems

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a) for HVAC and filtration at the GROA surface facilities, the applicant described the HVAC systems in SAR Sections 1.2.2.3, 1.2.3.4, 1.2.4.4, 1.2.5.5, and 1.2.6.4. In addition, the applicant described the HVAC systems for the balance-of-plant facilities in SAR Sections 1.2.8.3.1 and 1.2.8.3.2. The applicant proposed to use HVAC systems during normal operations to (i) control flow from areas of lesser to greater contamination potential, (ii) control temperature for the health and safety of workers and proper equipment operation, (iii) limit the release and spread of airborne contamination in and from the surface facilities through filtration, and (iv) provide a release point to the atmosphere. In addition, the applicant stated that the HVAC system shall ensure reliable confinement and filtration of radiological releases from event sequences that involve breach of a

waste container or damaged SNF assembly. The applicant classified as ITS the HVAC system components required to mitigate the consequences of a radioactive release following an event sequence and provide cooling to ITS equipment.

The applicant described the HVAC systems for the surface facilities in terms of ITS or non-ITS subsystems serving confinement zones or nonconfinement zones. Note that design review and evaluation of the ITS portion of the HVAC systems are presented in SER 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 the applicant 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 I&C listed in SAR Table 1.2.2-14.

The applicant addressed the location and arrangement of the HVAC supply and exhaust equipment in SAR Sections 1.2.3.4, 1.2.4.4, 1.2.5.5, 1.2.6.4, and 1.2.8.3 for individual surface facilities. According to the applicant, the location and arrangement of the HVAC systems within the surface facilities ensure no interference with the safety functions of adjacent equipment and/or other systems. In addition, the applicant 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.

The applicant 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. The applicant 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, the applicant 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 SER Section 2.1.1.7.3.1.2.2. In addition, according to the applicant, 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.

The applicant defined the materials of construction for the HVAC systems as follows: minimum 18-gauge 304L stainless steel for ductwork and 14-gauge 304L stainless steel 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).

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

NRC Staff Evaluation: The NRC staff evaluated the surface HVAC system information using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.5 Mechanical Handling Equipment

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant 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. (b)(5)

(b)(5)

(b)(5) In addition, the applicant provided information on other ITS mechanical handling equipment that is not specialized or one-of-a-kind as the aforementioned major ITS mechanical handling equipment.

(b)(5)

Canister Transfer Machine

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.

The applicant 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. The applicant 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, the applicant considered the following load cases: normal operation load combinations (including testing and operating events) as in ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa), site-specific ground motions (DBGM-2), extreme wind (IHF only), and collision.

The applicant 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, the applicant 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 YMRP Section 2.1.1.2. (b)(5)

(b)(5)

Cask Handling Crane

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). The applicant (i) detailed the CHC equipment in SAR Section 1.2.4.2.1.1.3; (ii) provided mechanical design details of the CHC specific to the IHF in SAR Figure 1.2.3-19, to the CRCF in SAR Figures 1.2.4-34 and 1.2.4-35, and to the 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. The applicant 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, the applicant considered the following load cases: normal operation load combinations (including testing and operating events) as in ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa), site-specific ground motions (DBGM-2), extreme wind (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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

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. The applicant described the SFTM in SAR Section 1.2.5.2.2.1.3. The applicant provided mechanical design drawings of the SFTM in SAR Figure 1.2.5-47, the process and instrumentation diagram in SAR Figure 1.2.5-48, the logic diagram for the SFTM mast hoist in SAR Figure 1.2.5-49, and the logic diagram for the SFTM grapple in SAR Figure 1.2.5-50.

The applicant 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 the applicant described the load combinations used in the design: normal operation (including testing and operating events) load combinations as in ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa), site-specific ground motions (DBGM-2), extreme wind (IHF only), and collision. The applicant 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, the applicant relies on ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa) for the material properties, specifications, and analytical and design methods.

NRC Staff Evaluation: The NRC staff evaluated the description of the SFTM system using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

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. The applicant 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. The applicant described the CTT of the IHF in SAR Section 1.2.3.2.1.1.3.1, as illustrated in SAR Figure 1.2.3-20. 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, the applicant ~~also~~ 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).

The applicant provided approximate dimensions for the CTT in BSC Section 6.2.2.7 (2008bg). The applicant 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, the applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

Waste Package Transfer Trolley

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

The applicant selected ASME NOG-1–2004 (American Society of Mechanical Engineers, 2005aa) as the main design and materials construction code and standard for the WPTT. The 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.

The applicant 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, the applicant provided additional details on the interactions of the WPTT with other SSCs. In SAR Section 1.2.2.2.1, the applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

Other ITS Mechanical Handling Equipment

The applicant 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. 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)]. The applicant 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. The applicant 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. The applicant stated that the shield and confinement doors will be designed in accordance with ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). The applicant described shield and confinement doors in SAR Section 1.2 4.2.1.1.3.1, with design drawings presented in SAR Figures 1.2.4-19 and 1 2 4-22 and process and instrumentation diagrams illustrated in SAR Figures 1.2.4-20 and 1.2.4-23. The applicant stated that the shield doors are commonly used in nuclear facilities to prevent personnel exposure to radiation.

Rails are intended to support the WPTT, as well as the TEV, to transport casks from one location to another. In SAR Section 1.2.4.1.6, the applicant 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. The applicant will use the methods and practices provided in American Institute of Steel Construction (1997aa) to design the platforms and include seismic considerations. The applicant described platform categories in SAR Sections 1.2.3.2.1.1.2 (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). The applicant 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. The applicant stated that racks will be designed in accordance with the applicable provisions of ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa). The applicant 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. The applicant 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. The applicant described the 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. The applicant described the waste package closure subsystem in SAR Section 1.2.4.2.3. The applicant classified this subsystem as non-ITS, with the exception of the remote handling system bridge, which the applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.2.3.2.6 Shielding and Criticality Control Systems

To demonstrate compliance with 10 CFR 63 21(c)(3)(i) and 63.112(a), the applicant 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

The applicant 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)

The applicant described the shielding design by providing the shielding design objectives (SAR Section 1.10.3.1) and shielding design considerations (SAR Section 1.10.3.1.1). The applicant's shielding design objectives were taken from NRC Regulatory Guide 8.8 (NRC, 1978ab). The shielding design considerations the applicant provided 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.

The applicant used radiation sources (SAR Figure 1.10-18) and bounding terms (SAR Section 1.10.3.4) to approximate the geometry and physical condition of sources in various repository facilities. In addition, the applicant used flux-to-dose-rate conversion factors taken from ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa) to develop dose rates. The applicant assessed the basic design regarding protection of workers and the public using commonly accepted industry codes and standards, such as MCNP and SCALE codes. The applicant'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 YMRP Section 2.1.1.2.2 to determine the adequacy of the descriptions for the shielding system location, functional arrangement, and interactions with other SSCs within each facility. (b)(5)

(b)(5)

(b)(5)

Criticality Control

In SAR Table 1.14-2, the applicant identified several parameters that may need to be controlled to prevent criticality, as discussed next. The applicant described how the other parameters (geometry and reflection) are bounded in the analysis in SAR Section 1.14.2.3. The applicant's design does not include a criticality accident alarm system.

In SAR Section 1.14.2.3.2.4 the applicant stated that the waste form characteristic of fissile isotope concentration was the only criticality control parameter important for HLW glass. (b)(5)

(b)(5)

(b)(5) The applicant 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 the applicant 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. The applicant 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.

The applicant plans to use fixed neutron absorbers as part of the canister internals and as part of the SNF staging racks in the WHF Pool.

During wet handling operations, the presence of 2,500 mg/L [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, the applicant 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 the applicant's description and design information for the criticality control systems using guidance in YMRP Section 2.1.1.2.2. The NRC staff also reviewed information in the applicant's SAR and RAI responses (DOE, 2009dh) concerning the availability of sufficient enriched soluble boron. The applicant will develop

procedures for manual inspection of the boron concentration and enrichment, and manual operation of the boric acid makeup subsystem. (b)(5)

(b)(5)

Criticality Monitoring and Alarms

The applicant did not use a criticality accident alarm system to control the consequences of a potential criticality event. Instead the applicant relied on its screening of criticality as beyond Category 2 to justify that a criticality accident alarm system is not needed. The applicant also stated that because of the risk of false alarms and potential injury due to unnecessary evacuation, a criticality accident alarm system is considered to have a net adverse effect on worker safety (DOE, 2009di). This statement was supported by accounts of false alarms. (b)(5)

(b)(5)

NRC Staff Evaluation: The NRC staff reviewed the applicant's information regarding criticality monitoring and alarms using guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5) In SER Section 2.1.1.3.3.2.7.5, the NRC staff reviewed the applicant's screening of initiating events that could lead to criticality. In addition, in SER Section 2.1.1.4, the NRC staff reviewed the applicant's calculation of the probability of event sequences that may lead to an end state important to criticality. (b)(5)

(b)(5)

Comment (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.7 Fire Safety Systems

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant 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. The applicant 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.

The applicant'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.

The applicant 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. The applicant also included containment volume for the largest anticipated vessel spill within each handling facility. Supplemental design data were provided in the applicant's RAI response (DOE, 2009dm). The applicant 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, the applicant 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. The applicant 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.

The applicant 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. The applicant 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, the applicant stated that units will be sized and installed per NFPA 10 (National Fire Protection Association, 2007aa). In addition, the applicant 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, preengineered suppression system to protect the unit from electrical fires. The applicant referenced BSC (2007bf) in the SAR. (b)(5)
(b)(5)
(b)(5) The suppression system is non-ITS, because the consequence of such a fire is limited given the fail-safe design of the TEV.

As the applicant 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 SER Section 2.1.1.7.3.8.

(b)(5)

Information the applicant provided in response to an RAI (DOE, 2009dm) indicated that the HVAC control function of the fire alarm system must be closely reviewed for conflicts with the containment requirements of the HVAC system. (b)(5)
(b)(5)

(b)(5) 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 license application will be updated to reflect this hierarchy in HVAC control. Although the final design may not fully comply with NFPA 90A (National Fire Protection Association, 2002aa), this variance is necessary to ensure that the confinement takes precedence.

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). The design of passive fire barriers using national design standards is acceptable because GROA operations do not pose any unusual fire loads or thermal challenges to fire barriers (e.g., no high density flammable liquids or other combustible materials storage). Furthermore, the applicant 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 the applicant's description of the fire safety systems and applicant's commitment to update the license application (DOE, 2009dm) using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

Commitment

As stated in DOE (2009dm), the applicant committed to 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. This 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

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.8 Piping and Instrumentation Diagrams

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described and discussed 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 the applicant 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.

Cask Cavity Gas Sampling Subsystem

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. The applicant also provided a simplified table of valve positions (SAR Figure 1.2.3-40) showing the valve layout of the cask cavity gas sampling subsystem and orientation for different operation modes. In addition, in SAR Section 1.2.4.3.1.3, the applicant described the design methods and applicable codes and standards. The applicant stated that it will use methods and practices in ANSI/ANS-57.7–1988 (American Nuclear Society, 1988aa) and ANSI/ANS-57.9–1992 (American Nuclear Society, 1992aa).

NRC Staff Evaluation: The NRC staff evaluated the applicant's description and P&ID for the IHF cask cavity gas sampling using guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

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. The applicant classified this subsystem as non-ITS. The subsystem contains primary piping that transfers waste water effluents to the liquid LLW sampling tank. The effluents 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. The applicant provided the design basis requirements for this subsystem in SAR Section 1.2.4.3 2. In addition, the applicant proposed to comply with Regulatory Guide 1.143, Table 1, excluding footnotes (NRC, 2001ab).

NRC Staff Evaluation: The NRC staff evaluated the applicant's description and P&ID for the Liquid LLW subsystem using guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

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. The applicant classified this subsystem as non-ITS. The applicant 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).

The applicant stated that the waste package and its closure system were selected for prototyping programs prior to their use in the repository. The applicant 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. The applicant 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. The applicant recorded recommendations and lessons learned, and planned a retest. The applicant stated that potential design modifications may be required. In addition, to clarify how the performance capability of the waste package closure system would be verified during demonstration testing, the applicant will perform capability demonstrations (full system qualification testing) to ensure conformance with repository safety requirements (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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

WHF Pool Water Treatment and Cooling System

The applicant presented the P&ID for the pool water treatment and cooling system in SAR Figures 1.2.5-58 to 1.2.5-63. The pool water treatment and cooling system and its subsystems are classified as non-ITS. In SAR Section 1.2.5.3.2, the applicant provided information on functions, location, and components for the pool water treatment and cooling system that 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.

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

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the pool water treatment and cooling subsystem using the guidance in YMRP Section 2.1.1.2.2.

(b)(5)

Cask Decontamination Subsystem

The applicant presented the P&ID for the cask decontamination subsystem in SAR Figure 1.2.5-67. The cask decontamination subsystem uses deionized water to rinse casks when removed from the WHF pool. The cask decontamination subsystem 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

Cask Cooling and Filling Subsystem

The applicant described the cask cooling and filling subsystem in the cask preparation area of the WHF in SAR Section 1.2.5.3.4. SAR Figures 1.2.5-69 through 1.2.5-72 presented P&IDs of this subsystem. The function of this subsystem is to cool the inside of 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. The applicant committed to 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5) The applicant committed in response to an RAI (DOE, 2009du) to revise 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."

(b)(5)

(b)(5)

WHF TAD Canister and Shielded Transfer Cask Drying and TAD Canister

Inerting Subsystem

In SAR Section 1.2.5.3.5, the applicant described the WHF TAD canister and shielded transfer cask drying and TAD canister inerting subsystem. The applicant 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, the applicant described the operational process. In SAR Figure 1.2.5-73, the applicant presented a simplified P&ID that shows components and their arrangement in this non-ITS subsystem.

The codes, standards, and regulatory guidance that the applicant proposed to use are (i) ANSI/ANS–57.7–1988 (American Nuclear Society, 1988aa), (ii) ANSI/ANS–57.9–1992 (American Nuclear Society, 1992aa), (iii) ANSI N14.5–1997 (American National Standards Institute, 1998aa), and (iv) NUREG–1536 (NRC, 1997ae). 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

WHF Water Collection Subsystems

In SAR Section 1.2.5.3.6, the applicant 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. The applicant provided the codes and standards generally applicable for these systems in SAR Sections 1.2.3.3 and 1.2.4.3.

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information of the WHF water collection systems using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

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, the applicant 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, the applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

ITS Diesel Generator Air Start System Train A: In SAR Section 1.2.8.2.2 and Figure 1.2.8-20, the applicant described the diesel generator air start system. 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 YMRP Section 2.1.1.2. (b)(5)

(b)(5)

(b)(5)

Diesel Generator Water Jacket Cooling System: In SAR Section 1.2.8.2.3 and Figure 1.2.8-22, the applicant described the ITS diesel generator jacket water cooling system. 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 YMRP Section 2.1.1.2.2.

(b)(5)

(b)(5)

Diesel Generator Lubricating Oil System Train A: In SAR Section 1.2.8.2.4, the applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

ITS Air Intake and Exhaust System Train A

In SAR Section 1.2.8.2.5, the applicant 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, the applicant stated that the ITS air intake and exhaust system is designed in accordance with requirements specified in ASME B31.3–2004 (American Society of Mechanical Engineer, 2004ab) and NUREG/CR–0660 (Boner and Hanners, 1979aa).

NRC Staff Evaluation: The NRC staff evaluated the description and P&ID information for the air intake and exhaust system using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

Commitments

- The applicant commits to revise the text in SAR Section 1.2.5.3.4 to read 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, 2009du).

- The applicant commits to update the l~~L~~icense a~~A~~pplication to ensure a consistent statement of system and subsystem safety classification ~~between~~ among the SAR text, ~~as well as the~~ tables, ~~and~~ figures, and SAR Table 1.9-1 (DOE, 2009ec).

- The applicant commits to provide detailed design information on system interfaces and the boundary between ITS and non-ITS components in the P&IDs for the emergency diesel generator (DOE, 2009du).

- The applicant commits to address in the detailed design potential lubricating oil leak prevention, detection, containment, and isolation features of the lubrication oil system and seismic design consideration for engine-mounted components on emergency diesel generators (DOE, 2009dt,ec).

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.2.9 Decontamination, Emergency, and Radiological Safety Systems

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described and discussed the design of the decontamination, emergency, and radiological safety systems.

In SAR Section 1.2.1.3, the applicant provided an overview of the decontamination systems. The applicant noted that each of the handling facilities will be capable of performing activities, including (i) decontaminating exterior surfaces of casks, waste packages, and canisters; (ii) decontaminating the interior surfaces of casks in a dry environment; and (iii) in the case of WHF, decontaminating underwater using the cask decontamination subsystem. Other than minor decontamination in the CRCF and RF, if surface contamination levels exceed acceptable limits, canisters will be sent to the WHF for decontamination in the cask decontamination subsystem. In SAR Section 1.2.5.3.3, the applicant described the cask decontamination subsystem of the WHF. The 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, the applicant 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 SER Section 2.5.7. The applicant's emergency and radiological safety systems were described in SAR Section 5.11 as part of the operational radiation protection program description. The applicant has committed to setting 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.

The applicant has committed to periodic testing and calibration of the RMS using the practices found in ANSI/ANS-HPSSC-6.8.1–1981 (American Nuclear Society, 1981aa) and ANSI N42.18–2004 (American National Standards Institute, 2004aa). The applicant has committed to the 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 YMRP Section 2.1.1.2.2.

(b)(5)

NRC Staff Conclusion

(b)(5)

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

In this SER section, the NRC staff evaluates the applicant's description of the subsurface facility SSCs and operational process activities on the basis of information in SAR Section 1.3. (b)(5)

(b)(5)

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

(b)(5)

(b)(5)

2.1.1.2.3.3.1 Subsurface Facility Layout and Development Plan

Subsurface Facility Layout

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(a) in the context of the GROA subsurface facility, the applicant described the layout of subsurface facility structures in SAR Sections 1.3.1.1 and 1.3.2.2.1. According to the applicant, the subsurface facility 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 the applicant 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).

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

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.2.2 to evaluate the applicant's description of the subsurface facility layout. (b)(5)

(b)(5)

Subsurface Facility Development Plan

To demonstrate compliance with 10 CFR 63.21(c)(2) and 63.21(c)(3)(i), the applicant described the subsurface facility development plan in SAR Section 1.3.1. The applicant stated that operations in the subsurface facility will be preceded by a period of initial construction during which three emplacement drifts will be built and commissioned to receive waste. The start of waste emplacement will mark the end of the period of initial construction and the beginning of repository operations in the subsurface facility. The applicant plans for a period of operations, also referred to as the preclosure period, of approximately 100 years (SAR Section 1.3.1). The preclosure period would consist of 50 years of waste emplacement, including an initial period of 24 years of concurrent repository development, and 50 years of postemplacement monitoring. The subsurface facility 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 YMRP Section 2.1.1.2.2 to review the applicant's description of the subsurface facility development plan. (b)(5)

(b)(5)

Thermal Load Design

To demonstrate compliance with 10 CFR 63.21(c)(2) and 63.21(c)(3)(i), the applicant described its approach to thermal management in SAR Section 1.3.1.2.5. The applicant 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. The applicant 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. The applicant stated that the actual waste stream that will be emplaced in the drifts depends on a number of variable and unspecified factors. The applicant indicated that a customized loading plan will be developed for each emplacement drift to meet the overall repository thermal goals after definitive shipping schedules for SNF from utilities and other sources become available.

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.2.2 to review the applicant's description of the approach to managing the repository thermal load. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.3.2 Nonemplacement Areas of the Subsurface Facility

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(a), the applicant described the nonemplacement areas of the subsurface facility in SAR Section 1.3.3. The applicant stated that the nonemplacement areas of the subsurface facility will consist of all underground openings and their SSCs, except the emplacement drifts. The NRC staff understanding of the functions of underground openings and their inverts in the nonemplacement areas of the subsurface facility during the preclosure period is summarized in SER Table 2-1. The functions support operation of ITS TEV or operations and activities, such as thermal management, that control parameter values that the applicant used for postclosure performance assessment.

North Portal and North Ramp

The applicant described the North Portal and North Ramp in SAR Section 1.3.3.1.1. The North Portal 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 YMRP Section 2.1.1.2.2 to review the applicant's description of the North Portal and North Ramp designs. (b)(5)
(b)(5)
(b)(5) The NRC staff's evaluation of the North Portal and North Ramp designs' capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7. (b)(5)
(b)(5)

Access Mains

The applicant described the access mains in the SAR Section 1.3.3.1.2. The applicant 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 YMRP Section 2.1.1.2.2 to review the applicant's description of the access mains. (b)(5)
(b)(5)
(b)(5) The NRC staff evaluation of the capability of the access mains' design to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7. (b)(5)
(b)(5)

Turnouts

The applicant described the turnouts design in SAR Section 1.3.3.1.4. The applicant 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. The applicant 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 YMRP Section 2.1.1.2.2 to review the applicant's description of the turnouts. (b)(5)

(b)(5)

(b)(5) The NRC staff's evaluation of the turnouts design's capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7. (b)(5)

(b)(5)

Exhaust Mains

The applicant described the exhaust mains design in SAR Section 1.3.3.1.3. The applicant 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 the applicant, 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]. The applicant 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. The applicant 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 the applicant (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 the applicant's thermal management strategies to satisfy repository thermal performance requirements the applicant established. According to the applicant, the exhaust mains also will be used for remote access for inspection and maintenance. Ground support for the exhaust mains will consist of fully grouted rock bolts with welded wire fabric (SAR Section 1.3.3.3.1). The rock bolts will have a nominal length of 3 m [9.8 ft] in a square-grid pattern at 1.25 m [4.1 ft] center to center. Ground support where an exhaust main and an emplacement drift intersect will consist of fully grouted rock bolts with steel-fiber-reinforced shotcrete and occasional lattice girders. The rock bolts will be approximately 5 m [16.4 ft] long and placed in the same square-grid pattern. The exhaust mains may have an invert to facilitate mobile equipment access as the applicant stated. The applicant explained in SAR Section 1.3.1.2.1.6 that the exhaust mains, like the exhaust shafts, will be nonaccessible because of high temperature and potential high radiation levels.

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.2.2 to review the applicant's description of the exhaust mains. (b)(5)

(b)(5)

(b)(5) The NRC staff's evaluation of the exhaust mains' design capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7. (b)(5)

(b)(5)

Shafts and Ventilation Raises

The applicant described the design of shafts and ventilation raises in SAR Section 1.3.3.1.5. The applicant stated that the subsurface facility 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. The applicant stated that plain concrete will provide adequate support for the shaft walls because the liner will be applied after full relaxation of the walls following excavation; the applicant implied the concrete liner will be stress free. As the applicant 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. The applicant stated in the 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 the

applicant, the subsurface facility also includes two short vertical openings, referred to as raises, as described in SAR Section 1.3.3.1.5. The applicant stated that one raise will connect the exhaust main of emplacement drift Panel 1 to the 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 YMRP Section 2.1.1.2.2 to review the applicant's description of shafts and ventilation raises. (b)(5)

(b)(5)

(b)(5) The NRC staff's evaluation of the shafts' and raises' design capability to perform the functions through the preclosure period is presented in SER Section 2.1.1.2.3.7. (b)(5)

(b)(5)

Subsurface Facility Openings Dedicated to Performance Confirmation

The applicant described the design of underground openings dedicated to performance confirmation in the SAR Section 1.3.3.1.6. The applicant stated that the subsurface facility will include an observation drift and three alcoves dedicated to performance confirmation. The observation drift and one alcove 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. The applicant 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 the applicant, the subsurface facility will also include two alcoves for monitoring unsaturated zone seepage: one in the nonlithophysal rock zone and another in the lithophysal zone. The applicant stated that the alcoves will be located using fracture mapping data from early stages of repository development and will be excavated off the access mains or the ECRB cross drift.

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.2.2 to review the applicant's description of the underground openings dedicated to performance confirmation. (b)(5)

(b)(5) The NRC staff's evaluation of the capability of the observation drift and alcoves' design to perform the functions through the preclosure period is presented in

SER Section 2.1.1.2.3.7 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.3.3 Emplacement Areas of the Subsurface Facility

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(a), the applicant described the emplacement areas of the subsurface facility in SAR Section 1.3 4. The applicant stated that the emplacement areas of the subsurface facility 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). The applicant stated that this drift orientation favors drift stability considering the prevalent orientation of rock joints. According to the applicant, the drifts will be laid out in a parallel pattern and spaced 81 m [266 ft] horizontally from centerline to centerline. The applicant stated that this drift spacing was chosen to prevent thermal interaction between adjacent drifts and to allow natural and thermally mobilized water percolation to drain between the drifts (SAR Section 1.3.4.2.3). The drift opening has a circular cross section with a nominal excavated diameter of 5.5 m [18 ft] (SAR Figure 1.3.4-4). The applicant stated that the total length of disposed waste packages in a drift including an end-to-end spacing of 10 cm [3.9 in] between adjacent waste packages will be limited to 800 m [875 yd] to maintain the applicant-specified ventilation efficiency. Other features of the emplacement drift design described in SAR Section 1.3.4.2.3 include the emplacement drift invert with a horizontal grade at the same elevation as the invert of the connected exhaust main, emplacement drifts excavated using a tunnel boring machine, and drift mapping after installation of the initial ground support and before installation of final ground support. According to the applicant, geologic

mapping of drifts will include documentation of fractures, fault zone characteristics, stratigraphic contacts, and lithophysal content.

The applicant described the emplacement drift ground support in SAR Section 1.3.4.4.1. According to the applicant, an initial ground support and a final ground support will be installed in each emplacement drift. The initial ground support 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. The applicant 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. The applicant stated in SAR Section 1.3.4.5.1 that the steel invert structure and crane rail will be fabricated from corrosion-resistant steel. The applicant also mentioned a third rail that will be used for power supply, but the third rail was not shown in the illustrations provided in the SAR. The crushed tuff ballast will fill the void space formed by the steel invert structure and surrounding rock. The applicant stated that the top of the ballast will coincide with the top of the steel structure.

The steel invert structure 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 YMRP Section 2.1.1.2.2 to review the applicant's description of the emplacement areas of the subsurface facility (b)(5)

(b)(5)

(b)(5) The NRC staff's evaluation of the capability of the emplacement drifts, invert structure, and ground support designs to perform the functions through the preclosure period is documented in SER Section 2.1.1.2.3.7 (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.3.4 Waste Package Transportation and Emplacement System

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described and discussed the TEV design in SAR Sections 1.2 and 1.3. The applicant used this information in the PCSA and iterative design of the TEV (SAR Sections 1.6 through 1.9). The applicant designated the TEV as ITS.

General Description of TEV and Functions

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

For the surface facilities, the applicant provided the layout of the surface rails that illustrated the specific routes of the TEV at the surface (SAR Figures 1.2.1-1 to 1.2.1-3). The applicant also provided general descriptions related to the role of the TEV in these surface facility areas, such as the CRCF and IHF. When appropriate, the applicant provided specific functions and interactions between the TEV and other SSCs in the individual rooms within these facilities. It also briefly discussed contamination surveying and interlocking system requirements prior to the TEV exiting from both the surface and subsurface facilities.

Similarly, the applicant provided the routes the TEV will follow in the subsurface facility. This information was described in the form of layouts of the facility's rail system, shown in SAR Figures 1.3.3-9 to 1.3.3-11. The applicant discussed the subsurface crane rail, which is an integrated rail system that connects the IHF and CRCF buildings to the subsurface emplacement areas. The applicant provided detailed location, length, direction, and magnitude of expected slopes that the TEV 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. The applicant also provided estimates of the TEV travel distances and travel times, such as the minimum one-way travel distance of the TEV {2,760 m [9,055 ft]} in

60 minutes and maximum one-way distance {7,200 m [23,622 ft]} corresponding to a travel time of 160 minutes, excluding stops and delays.

NRC Staff Evaluation: The NRC staff reviewed the general description of the TEV using the guidance in YMRP Section 2.1.1.2.2. The NRC staff also reviewed the applicant's description of the locations, both in the surface and subsurface facilities, in which the TEV will operate. (b)(5)

(b)(5)

TEV Design Information

The applicant 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 baseplate, third rail power with shielded conductors, sensors (speed, range, and temperature), fire-suppression system, communication devices, PLCs, interlock switches, and video cameras.

The applicant 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. The applicant specified commercially available "thruster" brakes because of their ability to utilize the TEV's own weight and motion to exert a vertical force (directly proportional to a braking force) to the top of the rail to prevent TEV movement. The applicant specified that the TEV rail wheel material will be of a lower hardness than the subsurface crane rail. This will result in more wear of TEV wheels rather than the subsurface rails, which are more difficult to repair inside the radiation environment of the drifts.

In addition to design information in the SAR, the applicant provided other supplemental references; in particular, TEV drawings, dimensions, weight, materials of construction, and subsystem specifications (BSC, 2008bz,cb, 2006aj). The applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

TEV Structural and Thermal Analysis

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

The applicant 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. The applicant'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.

The applicant provided analyses predicting the temperature within the TEV during its emplacement operations. The analyses (BSC, 2007ce) considered the geometry of the TEV, a range of heat generation of the waste packages (11.8 kW to 30 kW TAD), heating from solar energy incidents {200 cal/cm^2 [1,290 cal/in^2]} on the TEV surface, and 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 the applicant's quantification of the temperature harshness within the emplacement drifts. The applicant 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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.3.5 Waste Package Emplacement Pallet System

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described and discussed the waste package emplacement pallet design in SAR Section 1.3.4.6. The applicant proposed to use the waste package emplacement pallet to support the waste package for handling, transportation, and emplacement during the preclosure and postclosure periods. The applicant classified the waste package emplacement pallet as non-ITS because it 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, the applicant 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.

The applicant described the design and design drawings for two waste package emplacement pallet configurations that will be used at the repository: (i) the standard waste package emplacement pallet, which is designed to accommodate all waste package configurations except the 5-DHLW/DOE short waste package, and (ii) the short waste package emplacement pallet, which is specifically designed to accommodate the 5-DHLW/DOE short waste package. Both waste package emplacement pallet configurations 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), the applicant 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, the applicant 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 the applicant, using the same material on the waste package supports and the waste package outer corrosion barrier would minimize the potential for galvanic corrosion at the areas of contact between the two. The material used for the square tubes is Stainless Steel Type 316, as described by the American Society of Mechanical Engineers SA–240, UNS S31600 (2001aa). According to the applicant, this was selected because the general corrosion rate of the stainless steel tubes in the repository-relevant environment is low.

The applicant stated that the waste package emplacement pallet 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 YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.3.6 Drip Shield Emplacement System

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described the DSEG design in SAR Section 1.3.4.7.2. The applicant 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, the applicant 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.

The applicant 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. The applicant provided a more extensive discussion on the drip shield emplacement operations and its conceptual design including drive system, electrical and control systems, braking controls, cooling system, vision system, thermal and radiation monitoring system, fire protection, and communication systems in a supplemental document (BSC, 2007cf). The applicant also provided the specific routes for the DSEG from the surface to the subsurface facility. Furthermore, the applicant described the motion of the DSEG including stops, inspection, and calibration of its precise position, which closely resembles the TEV operational sequence toward the subsurface.

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

The applicant 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. The applicant'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. The applicant 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 YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.4 Description of Waste Form Characteristics

In accordance with 10 CFR 63.21(c)(4), the applicant must provide 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

To demonstrate compliance with 10 CFR 63.21(c)(4) and 63.112(a), the applicant 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). The applicant discussed how it uses the CSNF characteristics in the PCSA (SAR Section 1.8.1.3.1) and creates representative CSNF characteristics for the applicant's ALARA analysis (SAR Section 1.10.3.4.1).

The applicant 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]}. The applicant summarized the physical, thermal, nuclear, and source term characteristics in SAR Sections 1.5.1.3.1.1, 1.5.1.3.2, 1.5.1.3.3, and 1.5.1.3.4, respectively. The physical, thermal, nuclear, and source term characteristics of naval SNF were discussed in SAR Sections 1.5.1.4.1.1, 1.5.1.4.2, 1.5.1.4.3, and 1.5.1.4.4.

Physical Characteristics

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, the applicant discussed these isotopes, which are available for instantaneous release when the cladding is breached.

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). The applicant used 500 kg [0.5 MTHM] per canister equivalence for DOE HLW to determine how many canisters can be disposed of at the repository. The applicant expects to receive approximately 9,300 DOE HLW canisters containing a total of 4.7 × 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.

The DOE SNF waste form comes from a range of backgrounds with a variety of fuel types, moderators, enrichments, shapes, and chemistries. The approximately 2.3 × 10^6 kg [2,265 MTHM] of DOE SNF proposed for disposal at the Yucca Mountain site may be stored in 2,500 to 5,000 DOE canisters. In SAR Section 1.5.1.3.1.1.1, the applicant developed 34 groups of DOE SNF, including naval SNF, that are based on the characteristics the applicant believes have the greatest impact on release and criticality. SAR Table 1.5.1-23 listed these 34 fuel groups and described how they 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. 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 the applicant's description of physical characteristics of HLW using the guidance in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Thermal Characteristics

SAR Table 1.5.1-11 provided the average (25 years' cooling time) and maximum thermal power (5 years' cooling time) for PWR and BWR assemblies. SAR Figure 1.5.1-6 showed thermal power per assembly as a function of time. The applicant chose to analyze limiting values so that the uncertainties are bounded by the maximum cases.

The applicant calculated the heat generation rate from the HLW radionuclide inventory and displayed the results in SAR Table 1.5.1-19. The applicant 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. The applicant 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 the applicant, naval SNF canisters will not be shipped until the heat output, when received at the GROA is less than or equal to 11.8 kW. Those that do not meet this limit will not be disposed of (SAR Table 5.10-3). For preclosure, the 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 the applicant's description of thermal characteristics and heat generation rate using the guidance in YMRP Section 2.1.1.2.2 (b)(5)
(b)(5)

Nuclear Characteristics

The applicant 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 the applicant used for its HLW glass consequence analysis.

The applicant 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, the applicant sorted the DOE SNF into nine groups with a representative DOE SNF for each group. The groups and their representatives were listed in SAR Section 1.5.1.3.1.1.3 and analyzed in SAR Sections 1.14.2.3.2.3 and 2.2.1.4.1 for preclosure and postclosure, respectively. The applicant listed the postclosure critical limits for the nine groups of DOE SNF analyzed for criticality purposes in SAR Table 2.2-11.

SAR Table 1.5.1-32 presented an initial radionuclide inventory developed for a representative naval SNF canister, with an assumed cooling time of 5 years

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of the radionuclide inventories using the guidance in YMRP Section 2.1.1.2.2 (b)(5)
(b)(5)

Source Term Characteristics

The applicant considered the PWR fuel assembly to be bounding and used it in shielding design (e.g., worker dose assessments, process facility design, ALARA) and repository consequence analysis for preclosure. In SAR Table 1.10-18, the applicant provided the radiation sources from the maximum PWR assembly {5 wt% initial enrichment, 0.08 GWd/kg [80 GWd/MTU] burnup, and 5-year cooling}, which 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 WGd/kg [60 GWd/MTU] burnup, 10-year cooling}, which the applicant 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, the applicant used representative PWR {4.2 wt% initial enrichment, 0.05 GWd/kg [50 GWd/MTU] burnup, and 10-year cooling} and BWR {4 wt% initial enrichment, 0.05 GWd/kg [50 GWd/MTU] burnup, and 10-year cooling} assemblies to generate the radionuclide inventories in SAR Table 1.8-3. For airborne releases from Category 1 and Category 2 event sequences, radionuclide inventories (see SAR Table 1.8-3) from the maximum assemblies were used.

The applicant 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. The applicant stated that the maximum per canister inventories were provided in SAR Table 1.5.1-21 and used in the PCSA. The applicant also stated that the Savannah River Site HLW canister 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 SAR Table 1.8-30 and 31 dose limits are met.

The applicant discussed DOE SNF shielding sources term characteristics in SAR Section 1.10.3. Neutron and gamma energy spectra and source intensity and fuel composition the applicant used in shielding calculations for a homogenized TRIGA-FLIP fuel 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, the applicant discussed the gamma and neutron source terms for the naval SNF canisters, which the applicant 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. The applicant 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 the applicant's description of the radiation source term using the guidance in YMRP Section 2.1.1.2.2. (b)(5)
(b)(5)
(b)(5) The NRC staff's evaluation of the source terms is documented in SER Section 2.1.1.5.3.2 (b)(5)
(b)(5)

(b)(5)

The applicant made numerous assumptions about the characteristics of the waste forms as part of its licensing basis, which are provided in various parts of the SAR. (b)(5)

(b)(5)

NRC Staff Conclusion

On the basis of the NRC staff's evaluation of the applicant's description of HLW, CSNF, and naval SNF to be disposed at the Yucca Mountain repository (b)(5)

(b)(5)

2.1.1.2.3.4.2 Description of Low-Level Radioactive Waste

The applicant 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, solid LLW was discussed in SAR Section 1.4.5.1.1.1, liquid LLW in SAR Section 1.4.5.1.1.2, and gaseous LLW in SAR Section 1.4.5.1.1.3. SAR Table 1.4.5-1 listed the expected annual LLW volumes. SAR Table 1.4.5-2 listed the expected LLW radionuclide concentration. SAR Section 1.10.3.4.5 provided the LLW source terms. The applicant 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. The applicant 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 YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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 the applicant'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

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described and discussed the waste package design in SAR Section 1.5.2 and other applicable sections of the SAR (e.g., 1.2.1.4.1, 1.2.4.2.3.1.3, 1.3.1.2.5, and 2.3.6.7.4). The applicant proposed to use the waste package as an engineered barrier for disposal of SNF and HLW. The waste packages would be loaded with TAD, HLW, DOE, and naval SNF canisters at the surface facilities. The applicant classified waste packages as ITS because they 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, the applicant classified waste packages as ITWI because, after repository closure, they are relied upon to comply with postclosure performance objectives.

The applicant described six waste package configurations: (i) waste package loaded with one 21-PWR/44-BWR TAD canister, (ii) waste package loaded with five short HLW canisters and one short DOE SNF canister in the center location (5-DHLW/DOE short codisposal), (iii) waste package loaded with five long HLW canisters and one long DOE SNF canister in the center location (5-DHLW/DOE long codisposal), (iv) waste package loaded with two DOE multicanister overpacks (MCO) and two long HLW canisters (2-MCO/2-DHLW), (v) waste package loaded with one short naval SNF canister, and (vi) waste package loaded with one long naval SNF canister.

The approximate percentage by waste package configuration was provided in SAR Table 1.5.2-2. According to this table, the configuration of the waste package with one 21-PWR or one 44-BWR TAD canister is the most commonly used configuration, which accounts for approximately 71 percent of all waste packages. Also, the waste package configuration with five long HLW and one long DOE SNF canisters or five short HLW and one short DOE SNF canisters accounts 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).

(b)(5) As the applicant 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). The applicant stated the inner vessel will be stamped with an N symbol

and, therefore, is identified as a pressure vessel. (b)(5)
(b)(5)
(b)(5) However, according to the applicant, 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].

The applicant 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 the applicant, 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 YMRP Section 2.1.1.2.2. (b)(5)
(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.5.2 Waste Canisters

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described the design of waste forms and waste packages in SAR Section 1.5.1. The applicant used this information in its PCSA and design of the waste canisters (SAR Sections 1.6 through 1.9). The SNF and vitrified HLW will be shipped to Yucca Mountain in TAD canisters, DOE standardized canisters, HLW canisters, and DPCs. On the basis of the PCSA, the applicant designated these waste canisters as ITS.

TAD Canisters

The applicant provided the performance specifications in the 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, the applicant also provided a conceptual drawing of the TAD canister. The information the applicant provided included key dimensions, weights, fabrication specifications and materials of construction, sealing (welding) specifications, and drying processes. In SAR Section 1.5.1.1.1.2.1.4, the applicant presented the principal physical characteristics of the proposed TAD canister. The TAD canister will have a diameter of 1,689 mm [66.5 in], a minimum height of 4,724 mm [186.0 in], and a maximum height of 5,385 mm [212.0 in]. For a TAD canister 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, the applicant 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.

The applicant specified that the fabrication of the TAD canister shell and lid will follow 2004 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (American Society of Mechanical Engineers, 2004aa). The TAD will be constructed of Type 300-series stainless steel, as per ASTM A 276–06 (ASTM International, 2006ab), for the canister shell and structural internals. The applicant chose this material because of its resistance to degradation. The TAD canister and its basket materials are required to be compatible with either borated or unborated pool water because the canister will be submerged during fuel loading at the repository and/or reactor sites. With respect to the canister internals, the applicant specified that the neutron absorbers, necessary for criticality safety control, will be fabricated from borated stainless steel with a boron content of 1.1 wt% to 1.2 wt% and will meet ASTM A 887-89, Grade "A" alloys (ASTM International, 2004ab). The neutron absorber plates 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, the applicant described the TAD canister containment characteristics pertaining to welding of the TAD lid. The applicant 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).

The applicant 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 the applicant's description and design information for the TAD canisters using the guidance in YMRP Section 2.1.1.2.2. 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.

(b)(5)

(b)(5)

DOE Standardized Canister

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 the applicant stated that the functions and requirements are the same. The applicant 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]. The applicant stated that these two standardized canisters will have two lengths: 3.1 and 4.6 m [10 and 15 ft]. The maximum allowable weight of the standardized canister including its contents 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, the applicant described the operational processes used for drying, sealing, inerting, and leak testing the canisters. The applicant specified that the inerting process will utilize an inert gas such as helium. In terms of sealing, the standardized canister boundary components are joined with full-penetration welds. The applicant 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). The applicant 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. The applicant will demonstrate leak tightness by utilizing a helium leak test in accordance with 2001 ASME Boiler and Pressure Vessel Code, Section V, Article 10, Appendix IV (American Society of Mechanical Engineers, 2001aa).

In SAR Section 1.5.1.3.1.2.6.1, the applicant presented information on the structural design methodology used for the standardized canister. The applicant stated that the standardized canisters have been designed to the 1998 ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998aa). The applicant also described the finite element analyses performed on the canisters by which the design can be evaluated.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of DOE standard canisters using the guidance in YMRP Section 2.1.1.2.2. (b)(5)
(b)(5)

HLW Canister

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, the applicant 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. The applicant provided the materials of construction, welding, weld testing, and leak testing in

SAR Table 1.5.1-18 for the HLW canisters. Specifically, the canisters will be fabricated from 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 the applicant's description of HLW canisters using the guidance in YMRP Section 2.1.1.2.2. (b)(5)
(b)(5)

(b)(5) 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. (b)(5)
(b)(5)
(b)(5) In SAR Section 1.7.2.3.1, the applicant discussed a full-scale experimental testing of HLW canisters. (b)(5)
(b)(5)

Dual Purpose Canister

The applicant 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, the applicant 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 the applicant 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 the applicant's description of DPC using the guidance in YMRP Section 2.1.1.2.2. (b)(5)
(b)(5)

(b)(5) The evaluation of design and analysis of the DPC is in SER Chapter 2.1.1.7.

Naval Canister

In SAR Section 1.5.1.4.1.2.1, the applicant 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. The applicant 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, a 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, the applicant 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 the applicant's description of the naval SNF canisters using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.5.3 Aging Overpack and Shielded Transfer Casks

To demonstrate compliance with 10 CFR 63.21(c)(3(i) and 63.112(a), the applicant 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.

Aging Overpacks

Aging overpacks were detailed in SAR Section 1.2.7, which discussed the proposed AF. The applicant 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. The applicant 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, the applicant 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, the applicant provided a general description of vertical aging overpacks. The applicant described the vertical aging overpack as a cylinder with a metal liner surrounded by steel-reinforced concrete that 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.

The applicant described the horizontal aging module to be a boxlike, thick-walled reinforced concrete structure having a minimum concrete shielding thickness of 0.9 m [3 ft], a maximum height of 6.4 m [21 ft], a maximum width of 2.6 m [8.5 ft], and a minimum length {with the minimum of 0.9 m [3 ft] of shielding} of 7.1 m [23 ft 4 in]. A shield wall 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. The applicant stated that the concrete used to construct the aging overpacks will follow ACI 349–01/349R–01 (American Concrete Institute, 2001aa) and the reinforcing steel should comply with ASTM A 706/A 706M–06a or ASTM A 615/A 615M–06a

(ASTM International, 2006ac,ad). In addition, the aging overpack design will follow ASCE/SEI 43–05 (American Society of Civil Engineers, 2005aa); ACI 349–01/349R–01 (American Concrete Institute, 2001aa) and ANSI/ANS-6.4–1997, as described in American Nuclear Society, Appendix A (1997aa).

NRC Staff Evaluation: The NRC staff reviewed the aging overpack information using the guidance in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

Shielded Transfer Cask

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. The applicant 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. The applicant classified shielded transfer casks as ITS, and the nuclear safety design bases and design criteria were given in SAR Table 1.2.5-3.

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. The applicant 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 YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.5.4 Drip Shield

To demonstrate compliance with 10 CFR 63.21(c)(3(i) and 63.112(a), the applicant described and discussed the drip shield design in SAR Section 1.3.4.7. The applicant 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. The applicant classified the drip shield as non-ITS because it is not relied on to prevent or mitigate Category 1 and Category 2 event sequences. The applicant classified the drip shield as ITWI because it is relied upon to prevent or substantially reduce the rate of movement of water and radionuclides.

The applicant 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, the applicant provided drip shield main assembly, subassemblies, and components drawings (DOE, 2009dr).

According to the applicant, the drip shields are designed to form a continuous barrier throughout the entire length of the emplacement drift by interlocking the drip shield segments. The applicant stated that the drip shield 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, the applicant provided a sequence of isometric sketches that illustrated the drip shield interlocking process and figures that demonstrated the height clearance required to interlock two drip shields (DOE, 2009dr).

The applicant stated that, except for the attachment to the Alloy 22 base, drip shield components are intended to be connected to each other by welding. According to the applicant, 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. The applicant included codes and standards governing physical and mechanical properties (e.g., density, elongation, yield and ultimate tensile stresses) of Titanium Grades 7 and 29 in SAR Table 1.3.2-5. In response to the NRC staff RAI (DOE, 2009dr) regarding the codes and standards for the drip shield design and fabrication, the applicant extracted codes and standards as applicable for materials, welding, postweld heat treatment, and postweld nondestructive examination of the drip shield from BSC (2007bu). The applicant stated that the codes and standards cited in the prototype specification were adopted from the ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa) and American Welding Society standards for welding. In addition, the applicant stated that the prototype program will be used to demonstrate and confirm the design suitability and progressively develop and refine the production fabrication process.

According to the applicant, for similar welds including Titanium Grade 7 to Titanium Grade 7 and Titanium Grade 29 to Titanium Grade 29, the filler metal matching the base metal will be used. However, for Titanium Grade 7 to Titanium Grade 29 welds, Titanium Grade 28 filler material will be used. In 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, the applicant stated that this dissimilar welding joint mitigates hydrogen embrittlement. In addition, the applicant cited two examples from the literature (American Welding Society, 2007aa; Boyer, et al., 1994aa) to indicate that welding joints similar to Titanium Grade 7/Grade 28/Grade 29 have been used in the industry.

NRC Staff Evaluation: The NRC staff reviewed the drip shield information using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.6 Description of Geologic Repository Operations Area Processes, Activities, and Procedures, Including Interfaces and Interactions Between Structures, Systems, and Components

In this section, the NRC staff evaluates the applicant's description of operational processes in the surface and subsurface facilities of the GROA and onsite transportation to determine compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a). 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 the applicant stipulates. These topics are covered in SER 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 SER Section 2.1.1.2.3.6.2.

2.1.1.2.3.6.1 Operational Processes

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described GROA process activities in SAR Section 1.2 to review identification of hazards and event sequences in the PCSA. The applicant'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 SER Sections 2.1.1.2.3.2.2 and 2.1.1.2.3.2.5, respectively.

Surface Facility Operations

The applicant presented its overview of the CRCF operational processes in SAR Section 1.2.4.1.2 and the detailed description of the operations for the cask handling, canister transfer, waste package closure, and waste package load-out subsystems in SAR Sections 1.2.4.2.1.2, 1.2.4.2.2.2, 1.2.4.2.3.2, and 1.2.4.2.4.2, respectively. The overview of the operational processes for the IHF 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.

The applicant 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.

In the cask handling subsystem, transportation casks on rail- or truck-based trailers are received and after removing the impact limiters moved onto the cask transfer trolley. 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), the applicant 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.

Comment (b)(5)
(b)(5)

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 YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5) The detailed review of the operations as it pertains to the identification and quantification of initiating events is described in SER Sections 2.1.1.3.3.2.1 and 2.1.1.3.3.2.3.4.

(b)(5)

(b)(5) The applicant, 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 SER Section 2.1.1.4.3.3.1.1). (b)(5)

(b)(5)

Intrasite Surface Operational Processes

The applicant described the intrasite operations in SAR Section 1.2.8.4 and in supplemental documents (BSC, 2008at,au). These documents described activities related to the 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.

The applicant 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. The applicant provided a flow diagram showing the flow path of each type of waste container and the transportation equipment used to move the waste container from one surface facility to another. The surface facilities included in the flow diagram were the Cask Receipt Security Station, truck buffer area, railcar buffer area, IHF, CRCF, WHF, RF, and AF.

The applicant identified and described that several types of site transportation SSCs would be utilized: the site transporter (SAR Section 1.2.8.4.1), the cask tractor and cask transfer trailer (SAR Section 1.2.8.4.2), and the prime mover (SAR Section 1.2.8.4.3). All three are ITS and are further described and evaluated in SER Section 2.1.1.7. In BSC Appendices B1.4, B2.3, and B3.3 (2008au), the applicant also listed the dependencies and interactions associated with each of the three transportation systems mentioned previously. It included functional, environmental, spatial, human, and external events interactions.

For human-related operations, the applicant listed human interactions and described human-induced failures during intrasite operations. For the site transporter, in particular, the applicant provided additional detail on human interfaces. This 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 (b)(5)

(b)(5)

Subsurface Operational Processes

The applicant 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. The applicant provided a process flow diagram in a 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. The applicant also provided a process flow diagram for the drip shield emplacement operations in BSC (2008bj) and indicated that the only ITS SSC associated with the subsurface operations 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 baseplate, 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).

The applicant also briefly described the construction operations that will occur at the same time as emplacement operations during a portion of the preclosure period, 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.

The applicant 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. The applicant described the use of high-intensity lights and a camera onboard the TEV to provide feedback to operators in the control center. The applicant also described the use of PLCs that accept initiating commands from the operators to execute predefined, preprogrammed instructions and maneuvers. The applicant 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 YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

Waste Form Throughput

According to the applicant, 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. The applicant 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. The applicant indicated in SAR Section 1.5.1 and described in BSC (2007bh) that about 90 percent of the CSNF will be loaded in TAD canisters prior to being placed in transportation casks for shipping. The remaining 10 percent will be either in the form of bare fuel assemblies (uncanistered) or loaded in DPCs before being shipped to the repository in transportation casks. 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 the applicant (BSC, 2007bh), most canistered waste will be either (i) transferred into aging overpacks for aging and transferred into waste packages for disposal after the waste is aged or (ii) directly transferred into waste packages for emplacement 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 the applicant'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. The annual rate information is necessary for the applicant to assess normal operational exposures for demonstrating compliance with 10 CFR 63.111(a)(1).

NRC Staff Evaluation: The NRC staff reviewed the applicant's waste throughput information using the guidance in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

Comment (b)(5)

Operational Period

The applicant indicated in the License Application 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. The applicant 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 SER Section 2.1.1.3.

NRC Staff Evaluation: The NRC staff reviewed the information on the operational period the applicant proposed using the guidance in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.6.2 Instrumentation and Control Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3)(i), and 63.112(a), the applicant provided information on instrumentation and control (I&C) and related communications systems in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.1–1.3.6, 1.4.2, 1.9, 1.13, 5.5, and 5.6 for the surface and subsurface facilities. This information also 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 the applicant adequately described (i) control philosophy, conceptual process diagrams, and digital control logic diagrams for ITS and non-ITS controls; (ii) design codes, standards, and acceptable industry practices used for ITS and non-ITS controls; and (iii) plans and procedures for initial startup, operation, maintenance, and periodic testing of ITS and non-ITS controls. For I&C and related communication systems to be used in the underground nonaccessible areas, the NRC staff also evaluated the adequacy of the I&C and related communication systems design descriptions to support potential operations and/or waste retrieval subject to 10 CFR 63.111(e).

Surface and Subsurface Facility Instrumentation and Control

Most normal facility operations 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 may be found 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. The applicant 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. The applicant 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), the applicant identified the ITS controls and the related safety functions that will be implemented for the CRCF. These ITS controls were considered representative of designs for ITS controls for other surface facilities. The applicant provided supplemental information regarding the applicability of cited principal codes and standards (DOE, 2009dl) and discussed the proposed application of specific sections of principal codes and industry standards that the applicant intends to apply to the final design of the ITS controls (DOE, 2009do). The applicant proposed to use IEEE 308, 379, 384, and 603 (Institute of Electrical and Electronics Engineers, 2001aa,ab 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. The applicant 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. The applicant proposed that ITS I&C equipment associated with ITS cranes will be environmentally qualified in accordance with IEEE 323–2003 (Institute of Electrical and Electronics Engineers, 2004aa) and the equipment qualification program will be developed consistent with Regulatory Guide 1.89 (NRC, 1984aa). SAR Sections 5.5 and 5.6 stated that channel functional tests and channel calibrations for control systems will be performed (specifically, tests for cranes, trolleys, HVAC systems, and TEV ~~are~~ 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 before receipt and possession of HLW (DOE, 2009dk,dm). The applicant also stated that preventive maintenance of hardwired ITS interlocks will be based upon manufacturer's recommendations, industry codes and standards, and equipment qualification and reliability requirements from the PCSA, as identified in SAR Section 1.9. The applicant further stated that potential future upgrade of the ITS interlocks will be based 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. The applicant provided assurance 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 the applicant's description of I&C SSCs using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

(b)(5) A safety evaluation of the design of ITS I&C SSCs is included in SER Sections 2.1.1.6.3.2.8.2.1 and 2.1.1.7.3.5.

(b)(5)

(b)(5)

(b)(5) More specific information regarding the function and use of particular non-ITS I&C SSCs was generally provided within descriptions of other GROA non-ITS facilities and operating systems in the SAR. (b)(5)

(b)(5)

Special Vehicle Instrumentation and Control

The ITS TEV was described in SAR Sections 1.3.2.1, 1.3.3.5.1.1, and 1.3.4.8. The ITS electrical components of the TEV include the mechanical location switch and shield door motors (DOE, 2009dp). The applicant confirmed that design criteria in IEEE 384–1992 and IEEE 603–1998 (Institute of Electrical and Electronics Engineers, 1998aa,ab) will be used for the TEV ITS electrical components and interlocks (DOE, 2009dp).

PLCs are used for remote-controlled, non-ITS operations of the TEV. In response to an NRC staff RAI (DOE, 2009dm), the applicant confirmed the use of ASME NOG-1–2004 Sections 6410 to 6419 (American Society of Mechanical Engineers, 2005aa), NUREG/CR–6090 (Wyman, 1993aa), and portions of International Electrotechnical Commission (IEC) 61131-3 (Part 3) (International Electrotechnical Commission, 2003aa) for the PLC design. The applicant stated that the onboard PLC 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. The applicant 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. ~~Since~~ 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 and are not anticipated ~~prior to commencement of~~before waste emplacement operations begin.

In SAR Section 4.2.1.8, the applicant indicated that while the technology to remotely inspect emplacement drifts is available, the high-temperature and high-radiation environments representative of postemplacement conditions within the drifts will require developing a first-of-a-kind application of existing technologies to build ROVs able to perform the intended operations and inspections. There are similarities between the design of the emplacement drift ROV and the design of the TEV and the DSEG pertaining to how the ROV 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 the applicant's description of special vehicle I&C using the guidance in YMRP Section 2.1.1.2.2. (b)(5)
(b)(5)

(b)(5) A safety evaluation of the design of TEV ITS I&C SSCs is included in SER Sections 2.1.1.6 and 2.1.1.7.3.5.

(b)(5)

The NRC staff reviewed descriptions, outline drawings, and other diagrams for the planned ROVs provided in the SAR and BSC (2008ca). (b)(5)
(b)(5)

The applicant described these ROVs as "yet-to-be-designed or specified (BSC, 2008ca)"
(b)(5)

Subsurface Ventilation Instrumentation and Control

Drift temperature, pressure, and relative humidity are very important parameters indicating the effectiveness of the ventilation system and are closely monitored. The applicant indicated that the proposed sensors/monitors needed to determine the effectiveness of the subsurface ventilation system will not be required to operate under extreme environmental conditions (DOE, 2009dm), 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, the applicant indicated that selection of these industrial grade components will be based on the guidance provided within Regulatory Guide 1.23 (NRC, 2007aa) and in accordance with applicable sections of ANSI/ANS-3.1 1–2005 and EPA-454/R-99–005 (American Nuclear Society, 2005ab; EPA, 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 ventilation system provided in the SAR using the guidance in YMRP Section 2.1.1.2.2. (b)(5)

(b)(5)

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.

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

The DCMIS architecture 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). The applicant 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. The applicant 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. The applicant 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), although the design time period for such sustained operations was not provided.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of DCMIS using the guidance in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

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).

The applicant identified codes and standards applicable to the communications systems in SAR Section 1.4.2.4.3.

Radio-frequency wireless transmission communications systems 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, the applicant (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.

The applicant 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. The applicant 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.

The applicant 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. The applicant also suggested use of a slotted microwave guide system (DOE, 2009ee) as an alternate communications approach. The system, according to the applicant, would provide payload data rates of up to 54 Mbps and should provide adequate capacity for communications with the TEV. The applicant also described an additional alternative communication system using power line carrier transmission techniques operating through vehicle sliding electrified third rail contacts. Bandwidth of up to 100 Mbps through static power lines is based on proposed IEEE P1901 (Institute of Electrical and Electronics Engineers, 2010aa). The applicant committed to 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. The applicant'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.

The SAR contained no description of communications 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). 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 the applicant's description of the non-ITS communications systems in the GROA facilities using the guidance in YMRP Section 2.1.1.2.2. (b)(5) (b)(5) The NRC staff reviewed the applicant's response to an NRC staff RAI (DOE, 2009ee) (b)(5) (b)(5) (b)(5) (b)(5) 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. (b)(5) (b)(5)

The applicant committed to 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) (b)(5) (b)(5) The applicant described the ROVs as "yet-to-be-designed or specified." (b)(5) (b)(5) (b)(5)

(b)(5)

(b)(5)

Radiation/Radiological Monitoring System

The function of the RMS is radiological monitoring of both surface and subsurface facilities during the preclosure period. The applicant categorized the RMS as non-ITS. SAR Figure 1.4.2-3 provided the RMS functional block diagram for the GROA. The major components 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 These codes and standards contain recommendations for obtaining valid samples of airborne radioactive material in effluents and the guidelines for sampling from ducts and stacks. The applicant stated (DOE, 2009dm) that evaluation and selection of area radiation monitors and continuous air monitors would follow the guidance of ANSI/ANS-HPSSC-6.8.1–1981, ANSI N42.17B–1989, and ANSI N42.18–2004 (American Nuclear Society, 1981aa,1989aa,2004aa). The applicant confirmed that only the sensor probes for the instruments located at the ventilation shaft collars will be exposed to greater than ambient temperatures, as the exhaust air 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 the applicant's description of RMS using the guidance in YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

(b)(5)

Environmental/Meteorological Monitoring System

The environmental/meteorological monitoring system, which the applicant 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. The applicant 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 YMRP Section 2.1.1.2.2 (b)(5)

(b)(5)

Commitments

- The applicant commits to 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 ~~which~~ that are planned to operate in nonaccessible areas (BSC 2008ca).

- The applicant commits to provide a complete design description on communications SSCs located in nonaccessible areas including design, construction, connections, configurable switching (i.e., at turnouts), and plans for inspection, observation, and

maintenance and repair, if needed, during the preclosure period for all ROVs ~~which~~ that are planned to operate in nonaccessible areas and must communicate with remote operators/inspectors (DOE, 2009ee).

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.7 Design of Subsurface Facility Structures, Systems, and Components

This SER section evaluates the design of subsurface structures and systems to determine their capability to perform functions that the applicant credited to the structures and systems to satisfy the NRC regulatory requirements in 10 CFR 63.111, 63.112, 63.113, and 63.114. The NRC staff's evaluation of the design of SSCs ITS is presented in SER Chapter 2.1.1.7. The evaluation in this section focuses on the design of structures and systems that are not evaluated in SER Chapter 2.1.1.7 but which the applicant relied on to perform functions important to subsurface facility operations relevant to the applicant's demonstration of compliance with NRC regulations.

To address the GROA performance objectives in 10 CFR 63.111 and 63.113, the applicant described the subsurface facility operations in SAR Section 1.3.1.2. The applicant 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. The applicant 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, the applicant indicated that underground openings and the inverts will remain stable and retain their as-designed alignments and grades through the preclosure period. The applicant (DOE, 2009ed) stated that it established 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 SER Section 2.1.2, the applicant 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, the applicant 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, the applicant relied on the functions to support capability to install drip shields to satisfy the geometry and interlocking that the applicant specified for postclosure performance assessment.

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

2.1.1.2.3.7.1 Thermal Load and Ventilation Design

To demonstrate compliance with 10 CFR 63.21(c)(3)(i) and 63.112(a), the applicant described and discussed the thermal management and loading strategy in SAR Section 1.3.1.2.5, and the subsurface facility ventilation design in SAR Section 1.3.5. The applicant categorized the subsurface ventilation system as non-ITS, because it 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

The applicant 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 ~~based on~~ on the basis of several inputs such as waste inventories at utilities and queuing priorities established through agreement between, for example, the applicant and utilities (BSC, 2007cb). The applicant's analysis also assumed that TAD canisters having a heat load as high as 22.0 kW 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, the applicant used the estimated representative limiting waste stream to determine the waste package emplacement sequence that would result in meeting the local thermal loading condition, such as the midpillar index temperature (BSC, 2007cc).

The third and final step of the applicant's analysis involved evaluating the thermal-hydrologic, geomechanical, and geochemical response to the loading arrangement determined in the previous step (SNL, 2008ai). The applicant applied a number of criteria (i.e., waste package heat load at receipt and emplacement, waste package canister types, and line load limit) that were described in SAR Section 1.3.1.2.5. In its design analysis (SNL, 2008ai) and in response to the NRC staff RAI (DOE, 2009ea), the applicant stated that emplacement will take place with three constraints: (i) the seven-waste-package running average midpillar temperature will be a maximum of 96 °C [205 °F], (ii) the maximum thermal load per waste package will be 18 kW, and (iii) the maximum average line load will be 2.0 kW/m [0.61 kW/ft]. According to the applicant, the proposed thermal emplacement loading plan will result in complying with 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 the applicant 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), the applicant asserted that ventilation efficiencies were calculated for a higher heat load of 2kW/m [0.61 kW/ft] that justified using an integrated efficiency value of 86 percent (BSC, 2008by). The proposed waste package arrangement analysis assumed phased, time-dependent emplacement with ventilation lasting up to 100 years. A waste package will be subjected to a minimum of 50 years to a maximum of 100 years of cooling by ventilation, depending on the emplacement time of the waste package. The applicant performed a thermal analysis to show that the thermal load reference case bounds any thermal loading scenario on the basis of the proposed emplacement thermal load strategy. Results of a sample analysis were shown in SAR Figure 1.3.1-6.

NRC Staff Evaluation: The NRC staff reviewed the information on thermal loading strategy and thermal management using the guidance in YMRP Section 2.1.1.7.2.3. The NRC staff evaluated the applicant-provided information on design assumptions, constraints, design technical basis, uncertainty, and analytical or modeling techniques employed.

(b)(5)

(b)(5)

(b)(5) the applicant 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. (b)(5)

(b)(5)

Subsurface Facility Ventilation System

The applicant 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, the applicant 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. The applicant plans to provide a nominal airflow rate of 15 m^3/s [32,000 cfm] in each emplacement drift with thermal loading of up to 2.0 kW/m [0.61 kW/ft] and, if required, can vary the drift airflow rate between 0 and 47 m^3/s [0 and 100,000 cfm]. The applicant stated that the total power required for ventilation fans at the exhaust shaft will be approximately 1,343 kW [1,800 hp].

The applicant provided information on the operability of ventilation system components under normal and off-normal conditions. According to the applicant, large-diameter exhaust shafts will normally have two fans operating simultaneously, and each of the fans individually is capable of producing approximately 70 percent of the required airflow rate if the other fails. The applicant 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). The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Comment (b)(5)

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. (b)(5)
(b)(5)
(b)(5) the applicant provided an analysis in SAR Section 1.3.5.3.2.1 concluding that in the absence of any ventilation for 30 days, in-drift components will not exceed their limiting temperatures. During a power failure, the emplacement access door will temporarily stop operation as the motorized actuator needs electrical power to function. (b)(5)
(b)(5)

Subsurface Ventilation System Maintenance

The applicant described the subsurface ventilation system maintenance considerations in SAR Section 1.3.5.1.5. It asserted that ventilation fans will be monitored and maintained according to manufacturer guidelines and the fans will be located on the surface, providing easy access for maintenance. According to applicant, emplacement access doors will require regular periodic inspection with the bulkhead and frame requiring minimal maintenance. The applicant stated that emplacement door components will have a modular design that facilitates easy replacement. The applicant does not plan any routine maintenance activities for door actuators, which will be remotely monitored and replaced, if necessary. The applicant also anticipates that 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. The applicant considered three different cases: (i) analysis of complete ventilation shutdown in the absence of natural convection, (ii) naval SNF behavior under ventilation shutdown with natural convection, and (iii) thermal effect of drift obstruction In the first analysis, the applicant demonstrated that waste package components will not reach 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. The applicant 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, the applicant demonstrated that the ventilation system is capable of maintaining normal airflow with 94 percent localized blockage of a single emplacement drift. The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)
(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.7.2 Underground Openings in Accessible Areas

To demonstrate compliance with 10 CFR 63.111(d), 63.111(e), 63.112(d), 63.112(e), and 63.114(a), the applicant provided the design of underground openings in accessible areas of the subsurface facility in SAR Section 1.3.3. The applicant classified the subsurface facility into nonemplacement areas (SAR Section 1.3.3) and emplacement areas (SAR Section 1.3.4). In addition, in response to an NRC staff RAI on the applicant's approach to assure adequate functionality of the openings and their SSCs during the preclosure period, the applicant classified the openings as accessible or nonaccessible on the basis of personnel accessibility because of thermal and radiation conditions the applicant determined (DOE, 2009bb). According to the applicant (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 the applicant to assure adequate functionality of the openings. The applicant stated in SAR Section 1.3.3.2 that the horizontal openings will be excavated using tunnel boring machines and vertical openings with raise boring. The applicant also stated that it will monitor the performance of the accessible openings through regular visual inspection by qualified personnel and will implement a geotechnical instrumentation program to measure drift convergence, ground support loads, and potential overstressed zones. The monitoring and maintenance program will be performed using methods similar to those used in underground openings in civil and mining industries (DOE, 2009bb).

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.7.2.3 to review the applicant's design of underground openings in the accessible areas of the

subsurface facility. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.2.3.7.3 Underground Openings in Nonaccessible Areas

To demonstrate compliance with 10 CFR 63.111(d), 63.111(e), 63.112(d), 63.112(e), and 63.114(a), the applicant provided the design of underground openings in nonaccessible areas of the subsurface facility in the SAR Sections 1.3.3 and 1.3.4. According to the applicant (DOE, 2009bb), the nonaccessible openings consist of the emplacement drifts, turnouts, exhaust mains, exhaust shafts, and shaft access drifts. The applicant expects high radiation levels in the emplacement drifts and turnouts (SAR Figure 1.3.3-13) and thermal and radiological conditions in the openings on the exhaust-air side of the emplacement drifts (exhaust mains, exhaust shafts, and shaft access drifts) that are high enough to make these openings inaccessible to personnel. The applicant 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), the applicant 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 the applicant'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 the applicant's approach to the subsurface facility opening design. The applicant 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. The applicant 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, the applicant considered the effects of *in-situ* stress, thermal loads, and seismic ground motions and performed analyses to examine the stability of the openings with and without ground support. The applicant concluded, on the basis of the analyses, that the openings will be stable without ground support but the surrounding rock will

sustain stress-induced damage within a zone approximately 0.3–1.0 m [1–3.28 ft] from the circumference, around the entire opening in the lower quality rock categories but only in the roof areas for higher quality rock categories, as outlined in BSC Section 6.4.3 (2007an) and BSC Section 7.2 (2007ao). The applicant also concluded that the repository thermal loading and potential seismic ground motion will not have a significant effect on the damaged zone. According to the applicant in BSC 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). The applicant also assessed the effect of ground support on stability of the openings. The applicant'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, the applicant 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), the applicant will monitor the mechanical performance of the nonaccessible openings using remotely operated equipment to assess maintenance needs. According to the applicant, concrete liners in the exhaust shafts will be inspected for cracks and voids and any evidence of spalling; the exhaust mains and shaft access drifts will be inspected for indications of ground support damage or roof sagging; and shotcrete will be inspected for cracks, delamination, void development, spalling, or chemical alteration (DOE, 2009bb). For the emplacement drifts, the applicant will also monitor convergence of the drift circumference (DOE, 2009bb,ef) to ensure the integrity of equipment operating envelopes specified in the subsurface facility design (e.g., SAR Figure 1.3.4-18).

The applicant will inspect the entire length of the openings annually for the first few years and progressively less frequently if the applicant determines that the inspection frequency could be reduced (DOE, 2009bb). The applicant stated that maintenance of the openings will be performed only as a contingency measure in cases of significant failure or deterioration (DOE, 2009bb). The applicant may inspect areas of failed ground support more frequently to determine when to initiate repair or maintenance. The applicant also stated that maintenance (i) will be scheduled to preclude impacts to repository nuclear safety functions, (ii) may be performed using remotely operated equipment, and (iii) will be preceded by planning and design of remediation activities and controls to assure personnel safety when personnel access to the openings is necessary (DOE, 2009bb).

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.7.2.3 to review the applicant'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 the applicant established and the monitoring and maintenance programs will be adequate to assure satisfactory structural performance of the openings through the preclosure period.

(b)(5)

(b)(5)

Comment (b)(5)

(b)(5)

(b)(5) The applicant 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 (b)(5)

(b)(5)

(b)(5)

(b)(5)

For underground openings on the exhaust-air side of the emplacement drifts, the NRC staff requested the applicant 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. The applicant (DOE, 2009ea) stated that its analysis shows that blockage of an exhaust airway due to rockfall is unlikely and ~~that~~, if such blockage were to occur, the configuration of the subsurface facility will ensure sufficient airflow to maintain the design basis ventilation flow rates. The applicant 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 the applicant ~~to~~ clarify its plans to monitor convergence of the openings through measurements at preselected locations and clarify how the plan will ~~be~~ adequately ~~to~~ preserve operating envelopes and detect potential instability timely. The applicant (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. In response to an NRC staff request that the applicant clarify its approach for using monitoring data to plan for timely maintenance, the applicant (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.

(b)(5)

Commitment

The applicant commits to 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, (iii) sufficient airflow for waste package ventilation, and (iv) timely and safe repair of invert, rails, and power/communications equipment inside the emplacement drifts (DOE, 2009bb,ea,ef,gk).

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.2.3.7.4 Invert Structure and Rails

To demonstrate compliance with 10 CFR 63.21(c)((2), 63.21(c)(3)(i), and 63.112(a), the applicant 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 the applicant (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).

The applicant 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.

The applicant 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). The applicant 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), the applicant 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 The applicant 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 the applicant (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 the applicant proposed for the design of the invert structure and rails, using the guidance in the YMRP. (b)(5)

(b)(5)

Design Codes and Standards

The applicant 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 the applicant proposed for the design of the invert structure and rails, using the guidance in the YMRP. (b)(5)

(b)(5)

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). The applicant indicated (SAR Section 1.3.4.5.6) that site-specific acceleration response spectra were developed at the repository horizon in three orthogonal directions. The seismic loads for the invert structure 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 the applicant proposed for the design of the invert structure and rails, using the guidance in YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Design and Monitoring of the Invert Structure and Rails

The applicant 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, the applicant 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, the applicant indicated, in its response to an NRC staff RAI (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, the applicant 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 the applicant's information regarding the design and monitoring of the invert structure and rails using the guidance in YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5) In response to an NRC staff request that the applicant clarify its plan to ensure adequate functionality of the invert structure and rails through the preclosure period, the applicant (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. The applicant 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 the applicant will develop and implement a remediation method for each case

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.2.4 Evaluation Findings

The NRC staff has reviewed and evaluated the applicant's SAR and other information submitted in support of the license application, relevant to description of structures, systems, components, equipment, and operational process activities for the GROA. (b)(5)

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Comment (b)(5)

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(b)(5)

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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 the applicant'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 verify that the information the applicant presented is sufficient to identify 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 the applicant's response to the NRC staff requests for additional information (RAIs) (DOE, 2008ah; 2009ap,bg,dn,dx,dy,ed,ej,ey,fa,fe,fh,fi,fm,fn). This information included how the applicant 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 Regulatory Requirements and Regulatory Guidance

The regulatory requirements for identification of hazards and initiating events as they pertain to preclosure period are defined in 10 CFR 63.112(b) and 63.112(d). Specifically, the SAR must include a preclosure safety analysis (PCSA) of the GROA. The PCSA is defined in 10 CFR 63.2 to be a systematic examination of the site; the design; and the potential hazards, initiating events, and event sequences and their consequences. In addition, 10 CFR 63.2 defines an event sequence to include one or more initiating events and associated combinations of repository system and component failures. The Category 1 event sequences are those expected to occur one or more times before permanent closure of the GROA. Other event sequences having at least one chance in 10,000 of occurring before permanent closure are Category 2 event sequences.

The PCSA must include an identification and systematic analysis of naturally occurring and human-induced hazards at the GROA, in addition to a comprehensive identification of potential event sequences. The information must also include the technical basis for either including or excluding specific hazards in the safety analysis. For the purposes of this analysis, the applicant should assume that operations at the GROA will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application as required in 10 CFR 63.21(c)(5).

The NRC staff reviewed the applicant'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 acceptance criteria used in the NRC staff's review 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 Staff Review and Analysis

The NRC staff's review of the applicant's identification of hazards and initiating events is integrated with its review of site-related information in Safety Evaluation Report (SER) Section 2.1.1.1 and facility-related information in SER 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 the applicant's assessment of hazards and initiating events at the GROA facilities, as stated in the statement of consideration for the final rule of 10 CFR Part 63 (NRC, 2001aa). The NRC staff selected several hazards and initiating events based on 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 are selected for review, which 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 similar facilities 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

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant identified external initiating events in SAR Section 1.6.3.2. This information included a list of potential naturally occurring and human-induced external hazards compiled from various sources and 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 the applicant'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, the applicant identified potential naturally occurring and human-induced external hazards at the repository that could initiate event sequences (SAR Table 1.6-8). The applicant 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 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 the applicant's identification of potential naturally occurring and human-induced external hazards using the guidance in YMRP Section 2.1.1.3 to determine whether the specific hazards provided in SAR Table 1.6-8 constitute a comprehensive list of external initiating events for screening. (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.1.2 Screening Criteria

The applicant discussed the criteria used to screen external hazards in SAR Table 1.6-1. The applicant 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 over the 100-year preclosure period to initiate an event sequence. The screening criteria used were

(i) Can the external event occur at the repository?

(ii) Can the external event occur at the repository with a frequency greater than 10^{-6}/year, (i.e., have a 1 in 10,000 chance of occurring in the 100-year preclosure period)?

(iii) 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}/year?

(iv) 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}/year?

Additionally, the applicant 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 demonstrated that sufficient time would be available to eliminate the source of the threat or provide an adequate 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 YMRP Section 2.1.1.3 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.1.3 Screening Implementation

As described next, the NRC staff reviewed the applicant's screening process to assess whether the applicant screened each of the 89 potential hazards identified in SAR Table 1.6-8 appropriately. This includes review of the applicant's estimation of the annual frequency of occurrences of the hazards. The NRC staff's review of the applicant-identified potential hazards is organized into five areas on the basis of technical subject matter. SER Table 3-1 groups these hazards for the NRC staff's review.

Table 3-1. Grouped External Hazards Used in the NRC Staff's 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

Table 3-1. Grouped External Hazards Used in the NRC Staff's Review (continued)	
Review Area	**Specific Hazards**
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 applicant 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 SER Section 2.1.1.3.3.1.3.5.

The NRC staff's review of external hazards focused on the technical basis for excluding potential hazards. Using the guidance in YMRP Section 2.1.1.3, the NRC staff examined whether the technical bases for excluding external hazards and initiating events are defensible and consistent with site information. To do this, the NRC staff determined whether (i) the applicant used site-specific data appropriately to identify external hazards and (ii) the applicant's determination of the frequency or probability of occurrence of external hazards is acceptable.

2.1.1.3.3.1.3.1 Geological/Geotechnical Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant 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 geological/geotechnical hazards. The seismic hazards include (i) lateral spread, (ii) liquefaction, (iii) earthquake, (iv) surface fault displacement, and (v) subsurface fault displacement. The geological/geotechnical hazards are: (i) avalanche, (ii) coastal erosion, (iii) denudation, (iv) dissolution, (v) drift degradation, (vi) epeirogenic diastrophism, (vii) erosion,

(viii) fracturing–fractures, (ix) glacial erosion, (x) glaciation, (xi) landslide, (xii) mass wasting, (xiii) orogenic diastrophism, (xiv) rockburst, (xv) rock deformation, (xvi) sedimentation, (xvii) settlement, (xviii) soil shrink–swell consolidation, (xix) static fracturing, (xx) stream erosion, (xxi) subsidence, (xxii) tectonic activity–uplift and depression, (xxiii) undetected geologic features, (xxiv) undetected geologic processes, (xxv) lahar, (xxvi) volcanic activity, (xxvii) volcanism–intrusive igneous activity, (xxviii) volcanism–extrusive igneous activity, and (xxix) volcanism–ash fall.

The NRC staff assesses these 29 hazards according to the following groups: (i) seismic related, (ii) volcano-related, (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 SER Section 2.1.1.3.3.1.3.5.

Seismic Hazards

The applicant 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 causes lateral spread. Liquefaction of soil below the important to safety (ITS) structures would cause stability problems and may induce damage to the structures.

The applicant described the basis for screening of seismic hazards in BSC Section 6.1 (2008ai). The applicant screened in all seismic hazards except liquefaction and lateral spread (BSC, 2007bq). The applicant (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 the applicant's seismic hazard screening information using the guidance in YMRP Section 2.1.1.5. (b)(5)
(b)(5)

The NRC staff's evaluation of the applicant's earthquake information, including the probabilistic seismic hazard analysis, is provided in SER Sections 2.1.1.1.3.5 and 2.1.1.1.3.5.2. (b)(5)
(b)(5)
(b)(5) In addition, the NRC staff evaluated the seismically initiated event sequences at the repository facilities in SER Section 2.1.1.4.3.4.2. (b)(5)
(b)(5)
(b)(5) The NRC staff evaluated the applicant's information on seismically induced rockfall in SER Section 2.1.1.2.3.7.3. (b)(5)
(b)(5)

Because the applicant screened in all hazards except liquefaction and lateral spread, the focus of staff 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) (b)(5)

(b)(5)

Volcano-Related Hazards

The applicant presented five volcano-related hazards in SAR Section 1.6.3.4.3 and BSC Section 6.3 (2008ai). The applicant 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. The applicant provided information on five volcano-related hazards—namely, 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. The applicant estimated the mean frequency of intersection of the subsurface repository footprint by a volcanic event to be 1.7×10^{-8}/year and the mean frequency for the development of an eruptive conduit to the surface to be 4.7×10^{-9}/year. 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. The applicant estimated a mean annual frequency of 6.4×10^{-8} for a 10 g/cm^2 [20 psf] ash fall based on 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, the applicant concluded that the ash depth for a 10 g/cm^2 [20 psf] live load would be well below the 41-cm [16-in] distance to the bottom of the aging overpack vents. Furthermore, the applicant 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, the applicant excluded ash fall from further consideration.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information about volcano-related hazards using the guidance in YMRP Section 2.1.1.3 to determine whether

the applicant adequately estimated the frequency or probability of occurrence of volcanic hazard to include or exclude volcanic-related hazards as initiators of event sequences.

(b)(5)

(b)(5)

Slow Geologic Processes

The applicant 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, requiring a long-term solution. The applicant 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 remedial actions.

The applicant identified 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. The applicant 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, the applicant excluded diastrophism and tectonic activity from further consideration as external hazards (SAR Section 1.6.3.4.2; BSC, 2008ai).

The applicant 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. The applicant 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).

The applicant 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, the applicant excluded both denudation and erosion for further evaluation as external hazards (SAR Section 1.6.3.4.2; BSC, 2008ai). As there are no coastlines near the repository site, the applicant also excluded coastal erosion as a potential hazard. Currently, there are no intermittent or continuous flowing streams at the site. Consequently, the applicant excluded stream erosion as a potential hazard in the preclosure period.

The applicant stated that the current climatic conditions at the repository site would not allow glacier formation. Therefore, the applicant concluded that glacial erosion and glaciation would not be potential initiators of initiating events at the repository during the preclosure period (SAR Section 1.6.3.4.2; BSC, 2008ai).

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on exclusion of tectonic activity using the guidance in YMRP Section 2.1.1.3 to examine whether the applicant's technical basis for excluding tectonic activity, sedimentation, erosion, and glaciation as potential initiating events is defensible and consistent with site information. To do this, the NRC staff determined whether the applicant adequately demonstrated that these hazards are not credible for the repository facilities, because they would progress slowly over time, and allow remedial actions.

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(b)(5)

(b)(5)

(b)(5)

Hill Slope Processes

The applicant 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. The applicant 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. The applicant 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. The applicant screened out avalanches as not having potential to initiate an event sequence during the preclosure period, as described in the DOE Section 1.6.3.4.2 (2008ab). The applicant also excluded mass wasting and, therefore, landslide as a potential initiator of an event sequence because of the absence of suitable topography and geology (BSC, 2008ai).

NRC Staff Evaluation: The NRC staff reviewed the hill-process information using the guidance in YMRP Section 2.1.1.3 to determine whether the applicant used appropriate site-specific information and analysis to screen out avalanche, landslide, and mass wasting hazards from initiating event sequences.

(b)(5)

(b)(5)

(b)(5) The applicant 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. (b)(5)

(b)(5)

(b)(5)

The applicant excluded a mass wasting event on Exile Hill eastern slopes in SAR Section 1.1.4.1.2.2 on the basis of 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 in addition to an absence of suitable topography and geology. Design details of the diversion channels to mitigate mass wasting or landslide events in addition to probable maximum flood are reviewed in SER Section 2.1.1.7.3.1.3.

Processes Affecting Soil Stability

The applicant provided information about the potential impact of processes affecting soil stability in SAR Section 1.6.3.4.2 and in BSC Section 6.2 (2008ai). The applicant provided additional information in its response to NRC staff RAIs (DOE, 2009bg,ej,ey) and in BSC (2007ba). As listed in SAR Table 1.6-8, these hazards include settlement, soil shrink–swell consolidation, static fracturing, and subsidence. These hazards can potentially affect the surface facilities by compromising stability and integrity of the surface soil materials. The applicant screened out all these processes as not having potential to initiate an event sequence during the preclosure period, largely because these hazards would progress slowly over time, allowing necessary remedial actions to be taken to prevent event sequence development.

The applicant 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). The applicant (DOE, 2009ey) stated that any clay-rich soil at the repository site would not be exposed to sufficient wetting and drying due to the arid climate; therefore, any potential hazards associated with soil consolidation from shrink–swell can be eliminated. Additionally, the applicant stated that repository site subsidence would be localized (BSC, 2008ai). The applicant excluded subsidence on the basis of the overall slow progress that would allow necessary 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 YMRP Section 2.1.1.3 to determine whether the technical bases for excluding external hazards and initiating events are defensible.

The NRC staff reviewed the applicant's assessment of potential hazards from settlement of surface facility structures in SER Section 2.1.1.1.3.5.4. (b)(5)

(b)(5)

(b)(5)

Subsurface Drift Degradation Processes

The applicant provided information about drift degradation hazards in SAR Section 1.6.3.4.2 and in BSC Section 6.2 (2008ai). The applicant provided additional information in its response to an NRC staff RAI (DOE, 2009ey).

The applicant 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. The applicant assessed the potential degradation of the emplacement drifts during preclosure period and concluded that drifts will be stable without ground support (BSC, 2007ai). The applicant further stated that drifts could have spalling of the rock wall; however, such spalling will be mitigated by including a perforated stainless steel liner in the ground support system (DOE, 2009ed). The applicant concluded that these hazards would not cause any adverse effects on the GROA facilities during the preclosure period (BSC, 2008ai).

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on potential drift degradation using the guidance in YMRP Section 2.1.1.3 to examine whether the applicant's technical bases for excluding external hazards and initiating events are defensible and consistent with site information. (b)(5)

(b)(5)

(b)(5) The NRC staff's review of underground opening stability during the preclosure period is given in SER Sections 2.1.1.2.3.2 and 2.1.1.2.3.7.3. (b)(5)

(b)(5)

Undetected Geologic Processes and Features

The applicant discussed the impact of undetected geologic processes and features in SAR Section 1.6.3.4.2 and BSC Section 6.2 (2008ai). The applicant 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 potential 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 YMRP Section 2.1.1.3. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.1.3.2 Weather-Related Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant 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 determines how the applicant quantified the weather-related hazards that could affect the repository facilities' operations during the 100-year preclosure period. The NRC staff assessment is provided in three hazard groups: (i) high winds, (ii) tornado and tornado-induced missiles, and (iii) 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 SER Section 2.1.1.3.3.1.3.3.1.

High Winds

The applicant 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 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, the applicant collected wind data from Site 1, approximately 1.0 km [0.6 mi] south of the North Portal. The applicant estimated the maximum 3-second gust straight 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). The applicant 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, the applicant 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, the applicant 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 the applicant used to assess the straight-wind-related hazards using the guidance in YMRP Section 2.1.1.3. The NRC staff's review focus was to determine whether the applicant appropriately (i) characterized the straight wind speed and (ii) used site-specific data information in wind speed characterization. As appropriate, 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 the applicant's response to an NRC staff RAI (DOE, 2009fe). Specifically, the NRC staff evaluated how the applicant 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, (b)(5)

(b)(5)

(b)(5)

Tornado/Tornado-Generated Missiles

The applicant 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.

The applicant 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, the applicant 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.

The applicant 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, the applicant 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, the applicant 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. The applicant 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 to them during preclosure operations.

The applicant estimated the potential damage from tornado wind and tornado-generated missiles given a tornado strike with an annual frequency of occurrence of 10^{-6}. The applicant 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). The applicant considered a structure damaged if the surrogate for that structure sustained damage at a given wind speed. The applicant 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 estimated 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, the applicant 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, the applicant 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, the applicant 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), the applicant used the classification of tornado-generated missiles of Coats and Murray (1985aa) to assess their effects on ITS structures. On the basis of tornado wind speed estimated at an annual frequency of occurrence of 10^{-6}, the applicant 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, the applicant conservatively 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, the applicant indicated that the tornado-generated missiles are expected to be small debris onsite and 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. The applicant 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, the applicant eliminated tornado-generated missiles as initiators of event sequences during the preclosure period.

NRC Staff Evaluation: The NRC staff reviewed the information the applicant 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 YMRP Section 2.1.1.3. In addition, the NRC staff reviewed the applicant's response to an NRC staff RAI (DOE, 2009ey). As appropriate, 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).

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(b)(5) The applicant 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. (b)(5)

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As discussed, the applicant used the Bernoulli's equation to estimate the probability of failure of transportation casks and aging overpacks. (b)(5)

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Lightning

The applicant 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/year [0.16 to 1.04 strikes/mi^2/year]. A National Oceanic and Atmospheric Administration report stated that the strike density for the Yucca Mountain area is 0.2 flashes/km^2/year [0.52 flashes/mi^2/year] (BSC, 2008ai). Assuming the protected area of the GROA is 2.7 km^2 [1.04 mi^2], the applicant 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, the applicant proposed design features for the ITS structures and systems to withstand a lightning strike (SAR Section 1.6.3.4.6; BSC, 2008ai).

The applicant 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 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 much lower risk. The applicant 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. The applicant 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 [eq], the applicant (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 the applicant provided using the guidance in YMRP Section 2.1.1.3. Specifically, the staff evaluated how the applicant screened out initiating events arising from lightning strikes that could affect an ITS SSCs in addition to design features incorporated to withstand a lightning strike. Additionally, as appropriate, 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 was to determine whether the applicant (i) appropriately 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.

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NRC Staff Conclusion

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2.1.1.3.3.1.3.3 Aircraft Crash Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant provided activities involving aircraft in SAR Section 1.6.3.4.1 and BSC (2007ak,ap). The applicant provided additional information in its response to NRC staff RAIs (DOE, 2008ah, 2009fh,fi). The NRC staff reviewed the applicant's aircraft crash hazard analysis including the assumptions, data, and methodology the applicant used to assess the hazards from aircraft crashes that could affect ITS structures at the GROA. An ITS structure can sustain significant damage from crashing aircraft impact including fire damage resulting from the ignition of the aviation fuel carried onboard the aircraft. The applicant 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

The applicant listed potential sources of aircraft-related hazards within 160 km [100 mi] of the North Portal (SAR Section 1.6.3.4.1; BSC, 2007ap). These hazards include flights at nearby civilian, 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. The U.S. Department of Energy (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).

Civilian, DOE-Controlled, and Military Airports, and Helipads

The applicant 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. The applicant screened out any hazards associated with the aircraft landing at and taking off from these airports because these airports are a sufficient distant away from the GROA. Consequently, 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. The applicant indicated that helicopters flying to these helipads may come close to the repository facilities. The applicant 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, the applicant 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 YMRP Section 2.1.1.3, (b)(5)

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Additionally, (DOE, 1996aa) was developed for generic safety analysis of DOE facilities. (b)(5)

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Jet Routes and Federal Airways

The applicant identified the jet routes and federal airways within 160 km [100 mi] of the GROA in BSC Table 6.5 (2007ap). The applicant 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 YMRP Section 2.1.1.3. (b)(5)

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(b)(5) There are several airports nearby, as listed in BSC Table 7–1 (2007ap) (b)(5)

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Beatty Corridor

The applicant 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. The applicant 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 YMRP Section 2.1.1.3. (b)(5)

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Military Operations Areas

The applicant indentified 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). The applicant 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 YMRP Section 2.1.1.3. (b)(5)

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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). The applicant (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. The applicant 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 YMRP Section 2.1.1.3. (b)(5)

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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. The applicant 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. The applicant has screened out flight-related activities in these military routes as 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 YMRP Section 2.1.1.3. (b)(5)

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Ordnance, Dropped Objects, and Ground-To-Ground Missile Testing at Nevada Test Site

The applicant 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). The applicant 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, the applicant 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, the applicant 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 YMRP Section 2.1.1.3. (b)(

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Radar and Communication Jamming Activities

The applicant 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. The applicant 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, the applicant 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 YMRP Section 2.1.1.3. (b)(5)

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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 YMRP Section 2.1.1.3. (b)(5)

(b)(5) Although the Yucca Mountain ridgeline provides a visual separation, as outlined in BSC Appendix F (2007ap), the applicant 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

The applicant 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), the applicant assumed a flight-restricted airspace surrounding the North Portal with a radius of 9 km [4.9 nautical mi or 5.6 statute mi] extending from the ground surface to 4,267 m [14,000 ft] above mean sea level. 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).

The applicant currently allows military aircrafts to transit R-4808N but requires that pilots observe certain avoidance areas. On the basis of interactions with the applicant, 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 future implementation date before receiving and possessing of nuclear waste. The applicant will 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 the applicant provided on the flight-restricted airspace using the guidance in YMRP Section 2.1.1.3. (b)(5)

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Effective Area

Using the equations from DOE–STD–3014–96 (DOE, 1996aa), the applicant 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, it would be considered that the crashing aircraft has impacted the structure, regarded 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 the applicant provided on the effective area using the guidance in YMRP Section 2.1.1.3. (b)(5)

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Aircraft Crash Rates

To estimate the annual frequency of aircraft crashes onto the repository facilities, the applicant 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). The applicant updated the crash rate of air taxi aircraft using information from the National Transportation Safety Board. The applicant 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, the applicant updated these crash rates with recent mishap information

from 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, the applicant relied on data from U.S. Department of the Air Force (2007aa) from 1990 through 2006. The applicant 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 substantially 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.

The applicant 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), the applicant 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/year-km^2 [7.3×10^{-5} crashes/year-mi^2], which the applicant rounded up to 2.9×10^{-5} crashes/year-km^2 [7.5×10^{-5} crashes/year-mi^2] in the actual calculations.

NRC Staff Evaluation: The NRC staff reviewed the information the applicant provided on aircraft crash rates using the guidance in YMRP Section 2.1.1.3 (b)(5)

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Crashes from Flights Through the Beatty Corridor

The applicant 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 to license NRC-regulated facilities. The result was 2.9×10^{-8} crashes/year onto the repository facilities from aircraft transiting this corridor.

NRC Staff Evaluation: The NRC staff reviewed the information the applicant provided on crashes from flights through the Beatty Corridor using the guidance in YMRP Section 2.1.1.3.

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Crashes from Military Flights Over Flight-Restricted Airspace

In its response to NRC staff RAIs (DOE, 2008ah, 2009fi), the applicant 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/year onto the repository facilities from these flights over the flight-restricted airspace.

NRC Staff Evaluation: The NRC staff reviewed the information the applicant provided on crashes of military flights over flight-restricted airspace using the guidance in YMRP Section 2.1.1.3. (b)(5)

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Crashes from Military Flights Outside Flight-Restricted Airspace

The applicant 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, the applicant 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. The estimated crash frequency was 9.4×10^{-7} crashes/year (DOE, 2009fi).

NRC Staff Evaluation: The NRC staff reviewed the information the applicant provided on crashes of military flights outside flight-restricted airspace using the guidance in YMRP Section 2.1.1.3 (b)(5)

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Total Annual Crash Frequency

Combining the crash frequencies from flights through the Beatty Corridor and military flights over and outside flight-restricted airspace, the applicant estimated the annual frequency of aircraft crashing onto the repository facilities to be 1.78×10^{-6} crashes/year. This combined crash frequency is slightly smaller than the screening criterion of 2×10^{-6}/year BSC used (2007ak). This screening criterion of 2×10^{-6}/year is based on a 50-year operational period for the surface facilities, as stated in SAR Section 1.6.3.4.1 and BSC (2007ak). The applicant 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 the applicant provided on the total annual crash frequency using the guidance in YMRP Section 2.1.1.3. (b)(5)

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Commitment

As stated in DOE (2008ah), the applicant commits to develop controls that will be in place to restrict maneuvering and other activities, as assumed in the analysis to estimate the annual aircraft crash frequency onto the repository facilities while transiting the flight-restricted air space. The NRC staff will review these controls before receipt and possession of radioactive waste starts.

NRC Staff Conclusion

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2.1.1.3.3.1.3.4 Industrial and Military Activity-Related Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant 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), the applicant 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 (author, year), 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. The land use information of the NTS 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). The applicant used this information and the analysis as the bases for evaluating activities at these nearby facilities that could pose a potential hazard to the repository during the preclosure period and initiate an event sequence.

Although the applicant 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 sea shore, 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

The applicant 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] from the repository that would store 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, the applicant (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, the applicant (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), the applicant estimated that 5,900 ktonnes [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. The

applicant (BSC, 2008an) stated that 92 ktonnes [2×10^8 lb] would most likely exceed any TNT inventories of the NTS facilities. Therefore, the applicant 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 through 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. The applicant will construct a new rail line connecting the repository operations area with the commercial line. The applicant (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 the applicant's induced air overpressure information using the guidance in YMRP Section 2.1.1.3. 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.

(b)(5)

(b)(5) The applicant 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]. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Induced Seismic Motion

The applicant 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). The applicant 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). The applicant 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. The applicant 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, 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, 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 that of a nuclear blast.

NRC Staff Evaluation: The NRC staff reviewed the induced seismic motion information using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Release of Radiological Materials and Toxic Chemicals

The applicant 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). The applicant 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). The applicant 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), the applicant concluded, using an analysis conducted by Lawrence Livermore National Laboratory, that the worst consequences to the environment from these experiments would be minor local contamination from radioactive materials and, therefore, there is no adverse consequence to the repository at 32 km [20 mi] away.

The applicant 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, the applicant 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. This facility, classified as a Hazard Category 2 nuclear facility (potential for onsite consequences), 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. The activities at this complex are not expected to release any radioactive materials (BSC, 2008an) and, consequently, will not be a hazard to the repository operations.

The Nonproliferation Test and Evaluation Complex in Area 5 of the NTS, approximately 40 km [25 mi] from the repository, tests large- and small-scale release of hazardous and toxic materials and biological simulants in a controlled environment. Most tests are conducted when the wind is blowing away from the repository site (BSC, 2008an). On the basis of the distance from the repository, the applicant (BSC, 2008an) concluded that there would not be any impact.

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 YMRP Section 2.1.1.3. The NRC staff also consulted some of the environmental impact statements or environmental assessments that BSC (2008an) referred to in its 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. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Waste Management Programs

The applicant 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. The applicant could 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 YMRP Section 2.1.1.3. (b)(5)

(b)(5)

Mining

The applicant provided information on hazards from mining-related activities near the repository facilities in SAR Sections 1.6.3.4.8 and 1.1.1.3 and BSC (2008an). 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 is no sand or gravel quarrying operations within a 8-km [5-mi] radius of the repository, and any activities that may cause significant impact will not be permitted. Therefore, the applicant concluded that the nearby mining operations would not have any impact to the repository and its operation, because of 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, the applicant'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 YMRP Section 2.1.1.3. Specifically, the NRC staff evaluated the locations of the nearby mines and description of their activities to assess the potential hazards.

The applicant obtained the locations of the nearby active mines from Driesner and Coyner (2006aa). (b)(5)

(b)(5) 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. (b)(5)

(b)(5)

Commercial Rocket Launch and Retrieval

The applicant 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 recover 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, the applicant 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 YMRP Section 2.1.1.3. The NRC staff evaluated whether the hazard currently exists and what would be done if these activities start in 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). (b)(5)

(b)(5)

Release of Onsite Hazardous Materials

The applicant assessed hazards to the repository facilities associated with potential onsite release of hazardous materials in SAR Section 1.6.3.4.9 and BSC Section 6.11 (2008an). The applicant 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). The applicant also stated that if any operation room needed to be abandoned because of inhabitable conditions from a release of chemicals, remote monitoring equipment installed at the repository facilities would continue to monitor the safety-related functions. Consequently, the applicant concluded that an accidental release of hazardous materials would not affect the safety-related functions of the repository due to paucity of onsite hazardous chemical sources.

NRC Staff Evaluation: The NRC staff reviewed the information on potential release of onsite hazardous materials using the guidance in YMRP Section 2.1.1.3 (b)(5)

(b)(5)

Turbine-Generated Missiles

The applicant assessed hazards to the repository facilities associated with potential turbine-generated missiles in SAR Table 1.6-8 and BSC (2008an). SAR Table 1.6-8 identified turbine-generated missiles as a potential hazard for the repository facilities. The applicant (BSC, 2008ai) stated that the hazard from turbine missiles is generally associated with large turbines in nuclear power plants. As there are no nuclear power plants near the repository facilities, the applicant 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 YMRP Section 2.1.1.3 (b)(5)

(b)(5)

Fog

The applicant 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 YMRP Section 2.1.1.3. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

(b)(5)

(b)(5)

2.1.1.3.3.1.3.5 Other Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant provided information for the remaining 14 hazards (SAR Table 1.6-8). These hazards were further classified into the following groups: (i) external flooding; (ii) loss of power; (iii) loss of cooling capability; (iv) external fire; (v) explosions; (vi) extraterrestrial activity; (vii) waste and rock interaction, geochemical alterations, and dissolution; (viii) perturbation of groundwater system; (ix) improper design and operation; (x) undetected past human intrusions; and (xi) security-related hazards (namely, sabotage, terrorist attack, and war).

External Flooding

The applicant provided information on external flooding at the repository facilities in SAR Section 1.6.3.4.5, SAR Table 1.6–3, and BSC Section 6.5 (2008ai). The applicant identified 15 events that may cause external floods at the repository site: dam failure, external flooding, extreme weather and climate fluctuations, high lake level, high tide, high river stage, hurricane, ice cover, rainstorm, river diversion, seiche, snow, storm surge, tsunami, and waves. Because no rivers or streams flow past the site, there are no upstream dams. Therefore, 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, external flooding due to high lake level or dam failure cannot occur at the GROA.

Only external flooding resulting from rainstorms is further evaluated 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]. Potential flooding due to melted snow and ice is determined to be less severe and less frequent than from rainstorms. Thus, the applicant evaluated rainstorm as a bounding case for potential flooding resulting from storm precipitation.

The applicant'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, the applicant will implement a standard maintenance practice on the flood protection features. The applicant estimated the probable maximum flood (PMF) frequency as 1.1×10^{-9}/year (BSC, 2008ai). In the final disposition, the applicant screened out external floods from further consideration because the PMF frequency is less than 10^{-6}/year and the building roof drainage system is designed to accommodate rainfall criteria.

NRC Staff Evaluation: The NRC staff evaluated information and analysis the applicant provided on external flood hazards using the guidance in YMRP Section 2.1.1.3. The NRC staff also reviewed the document describing the flood hazard curve (BSC, 2008cd).

(b)(5)

(b)(5) Because intense precipitation can occur at Yucca Mountain, external flooding at the GROA due to rainstorms is evaluated using a probabilistic flood hazard curve to estimate annual flood frequencies. In the process of developing the flood hazard curve, a probable maximum precipitation is estimated based on annual precipitation values the National Oceanic and Atmospheric Administration reported for the semiarid Southwest. In addition, a PMF for the GROA is estimated based on flood peak simulations using the HEC–1 hydrologic model the U.S. Army Corps of Engineers, Hydrologic Engineering Center developed. (b)(5)

(b)(5)

(b)(5)

Loss of Power

The applicant provided information on loss of power to the repository facilities in SAR Section 1.6.3.4, SAR Table 1.6-8, and BSC Section 6.7 (2008ai). The applicant 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.

The applicant (BSC, 2008ac,as,be,bk,bq) estimated the frequency of a loss of electrical power event occurring at the Yucca Mountain facilities to be 3.6×10^{-2}/year 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}/year. Assuming that the waste handling operations would continue for the first 50 years of the preclosure period, the applicant 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). The applicant 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 YMRP Section 2.1.1.3. Specifically, the NRC staff evaluated the annual frequency of occurrence of a loss of offsite power event at the repository to determine whether the applicant used appropriate analysis and information to develop this annual frequency.

(b)(5)

Loss of Cooling Capability

The applicant provided information on loss of cooling capability to the repository waste handling facilities in SAR Section 1.6.3.4.7 and BSC 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, the applicant concluded that dam failure, ice cover, low lake level, low river level, and river diversion would not result in loss of cooling capability at the repository facility (BSC, 2008ai). In addition, 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. The applicant 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, 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, the applicant (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 YMRP Section 2.1.1.3. (b)(5)

(b)(5)

External Fire

The applicant 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), the applicant summarized an analysis to establish the required firebreak width to keep the ITS SSCs safe from potential wildfires.

The applicant indicated that the U.S. Forest Service collected information on wildfires from 1970 through 2000 based on the 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/year in the GROA with a surface area of 2.7 km^2 [1.0 mi^2] (BSC, 2008ai). The applicant [SAR Section 1.6.3.4.10; BSC 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). The applicant 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. The applicant 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 radiative 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], the applicant 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. The applicant further concluded that the buffer zone width would be sufficient 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 findings, the applicant screened out external fire as a potential initiator of event sequences in the repository facilities because the ITS SSCs would be surrounded by an 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 the applicant presented using the guidance in YMRP Section 2.1.1.3. Specifically, the NRC staff evaluated the description of potential vegetation characteristics near the repository site and the analysis the applicant provided to screen out initiating events arising from fires originating outside the GROA boundary (vegetation or wildfire).

(b)(5)

(b)(5)

(b)(5)

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. (b)(5)

(b)(5)

Explosions

The applicant provided information on explosion hazards to the repository facilities in SAR Section 1.7.1.2.2 and BSC Section 6.0.5 (2008au). The applicant 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. The applicant 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], the applicant (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 be developed from an accidental explosion of the tanker truck, the applicant (BSC, 2008au) concluded that the transportation cask would not suffer any adverse effects. Therefore, the applicant 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 (SPM) citing a design requirement of explosion-proof fuel tanks. The applicant 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 the applicant's assessment of potential explosion hazards using the guidance in YMRP Section 2.1.1.3. Specifically, the NRC staff evaluated the description of explosion sources and the analysis the applicant 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.

(b)(5)

Extraterrestrial Activity

The applicant provided information on extraterrestrial activity at the repository site in SAR Section 1.6.3.4.11 and SAR Table 1.6-8. The applicant also discussed the potential impact on the repository facilities by the extraterrestrial objects in BSC 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, the applicant indicated that the frequency of asteroid impacts is relatively small. The return periods for smaller asteroids are hundreds to thousands of years (BSC, 2008ai) 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. The applicant (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, based on 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. The applicant therefore considered the

iron and hard stone meteorites within the size range of 10 to 1,000 kg [22 to 2,204 lb] for potential impact to the repository.

Using information on the number of meteorites impacting the Earth's surface from Bland and Artemleva Table 2 (2006aa), the applicant (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}/year. Hard stone meteorites will impact the GROA at a frequency varying between 6.4×10^{-7} and 1.2×10^{-9}/year 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, the applicant (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. The applicant (BSC, 2008ai) estimated that about one object reenters the atmosphere every day and one to two objects with a 1-m^2 [11-ft^2] radar cross section reenter the atmosphere each week. The applicant assumed, for conservatism, 4 objects with radar cross sections exceeding 1 m^2 [11 ft^2] per week or up to 210 objects per year 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 considerations (BSC, 2008ai).

NRC Staff Evaluation: The NRC staff reviewed the information and analysis on extraterrestrial activity presented in the SAR Section 1.6.3.4.11 and supporting documents to screen out initiating events arising from extraterrestrial object impact with a safety-related structure using the guidance in YMRP Section 2.1.1.3.

(b)(5)

(b)(5)

(b)(5)

Waste and Rock Interaction, Thermal Loading, Geochemical Alterations, and Dissolution

The applicant provided information on waste and rock interaction, geochemical alterations, thermal load, and dissolution at the repository site in SAR Table 1.6-8, BSC Section 4.4 (2008ai), and DOE (2009ey). The applicant 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, the applicant stated in SAR Section 1.3.5 that forced ventilation during the preclosure period would moderate any temperature rise. The applicant identified in SAR Section 1.1.8.4 that geochemical alteration and dissolution are slow-acting geological processes, and heat from the waste package and forced ventilation during the preclosure period would further limit geochemical alteration and dissolution in the rock by drying out the near-field rock mass and moderating the temperature rise.

NRC Staff Evaluation: The NRC staff reviewed the information and technical basis the applicant provided on waste and rock interaction, thermal loading, geochemical alterations, and dissolution hazards using the guidance in YMRP Section 2.1.1.3. Specifically, the NRC staff evaluated the information and rationale the applicant 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 the applicant provided on dissolution in the postclosure screening analysis of features, events, and processes in SAR Table 2.2-5 (solubility, speciation, phase changes, precipitation/dissolution) and in SAR Section 2.3.5.3.3.

(b)(5)

Perturbation of Groundwater

The applicant provided information on perturbation of groundwater at the repository site in SAR Table 1.6-8, BSC Section 4.4 (2008ai), and DOE (2009ey). The applicant determined that the hazard associated with perturbation of groundwater or availability of groundwater in the long term would not initiate event sequences, because there would be sufficient time to develop alternate sources for additional water demand at the repository facilities.

NRC Staff Evaluation: The NRC staff reviewed the information and rationale the applicant provided on the hazard associated with perturbation of groundwater using the guidance in YMRP Section 2.1.1.3.

(b)(5)

Improper Design and Operation

The applicant provided information on improper design and operation of the GROA facilities in SAR Table 1.6-8 and BSC Section 4.4 (2008ai). The applicant identified improper design and operation as one of the hazards unique to the Yucca Mountain repository. However, the applicant 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 the applicant provided on hazards associated with improper design and operation of the GROA facilities using the guidance in YMRP Section 2.1.1.3. (b)(5) (b)(5) The NRC staff separately reviewed the applicant's assessment of operational hazards in the repository facilities from improper design and operation in SER Section 2.1.1.3.3.2. (b)(5) (b)(5)

Undetected Past Human Intrusions

The applicant provided information on undetected past human intrusions at the repository site in SAR Table 1.6–8 and BSC Section 4.4 (2008ai). The applicant described undetected human intrusions as potential hazards associated with undiscovered boreholes or mine shafts. Any undetected boreholes or mine shafts that are directly connected to the subsurface facilities may act as direct conduits for radionuclide release in addition to preferential flow paths for air and water. The applicant 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. The applicant considered undetected human intrusions (open site investigation boreholes or open mine shafts) in a screening of relevant features, events, and processes for the postclosure period in SAR Table 2.2-1 and SNL Table G-1 (2008ab).

NRC Staff Evaluation: The NRC staff reviewed the information and technical basis the

applicant provided on hazards associated with undetected past human intrusions using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

Security-Related Hazards

The applicant provided information on security-related hazards at the repository site in SAR Table 1.6-8 and BSC Section 4.4 (2008ai). The applicant identified sabotage, terrorist attack, and war as security-related events and screened out this hazard as not having the potential to initiate an event sequence in the repository facilities during the preclosure period. Therefore, the applicant did not evaluate these events in its PCSA.

NRC Staff Evaluation: The NRC staff reviewed the information on security-related hazards and the applicant's classification of these events using the guidance in YMRP Section 2.1.1.3.

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2 Operational (Internal) Hazards and Initiating Events

The applicant identified internal hazards and initiating events at the GROA in SAR Section 1.6.3. 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 the applicant'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 the applicant's quantification and screening into drop and collision hazards, direct exposure hazards, fire hazards, internal flood hazards, criticality hazards, and subsurface hazards (SER Section 2.1.1.3.3.2.3.4).

2.1.1.3.3.2.1 Identification of Internal Initiating Events

To demonstrate compliance with 10 CFR 63.112(b), the applicant performed a systematic safety analysis to identify initiating events involving equipment failures and human actions for the GROA facilities. The applicant described its approach for identifying initiating events in SAR Section 1.6.3 and tabulated the internal initiating events developed from this analysis in SAR Table 1.6-3. The applicant tabulated initiating events separately for the WHF, CRCF, RF, IHF, intrasite and balance of plant, and subsurface facilities. For this identification of internal initiating events, the applicant used NUREG/CR–2300 (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa).

The applicant developed internal initiating events from equipment failures and human actions involving processes or operations. To identify initiating events, the applicant performed a master logic diagram (MLD) analysis and a hazard and operability (HAZOP) evaluation. The applicant used the HAZOP evaluation to supplement the MLD analysis and identify any additional initiating events not identified using the MLD analysis.

The applicant developed process flow diagrams from facility block flow diagrams to delineate the facility processes and operations and to guide its MLD development and HAZOP evaluation. The process flow diagrams depict the flow and relationships among major operations and are broken down into nodes (i.e., groups of sequential steps). The applicant evaluated activities within a node in terms of their potential to initiate an event.

The applicant identified several initiating events in SAR Table 1.6-3, many of which are common among the surface facilities. For example, the applicant identified canister transfer machine (CTM) and cask handling crane drops and collisions for the surface facilities. In addition, the applicant identified fire-related initiating events across all of the surface facilities, intrasite and balance of plant, and the subsurface facilities.

The applicant also identified initiating events that capture equipment and operations specific to a facility [e.g., spent fuel transfer machine (SFTM) drops in the WHF where the SFTM is specific to the WHF; bare fuel handling in a pool is also specific to the WHF]. In its response to an NRC staff RAI (DOE, 2009dy), the applicant stated that fuel assembly collisions during bare fuel handling were not prevented and they might occur, but their consequences are bounded by fuel assembly drops. For the intrasite and balance of plant, the applicant identified initiating events involving LLW as well as impacts to an aging overpack, transportation cask, or shielded transfer cask.

For the subsurface, the applicant identified several initiating events involving the TEV to include derailment or collision and prolonged immobility. In response to an NRC staff RAI (DOE, 2009ey), the applicant clarified how identically described initiating events from SAR Table 1.6-3 can be traced using their sequential numbering and their order of appearance in the facility source document table.

The applicant incorporated operations into its identification of initiating events. The applicant described operational processes associated with various systems and operations in terms of the nodes on an associated process flow diagram. For example, for the CRCF, the mechanical handling system was described in SAR Section 1.2.4.2 and the operational processes for the cask handling subsystems in SAR Section 1.2.4.2.1.2. The operational summary for the CRCF was described in the applicant's facility event sequence development analysis [e.g., BSC Attachment B (2008ab)] using the nodes from the associated process flow diagram [e.g., BSC Figure 15 (2008ab)]. The applicant showed how it included operations in its identification of initiating events in BSC Figure 3 (2008ab). This figure showed how the applicant reviewed

operations and processes, developed a simplified process flow diagram, and established nodes prior to performing the MLD and HAZOP analyses to identify initiating events.

NRC Staff Evaluation: The NRC staff reviewed the applicant's methodology and its implementation for identifying internal initiating events using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

After reviewing the applicant's response to an NRC staff RAI (DOE, 2009ey), (b)(5)

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(b)(5)

Proposed Condition of Construction Authorization

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.2 Screening and Grouping of Internal Initiating Events

To demonstrate compliance with 10 CFR 63.111(c) as part of its PCSA, the applicant described its methodology for grouping and screening initiating events in SAR Section 1.6.1.2. Additionally, the applicant described its approach to grouping initiating events in SAR Section 1.7.1.1 and screening initiating events in SAR Section 1.7.1.2.1. The applicant organized internal initiating events into groups and depicted this grouping in ESDs. The ESDs contain both initiating events and responses to the initiating events (i.e., pivotal events). Grouping allows event sequence analysis to be more efficient because several initiating events can be included in the same ESD. Groups in the same ESD indicate that the facility, SSCs involved, and human response are qualitatively the same for each group. The applicant represented different groups (or types) of initiating events as small bubbles that point to one larger bubble (i.e., aggregated initiating event). In several cases, the applicant included more than one initiating event within a small bubble; these would be associated with both the same pivotal events and the same conditional probabilities for those pivotal events.

The applicant screened initiating events from further analysis after grouping them and placing them in ESDs. The applicant identified three methods for screening initiating events:

(i) They could be screened out using a non-probabilistic engineering analysis to determine that the event could not occur.

(ii) They could be identified as bounded by another initiating event.

(iii) They could be screened out based upon a probabilistic evaluation.

For the third case involving a probabilistic evaluation, the applicant screened out initiating events if they fell below the threshold for Category 2 event sequences.

NRC Staff Evaluation: The NRC staff reviewed the applicant's methodology for grouping initiating events using the guidance in YMRP Section 2.1.1.3 (b)(5)

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(b)(5)

(b)(5)

The NRC staff's evaluation of the applicant's implementation of this approach is discussed in SER Section 2.1.1.3.3.2.4 and its associated subsections. For example, SER Section 2.1.1.3.3.2.4.1 evaluates the screening of initiating events involving drops and collisions and SER Section 2.1.1.3.3.2.4.3 evaluates the screening of internal flooding.

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3 Determination of Probability for Grouped Initiating Events

The applicant estimated the probability of occurrence for initiating events using direct probability assignments and fault tree analysis (BSC, 2008ac,as,au,be,bk,bq). The analysis for each facility was detailed in the files listed in Attachment H of these documents.

(b)(5)

Proposed Condition of Construction Authorization

(b)(5)

2.1.1.3.3.2.3.1 Initiating Events Quantification

To demonstrate compliance with 10 CFR 63.111(c), the applicant presented its methodology for quantifying initiating events as part of event sequence quantification and categorization in SAR Section 1.7.1, as part of its PCSA. The applicant summarized the quantification results (i.e., mean, median, and standard deviation) for event sequences initiated by internal events in SAR Tables 1.7-7, 1.7-9, 1.7-11, 1.7-13, 1.7-15, and 1.7-17 for the IHF, RF, CRCF, WHF, intrasite operations and balance of plant, and subsurface facilities, respectively. In some of these event sequences, the applicant indicated that there were no pivotal events; therefore, the

value in the table represented the number of occurrences over the preclosure period that was obtained from the probability for the initiating event combined with the results from a throughput analysis (e.g., number of canisters processed).

The applicant quantified initiating events using direct probability assignments and fault tree analysis. The applicant followed the methodology in NUREG–0492 (NRC, 1981ab) for developing fault trees. As part of fault tree quantification, the applicant developed the failure probability of individual components or individual pieces of equipment and included equipment and component failures as basic events. The applicant also developed human failure events (HFE) and included these failures as basic events in fault trees.

The applicant separated its initiating events by facility and listed them in its supporting documents. For example, the applicant listed them in BSC Table 10 (2008bo) for the WHF. This table cross-referenced initiating events to MLDs, HAZOP analyses, and ESDs. Excluding fire and internal flooding, which are addressed in SER Sections 2.1.1.3.3.2.3.4.3 and 2.1.1.3.3.2.3.4.4, respectively, the applicant identified 109 initiating events for the CRCF in BSC Table 10 (2008ab). With the exception of initiating event CRC–1806 (i.e., a collision between the site transporter and another moving vehicle), the applicant mapped this list of initiating events onto 20 ESDs included in BSC Attachment F (2008ab). In addition, excluding fire and internal flooding, the applicant identified 129 initiating events for the WHF in BSC Table 10 (2008bo). With the exception of initiating event WHF-1701 [i.e., exposure due to an overpressurization condition in the dual purpose canister (DPC)], the applicant mapped this list of initiating events onto 31 ESDs included in BSC Attachment F (2008bo). Additionally for the WHF, the applicant described initiating event WHF-1106 in BSC Table 10 (2008bo) in terms of a canister collision due to CTM malfunction or misoperation leading to an impact. However, the small bubble in its corresponding ESD [i.e., BSC Figure F-13 (2008bo)] refers to a canister drop inside the CTM shield bell. In addition, BSC Table 10 (2008bo) described initiating event WHF-503 in terms of an unplanned conveyance movement and the ESD [i.e., BSC Figure F-5 (2008bo)] includes this initiating event in a small bubble that refers to a cask drop. However, BSC Section A4.5.1 (2008bq) associated unplanned conveyance movement with a side impact. For the IHF and RF, excluding internal flooding and excessive temperature events, the applicant listed 72 and 98 initiating events, respectively, in BSC Table 10 (2008ao,bd). The applicant mapped all of these initiating events to ESDs. In addition, for the intrasite operations and subsurface, the applicant mapped all of its initiating events to ESDs.

After grouping initiating events and placing them in ESDs, the applicant quantified these grouped initiating events (i.e., small bubbles in the ESDs) in its SAPHIRE models. For example, the applicant included these models for the WHF in BSC Attachment H (2008bq). In response to an NRC staff RAI (DOE, 2009fk) to clarify its initiating event grouping, the applicant corrected the ESD in BSC Figure F-16 (2008bo) where some initiating events were grouped incorrectly. Additionally, the applicant stated in its response (DOE, 2009fk) that it grouped initiating event WHF-1604 with a sample line break but could have instead grouped it with overpressurization of a transportation cask. In another response to an NRC staff RAI (DOE, 2009dn), the applicant clarified how it accounted for individual initiating events for a group in its fault tree quantification.

NRC Staff Evaluation: The NRC staff reviewed the applicant's methodology for quantifying initiating events and its implementation of that methodology using the guidance in YMRP Section 2.1.1.3. (b)(5)

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(b)(5)

In terms of the applicant's implementation of its methodology for quantifying initiating events, the NRC staff reviewed the applicant's list of initiating events [e.g., BSC Table 10 (2008bo)], their inclusion on MLDs [e.g., BSC Attachment D (2008bo)], and its grouping of initiating events [e.g., BSC Attachment F (2008bo)]. In addition, the NRC staff reviewed the SAPHIRE models [BSC Attachment H (2008bq)].

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(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3.2 Equipment Reliability Quantification

To demonstrate compliance with 10 CFR 63.111(c), the applicant estimated the probability of failure for various components and then used these estimates in its fault tree analysis to develop initiating event probabilities as part of its PCSA. The applicant provided its approach in SAR Section 1.7.2.2 and in its supporting documents (BSC, 2008ac,as,au,be,bk,bq). For example, the applicant described its approach for the CRCF in BSC (2008ac).

SAR Section 1.7.2.2 described how the applicant developed active system or component reliabilities and defined an active system or component as one that changes position and, by doing so, modifies the system behavior. The applicant 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, the applicant referenced Droguett, et al. (2004aa) and Siu and Kelly (1998aa). Additionally, the applicant used the alpha factor method described in NUREG/CR–5485 (Mosleh, et al., 1998aa) to quantify common-cause failure.

In some cases, when more than one data source was available, the applicant quantified a component's failure probability by selecting one failure distribution. The applicant used this approach to quantify, for example, the interlock failure on demand. The applicant indicated that it used the data source yielding the most diffuse information, which would produce the largest uncertainty, and the median of the five data sources as representative of the mean failure rate (DOE, 2009dy). The applicant selected one of the five distributions having a peak value that coincided with the combination distribution peak for interlock failure on demand.

For cases with only one data source (e.g., air handling unit failure to run and pressure sensor failure on demand), the applicant used a single data source and then updated the value using a Jeffreys noninformative prior distribution in accordance with NUREG/CR–6823 (Atwood, et al., 2003aa).

NRC Staff Evaluation: The NRC staff conducted its review using the guidance in YMRP Section 2.1.1.3 to examine the applicant's approach and evaluate selected component failure probability distributions for the CRCF, intrasite, and the subsurface. Because the applicant used the same approach to develop failure probability distributions for active systems and components for all surface facilities, only component failure distributions for the CRCF were reviewed in detail.

(b)(5)

(b)(5)

The NRC staff reviewed the applicant's implementation of its methodology for estimating reliability when only one data source was available (b)(5)

(b)(5)

(b)(5) For pressure sensor failure on demand, the applicant described its equations in "YMP Active Comp Database.xls" (i.e., Attachment H to BSC, 2008ac) (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3.3 Quantification of Human Failure Events

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d) with respect to HFE at the repository facilities during the preclosure period in the PCSA, the applicant provided information in SAR Section 1.7.2.5 and BSC Attachment E, Sections 4.3.4 and 6.4 (2008ac,as,au,be,bk,bq).

The treatment of operator errors is important to address for preclosure activities because many waste handling activities are accomplished through manual actions by human operators. Interactions between operators and preclosure systems, components, and equipment create opportunities for human-induced failures that could result in hazards or other negative consequences. Consequently, it is important to identify and address such opportunities for human-induced hazards with a systematic analysis approach. If the potential for human-induced hazards is ignored, important contributions to potential preclosure hazards will be omitted, resulting in an incomplete safety analysis.

The applicant outlined a nine-step approach for conducting the human reliability analysis (HRA) that it considered to be consistent with HLWRS–ISG–04 (NRC, 2007adx), ASME RA–Sb–2005 [American Society of Mechanical Engineers Section 4.5.5 (2005ad,)], and NUREG–1624 (NRC, 2000ai). The approach the applicant outlined is an iterative process that begins with a definition of the scope of the analysis, works through an identification of potential HFE, conducts a preliminary analysis for initial quantification of the HFE, and performs a detailed analysis of HFE deemed to be of high significance. The applicant quantified the HFE 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).

BSC Attachment E (2008ac,as,au,be,bk,bq) discussed the applicant's general HRA method selection including selecting four HFE quantification methods to treat operator errors. The applicant's specific selection of HRA quantification methods for specific HFE was described in conjunction with the analysis of these HFE.

The results of the human error analysis were a list of HFEs that were incorporated into the PCSA as basic events in the SAPHIRE model. Further details on the applicant's HRA were given in BSC Section 4.3.4 (2008ac,as,au,be,bk,bq). The applicant discussed the results of the analyses in Section 6.4 of these documents, and the list of resulting HFEs were included in BSC Table 6.4-2 (2008ac,as,be,bq) and BSC Table 6.4-1 (2008au,bk). The analyses are detailed in Appendix E of these documents.

NRC Staff Evaluation: The NRC staff reviewed the information on quantification of HFE the applicant provided using the guidance in YMRP Section 2.1.1.3 and HLWRS-ISG–04 (NRC, 2007ad) to evaluate the applicant's treatment of operator errors to determine whether (i) the methodology the applicant used to assess the potential for operator errors was acceptable and (ii) the applicant acceptably implemented its methodology.

The NRC staff examined the applicant's methodologies documented in the SAR and BSC Attachments A, B, and E (2008ac,as,au,be,bk,bq). (b)(5)

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(b)(5) 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 HRA quantification were discussed for each specific HFE. (b)(5)

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NRC Staff Conclusion

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(b)(5)

2.1.1.3.3.2.3.4 Quantification and Screening of Hazards

In this section, the NRC staff reviewed the applicant's quantification and screening of internal initiating events. The NRC staff also reviewed the applicant's grouping of initiating events and determination of the probability for the grouped initiating events.

2.1.1.3.3.2.3.4.1 Drop/Collision Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant identified initiating events involving drops and collisions and provided its technical bases for including or excluding these initiating events. To demonstrate compliance with 10 CFR 63.111(c) as part of its PCSA, the applicant quantified initiating events involving drops and collisions. The applicant listed the internal initiating events involving drops and collisions in SAR Table 1.6-3. In addition, the applicant included its quantification analyses in its supporting documents (BSC, 2008ac,as,au,be,bk,bq).

The applicant identified drop of a canister from the CTM, drops of heavy objects such as a lid onto a canister, and drops of fuel assemblies from the SFTM (SAR Table 1.6-3). The applicant identified impacts to a canister from movement of the Waste Package Transfer Trolley (WPTT), Cask Transfer Trolley, or site transporter during a loading or unloading operation. The applicant identified impacts to a waste package from the malfunction of the WPTT or Waste Package Transfer Carriage and impacts to a waste package from the drop of a heavy object, such as an empty waste package, from the waste package handling crane. The applicant identified collisions involving the Cask Transfer Trolley within a facility and collisions involving the Site Prime Mover (SPM) while transferring a transportation cask to a facility.

Technical Bases for Screening Out Initiating Events

The applicant provided its technical bases for screening out individual internal initiating events involving drops and collisions in its supporting documents. For example, for the CRCF, the applicant provided its technical bases for screening out internal initiating events in BSC Table 6.0-2 (2008ac). The applicant screened out the drop of a canister inside the shield bell of the CTM with the slide gate closed, indicating that such drops would be subsumed by drops from the operational lift height, which has a higher drop height. In addition, the applicant screened out the drop of a heavy object (such as a lid) onto an HLW canister or onto a DOE spent nuclear fuel (SNF) canister on the basis of having divider plates in the codisposal waste package and the design of transportation casks. The applicant screened out damage to a canister from a side impact of the slide gate during CTM transfer operations. The applicant screened out a TEV collision with the WPTT in the CRCF because (i) the docking station separates the two pieces of equipment, (ii) the TEV is parked when the rail-based WPTT enters the room, and (iii) the vehicle speeds are extremely slow.

NRC Staff Evaluation: The NRC staff reviewed the applicant's technical bases for screening out initiating events using the guidance in YMRP Section 2.1.1.3. (b)(5)

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(b)(5)

(b)(5)

(b)(5) In its response to an NRC staff RAI (DOE, 2009dy) involving the drop of a canister inside the shield bell of the CTM, the applicant stated that the CTM slide gate and supporting structures are designed to withstand a 30-cm [12-in] vertical drop of the heaviest canister. In addition, the applicant stated that the CTM shielding would continue to perform its safety function. (b)(5)

(b)(5)

Use of System Information

The applicant incorporated system information into initiating event quantification by identifying (i) aspects of the operations (e.g., number of lifts per canister) associated with the initiating events, (ii) human interactions (e.g., resetting an interlock), and (iii) specific equipment operating modes (e.g., SPM in-facility versus intrasite operation).

The applicant included system information pertaining to operations as part of its event sequence development analysis. For example, for the CRCF, the applicant included operational descriptions in BSC Section 6.1 and Attachment B (2008ab). In its response to an NRC staff's RAI (DOE, 2009dy), the applicant clarified how the quantification of individual basic events in a fault tree is tied to the overall operation being modeled. For a canister transfer using the CTM, the applicant included information on the number of port slide gate operations, number of demands placed on the holding brake, and number of object lifts associated with a canister transfer. To quantify the initiating event probability involving canister transfers, the applicant included the number of object lifts during transfer directly in the initiating event fault tree. Alternatively, in the case of an object dropped on a canister, the applicant included the number of object lifts per canister. In its responses to the NRC staff's RAI (DOE, 2009dy) pertaining to

an object drop (e.g., 060-CTMOBJLIFTNUMBER), the applicant included the drop of a canister (e.g., the drop of an HLW canister or DOE SNF canister onto an HLW canister), the drop of a grapple and the drop of a transportation cask lid and waste package inner lid.

The applicant included system information for human interactions pertaining to maintenance and operations. For example, the applicant described the failure to reset the CTM slide gate interlock following maintenance when evaluating a shielding compromise during a DOE standardized canister transfer. It did not, however, similarly identify this reset failure when evaluating a shielding compromise for other canisters. The applicant indicated in its response to an NRC staff's RAI (DOE, 2009dy) that a basic event involving a human failure to reset the interlock was inadvertently omitted for canisters other than the DOE standardized canister. For the omission, the applicant showed in DOE Table 1 (2009dy) that the event sequence frequencies increased by approximately a factor of two as a result of this error. The applicant also listed in DOE Table 2 (2009dy) interlock failure basic events along with an indication of whether a failure to restore event was modeled explicitly; the applicant provided this table to illustrate consistency in modeling interlock failures.

The applicant included system information related to specific equipment operating modes. For example, the applicant described SPM operations in terms of intrasite operations and in-facility operations. For intrasite operations, the SPM is driven via a diesel engine; however, inside facilities, the SPM is electrically driven using the facility's main electrical supply and is connected to the supply by an umbilical cable. In addition, during intrasite operations, an operator controls the SPM at a console inside the SPM, whereas inside facilities, the operator controls the SPM using a remote (wireless) control or a pendant attached to the vehicle. The applicant explained its quantification for the basic events in its fault tree for SPM collision while entering the CRCF and explained its inclusion of speed control (governor) failure in the fault tree (DOE, 2009dy). The applicant also clarified that in a facility, the governor is not involved; however, the applicant included the potential for governor malfunction before the SPM is transitioned to electrical power.

NRC Staff Evaluation: The NRC staff reviewed the applicant's use of system information including description of operations and initiating event fault tree models using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

(b)(5) For example, the applicant discussed how it considered both the diesel-driven and electrically driven operating modes for the SPM involving the collision of a SPM with facility structures while entering a facility. The applicant included governor failure to account for a potential malfunction during the SPM transition to electrical power (DOE, 2009dy). (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Use of Data Including Assumptions for Data Sources Used

The applicant provided data sources used in the reliability quantification for both components (e.g., CTM grapple-engaged limit switch) and systems (e.g., SFTM) in Microsoft® Excel® spreadsheets that are part of its supporting documents. For example, for the CRCF, BSC Attachment H (2008ac) contained the applicant's active component database, "YMP Active Comp Database.xls," which included data sources used to develop failure probabilities for active components. This spreadsheet is organized by component type and failure mode. In addition, the applicant provided its criteria for selecting data used in its analyses in the supporting documents. For the CRCF, the applicant described its criteria in BSC Section C1.2 (2008ac). The applicant indicated that when selecting data, it (i) matched the component type and failure mode from a data source to the basic events in the fault tree, (ii) verified that the data source used was not proprietary, and (iii) considered the operating conditions including maintenance and testing.

Within its active component database, the applicant provided descriptions (including references) of the data sources and associated assumptions used to quantify the probability of failure for active components. For example, the applicant developed the failure probability of interlocks by estimating an average duration between operational demands of 8 hours throughout the facilities (DOE, 2009dy). For SPM speed control and brake failure, the applicant used data for a 4,536-kg [5-ton] cargo truck that was converted from miles to hours using a conversion factor of 3.2 km/h [2 mph] (DOE, 2009dy). It also indicated that this conversion factor was based on the design speed limit for most vehicles at the time the database was developed and that the uncertainty for governor failure captures a higher SPM speed of 14 km/h [9 mph]. Additionally, the applicant stated that its quantification of collisions involving the SPM was conservative because vehicles on intrasite roadways would travel at slower speeds and use speed-limited

escort vehicles (DOE, 2009dy).

To estimate the probability of a drop, the applicant used fault tree analysis for the CTM. For other cranes, such as the Waste Package Handling Crane and the SFTM, the applicant developed the probability directly from published sources. The applicant identified the drop events it used from NUREG–1774 (NRC, 2003ai) to develop the failure or drop probabilities for the Waste Package Handling Crane and the SFTM (DOE, 2009dy). Additionally, the applicant indicated that to develop the drop probabilities of the SFTM, it (i) did not consider the age of the fuel when selecting data, (ii) considered events since 1969 for the upper bound on the failure probability, (iii) did not include fuel assembly slips because they do not have the potential for a release, and (iv) did not find post-2001 data that it considered applicable. Although the CTM is an ASME NOG-1 (American Society of Mechanical Engineers, 2005aa), crane, the applicant used data from systems that are not NOG-1 cranes. For example, the applicant used data for an AN/BQQ–5 sonar system for the CTM holding brake failure and data collected from a hotel, motel, casino, or resort facility in Coronado, California, for failure of the CTM load cell pressure sensor.

NRC Staff Evaluation: The NRC staff reviewed the criteria that the applicant used for selecting data using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

The NRC staff reviewed several of the data values that the applicant developed to understand how the applicant applied its selection criteria and to understand the applicant's assumptions when quantifying component failure. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3.4.2 Direct Exposure Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant identified initiating events involving direct exposures and provided its technical bases for including or excluding these initiating events. To demonstrate compliance with 10 CFR 63.111(c), the applicant quantified initiating events involving direct exposures as part of its PCSA. The applicant listed the internal initiating events involving direct exposures in SAR Table 1.6-3. In addition, the applicant included its quantification analyses in its supporting documents (BSC, 2008ac,as,be,bq).

The applicant quantified the direct exposure initiating events involving temporary loss of shielding (e.g., CTM shield skirt raised and slide gate open) during CTM transfer activities using fault tree analysis. The applicant listed the initiator event trees in BSC Table A4.18-2 (2008ac) and included its SAPHIRE models in BSC Attachment H (2008ac). The applicant used two separate fault trees for the various waste forms. The applicant used one fault tree (i.e., ESD18-TMP-SHLD-LOSS-DSTD) for canister transfers involving the DOE standardized canister. The applicant used a different but similar fault tree (i.e., ESD18-TEMP-SHIELD-LOSS) for the remaining waste form containers (i.e., DPC; HLW; multicanister overpacks; and transportation, aging, and disposal canister). For the DOE standardized canister fault tree, the applicant showed additional human failure basic events. One of these basic events involved operator failure during canister staging (i.e., 060-OPSTAGERACK1-HFI-NOD), and another involved human failure to reset the CTM slide gate interlock following maintenance (i.e., 060-OPFAILRSTINT-HFI-NOM).

The applicant updated its fault tree analysis for direct exposure initiating events involving temporary loss of shielding in its supplemental response to an NRC staff RAI (DOE, 2009dx). It added a function to an existing ITS interlock to prevent raising the CTM shield skirt if any port slide gate was open and showed this modification in its fault tree analysis in DOE Attachment 2 (2009dx). In this analysis, the applicant accounted for 100 percent staging of the DOE standardized canisters, HLW canisters, and multicanister overpacks, as shown in DOE Table 3 (2009dx). In addition, it modeled additional basic events to account for operator failure to follow administrative controls, such as an operator potentially entering the canister transfer room in violation of the radiation protection program.

The applicant summarized the event sequence frequencies with the additional interlock function and obtained a combined frequency for DOE standardized canisters, HLW canisters, and multicanister overpacks of 8.4×10^{-3} and a combined frequency for transportation, aging, and disposal canisters and DPCs of 4.8×10^{-3}, as outlined in DOE Table 4 (2009dx). To estimate these frequencies, the applicant used total demands for each of the canisters in the CTM transfer operations, where a demand is defined as an operation having a potential for exposure. The applicant showed the total demands as the product of the number of canisters and the total transfer operations per canister, as described in DOE Table 3 (2009dx). In addition, the applicant stated that it would review analogous direct exposure event sequences in the CRCF, RF, WHF, and IHF and make changes, if necessary, to achieve similar margins to the Category 1 threshold as those in ESD18 (DOE, 2009dx).

NRC Staff Evaluation: The NRC staff reviewed the applicant's inclusion of initiating events involving direct exposures and quantification of these initiating events using the guidance in YMRP Section 2.1.1.3. The NRC staff concentrated on the direct exposure event sequences involving CTM operations because the applicant quantified several of these event sequences as Category 2 [e.g., BSC Table 6.8-3 (2008ac)]. Because the CRCF, IHF, RF, and WHF have CTMs and all facilities have similar transfer operations using the CTM, the NRC staff primarily reviewed the direct exposure initiating events in the CRCF and the applicant's quantification of

these initiating events.

(b)(5)

Commitment

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3.4.3 Fire Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d) with respect to potential fire hazards at the repository facilities during the preclosure period, the applicant 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 SER section. However, potential hazards from wildfires are evaluated in SER Section 2.1.1.3.3.1.3.5.4.

Methodologies Used

The applicant 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. The applicant described the methodology in BSC Attachment F (2008ac,as,au,be,bk,bq). The methodology requires the applicant 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.

The applicant 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 described by Tillander (2004aa). In addition, the applicant 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 YMRP Section 2.1.1.3, the NRC staff reviewed information the applicant 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 appropriateness 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 the applicant's overall approach and subsequently selected specific facilities for detailed review. The fire analysis for the CRCF, as described in BSC (2008ac), was selected for detailed review because (i) the activities at this facility adequately represented the activities at other waste handling facilities and (ii) the methodology used to derive fire-related initiating event frequencies was similar to the methodology used for the other waste handling facilities (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., BSC 2008ac,as,au,be,bk,bq). The NRC staff's review focus was to determine whether the applicant (i) used appropriate methodologies to assess fire hazard, (ii) applied the methodologies appropriately, (iii) used facility-specific data and system information appropriately, and (iv) quantified fire-initiating event frequencies appropriately.

(b)(5)

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. The applicant divided the number of reported fires at industrial and chemical facilities from Ahrens (2000aa) by the estimated number of industrial and chemical facilities in operation, as reported by the U.S. Census Bureau (2000ab). This generated a probability of fire per facility. The applicant considered intrasite operations and subsurface operations separate facilities to determine ignition frequency. The applicant then apportioned the overall frequency of fire to seven different categories (e.g., storage areas, trash areas, vehicle fires) on the basis of data from Ahrens (2000aa). To estimate the probability of fire affecting a waste form, the applicant assumed that three of the seven categories of fires (vehicle fires, fires in receiving areas, and fires in storage areas) could expose waste forms within the intrasite operations area, and only vehicle fires (e.g., fires originating from the TEV) had sufficient potential to expose waste forms in the subsurface facility. The applicant 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 based on waste form residence times or fire propagation probabilities. These resulting ignition frequencies were directly used in the event sequence analysis, as reviewed in SER Section 2.1.1.4.3.1.3.

NRC Staff Evaluation: The NRC staff reviewed information the applicant provided on data sources using the guidance provided in YMRP Section 2.1.1.3. 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 the applicant used to determine whether the applicant used site-specific data and system information appropriately. Ignition frequency was based on a number of data sets (e.g., data from U.S. Census Bureau, NFPA, and Finnish databases). (b)(5)

(b)(5)

Fire propagation probabilities were also determined using data from radioactive material working facilities and nuclear energy plants of noncombustible construction. (b)(5)

(b)(5)

Application of Methodologies

The applicant estimated the number of expected fires annually in each of these facilities on a

per unit floor area (fires/unit area/year) basis using the data derived from industrial buildings having floor areas larger than 1,000 m² [10,764 ft²], as Tillander (2004aa) reported. The applicant also estimated the confidence limits, as provided in BSC Table F.III-2 (2008ac,as,au,be,bk,bq). 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), the applicant distributed the overall facility ignition frequency to each room, on the basis of the number and types of ignition sources that would be present in the room. The applicant relied on ignition source data from Ahrens (2007aa), which described the likelihood that a fire would originate from a particular class of equipment (e.g., welders, motors, internal combustion engines), and a scoring methodology, described in Electric Power Research Institute (2005aa), to determine the ignition frequency on a "per room" basis.

NRC Staff Evaluation: The NRC staff reviewed information the applicant provided on application of methodologies to access the fire potential at the repository facilities using the guidance provided in YMRP Section 2.1.1.3. In addition, the NRC staff also 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.

(b)(5)

The NRC staff reviewed the classification of ignition frequency by ignition sources (e.g., number of fires attributed to welding equipment, electrical equipment, vehicles) (b)(5)

(b)(5)

(b)(5)

The NRC staff reviewed how the applicant distributed the overall building ignition frequency to each room of each facility. (b)(5)

(b)(5)

The NRC staff also reviewed how the applicant assessed the propagation probabilities of fire inside a waste handling facility. (b)(5)

(b)(5)

(b)(5)

(b)(5) The applicant 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). The applicant justified the use of information on chemical, plastic, or petroleum products plants for the waste handling facilities (DOE, 2009f). (b)(5)

(b)(5)

(b)(5)

(b)(5)

For fires in the intrasite and subsurface areas, the applicant 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 a credible initiating event. (b)(5)
(b)(5)

Initiating Event Frequencies

The applicant used site-specific information regarding the quantity and types of equipment in each room to assign room fire ignition probabilities. After assessing the ignition frequency for each room, the applicant used fire growth data from Ahrens (2007aa) to evaluate the likelihood of fire propagating from the room or area of origin to adjacent rooms or areas. The 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. The applicant discounted the performance of passive fire resistance-rated wall construction within the facility. The applicant also assumed no benefit from automatic suppression systems and applied fire propagation data for fires where no sprinkler system was present or the system failed to operate. The applicant summarized the results of these analyses in SAR Section 1.7.1.2.2.

The applicant developed an uncertainty distribution for the ignition frequency, a conditional probability based on the extent of flame damage using data from Ahrens (2007aa), and a categorization of ignition sources by types of equipment using data from Ahrens (2007aa). The selected distribution for ignition frequency is lognormal; however, normal distribution was selected for the other two parameters. In addition, the applicant 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 the applicant provided on quantification of fire event frequencies using the guidance provided in YMRP Section 2.1.1.3. The applicant quantified the initiating event frequencies for waste forms by evaluating the potential location of various waste forms in each building. By using a compilation of ignition and propagation probabilities, the applicant was able to determine which fires had the potential of reaching a waste form and serving as an initiating event. (b)(5)
(b)(5)

The applicant dealt with "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. (b)(5)
(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3.4.4 Internal Flood Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 10 CFR 63.112(d), the applicant identified internal flooding initiating events, and provided its technical bases for screening out internal flooding. For the surface facilities, the applicant identified (i) the potential for internal flooding caused by actuation of the fire protection system and piping or valve failure in SAR Table 1.6-3 and (ii) the potential for subsurface flooding due to a construction-related accident (e.g., water supply piping to the Tunnel Boring Machine) in SAR Section 1.6.3.5. For both the surface and subsurface facilities, the applicant screened out internal flooding as an initiating event. However, the applicant stated in SAR Section 1.7.1.2.3 that moderator entering a breached waste container and contributing to the pivotal event of an event sequence was considered. This pivotal event is addressed in SER Section 2.1.1.4.3.3.2.2 under moderator intrusion control.

The applicant screened out internal flooding in the surface facilities on the basis of design (SAR Section 1.7.1.2.3; DOE, 2009fn). The applicant stated that a waste form container exposed to water will not lose its structural integrity or shielding capability. The applicant 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, the applicant 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. The applicant also referred to discussions in SAR Sections 1.14.2.3.3.1.4, 1.14.2.3.2.1.5, 1.14.2.3.2.3.4, and 1.14.2.3.2.3.5 with regard to the criticality potential for a sealed canister surrounded by water.

As discussed in its response to an NRC staff RAI (DOE, 2009fm) and in BSC Section 6.0.4 (2008bk), the applicant 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, the applicant 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 the technical bases for screening it out using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.3.2.3.4.5 Criticality Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d) with respect to criticality hazards at the repository facilities during the preclosure period, the applicant provided information in SAR Sections 1.6.1.6 and 1.14 and BSC (2008ba). The applicant provided an overview of its criticality safety analysis process in SAR Section 1.6.1.6. To determine the criticality potential hazard for different waste forms and operations, the applicant performed criticality sensitivity analyses to evaluate the impact on system reactivity of variations in those parameters important to criticality. In SAR Sections 1.7 and 1.14, the applicant provided the results from these analyses.

To demonstrate that waste forms will remain subcritical during the preclosure period and thus screen out all criticality-related initiating events during the preclosure period, the applicant identified seven parameters as important to criticality in SAR Table 1.14-2: (i) waste form characteristics, (ii) moderation, (iii) soluble neutron absorbers, (iv) fixed neutron absorbers, (v) geometry, (vi) interaction, and (vii) reflection. For each of these parameters, the applicant determined the impact of their variations on reactivity by performing sensitivity analyses. On the basis of the output of these sensitivity calculations, the applicant deterministically determined whether 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). These deterministic screening evaluations were performed for commercial SNF (CSNF) and a representative fuel from each of the DOE SNF groups to cover the possible operational conditions to which the waste form may be exposed at the GROA.

For CSNF, moderation was determined to be the primary control parameter during dry operations, and the concentration of soluble neutron absorber was determined to be the primary control parameter during WHF pool and fill operations. For DOE SNF, moderation and interaction were determined to be the primary control parameters, as shown in SAR Table 1.14-2. HLW glass canisters remain subcritical due to licensing basis limits on fissile isotope concentration, as discussed in SER Sections 2.1.1.2.3.4.2 and 2.1.1.7.3.4. Naval fuel is considered a group of DOE SNF, as described in SAR Section 1.5.1.

On the basis of these determinations, the applicant identified, developed, quantified, and categorized event sequences involving (i) introduction of moderator into breached or open transportation casks, DPCs, and transportation, aging, and disposal canisters containing CSNF; (ii) multiple DOE SNF canisters in the CRCF in the same general location; and (iii) the presence of sufficient soluble boron in the WHF pool. The applicant's methodology requires it to identify, develop, quantify, and categorize event sequences that impact conditional criticality control parameters only if there are categorized event sequences that result in the control violation of the primary criticality control parameter.

The applicant marked the end states of some event sequences as "important to criticality." Such possible criticality configurations were analyzed in or bounded by analyses in BSC (2008ba) to determine whether reactivity would exceed the upper subcritical limit. This analysis was performed in conjunction with the design process. If an initial criticality calculation resulted in

exceeding the upper subcritical limit, the design was modified or procedural safety controls (PSCs) were employed to prevent such event sequences. As a result, no "important to criticality" categorized event sequences were presented in the SAR Section 1.7. The applicant focused on event sequences that result from canister breach and the subsequent introduction of moderator.

To identify the initiating events from GROA facilities, the applicant used the MLD method supplemented by an HAZOP evaluation (e.g., BSC, 2008bo). Criticality was accounted for in the criticality calculations used to identify hazards and initiating events. 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. The applicant indicated that no Category 1 or Category 2 event sequences that required crediting fixed neutron absorbers were identified (DOE, 2009ey). Therefore, material selection errors during manufacturing of such absorbers need not be considered for preclosure criticality safety. The applicant further pointed out 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, 2009dy). The applicant presented its analysis of the external initiating events in BSC (2008ai).

The applicant screened out the interaction between more than two naval canisters in the IHF from consideration in the PCSA 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 (DOE, 2009ey).

The applicant provided the technical basis for screening the neutronic interaction of more than four DOE SNF casks/canisters by referencing criticality calculations (BSC, 2008cm) that were used as input to the screening evaluation (DOE, 2009ey). 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.

To ensure sufficient concentrations of enriched boron in the pool, the applicant developed PSC–9, as outlined in SAR Table 1.9-10. The applicant indicated that this control provided the initial conditions in the WHF pool to ensure that a critical configuration could not be created in the pool. SAR Section 1.14.2.3.3.3 stated that boron dilution was screened out as an initiating event and a PSC was relied upon to ensure the needed concentration. The applicant assigned the PSC with a failure probability of 1×10^{-6}; this was based primarily on the required boron concentration fraction needed to maintain subcriticality being less than 53 percent of the boron normally available in the pool. The applicant indicated that even if enough unborated water was added to fill the pool to the brim to increase the concentration fraction to 91 percent (DOE, 2009dy), it is still not enough dilution to allow a criticality event. The applicant stated that the only sources of unborated water large enough to fill the pool to overflowing were the fire suppression system and the potable water system, as described in DOE 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.

In SAR Section 6.4.4, the applicant screened out internal flooding as an initiating event on the basis of the conclusion that internal flooding could not initiate an event sequence that results in increased levels of radiological exposure. The applicant explained that the deionized water system was the only water source considered in the boron dilution initiating event screening 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 (DOE, 2009dy).

The applicant did not consider the possibility of boron dilution in the HAZOP or MLD evaluations of the DPC fill water, because the boron dilution initiating event was screened out. 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 (DOE, 2009dy). Therefore, a moderator could only be introduced into a DPC during filling while the DPC siphon and vent ports were open and before it was filled with borated water (SAR Section 1.2.5.2.3.2).

The applicant indicated that the DOE SNF interaction outside the CRCF was not a concern, because the analysis (BSC, 2008ba) showed that the transportation casks and waste packages would provide adequate separation to prevent criticality. In addition, naval SNF interaction outside the IHF is not a concern, because the transportation casks and waste packages would provide adequate separation to prevent criticality.

The applicant assessed criticality-initiating events during subsurface operations using the same methodology for subsurface operations as for other facilities of the GROA. It credited the canister barrier with preventing introduction of moderator to prevent criticality. The waste packages and the shape of the emplacement drifts control criticality and prevent a criticality arrangement of DOE SNF canisters from occurring (BSC, 2008ba).

In SAR Section 1.6.3.5, the applicant considered that construction hazards are bounded by internal initiating events. The applicant credited the distance between construction operations and canisters with reducing the effects of impacts to less than those considered for internal initiating events. Additionally, the construction would be scheduled such that it would not interfere with operations.

NRC Staff Evaluation: The NRC staff conducted its review by evaluating the information the applicant 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 YMRP Section 2.1.1.3. In addition, the NRC staff reviewed supporting documents (BSC, 2008ai,bj,bo) and the applicant's response to the NRC staff RAIs (DOE, 2009dy,ey).

The NRC staff evaluated the rationale the applicant used to assess the criticality-related hazards originated outside the GROA. (b)(5)

(b)(5)

(b)(5)

The NRC staff examined the rationale the applicant used to assess criticality-related initiating events at the IHF and the interaction among more than two naval canisters in the IHF, which was screened out from further consideration in the PCSA by crediting the design of the mechanical handling capabilities of the IHF. (b)(5)

(b)(5)

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. (b)(5)

(b)(5)

The NRC staff examined the rationale the applicant used to screen out the boron-dilution-related criticality initiating events at the WHF by applying a PSC to ensure the needed boron concentration. (b)(5)

(b)(5)

(b)(5)

The NRC staff examined the rationale the applicant used to assess criticality-initiating events during intrasite operations. (b)(5)

(b)(5)

The NRC staff examined the rationale the applicant used to assess criticality-initiating events at the GROA during construction. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.3.3.2.3.4.6 Subsurface Hazards

To demonstrate compliance with 10 CFR 63.112(b) and 63.112(d), the applicant 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). The applicant 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). The applicant 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 was provided in BSC (2008bk) and the applicant's response to the NRC staff RAIs (DOE, 2009dy,ey). The applicant used this assessment of initiating events as the basis for evaluating whether any of these initiating events would develop into an event sequence. The applicant further detailed the results of the MLD and HAZOP processes in BSC Attachments D and E (2008bj). The applicant 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. The applicant performed a screening analysis on the aggregated ESDs, as described in SAR Section 1.7.1.2 and in BSC Section 6.0 (BSC, 2008bk). To quantify the probability of occurrence of the event sequence, the applicant used fault tree analyses as described in BSC Sections 4.3.2 and 6.2.2 and Attachment B (2008bk). The applicant 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 appropriately considering site-specific and facility information.

Use of System Information and Methodology

For the initiating event "TEV Impact During Transit," the applicant 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, the applicant developed the fault tree model for the initiating event. The applicant identified three potential failure modes: (i) another vehicle being driven into the TEV on the surface, (ii) uncontrolled descent of the TEV down the North Ramp resulting in an impact with the tunnel wall, and (iii) TEV impact with another objects along the rail line due to either spurious signal from the drive controllers or failure of the manual control switch (BSC, 2008bk). The fault tree comprised three subfault trees, one for each failure mode.

The applicant modeled the initiating event "TEV Stops for an Extended Period of Time" in a fault tree, SHIELD–STOP, as described in BSC 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, the applicant used the alpha-factor method to analyze common-cause failure of the speed sensor of the motors.

NRC Staff Evaluation: The NRC staff reviewed the applicant's assessment of initiating events of "TEV Impact During Transit" and "TEV Stops for an Extended Period of Time" using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

(b)(5)

Data Use

In conducting the fault tree analysis for the initiating event, labeled as TRANSIT–IMPACT (BSC, 2008bk), the applicant used the reliability data for component failure from NPRD–95 (Denson, et al., 1994aa). For the initiating event "TEV Stops for an Extended Period of Time," detailed in BSC Section B1.4.5 (2008bk), the applicant 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). The applicant 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 the applicant used in the fault tree analysis for the initiating event, labeled as TRANSIT–IMPACT, using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

Estimation of Annual Frequency

The applicant 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, the applicant proposed to install special crossing barricades and signals at all surface intersections. The applicant will restrict all traffic from the area of a loaded TEV in the subsurface facilities. Additionally, the TEV travels slowly {roughly 3.2 km/hour [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 the applicant used to determine the annual frequency of occurrence of the initiating event "TEV Impact During Transit" using the guidance in YMRP Section 2.1.1.3. The applicant concluded 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. (b)(5)

(b)(5)

Technical Basis for Screening

The applicant 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). The applicant also considered the uncertainties associated with the parameters. Following the standard approach, the applicant (BSC, 2008bk) assumed a lognormal distribution representing the uncertainties of the parameters. The estimated mean and standard deviation of the annual frequency of the initiating event were 2.94×10^{-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); however, it should be for this case.

The applicant 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 the applicant established a requirement that the shielding be designed to sustain the thermal loading for all waste package loadings for 30 days without significant degradation of the shielding function (DOE, 2009ey). Additionally, the applicant stated that at the limiting waste package power output for emplacement (as per SAR Section 1.3.1.2.5 and Table 5.10-3, 18 kW per waste package for CSNF or 11.8 kW per waste package for naval 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 TEV. The applicant 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), the applicant plans to 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 the applicant used to

screen the initiating events "TEV Impact During Transit" and "TEV Stops for an Extended Period of Time" using the guidance in YMRP Section 2.1.1.3. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.3.4 Evaluation Findings

The NRC staff reviewed the applicant's SAR and other supporting information relevant to internal and external hazards and initiating events at the GROA facilities. The NRC staff reviewed the applicant's identification, screening, and quantification of external initiating events. In addition, NRC staff reviewed the applicant's identification, screening, and grouping of internal initiating events and the applicant's quantification of these grouped initiating events. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

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CHAPTER 4

2.1.1.4 Identification of Event Sequences

2.1.1.4.1 Introduction

This chapter provides the U.S. Nuclear Regulatory Commission (NRC) staff review of information on identification of event sequences to determine compliance with the regulatory requirements for preclosure safety. The objective of the review is to determine whether the applicant, in accordance with the preclosure safety analysis (PCSA), provided sufficient information and technical basis for developing, quantifying, and categorizing event sequences needed to demonstrate compliance with preclosure performance objectives. The NRC staff evaluated the information in the Safety Analysis Report (SAR) Section 1.7 (DOE, 2008ab); supplemental documents referenced in the SAR; and information the applicant provided in response to the NRC staff requests for additional information (RAIs) (DOE, 2009bl,dq,dx,dz, ed,ej,fg,fk,fl,fr,ft–z,ga–gi).

The evaluation presented in this chapter considers information reviewed in other Safety Evaluation Report (SER) chapters: site description in SER Section 2.1.1.1; the description of structures, systems, and components (SSCs), operational process, and throughput analysis in SER Section 2.1.1.2; identification of hazards and initiating events in SER Section 2.1.1.3; and design of SSCs ITS in SER Section 2.1.1.7. The output from this chapter includes event sequences and their associated categorizations that will be used in SER Section 2.1.1.5 to assess compliance with preclosure performance objectives.

2.1.1.4.2 Regulatory Requirements and Guidance

The regulatory requirements for identifying and categorizing event sequences as they pertain to preclosure period are defined 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 stated in the application. 10 CFR 63.111(c) requires that the applicant demonstrate compliance with preclosure performance objectives for Category 1 and Category 2 event sequences in accordance with 10 CFR 63.111(a) and (b). Specifically, 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 the applicant's identification of event sequences and the categorization using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.4 (NRC, 2003aa). The acceptance criteria used in the NRC staff's review 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 Staff Review and Analysis

The applicant stated that, to demonstrate compliance with 10 CFR 63.21(c)(5), 63.111(c), and 63.112(b), it used the regulatory acceptance criteria in YMRP Section 2.1.1.4. The NRC staff's review focused on evaluating technical bases and justification for methods selected, assumptions made, and site-specific data the applicant used to identify event sequences. The NRC staff verified whether event sequence development is based on adequate 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 on 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 is defensible and the categorization of event sequences is appropriate.

The applicant's identification of event sequences stems from the identification of naturally occurring and human-induced external and internal hazards (SAR Figure 1.7-1). The applicant developed a list of internal and external events in SAR Section 1.6. The NRC staff evaluates, in SER Section 2.1.1.3, the applicant's screening of hazards and initiating events and the associated frequency of occurrences for event sequence analyses. As discussed in SER 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 appropriateness of screening and frequency of occurrence of initiating events reviewed in SER Section 2.1.1.3.

On the basis of the initiating events identified for event sequence analysis, the review in 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 follows the YMRP guidelines and 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 the applicant used to develop and categorize event sequences in the PCSA is evaluated in this section. In accordance with 10 CFR 63.112(b) and 63.111(c), the applicant must (i) identify potential event sequences through identification and systematic analysis of naturally occurring and human-induced hazards and (ii) categorize these event sequences to demonstrate compliance with preclosure performance objectives. The applicant 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 YMRP Section 2.1.1.4, HLWRS–ISG–01 (NRC, 2006ad), and HLWRS–ISG–02 (NRC, 2007ab), the NRC staff evaluated the applicant's methods for event sequence identification. The NRC staff reviewed the applicant'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 acceptable, (ii) methods for developing event sequences are acceptable, (iii) methods for modeling event sequences are acceptable, (iv) the model appropriately represents events sequences, and (v) the methodology for categorization for event sequences is acceptable. The review is structured into the four following subsections: methodology for internal events (SER Section 2.1.1.4.3.1.1), seismically initiated events (SER Section 2.1.1.4.3.1.2), fire-initiated events within the GROA (SER Section 2.1.1.4.3.1.3), and event sequence categorization methodology (SER Section 2.1.1.4.3.1.4).

2.1.1.4.3.1.1 Internal Events

In SAR Section 1.7.1, the applicant described the methodology for development of event sequence for internal events initiated by random failure of equipment and human errors during preclosure operations to demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b). 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 the applicant's approach to considering preclosure-operations-related initiating events and their progression leading to potential consequences or end states. The applicant 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 or human errors providing containment, shielding, confinement, and criticality control functions as pivotal events. 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. The applicant 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. The applicant 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 WHF, event sequences were analyzed for transfer of spent nuclear fuel (SNF) assemblies. Thus, all internal event sequences quantified were associated with 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 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, the applicant 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 provided in YMRP Section 2.1.1.4.2. The NRC staff evaluated the applicant's overall approach to develop event sequences. (b)(5)

(b)(5)

(b)(5)

The NRC staff evaluated the applicant's methods for modeling and quantifying internal event sequences. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.1.2 Seismic Events

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant 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). The applicant screened in seismicity as a credible natural hazard for PCSA. The applicant 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. The applicant 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. The applicant 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 exceedence associated with horizontal peak ground acceleration (PGA) for the surface facilities. The applicant summarized, in SAR Section 1.7.2.4, the methodology used to develop mean fragility curves, 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. The applicant 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, the applicant evaluated the mean annual probability of failure of SSCs, which is the initiating event frequency.

The applicant used event tree and fault tree techniques for quantitative analysis of the event sequences. Seismically initiated event sequences were developed for each type of waste form in each of the four types of handling facilities, and intrasite and subsurface facilities. The event sequences, in the applicant'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. The applicant did not credit confinement of the heating, ventilation, and air conditioning (HVAC) system in the seismic event sequence analysis. The applicant 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 the applicant's methodology for seismically induced event sequences using the guidance in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.1.3 Fire Events

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant discussed fire initiated event sequences in SAR Section 1.7.1.2.2. The applicant 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).

The applicant 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. The applicant identified areas in each facility where fires could play a role in either directly exposing a waste form or affecting an SSC. The applicant developed initiating event probabilities for local fires that 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 SER 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 SER 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 SER 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.

The applicant 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 incorporated the appropriate initiating event and pivotal event probabilities for that particular waste form and facility.

Because the applicant's fire-related event sequence development methodology was based on a per unit probability, the applicant used throughput data in SAR Table 1.7-5, BSC (2008ac,as,au, be,bk,bq), and DOE (2009ga) to convert event frequencies into a total number of occurrences related to that waste form. The applicant 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 provided in YMRP Section 2.1.1.4.2 (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.1.4 Event Sequence Categorization Methodology

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant described its overall approach to event sequence categorization in SAR Section 1.7.5. For internal, seismic, and fire events, the applicant quantified each event sequence in an ESD by calculating an expected number of occurrences. The applicant 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 the applicant's methodology for categorizing event sequences using the guidance provided in YMRP Section 2.1.1.4.2 (b)(5)
(b)(5)
(b)(5) For Category 2 event sequences, the applicant 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. (b)(5)
(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.2 Event Sequences Development

In accordance with 10 CFR 63.21(c)(5) and 10 CFR 63.112(b), the applicant is required to develop, quantify, and categorize event sequences initiated by naturally occurring and human-induced hazards to show compliance with preclosure performance objectives. This section documents NRC staff's evaluation of the applicant's technical basis for event sequence development and modeling. The scope of this section includes a review of the appropriateness 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: SER 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 determine whether event sequences were developed appropriately and modeled consistent with the methodology reviewed in SER Section 2.1.1.4.3.1, the NRC staff's review 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

The applicant discussed the event sequences of internal events for the surface and subsurface facilities in SAR Section 1.7.1 and BSC (2008ab,ac,ao,as,at,bd,be,bj,bk,bo,bq). The internal event sequences are initiated by random failure of equipment or human error during waste handling operations. The NRC staff reviewed the information discussed in these documents and reviewed the SAPHIRE models to evaluate the applicant's event sequence development.

The 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 SER Section 2.1.1.4.3.2.1.1. The review of wet handling operations is discussed in SER Section 2.1.1.4.3.2.1.2 and handling of waste packages during subsurface operations is evaluated in SER Section 2.1.1.4.3.2.1.3.

2.1.1.4.3.2.1.1 Canister and Cask Handling Operations at Surface Facilities

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant 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 Table G–2(BSC 2008ab,ao,at,bd,bo) summarized the relationship between the ESDs and the initiator event trees and the system response trees.

At the CRCF, IHF, RF, and WHF, the applicant 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. The applicant 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, the applicant 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. The applicant, 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 (BSC 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. The applicant 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 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 control for important to criticality. 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, the containment function was credited to the transportation cask, as indicated in the system response tree RESPONSE–TCASK–CSNF. The initiating events were modeled as fault trees and linked 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, 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 provided in YMRP Section 2.1.1.4.2 and HLWRS–ISG–02 (NRC, 2007ab). (b)(5)

(b)(5)

sequence probability.

(b)(5)

(b)(5) The applicant indicated that the SAR did not support quantification of event sequences involving multicanister overpacks (DOE, 2009b). (b)(5)
(b)(5)

The NRC staff evaluated the transfer of branches from the initiator event trees to response trees.
(b)(5)

The NRC staff evaluated whether end states are consistent with the success or failure of the safety functions of the SSCs. (b)(5)
(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.2.1.2 Wet Handling Operations

To demonstrate compliance with 10 CFR 63.112(b) and 63.111(b), the applicant 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, the applicant 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.

The applicant 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, the applicant 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. The applicant stated that these initiating events were intended to allow consideration of drops onto the racks and other drops not onto the racks (DOE, 2009gc). The applicant 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; drops from the SFTM were included 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, the applicant credited the cask for maintaining containment if an event such as a drop or tipover occurs, and the applicant 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), the applicant specified Procedural Safety Control–6 requiring a minimum number of installed fasteners on the casks and the applicant 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, the applicant 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), the applicant 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, the applicant identified no heavy load drops onto a cask in the pool. (b)(5) (b)(5) In addition, the applicant described a heavy load drop from the jib crane in its response to an RAI (DOE, 2009gc) for WHF–ESD19–DPC. (b)(5) (b)(5)

(b)(5)

For event sequences involving direct exposure during pool operations, the applicant 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). The applicant 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), the applicant 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), the applicant used different response trees depending on whether the event (e.g., drop) occurs over the pool. For a drop over the pool, the applicant considered an unfiltered radionuclide release of gases including those important to criticality. For a drop over the floor, the applicant considered direct exposure and filtered and unfiltered releases including those important to criticality. Filtered and unfiltered releases pertain to the confinement pivotal event, which relates to the success or failure of the surface nuclear confinement HVAC system. The applicant 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. The applicant included the direct exposure from the shielding loss end state in BSC Figure G–9 (2008bo) for an STC being transferred from the pool. The applicant 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 the applicant's WHF event sequence development using the guidance in YMRP Section 2.1.1.4. (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.2.1.3 Subsurface Operations

To demonstrate compliance with 10 CFR 63.21(c)(5), 63.111(b), and 63.112(b), the applicant 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 the applicant identified that could occur during the operations.

The applicant 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. Event sequences considered by the applicant that 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. The applicant 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. The applicant 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.

The applicant 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. The applicant 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 the applicant plans to use monitoring and inspection programs to assess the need for maintenance to assure adequate functionality of the subsurface structures and systems. The NRC staff requested the applicant to clarify its approach to preventing or mitigating potential event sequences related to subsurface structures or systems failure, such as 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 an NRC staff RAI (DOE, 2009ed), the applicant stated that it established appropriate design criteria and bases 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 in a timely manner.

NRC Staff Evaluation: The NRC staff reviewed the event sequence development for subsurface operations using the guidance provided in YMRP Section 2.1.1.4.2 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.2.2 Seismic Events

To demonstrate compliance with 10 CFR 63.112(b) and 63.111 (b), the applicant provided information on the development of seismically induced event sequences for the GROA in BSC (2008bg). The applicant developed seismically initiated event sequences for the CRCF, IHF,

RF, WHF, and intrasite and subsurface operations. The NRC staff reviewed the information to determine whether the applicant 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.

Collapse of Surface Structures

The applicant 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, the applicant 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 YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

Waste Handling in Waste Handling Buildings

The applicant 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). The applicant 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, the applicant developed a fault tree model. A typical fault tree in the applicant's analysis consisted of 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 the applicant's analysis accounted for the amount of time the waste container is exposed to the seismic hazard. The applicant's calculation of exposure time was based on waste processing activity with an equipment and 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. The applicant considered HVAC to be always failed if the seismic event caused breach to the waste canister (BSC, 2008bg), therefore taking no credit for the "Confinement" pivotal event. The applicant 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, the applicant considered events resulting from failure of the WHF pool, collapse of the SNF staging rack, and failure of the HVAC integrity (e.g., contaminated ducts, filters in 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, the applicant 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 YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5) 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).

(b)(5)

(b)(5)

(b)(5)

(b)(5) 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. The applicant relied on the passive reliability of the waste form canisters for containment and transportation and aging casks for shielding. (b)(5)
(b)(5)

(b)(5) 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 the applicant relied on the availability of the pool for shielding, scrubbing of radioactive release, and boration. (b)(5)
(b)(5)

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, the applicant 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), the applicant 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), the applicant'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. The applicant discussed this assumption on the basis of an analysis provided in DOE (2009ej), and reviewed by the NRC staff in SER 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 provided in YMRP Section 2.1.1.4.2. (b)(5)
(b)(5)

(b)(5)

(b)(5)

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 the applicant's development of event sequences for the subsurface resulting from seismically initiated events using the guidance in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.4.3.2.3 Fire Events

To demonstrate compliance with 10 CFR 63.112(b) and 63.111(c), the applicant described fire-initiated event sequences in 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 was to determine whether the applicant provided a systematic analysis of naturally occurring and human-induced, fire-related hazards at the GROA, including a comprehensive identification of potential event sequences.

The applicant screened out external fire and explosion-related events and focused its analysis on internal fire events. The NRC staff reviewed the applicant's screening basis in SER Section 2.1.1.3.3.1.3.5 (b)(5)

(b)(5)

The applicant began its development of fire-related event sequences with the aggregation of fire-related initiating event probabilities, as discussed in SER Section 2.1.1.3.3.2.3.4.3. The number of fire-related initiating event trees for a particular event sequence was based 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.

The applicant 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 such as containment, shielding, 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).

The applicant performed an independent analysis of the reliability of some SSCs that play a role in pivotal events (e.g., canister reliability under thermal challenges, shield performance under thermal challenges). For other cases where a full probabilistic fault tree analysis of the pivotal event was not warranted (e.g., loss of low melting temperature shielding material during a fire, loss of HVAC confinement during a large fire), the applicant 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.

The LLWF was developed as a single initiating event that involved all combustible waste at the

LLWF. There was one response tree for the LLWF because the applicant identified only one initiating event for the entire facility.

NRC Staff Evaluation: The NRC staff reviewed the information on the applicant's development of event sequences resulting from internal fire events using the guidance provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5) The development of the corresponding fire-related response trees was reviewed by the NRC staff and described herein.

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.3 Reliability of Structures, Systems, and Components

The applicant 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. The applicant 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 the applicant'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 determine whether the applicant (i) appropriately 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 provided by the applicant, (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 (SER Section 2.1.1.4.3.3.1) and active systems (SER 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

This SER section contains the NRC staff's review of reliability of the passive, important to safety (ITS) SSCs. The applicant's determination of passive reliability can be broadly 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. The applicant estimated reliabilities (failure probabilities) of the containers to provide containment and shielding functions and used these failure probabilities as input to pivotal events containment and shielding in the system response event trees for internal and seismic event sequence quantification. The applicant 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 SER Section 2.1.1.4.3.3.1.1, seismic fragility for structural and mechanical systems is presented in SER Section 2.1.1.4.3.3.1.2, and thermal challenges will be covered in SER 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

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant provided information on reliability of passive SSCs in SAR Section 1.7.2.3. The applicant presented the passive equipment failure analyses (PEFA) and summarized the failure probabilities for each container in BSC (2008ac,as,au,be,bq,) for surface facilities (CRCF, RF, IHF, WHF) and intrasite operations. In addition, the applicant 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. The applicant used two approaches to evaluate the passive reliability of containers: (i) full-scale drop test and (ii) determination of applied load or demand and the capacity of the component. The applicant used the first approach to determine passive reliability of HLW canisters in which statistical analyses 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, the applicant used the probability of loss of containment or failure of canisters under structural challenges as a point estimate in the pivotal event in the response tree.

High-Level Waste Canisters

In SAR Section 1.7.2.3.1 and BSC Sections 6.3.2.2 and D1.3 (2008ac), the applicant evaluated the probability of failure of HLW canisters for drops from operational and beyond operational height. The applicant's methodology for determining the canister reliability is based on full-scale experimental drop tests; reliability was estimated on the basis of the number of canisters breached out of the total number of tests. HLW canisters were dropped from heights of 7 m [23 ft] (considered as operational height) 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 the applicant inspected the canisters using two standard test techniques (helium leak test and liquid dye penetrant test) to detect leaks and cracks. Although in some cases (e.g., around the top fill nozzle) significant plastic deformations were observed, the canisters made of stainless steel did not show any ruptures or surface cracks.

The applicant treated these test results as Bernoulli trials where the outcome was either breach or no breach. Because there was no breach (failure) from the tests, the applicant used a Bayesian approach to estimate failure probabilities separately for the two drop heights. The applicant based the Bayesian analysis on a beta-binomial conjugate distribution, which led to a beta posterior failure probability distribution. The applicant then used the drop test results to estimate the mean and standard deviation for the beta posterior failure probability distribution. Using this approach, the applicant determined that for the 7- and 9-m [23- and 30-ft] drop heights, the mean failure probability posterior distribution was 3.4×10^{-2} and 6.7×10^{-2}, respectively. The applicant used the mean values as point estimates in the event sequence analysis. The actual HLW canister failure probabilities used in the event sequence analysis were 3×10^{-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). The applicant 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 the applicant's reliability analysis of HLW canisters using the guidance provided in YMRP Section 2.1.1.4.3. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

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 (b)(5)

(b)(5)

Waste Package

In SAR Section 1.7.2.3 and BSC Sections 6.3.2.2 and D1.4 (2008ac), the applicant discussed the calculation of the waste package passive reliability.

The applicant defined the waste package as a passive component that may fail when it is subjected to loads that exceed its capacity (i.e., strength). Moreover, the applicant stated that, because the waste package is designed in accordance with the provisions of American Society of Mechanical Engineers, Section III, Division 1, Subsection NC (2001aa), a failure may only occur under loads that are greater than the design load. Although all waste package configurations consist of an Alloy 22 outer corrosion barrier and a 316 stainless steel inner vessel (see SER Sections 2.1.1.2.3.5.1 for more details), the applicant based the waste package passive reliability only on the capacity of the Alloy 22 outer corrosion barrier.

The applicant defined one waste package failure mode as a structural challenge causing loss of containment (breach). Structural challenges that may cause a waste package to lose 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. The applicant 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, the applicant calculated the time histories of the Von Mises effective stress and strain from the initiation of loading to the time of unloading. Following a simplified toughness index equation and using the maximum Von Mises effective stress and

strain, the applicant estimated the waste package demand as a wall-averaged expended toughness (BSC, 2007cq).

The applicant modeled the capacity of the waste package Alloy 22 outer corrosion barrier using a material toughness and determined the waste package capacity through calculating the material toughness index (BSC, 2007bi,cq). The applicant stated (BSC, 2007cq) that vendor averaged properties were used for the mean strength properties of the Alloy 22 outer corrosion barrier; thus, the applicant took into account the variability of the Alloy 22 material properties. Further, the applicant used a bilinear stress-strain curve to approximate the stress-strain behavior of Alloy 22. Additionally, because the elastic strains of Alloy 22 are negligible when compared to the ultimate tensile strain of the material, the applicant used a simplified toughness index equation (BSC, 2007bi,cq) for estimating the toughness index of Alloy 22.

To determine failure of the Alloy 22 outer corrosion barrier, the applicant calculated an expended toughness fraction, *ETF,* defined as a ratio of the waste package demand (i.e., wall-averaged expended toughness) to the waste package capacity (i.e., material toughness index). The applicant assumed that waste package damage occurs for values of $ETF \geq 1$. The applicant 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. The applicant summarized the waste package failure probabilities in BSC Tables D1.4-1 and 6.3-7 (2008ac).

The applicant 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. The applicant 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, the applicant used a failure probability of 10^{-8} in the event sequence analysis.

NRC Staff Evaluation: The NRC staff reviewed the applicant's reliability analysis of waste packages using the guidance in YMRP Section 2.1.1.4.3. (b)(5)

(b)(5)

The NRC staff reviewed the applicant's approach for calculating the demands on the waste package due to structural challenges (b)(5)

(b)(5)

(b)(5)

(b)(5)

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*. This equation required a coefficient of variation to be computed for the Alloy 22 to calculate the failure probability. (b)(5)

(b)(5)

The NRC staff reviewed the waste package failure probability data presented in BSC Table 6.3-7 (2008ac). (b)(5)

(b)(5)

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), the applicant discussed the passive reliability of the TAD canisters and DPCs. The applicant'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. The applicant calculated the demand in terms of the maximum effective plastic strain from each finite element drop simulation analysis. The applicant also determined the structural capacity of TAD canisters and DPCs on the basis of tensile elongation at failure data obtained from canister material tests. The applicant 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, the applicant calculated the probability of canister breach by relating the strain obtained from a finite element analysis to the fragility curve.

Although the applicant 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, the applicant did not directly evaluate the TAD canister's reliability under the structural challenges. Instead, the applicant utilized a representative canister to evaluate the failure probability. The representative canister was defined 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. The applicant 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 304 Stainless steel. The failure probability of the representative canister was used for TAD canisters and DPCs in the PCSA.

The applicant 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. The applicant presented details of the LS–DYNA finite element models in BSC Section 6.3.3 (2008cp). The canister shell was modeled with multiple solid brick (three-dimensional) finite elements. The applicant performed finite element mesh sensitivity studies for mesh refinement and contact friction effects. The sensitivity studies demonstrated that the mesh and friction parameters selected for performing the impact analyses converged to a stable solution. The applicant determined the demand due to impact using the maximum effective plastic strain of a single brick element through the thickness of the shell.

The development of the canister capacity (fragility) curve was based on the material used for the representative container, which was 304 stainless steel. The fragility curve, which represents probability of failure as function of true strain, was determined by fitting a probability density function to engineering tensile strain data for 304 stainless steel. 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. The applicant found the data to be nonnormally distributed, and stated that 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), the applicant shifted this initial fragility curve 8.3 percent to a lower value of minimum elongation to obtain a more conservative (i.e., higher) estimate of the failure probability. The applicant 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). The applicant 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).

The applicant included the probability of failure for a 4° off-vertical drop in BSC Table D1.2-3 (2008ac). The applicant 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), the applicant presented information on guide sleeve functions.

NRC Staff Evaluation: The NRC staff reviewed the applicant's reliability analysis of the TAD canister and DPC using the guidance in YMRP Section 2.1.1.4.3 to determine the appropriateness of the methodology used to estimate the reliability of the TAD canister and DPC. (b)(5)

(b)(5)

The NRC staff reviewed the acceptability of using a representative canister in the reliability estimate for DPC. (b)(5)

(b)(5) The applicant 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). (b)(5)

(b)(5)

(b)(5) In absence of a final TAD canister design, the applicant used the same representative canister to determine the failure probability. (b)(5)

(b)(5)

(b)(5)

(b)(5) The demand on the representative canister was measured in terms of the maximum effective plastic strain of a single solid element experienced during impact. (b)(5)

(b)(5)

(b)(5)

The NRC staff reviewed the applicant'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. (b)(5)

(b)(5)

(b)(5) 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), the applicant stated the effects of strain rate and temperature were not included in the fragility curve used in BSC (2008cp). (b)(5)

(b)(5)

The NRC staff reviewed failure probability values for several different cases in BSC Table 6.3-7 (2008ac) for the bare representative canister. (b)(5)

(b)(5)

The NRC staff reviewed calculations the applicant provided for determining the maximum canister drop angle that could be achieved when the guide sleeve is present (DOE, 2009fy). (b)(5)

DOE Standardized Canisters

The applicant discussed the determination of failure probabilities of DOE standardized canisters in BSC Section D1.2 (2008ac). The applicant'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 the applicant's analysis of the representative canister.

The applicant 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. The applicant 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. The applicant 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). The applicant demonstrated 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.

The applicant determined that the structural capacity of the canister depended on tensile elongation at failure obtained from canister material tests. The applicant 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).

The applicant calculated the probability of the DOE standardized canister breach by relating the maximum effective plastic strain obtained from a finite element analysis to the fragility curve. The maximum equivalent plastic strains, obtained at select locations in the canister model, were listed in BSC 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), the applicant calculated the failure probabilities using the maximum equivalent plastic strains.

The applicant summarized the failure probabilities for the DOE standardized canister in BSC Tables D1.2-7 and 6.3-7 (2008ac). For the case of vertical container drop from normal operating height {7 m [23 ft]}, the applicant stated the failure probability is equal to 10^{-8}, as shown in BSC Table D1.2-7 (2008ac). However, the applicant 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, the applicant 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, the applicant reported a failure probability of 10^{-5}.

NRC Staff Evaluation: The NRC staff reviewed the applicant's reliability analysis of the DOE-standardized canister using the guidance in YMRP Section 2.1.1.4.3 to assess the appropriateness of the methodology the applicant used.

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Transportation Cask

The applicant 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. The applicant's methodology for estimating the transportation cask reliability is the same as that for the representative canister. The applicant established a relationship between demand and capacity defined in terms of strain in the canister material.

The applicant stated in BSC (2008cp) that the transportation cask 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. The applicant used the fragility (capacity) curve developed for representative canister as shown in BSC Figure 6.3.7-3 (2008cp).

The applicant presented a number of drop scenarios at different heights and cask orientations. Two of these drop scenarios included the effects of slapdown. For all of the drop scenarios, an explicit finite element analysis using 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. The applicant 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).

The applicant 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.

The applicant summarized the failure probabilities for the transportation cask in BSC Table 6.3-7 (2008ac) and BSC Table 6.3.7.6-2 (2008cp). The applicant 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, the applicant reported failure probabilities of 10^{-5} in BSC Table 6.3-7 (2008ac) for use in the event sequence quantifications. The applicant 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 the applicant's reliability analysis of the transportation cask using the guidance in YMRP Section 2.1.1.4.3. (b)(5)

(b)(5)

The applicant has utilized nonlinear finite element analysis (i.e., LS–DYNA) to determine the demand on the internal representative canister (inside the transportation cask). (b)(5)

(b)(5)

The applicant followed the same methodology for determining structural capacity of the representative canister as detailed in BSC (2008cp). (b)(5)

(b)(5)

The NRC staff reviewed the applicant'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). (b)(5)

(b)(5)

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). (b)(5)

(b)(5)

(b)(5)

(b)(5) The transportation casks, which are classified as ITS, are certified under 10 CFR Part 71 as stated by the applicant in SAR Section 1.2.8.4.5.1. (b)(5)

(b)(5)

(b)(5)

Aging Overpack

The applicant presented the failure probabilities for loss of containment for AOs in BSC Table 6.3-7 (2008ac). The applicant followed the approach given in BSC (2008cp) in which the relationship between demand and capacity is defined in terms of strain in the internal representative canister contained within the AO. As 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, the applicant considered two different loading scenarios for the AO/canister model: (i) a 0.9-m [3-ft] vertical drop (normal operating height) onto a rigid surface and (ii) a slapdown from a vertical orientation while also having a 4 km/hour [2.5 mph] horizontal velocity.

The applicant summarized in BSC 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, the applicant calculated the probability of AO containment failure as 10^{-5}. In addition, the applicant specified a failure probability of 10^{-8} for the cases of low velocity impact/collisions.

NRC Staff Evaluation: The NRC staff reviewed the applicant's reliability analysis of loss of containment for the AO using the guidance provided in YMRP Section 2.1.1.4.3. The NRC staff reviewed the LS–DYNA analyses the applicant 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. (b)(5)

(b)(5)

The NRC staff reviewed failure probability results from the LS–DYNA analyses as given in BSC Table 6.3.7.6-1 (2008cp) (b)(5)

(b)(5)

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

The applicant 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

The applicant'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 given in BSC Equation D–26 (2008ac). The applicant 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, the applicant followed the methodology in BSC (2008cp). The applicant established the relationship between demand and capacity defined in terms of strain in the internal representative canister material. The demand is calculated in terms of the maximum effective plastic strain in the internal representative canister from each finite element analysis of a drop simulation. The applicant performed the explicit finite element analyses using the computer code LS–DYNA to calculate the demand on the internal representative canister, as discussed in

BSC Section 6.3.1 (2008cp). The finite element model consisted of an AO and the internal representative canister contained within the AO. The applicant stated that the internal SNF canister was the same as that used for the representative canister.

The applicant 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.

The applicant 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, the applicant calculated the AO shielding failure probability as 5.0×10^{-6}. In addition, the applicant specified a failure probability of 10^{-5} for the cases of low velocity impact/collisions and slapdown.

NRC Staff Evaluation: The NRC staff reviewed the applicant's reliability analysis of loss of shielding for the AO using the guidance provided in YMRP Section 2.1.1.4.3. The NRC staff reviewed the applicant's methodology for determining the AO shielding failure due to a structural challenge (b)(5)

(b)(5)

The NRC staff reviewed the finite element analyses the applicant performed to determine demand (b)(5)

(b)(5)

The applicant followed the same methodology for determining structural capacity of the representative canister as detailed in BSC (2008cp). (b)(5)

(b)(5)

The NRC staff reviewed the failure probabilities listed in BSC Table 6.3-7 (2008ac) for the AO (b)(5)

(b)(5)

(b)(5)

Transportation Cask

The applicant discussed in BSC Section D3 (2008ac) the degradation of shielding for a transportation cask when subjected to a structural challenge due to impact.

The applicant's methodology for estimating the failure probability utilized finite element analysis to determine the demand on the transportation cask subjected to transportation accident impacts. The applicant estimated the amount of damage by determining the amount of plastic strain in the transportation cask's inner shell as a function of the impact speed. On the basis of the impact speed, equivalent drop heights were calculated relating the maximum plastic strain as a function of drop height.

The applicant 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 transportation cask during impacts. The applicant 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, the applicant stated that lead slump occurs mainly for the end-impact orientations and, to a lesser extent, for center of gravity over corner impacts. For side impacts, the applicant stated that there is no significant reduction in shielding. Thus, the applicant focused only on the steel/lead/steel casks with the primary orientation being the end-impact condition. The applicant listed various impact speeds and the resulting maximum plastic strains for impacts onto an unyielding surface, as outlined in BSC Table D3.2-1 (2008ac). The applicant 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). The applicant 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), the applicant estimated the threshold velocities in which the loss of shielding (lead slumping) would occur. The applicant further stated that the 2 percent maximum plastic strain threshold for a truck cask would correspond to a 101 km/hour [63 mph] impact on a concrete surface, which translates into an equivalent drop height of 41 m [133 ft], as outlined in BSC Table D3.3-1 (2008ac). Thus, the applicant used an estimate of a median threshold for the failure drop height as 41 m [133 ft] (i.e., 2 percent plastic strain).

To calculate the transportation cask failure probability as a function of height, the applicant utilized BSC Equation 17 (2008ac). The applicant based the failure probability upon a median threshold of a 41-m [133-ft] drop, as discussed previously. In addition, the applicant 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). The applicant 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. The applicant stated that for all other casks, the only loss of shielding mechanism was by radiation streaming. Thus, the loss of shielding was equated to the probability of rupture of the cask due to closure failure, as outlined in BSC Section D3.4 (2008ac).

NRC Staff Evaluation: The NRC staff reviewed the applicant's reliability analysis of loss of shielding for the transportation cask using the guidance in YMRP Section 2.1.1.4.3.

(b)(5)

The NRC staff reviewed the failure probabilities listed in BSC Table 6.3-7 (2008ac) for the transportation cask (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.3.1.2 Passive Reliability for Structural Challenges Resulting From Seismic Events

This SER section contains the NRC staff review of reliability of the passive ITS SSCs for structural challenges resulting from seismic events. The NRC staff's review is presented in SER Section 2.1.1.4.3.3.1.2.1 for surface civil structures, and mechanical systems are covered in SER Section 2.1.1.4.3.3.1.2.2.

2.1.1.4.3.3.1.2.1 Surface Structural Civil Facilities

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(b), the applicant 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). The applicant 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. The applicant's approach was to demonstrate that the probability of structural failure is beyond Category 2 event sequences and therefore precludes radiological dose computation. The applicant presented the parameters used to develop fragility curves (i.e., median capacity and dispersion) and estimates of the annual failure probability for the surface facilities in BSC 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 adequacy of the information presented in the SAR with respect to (i) the approach used to generate fragility curves for ITS structural facilities, (ii) appropriateness of the capacity and uncertainty data used to develop fragility curves, and (iii) evaluation of annual probability of unacceptable performance.

For the ITS surface facilities, the applicant stated that seismic loading controls the structural performance (SER Section 2.1.1.7.3.1.1) and evaluated the structural performance only at the collapse limit state [i.e., Limit State A, according to American Society of Civil Engineers, Table 1-1 (2005aa)]. The applicant 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, the applicant 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 Unacceptable Performance

The applicant 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, the applicant 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). The applicant 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). The applicant also provided a table summarizing the computation of the probability of failure, P_f, for the CRCF in DOE Enclosure 3 (2009dz), which the applicant considered representative of the ITS buildings.

NRC Staff Evaluation: The NRC staff reviewed the methodology the applicant provided for computing the probability of unacceptable performance (or probability of failure) using the guidance provided in YMRP Section 2.1.1.4.3. 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 evaluated in SER Section 2.1.1.2.3.6.1.

(b)(5)

(b)(5)

Methodology for Generation of Fragility Curves

The applicant used a simplified methodology for developing seismic fragility curves for ITS surface facilities in SAR Section 1.7.2.4 and DOE (2007ab). In this methodology, seismic fragility curves were based 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, the fragility data were generated 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 is obtained from a simplified elastic model, Tier #1 (see SER Section 2.1.1.7.3.1.1 for a detailed discussion). Additionally, in its response to an NRC staff RAI (DOE, 2009dz), the applicant provided the fragility calculation only for the CRCF (BSC, 2007df).

NRC Staff Evaluation: The NRC staff reviewed the methodology the applicant provided to generate the fragility curves for the ITS surface facilities using the guidance provided in YMRP Section 2.1.1.4.3. (b)(5)

(b)(5)

Computation of $C_{1\%}$

To obtain $C_{1\%}$, the applicant performed elastic analyses at Beyond Design Basis Ground Motion (BDBGM) seismic levels to compute the High Confidence of Low Probability of Failure (HCLPF) capacity of the system, which was considered a reasonable approximation of $C_{1\%}$. The HCLPF capacity was computed based on the Conservative Deterministic Failure Margin method. For low-rise shear walls, the applicant 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).

The applicant designed the ITS surface facilities to withstand DBGM–2 seismic levels, which correspond to a mean annual probability of exceedence (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 SER Section 2.1.1.7.3.1.1. Tier #1 analyses were based on lumped-mass, multiple-stick models, in which floors were considered rigid slabs and soil–structure interaction was approximated using equivalent linear soil springs. For the IHF, the applicant used a finite element model, 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, the applicant 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. The applicant 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 request for technical bases on using Tier #1 models instead of more accurate finite element models, the applicant indicated 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), the applicant stated it will not provide nonlinear structural analyses of the surface facilities. Instead, the applicant 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 structure exhibited structural irregularities, the applicant considered that the recommended approach for buildings with structural irregularities is appropriate.

NRC Staff Evaluation: The NRC staff reviewed the information the applicant provided about the Tier #1 structural analysis used to obtain the probability of unacceptable performance, using the guidance in YMRP Section 2.1.1.4.3. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5) In response to the NRC staff's RAI in DOE Enclosure 4 (2009dz), the applicant indicated 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 required and the final reinforcement details will be developed in the detailed design for construction.[2] (b)(5)

(b)(5)

(b)(5)

Estimation of β_c

The applicant 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) (Campbell, et al., 1988aa; Kennedy, 2001aa; Shinozuka, et al., 2003aa), the applicant indicated that β_c estimates higher than 0.52 were not credible (DOE, 2009ge, Enclosure 8) and that β_c

[1]The soil demand-to-capacity ratio assumes that the large bearing capacity of 1,915 kPa [50 ksf] the applicant proposed is adequate.

[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.

estimates exceeding 0.4 were the result of simplified evaluations. The applicant 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. The applicant recommended using the lower bound value, β_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 instead used a β_c value of 0.4 to evaluate probability of failure, as outlined in BSC Section 6.2.1 (2008bg). The applicant stated that using β_c = 0.3 increased the P_f by about 50 percent as compared with using β_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 the applicant provided about the Tier #1 structural analysis used to obtain the probability of unacceptable performance, using the guidance provided in YMRP Section 2.1.1.4.3. (b)(5)

(b)(5)

Proposed Condition of Construction Authorization

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.3.1.2.2 Mechanical Equipment and Systems

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant provided information on seismic fragilities of the mechanical equipment in SAR Section 1.7. The applicant 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. The applicant used PGA to define the fragility curve. The applicant 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. The applicant considered a number of variables to quantify design margins and standard deviations (BSC, 2008bg).

The applicant 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). The applicant identified the failure modes of the equipment under seismic loads and provided median capacity, composite uncertainty, and annual probability of failure associated with the failure modes, as outlined in BSC Table 6.2.2 (2008bg). The annual probability of failure was calculated by convolving the fragility curve and the seismic hazard curve.

The applicant 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). The applicant expects the final equipment design will provide margin exceeding the preliminary design.

NRC Staff Evaluation: The NRC staff reviewed the information on the applicant's approach for assessing seismic fragility of mechanical systems using the guidance provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

The equipment fragility evaluation was based on a preliminary design (BSC, 2008bg). The applicant committed (DOE, 2009bl) to verify that the final equipment design and its associated fragility will satisfy the conclusions in BSC Table 6.2-2 (2008bg).

Commitment

The applicant commits to verify that the final equipment design and its associated seismic fragility will satisfy the conclusions in BSC Table 6.6-2 (2008bg).

NRC Staff Conclusion

On the basis of the NRC evaluation discussed previously and with the applicant's commitment, (b)(5)

2.1.1.4.3.3.1.2.3 Passive Reliability for Structural Challenges Resulting From Fire Events

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant provided information on the development of passive reliability probabilities for canister shielding and canister containment during fire-induced thermal challenges in SAR Sections 1.7.2.3.3 and 1.7.2.3.4 and BSC (2007ab,aw,bb,bf; 2008ae,ai,ap,bp).

The applicant developed passive reliability of SSCs subjected to fire challenges on the basis of either probabilistic analysis (e.g., probability of canister failure due to fire exposure) or basic design assumptions (e.g., assessment of concrete spalling under thermal challenges or the performance of low melting temperature shielding materials in a fire).

Dominant pivotal events in fire-related event sequences were the probability of canisters maintaining containment and the probability of maintaining shielding. The applicant estimated the probabilities of these pivotal events on the basis of an assessment of potential thermal challenges to various canisters and shielding configurations, and their predicted response to those exposures. SAR Section 1.7.2.3.3 summarized information on potential loss of containment or breach under thermal challenges, and SAR Section 1.7.2.3.4 summarized information on loss of shielding under thermal challenges.

Thermal Challenges and Loss of Containment

The applicant characterized the thermal demands on a canister as the canister wall temperature resulting from a fire exposure of a certain temperature for a certain duration. To quantify the demands, the applicant postulated fire exposure conditions and calculated the expected wall temperatures that could result from those exposures. The applicant used the fire data from large-scale tests conducted by different laboratories to develop a reasonable distribution of fire durations. 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. The applicant 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.

The applicant 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). The applicant took the fire temperature as the effective blackbody temperature of the fire. The applicant 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). The applicant also used flammable liquid fire data obtained from large-scale hydrocarbon fires involving railcars (Birk A.M. Engineering, 2005aa). The applicant reported effective temperatures ranging from 400 to 1,200 °C [752 to 2,192 °F] in fires involving solid fuel materials and temperatures from 927 to 1,327 °C [1,701 to 2,421 °F] in flammable-liquid pool fires. The applicant used these 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), the applicant indicated that 99.9 percent of all fires would have a temperature lower than or equal to approximately 1,330 °C [2,426 °F].

The applicant calculated the heat transfer to bare fuels and canisters inside casks using standard heat transfer models that were validated using finite element analyses. The ultimate wall temperature of a canister exposed to a fire is a function of fire duration, exposure

temperature, and the physical properties of the container. The applicant 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). The applicant 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.

The applicant evaluated the canister response to a thermal challenge on the basis of the canister's ability to withstand stresses induced by the elevated temperature. The applicant stated that creep-induced failure and limit load failure were two possible failure modes (BSC, 2008ac,as,au,be,bk,bq) and described the canister temperatures that could result in failures. The applicant evaluated both failure modes independently, and the lower of the two failure temperatures was assumed to govern.

Once the range of thermal demands and responses were identified, the applicant developed a demand curve using the range of canister wall temperatures resulting from a distribution of fire exposure temperatures and durations, and a corresponding response curve based on load limit and creep-induced failure probability as a function of canister temperatures. The superposition of the demand curve and the response curve yielded the number of expected failures that would occur in a given number of trials. The number of "observed" failures divided by the number of trials was taken as the failure probability.

Demand curves (showing resulting canister wall temperatures) were based on 100,000 to 1 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. The applicant assigned a failure probability of less than 10^{-6} for cases with no observed failures after 1 million trials.

The applicant 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). The failure probability of 5.4×10^{-4} for the bare fuel in a transportation cask was estimated 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 the applicant's evaluation of passive reliability of canisters under thermal challenges using the guidance provided in YMRP Section 2.1.1.4.2.

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Thermal Challenges and Loss of Shielding

The applicant 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.

The applicant 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). The applicant indicated that all transportation casks have separate gamma and neutron shields. Because the neutron shield is typically fabricated from a low melting point polymer, the applicant stated that its shielding function would be quickly lost when subjected to a thermal challenge; therefore, the neutron shield would govern the loss of shielding 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. The applicant 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, the applicant assigned a probability of failure of 1.0 for loss of transportation cask shielding due to a thermal challenge. The applicant also conservatively applied a failure probability of 1.0 to all transportation casks that do not use lead for shielding,

as shown in BSC Table D3.4-1 (2008ac,as,au,be,bk,bq).

The applicant 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. The applicant 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 unacceptable doses. As a result, the applicant assumed a loss of shielding probability of 0.0 for AOs.

The applicant 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. The applicant assumed that loss of CTM shielding probability was 0.0 because the shield bell components (primarily depleted uranium) have very high melting temperatures {3,400 °C [6,152 °F]}. Additionally, the applicant stated that fuel loading in proximity to the CTM made it highly unlikely that these melting temperatures would ever be achieved.

NRC Staff Evaluation: The NRC staff reviewed the information on the applicant's evaluation of the reliability of passive systems providing shielding under thermal challenges using the guidance provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5)

(b)(5)

For the AO, the applicant assumed various thicknesses of concrete to spall in a fire. (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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

To demonstrate compliance with 10 CFR 63.111(c) and 10 CFR 63.112(b), the applicant 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). The applicant included the confinement pivotal event in event sequences leading to a filtered radionuclide release end state for the surface nuclear confinement ITS HVAC systems in the CRCF and WHF. The applicant 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. The applicant 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. Note that the NRC staff evaluates the design of the ITS HVAC systems in SER Section 2.1.1.7.3.3 and the ability of the ITS HVAC systems to perform their safety functions in SER Section 2.1.1.6.3.2.8.2.2.

The applicant provided design information for the surface facilities HVAC systems in SAR Section 1.2.2.3. It described the surface nuclear confinement HVAC system specific to the CRCF in SAR Section 1.2.4.4 and BSC (2008ac) and specific to the WHF in SAR Section 1.2.5.5 and BSC (2008bq). 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 the HVAC supports the functioning of the emergency diesel generators and ITS electrical equipment. The applicant described the surface nonconfinement ITS HVAC system for the EDGF in SAR Section 1.2.8.3. During this review,

the applicant 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, 2008~~9a~~c~~b~~,bq~~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. The applicant provided a ventilation and instrumentation diagram in SAR Figure 1.2.4-101 (DOE, 2009av) for the operating train, which it refers to as Train A. This diagram showed one exhaust fan with an adjustable speed drive and an interlock. The interlock also connects to the standby train (i.e., Train B), which showed (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.

The applicant provided a fault tree model for loss of delta pressure in the CRCF in BSC Section B7.4 (2008ac~~9ab~~). This fault tree model accounts for both equipment failure and human error. The applicant stated that the top level of the fault tree in BSC Figure B7.4-3 (2008ac~~9ab~~) is incorrect for the CRCF and that BSC Figure B7.4-3 (~~2009ac~~2008bq) for the WHF is applicable to the ITS HVAC exhaust subsystem in the CRCF (DOE, 2009dq). The applicant showed in its fault tree in BSC Figure B7.4-3 (~~2009ac~~2008bq) 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, the applicant identified a potential leak from a HEPA filter plenum, as shown in BSC Figure B7.4-4 (~~2009ab~~2008ac), 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~~2008ac]. In addition, for the loss of confinement boundary, the applicant 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~~2008ac). For the loss of HEPA-filtered exhaust, the applicant 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, the applicant showed failure of the fan itself, failure of the adjustable speed drive for the fan, and loss of AC power. For loss of AC power, the applicant 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, the applicant 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~~2008ac) and continue to operate, as outlined in BSC Figure B7.4-19 (~~2009ab~~2008ac). Additionally, for the failure of the standby exhaust fan to startup, the applicant specified failure of the adjustable speed drive start logic and failure of the logic controller, as

shown in BSC Figure B7.4-9 (~~2009ab~~2008ac).

The applicant 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~~2008ac). The associated controlling parameter in SAR Table 1.9-3 (DOE, 2009av) is 4×10^{-2}. The applicant listed its dominant cut sets in BSC Table B7.4-3 (~~2009ab~~2008ac). In this table, the applicant 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 the applicant's surface nuclear confinement ITS HVAC exhaust system design criteria listed in SAR Table 1.2.4-4 (DOE, 2009av). The applicant 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.

The applicant provided its basic event probabilities and descriptions for these basic events in BSC Table B7.4-1 (~~2009ab~~2008ac). The applicant included associated active component reliabilities in BSC Attachment C (~~2009ab~~2008ac) and in its active component database (i.e., file "YMP Active Comp Database_final 8 August 2008.xls") included in BSC Attachment H (~~2009ab~~2008ac). The applicant's response to an NRC staff RAI (DOE, 2009fz) clarified how it quantified some 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], the applicant clarified 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 need not have been modeled. It clarified 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. The applicant 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. The applicant 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, the applicant indicated that it included control circuitry pertaining to fan tripped detection and failure of the ITS interlock in its exhaust fan failure to start basic event (DOE, 2009gi).

NRC Staff Evaluation: The NRC staff reviewed the applicant's HVAC system failure probability quantification using the guidance in YMRP Section 2.1.1.4. (b)(5)

(b)(5)

(b)(5)

In terms of the applicant's quantification of basic events used in its fault tree analysis, (b)(5)

(b)(5)

(b)(5)

Proposed Condition of Construction Authorization

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.3.2.2 Moderator Intrusion Control

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(b), the applicant 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)]. The applicant identified water from suppression systems, water from building service piping, and lubricating oil from overhead hydraulic equipment as potential moderator sources.

The applicant included specialized automatic suppression systems in the moderator control areas to reduce the potential for inadvertent water discharge due to spurious activation. The selected system is a double-interlock preaction (DIPA) system. The applicant identified these systems as ITS because of their role in preventing accidental moderator intrusion; a complete evaluation of the DIPA system is provided in SER Section 2.1.1.7.3.1.6. The applicant 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 provided a failure rate per unit length of pipe. The applicant used these data to derive a probability for other water sources introducing a moderator and included these potential sources in the moderator fault tree.

The applicant also described the fault tree associated with the probability of lubricating oil

leakage and the potential leakage path for lubricating oil to contact a breached canister.

The 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. The applicant derived the probability of moderator introduction using a simple fault tree that combined the failure probability of water and lubricating oil moderator sources (e.g., overhead sprinkler system, building service piping, or crane gearbox and containment pan) and evaluated their potential intrusion into a breached canister. The applicant provided the nuclear safety design basis for the prevention of DIPA system failure in SAR Table 1.4.3-2. The probabilities of inadvertent operation and water intrusion ranged from 5.0×10^{-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. The applicant 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, it must be assumed that a canister has been breached and is capable of accepting moderator. The applicant 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 the applicant's evaluation of reliability of the moderator control system using the guidance provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.4 Event Sequence Quantification and Categorization

In accordance with 10 CFR 63.21(c)(5), 63.111(c), and 63.112(b), the applicant is required to quantify and categorize event sequences to demonstrate compliance with preclosure performance objectives. The applicant 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 adequate the applicant's evaluation of probability of occurrence of event sequences and categorization in accordance with definitions of Category 1 and Category 2 event sequences if (i) the ESDs are implemented acceptably in the SAPHIRE model and (ii) data to quantify the event sequences are used appropriately.

In its review of event sequence quantification and categorization, the NRC staff evaluated the appropriateness of the approaches for (i) categorization of event sequences in SER Section 2.1.1.4.3.1 and (ii) development of ESDs in SER 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 SER Section 2.1.1.3.3.2.3, (ii) throughput numbers of the waste form in SER Section 2.1.1.2.3.6.1, and (iii) reliability data of active and passive systems in SER 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 (SER Section 2.1.1.4.3.4.1), Seismic Events (SER Section 2.1.1.4.3.4.2), and Fire Events (SER Section 2.1.1.4.3.4.3).

2.1.1.4.3.4.1 Internal Events

The NRC staff's review of the applicant's event sequence quantification and categorization for internal events 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

To demonstrate compliance with 10 CFR 63.111(c) and 63.112(b), the applicant 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. The applicant 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. The applicant 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 acceptably implement 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

The applicant'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. The applicant modeled the ESDs in SAPHIRE in an integrated fashion linking the initiator event tree, the system event trees, and the associated fault trees. The applicant 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 needed 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 appropriateness of 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 understand and verify the applicant's event sequence analysis. The NRC staff focused its review on the appropriateness of 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 used a vertical slice approach 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

The applicant quantified each initiating event ("small bubble") in the ESD using fault tree analysis. The input data were needed 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 the applicant used for initiating events to evaluate internal event sequence quantification and categorization related to structural challenges and direct exposure using the guidance in YMRP Section 2.1.1.4.2. The NRC staff reviewed the identification of initiating events and grouping of initiating events in ESDs in SER Section 2.1.1.3.3.2.1. In addition, evaluation of initiating event frequency quantification including the data used for fault tree modeling is reviewed in SER Section 2.1.1.3.3.2.3.1. (b)(5

(b)(5)

Throughput Numbers

The number of waste from 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, the applicant factored in the number of operations in the initiating event fault tree.

NRC Staff Evaluation: The NRC staff reviewed the throughput data the applicant used for event sequence quantification and categorization for internal events related to structural challenges and direct exposure using the guidance in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5) The NRC staff reviewed the applicant's throughput calculation in SER Section 2.1.1.2.3.6.1 1. (b)(5)

(b)(5)

The NRC staff reviewed event sequence CRCF–ESD09–TAD (BSC, 2008ac) to verify the throughput number for handling TAD canisters in the CRCF and consistency with the operational process described. (b)(5)

(b)(5)

Pivotal Events

The applicant 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. The applicant 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, the applicant also credited the containment function of the transportation cask containing bare SNF. The probability of loss or degradation of shielding under structural challenges was evaluated for the cask system (e.g., transportation cask, AOs, and STC) and used as input in the shielding pivotal event.

The applicant 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 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 the applicant used for pivotal events for internal event sequence quantification and categorization related to structural challenges and direct exposure using the guidance in YMRP Section 2.1.1.4.2. The NRC staff reviewed event sequence development in SER Section 2.1.1.4.3.2.1 (b)(5)
(b)(5)
(b)(5) The NRC staff reviewed the probability of containment and shielding failure of passive components in SER Section 2.1.1.4.3.3.1.1. (b)(5)
(b)(5)

The NRC staff reviewed the reliability of HVAC system providing a containment function in SER Section 2.1.1.4.3.3.2.1 (b)(5)
(b)(5)
(b)(5) In addition, the NRC staff reviewed the loss of moderator intrusion in SER Section 2.1.1.4.3.3.2.2 (b)(5)
(b)(5)

Exposure to Workers

The applicant concluded 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 require no radiation dose calculation in accordance with 10 CFR 63.111. Categorization of event sequences in the applicant's analysis can be classified into three groups: (i) direct exposure, (ii) loss or degraded shielding, and (iii) shield intact.

Direct Exposure

The applicant 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, the applicant categorized the event sequences as Category 2.

NRC Staff Evaluation: The NRC staff reviewed the applicant's event sequence quantification and categorization for internal events related to direct exposure using the guidance provided in

YMRP Section 2.1.1.4.2. The NRC staff reviewed the event sequences resulting from direct exposure identified for the CRCF (e.g., CRC–ESD–17, 18, 19). The applicant 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. (b)(5)

(b)(5)

Loss or Degraded Shielding

The applicant 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, the applicant relied on the shielding capabilities of the transportation cask, AO, STC, CTM bell, or WPTT. The applicant 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. The applicant 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 the applicant's internal event sequence quantification and categorization related to loss or degraded shielding using the guidance provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

Shielding Intact

The applicant 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. The applicant 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, the event sequences associated with exposure or direct radiation from loss or degradation of shielding to the workers were categorized as beyond Category 2.

NRC Staff Evaluation: The NRC staff reviewed the applicant's quantification and categorization of internal event sequences for the conditions with shielding to remain intact using the guidance provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

Radionuclide Release

The applicant'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, the applicant 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). The applicant evaluated dose consequences in accordance with the end states for filtered and unfiltered radionuclide releases. The applicant assigned dose consequence designators for the Category 2 event sequences involving structural challenges. The NRC staff evaluated the applicant's dose calculations in SER Section 2.1.1.5.

NRC Staff Evaluation: The NRC staff reviewed the applicant's quantification and categorization of selected internal event sequences related to radionuclide release using the guidance in YMRP Section 2.1.1.4. 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). (b)(5)

(b)(5)

(b)(5) The applicant used probability of canister failure as 10^{-5} for vertical drop. The NRC staff review of probability of TAD canister loss of containment in SER Section 2.1.1.4.3.3.1.1 (b)(5)

(b)(5)

The NRC staff reviewed the event sequences for radionuclide release that resulted from high frequency initiating events evaluated previously under loss of shielding or degraded shielding.

(b)(5)

(b)(5) The failure probability for drop, drop on, and tipover was 10^{-5} and for side impact was 10^{-8}. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.4.1.2 Wet Handling Operations

To demonstrate compliance with 10 CFR 63.111(c) and 63.112(b), the applicant described the WHF event sequence analysis in SAR Section 1.7.5.4. In addition, the applicant included ESDs in BSC Attachment F (2008bo) and event trees in BSC Attachments G and A, respectively (2008bo; 2008bq). Also, the applicant 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, the applicant 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 SER 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 SER Section 2.1.1.3.3.2.3.1.

SAR Table 1.7-13 showed a total of 48 event sequences that the applicant identified with (i) no Category 1, (ii) 22 Category 2, and (iii) 26 Beyond Category 2 event sequences. The applicant 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),

the applicant 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 adequate boron concentration from the ESD mapped to a pivotal event in the applicant'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 the applicant'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), the applicant 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). The applicant separated the ESD and the event trees to account for whether the initiating event (e.g., drop or impact) occurs over the pool or over the floor outside of the pool. 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 adequate boron concentration were shown on the ESD and mapped to pivotal events in the applicant'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 the applicant'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 the applicant's response tree, as described in BSC Figure A5–3 (2008bq). In addition to end states involving radionuclide release, the applicant specified a direct exposure in its ESD. These end states were shown in the applicant'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.

The NRC staff also reviewed 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, the applicant 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, the applicant'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), the applicant 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: the applicant did not credit the cask for maintaining containment for a drop above the operational height over the pool. The applicant 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. The applicant indicated that it performed the analyses to determine the minimum number of installed fasteners required. The applicant 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), the applicant 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 the applicant'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 the applicant'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 the applicant's WHF event sequence quantification and categorization using the guidance in YMRP Section 2.1.1.4. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5) The applicant 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, the applicant commits to 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).

(b)(5)

(b)(5) For failure to exclude moderator from entering a cask, the applicant specified a probability of failure of zero. (b)(5)

(b)(5)

(b)(5)

(b)(5)

Commitment

The applicant commits to 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).

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.4.1.3 Subsurface Operations

To demonstrate compliance with 10 CFR 63.111(c) and 63.112(b), the applicant 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. The applicant identified potential Category 2 and Beyond Category 2 event sequences (SAR Tables 1.7-17 and 1.7-18) and concluded that the Category 2 event sequences will not lead to any release. Also, the applicant 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.

The applicant relied on administrative controls to assign a beyond Category 2 frequency to the event sequence related to inadvertent entry into an emplacement drift. The applicant 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 the applicant, 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.

The applicant relied on administrative controls to assign a beyond Category 2 frequency to the event sequence related to worker proximity to a loaded TEV. The applicant 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).

The applicant 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. The applicant stated that the event sequence was Category 2. 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.

The applicant 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). The applicant 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 the applicant's quantification and categorization of the event sequence frequencies for subsurface operations except event sequences initiated by fire, which are reviewed in SER Section 2.1.1.4.3.4.3, using the guidance in YMRP Section 2.1.1.4. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.4.2 Seismic Events

To demonstrate compliance with 10 CFR 63.111(c) and 63.112(b), the applicant 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. The applicant detailed the seismic event sequence analysis in BSC (2008bg).

The applicant'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, the applicant considered rockfall in the emplacement drift initiated by seismic events. The applicant'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 buidling, and 1 in the subsurface). Six identified event sequences end in unfiltered radionuclide releases, which required consequence analyses to demonstrate compliance with the regulation. The single Category 2 subsurface event sequence the applicant identified involving direct exposure due to loss of TEV shielding by seismic failure required no dose calculation. The applicant calculated dose consequences to the public for radionuclide release scenarios and reviewed the information in SER Section 2.1.1.5.

The NRC staff reviewed the applicant's methodology for evaluation of seismically initiated events in SER Section 2.1.1.4.3.1.2 and development of event sequences in SER Section 2.1.1.4.3.2.2. The NRC staff also reviewed seismic fragility of SSCs in SER Section 2.1.1.4.3.3.2 and passive reliability of the containers in SER Section 2.1.1.4.3.3.1. In this section, the NRC staff reviews (i) the appropriateness of implementation of the methodology and the event sequences; (ii) whether the applicant considered appropriate 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 appropriate. 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.

Seismic Hazard Curve

The applicant developed site-specific seismic hazard curves for the surface and subsurface repository block based on 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. The applicant 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 provided in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

Structural Collapse of Facility Structures

The applicant evaluated the collapse of the ITS CRCF, WHF, RF, and IHF structures as potential seismically initiated events. The applicant 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} meet the applicant's nuclear safety design bases threshold of 2×10^{-6}/year as indicated in SAR Tables 1.9-2 to 1.9-4 (DOE, 2008ab). The collapse of surface structures could directly result in unfiltered radionuclide release. Thus, the applicant did not transfer the initiating event to a seismic response tree, because the applicant did not rely on any SSC to provide prevention or mitigation functions. The applicant (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, the applicant categorized seismic collapse of non-ITS LLW building breaching multiple LLW containers to be a Category 2 event sequence. For this event, the applicant further calculated the dose consequence, which is reviewed in SER Section 2.1.1.5.

NRC Staff Evaluation: The NRC staff reviewed the information on the applicant's evaluation of structural collapse of facility structures using guidance provided in YMRP Section 2.1.1.4.2.

(b)(5)

(b)(5) In SAR Section 2.2, the applicant 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. (b)(5)

(b)(5)

(b)(5) The NRC staff reviewed the fragility evaluation and the seismic performance of the surface structures in SER Section 2.1.1.4.3.3.1.2.1. (b)(5)

(b)(5)

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. The applicant 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. The applicant's seismic event sequence analysis was detailed in BSC (2008bg), which includes SAPHIRE model data files for each facility indicated in

Appendix J (BSC, 2008bg). The applicant 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. The applicant'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.

The applicant 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)]. The applicant 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. These 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. The applicant 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), the probability of waste container failure was considered as 10^{-5}.

The applicant used the exposure or residence time factor to demonstrate compliance with 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 provided in YMRP Section 2.1.1.4.2. The NRC staff reviewed the applicant's basis for the fragility parameters for

each failure mode and annual probability of failure in SER Section 2.1.1.4.3.3.1.2.2 (b)(5)
(b)(5)

The NRC staff reviewed the passive reliability of containers in SER Section 2.1.1.4.3.3.1.1 (b)(5)
(b)(5)

(b)(5) 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), the applicant assumed the waste containers to breach in most cases. For example, the probability of failure for the waste containers was assumed to be one for collapse of the crane, platforms, and staging racks. (b)(5) For collapse of the mobile platform, collapse of shield doors, and tipover of railcar trolleys on transportation cask, the applicant 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 an equivalent [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. (b)(5)
(b)(5)

(b)(5)

Subsurface

The applicant addressed the subsurface event sequences associated with rockfall impact in BSC Section 6.9.2.1 (2008bg). The applicant 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 the applicant's evaluation of event sequence quantification and categorization for subsurface seismic events using the guidance in YMRP Section 2.1.1.4.2. (b)(5)
(b)(5)

(b)(5)

Intrasite

For AO on an aging pad, the applicant considered in BSC Attachment E (2008bg) failure modes AO tipover, and sliding of AO and impact with other AO, and aging pad displacement and tipover of AO, and categorized event sequence as beyond category 2.

NRC Staff Evaluation: The NRC staff reviewed the information on the applicant's evaluation of event sequence quantification and categorization for intrasite seismic events using the guidance in YMRP Section 2.1.1.4.2. (b)(5)

(b)(5)

(b)(5) In addition, the applicant conducted a finite element analysis of AO with a canister inside for (i) a 0.914m [3 ft] vertical drop, and (ii) a slap down and having horizontal velocity of 1.117 m/s [2.5 mph] (BSC, 2008cp) on rigid (unyielding) ground. The applicant 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 and resulted in a failure probability of about 10^{-8} BSC Table 6.3.7-1 (BSC, 2008cp). (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.3.4.3 Fire Events

To demonstrate compliance with 10 CFR 63.21(c)(5), 63.111(c), and 63.112(b), the applicant quantified and categorized event sequences initiated by fires. The applicant 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 the applicant implemented the established fire-related ESDs correctly for the purposes of quantification.

Data Used in Fire Event Sequence Quantification

The applicant 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, the applicant relied on the derivation of passive reliabilities and other pivotal event probabilities to reduce the overall frequency of a particular event sequence.

The applicant 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. The applicant 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, the applicant 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, the applicant 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 the applicant'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).

The applicant typically assigned the probabilities of moderator intrusion to be either 0.0 or 1.0

for fire events. The 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). The applicant did identify 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 the applicant used to quantify its fire event sequences using the guidance provided in YMRP Section 2.1.1.4. The NRC staff evaluates the applicant's quantification for loss of confinement and moderator intrusion control in SER Section 2.1.1.4.3.3.2. In addition, the NRC staff evaluated the applicant's quantification for loss of containment (i.e., cask failure and loss of shielding) in SER Section 2.1.1.4.3.3.1.2.3.

(b)(5)

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, waste forms in transportation casks were assumed to suffer shielding failure during fire-initiated event sequences, while waste forms protected by the CTM shield bell were assumed to not fail. Furthermore, the calculation of event sequence probabilities used a basic event probability for HVAC confinement failures (3×10^{-2}). The applicant 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, the applicant indicated that moderator had no impact on HLW; however, the applicant 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. No direct exposure scenarios were identified 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. Loss of shielding was assumed to 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 YMRP Section 2.1.1.4 (b)(5)

(b)(5)

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. Five event sequences were categorized as beyond Category 2 because they resulted in 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). In accordance with SAR Section 1.7.5, events resulting in a frequency greater than 2×10^{-1} were to be screened for possible inclusion as a Category 1 event. (b)(5)

(b)(5)

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}.

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}. In accordance with SAR Section 1.7.5, events resulting in a frequency greater than 2.0×10^{-1} were to be screened for possible inclusion as a Category 1 event. (b)(5)

(b)(5)

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 the applicant'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. (b)(5)

(b)(5)

The fire-initiated events across the intrasite yielded identical failure probabilities for seven waste forms. (b)(5)

(b)(5)

(b)(5) Two event sequences (ISO09–UCSNF–SEQ3–RRU and ISO07–LLW–SEQ2–RRU) were identified as Category 2 events and were provided with consequence analyses. (b)(5)

(b)(5)

The fire-initiated event at the LLWF resulted in a single Category 2 event for the self-contained event sequence, as summarized in SAR Table 1.7-15. The highest number of occurrences of an end state was 7×10^{-2} for an unfiltered radionuclide release (ISO07–LLW– SEQ2–RRU).

The development of fire-initiated event sequences in the subsurface facilities resulted in one event sequence for the facilities that produced one Category 2 event (direct exposure due to shielding loss) and one beyond Category 2 event (unfiltered radionuclide release) for waste packages. The event sequence categorized as beyond Category 2 had a frequency of 3×10^{-6} and was correctly categorized as beyond Category 2.

Fire Event Sequence Categorization

The applicant 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, the applicant 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 the applicant's fire event sequence quantification and categorization using the guidance in YMRP Section 2.1.1.4. (b)(5)

(b)(5)

The NRC staff finds several fire event sequences (ESD20–TAD– SEQ2–DE, ESD12–TAD– SEQ2–DEL, and ESD31–TAD–SEQ2–DEL) had a frequency (2×10^{-1}) that approached the applicant's screening threshold for Category 1 events. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.4.4 Evaluation Findings

The NRC staff reviewed and evaluated the applicant's SAR and other supporting information related to the identification and categorization of event sequences in the GROA and (b)(5)

(b)(5)

(b)(5)

(b)(5)

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CHAPTER 5

2.1.1.5 Consequence Analysis

2.1.1.5.1 Introduction

This Safety Evaluation Report (SER) chapter provides the U.S. Nuclear Regulatory Commission (NRC) staff review of the applicant's consequence analysis to evaluate whether the repository design can meet 10 CFR Parts 20 and 63 radiation protection requirements. The objective of the review is to verify that the applicant adequately 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 the applicant'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. The applicant 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 the applicant'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 Regulatory Requirements and Regulatory Guidance

The regulatory requirements of 10 CFR 63.111(a), (b), and (c) specify the dose objectives for normal operations and Category 1 and Category 2 event sequences. Specifically, 10 CFR 63.111(a)(1) provides the preclosure performance objectives for the GROA of not exceeding the radiation exposure limits in 10 CFR Part 20. 10 CFR 63.111(a)(2) provides the preclosure performance objectives for the GROA, during normal operations and for Category 1 event sequences, in that the annual dose to any real member of the public, located beyond the boundary of the site, may not exceed 0.15 mSv [15 mrem].

10 CFR 63.111(b)(1) requires that the GROA be designed such that during normal operations and Category 1 event sequences, radiation exposures, radiation levels, and releases of radioactive materials will be maintained within the limits of 10 CFR 63.111(a). 10 CFR 63.111(b)(2) requires that the GROA be designed such that taking into consideration Category 2 event sequences, no individual located on or beyond any point on the site boundary will receive the more limiting of a total effective dose equivalent (TEDE) of 0.05 Sv [5 rem] or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent will not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin will not exceed 0.5 Sv [50 rem].

10 CFR 63.111(c)(1) requires the PCSA to meet the requirements specified at 10 CFR 63.112 and demonstrate that the radiation protection limits of 10 CFR Part 20 will be met. 10 CFR 63.111(c)(2) requires that the PCSA meets the requirements specified at 10 CFR 63.112 and demonstrates that the numerical guides for design objectives will be

met for GROA normal operations and Category 1 and Category 2 event sequences. Normal operations are considered those planned, routine activities in which monitored exposures are expected from the HLW process at the GROA. Category 1 event sequences include 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.

10 CFR 63.204 requires that, during the normal operation, the public in the general environment as defined in 10 CFR 63.202 will not receive an annual dose exceeding 0.15 mSv [15 mrem] from combined management and storage (defined in 40 CFR 191.2) and storage (defined in 10 CFR 63.202) of radioactive material inside the Yucca Mountain repository.

The guidance applicable to the review of consequence analysis is contained in the Yucca Mountain Review Plan (YMRP) Section 2.1.1.5 (NRC, 2003aa). 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

In SAR Section 1.8, the applicant stated that to demonstrate compliance with the regulatory requirements of 10 CFR 63.111 and 10 CFR 63.112, it addressed the acceptance criteria in YMRP Section 2.1.1.5. The NRC staff evaluation focuses on determining whether the applicant (i) used acceptable methodology and appropriate input parameters for dose calculation, (ii) used source terms consistent with those described in SAR Section 1.5, (iii) appropriately used the methodology for the worker and public dose determination, and (iv) demonstrated compliance with 10 CFR 63.111(a), 63.111(b), 63.111(c), and 63.204.

The applicant 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, the applicant defined an onsite member of the public (SAR Section 1.8.1) as any individual not receiving an occupational dose in performing duties. This included construction workers, delivery personnel, and public visitors within the preclosure controlled area. These individuals are subject to the dose limits in 10 CFR 20.1301. The Cind-R-Lite mining lease is located southwest of the surface facility GROA, within the preclosure controlled area, but near the site boundary. The mining personnel who periodically access this area are considered to be onsite members of the public.

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 that allows public access and is subject to the 0.15 mSv/year [15 mrem/year] limit in 10 CFR 63.111(a)(2) for normal operations and 0.05 Sv [5 rem] exposure for any single Category 2 event sequence per 10 CFR 63.111(b)(2). For the areas north and east of the site boundary, which are areas controlled by the Nevada Test Site and the Nevada Test and Training Range, access by general members of the public is restricted. These areas are referred to as "offsite but not in the general environment." Individuals working in these areas are subject to the 1 mSv/year [100 mrem/year] public dose limit in 10 CFR 20.1301 for normal operations and 0.05 Sv [5 rem] for any single Category 2 event sequence as required in 10 CFR 63.111(b)(2).

2.1.1.5.3.1 Dose Calculation Methodology and Input Parameter Selection

To demonstrate compliance with 10 CFR 63.111(a) and 63.111(b), the applicant discussed the methodology used to calculate dose consequences to site workers and members of the public, both onsite and offsite (SAR Section 1.8). The applicant considered the radiological doses to workers and the public for normal operations, off-normal events, and Category 2 event sequences for the GROA activities. The applicant did not identify any Category 1 event sequence that required evaluation of dose consequences to workers or the public.

The NRC staff review and evaluation of the applicant's dose calculation methodology and input parameter selection includes (i) dose calculation methodology, (ii) atmospheric dispersion determination, and (iii) assumptions and input parameter selection, as discussed next.

Dose Calculation Methodology: The applicant 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). The applicant estimated direct radiation dose rates at various distances from the GROA facilities, including the rail and truck casks at the buffer areas, using the MCNP computer program (Briesmeister, 1997aa). MCNP is an industry standard Monte-Carlo transport computer code that simulates particle transport through a three-dimensional modeling of the nuclear system. Direct radiation dose rates from the aging pads were calculated using the MCNP and the SCALE (Oak Ridge National Laboratory, 2000aa) computer codes. SCALE is an industry standard modular code system developed for NRC for performing standardized computer analyses for licensing evaluation. In addition, the applicant used SCALE to calculate doses from the aging pad assuming transportation, aging, and disposal (TAD) canisters were loaded with design basis fuel out to a distance of 1 km [0.6 mi]. Both MCNP and SCALE considered 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, the applicant indicated that direct exposure from the waste handling facilities was minor compared to the 2.5 μSv/hour [0.25 mrem/hour] shielding design limit. The onsite public doses were dominated by the transportation cask rail and truck buffer areas and the aging overpacks stored at the aging pads. The applicant indicated 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.

The inhalation dose resulting from the normal operational releases was determined 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, the external dose was determined 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. The TEDE for workers and the onsite public was determined

by summing the external dose due to direct radiation, the external dose due to submersion, and the internal dose due to inhalation. There are no agricultural activities in the onsite area and, thus, the applicant did not include any dose due to ingestion of foodstuffs in the calculations.

To calculate the airborne exposures to the offsite public, the applicant 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. Releases were assumed to be at ground level for the dose calculations at the site boundary.

For assessment of internal exposures, the methods proposed in EPA (1988aa, 1993aa, 1999aa) were used. The applicant used the GENII Gaussian statistical model, which implemented dosimetry models recommended by the ICRP and related guidance in EPA (1988aa, 1993aa, 1999aa). The applicant applied both deterministic and stochastic approaches to model the impacts of GROA operations. A majority of the calculations used a combination of deterministic bounding values coupled with stochastic parameters characterized by a mean value and distribution. The deterministic dose calculation methods used receptor characteristics that bound those of any real member of the public or a worker. Stochastic dose calculation methods used mean values and distributions for parameters including receptor-related parameters. Using sensitivity and uncertainty analysis techniques, the applicant determined a dose distribution. The dose to a real member of the public was then determined by using the maximum value obtained from the calculated dose distributions.

NRC Staff Evaluation: The NRC staff reviewed the applicant's dose calculation methodology using the guidance in YMRP Section 2.1.1.5. (b)(5)

(b)(5)

(b)(5)

Atmospheric Dispersion Determination: The applicant estimated airborne doses using the annual average onsite atmospheric dispersion coefficients (X/Q). To calculate the annual average onsite X/Q, the applicant used the guidance in Regulatory Guide 1.194 (NRC, 2003ah) and performed calculations using the ARCON96 (NRC, 1977ab) computer code. The applicant estimated airborne doses onsite and normal exposure to the offsite public using the annual average onsite X/Q values from a meteorological monitoring station located approximately 1 km [0.6 mi] south-southwest of the North Portal. Data used in the dose calculations were based on the period from January 1, 2001, through December 31, 2005. The applicant calculated X/Q values for the 16 meteorological sectors based on Regulatory Guide 1.111 (NRC, 1977ac) for annual releases and Regulatory Guide 1.145 (NRC, 1982aa) for hourly X/Q values for use with the Category 2 event sequences. The annual average and 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 the applicant's atmospheric dispersion determination using the guidance in YMRP Section 2.1.1.5. (b)(5)
(b)(5)

Assumptions and Input Parameter Selection: The applicant provided information concerning the assumptions and basis for the selection of models, source terms, exposure pathways, and dose coefficients used in calculating the radiological exposures (SAR Sections 1.8.1 and 1.8.2).

To estimate dose contributions from surface contamination, the applicant assumed that the entire external surface area of each transportation cask was contaminated at the regulatory limit [49 CFR 173.443(a), Table 9]. Contamination levels on the casks were assumed to be the 49 CFR 173.443(a) Table 9 limit for shipping of 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. For direct radiation exposures to individuals outside of a facility, the applicant established a limit of 2.5 μSv/hour [0.25 mrem/hour] as a shielding design limit for the various waste handling facilities.
The applicant indicated that the calculation results showed that resuspension of surface contamination was an insignificant contributor to the calculated total annual radiation worker dose. Time-motion calculations versus expected dose rates during the work activities were predicted and summed to generate an overall worker direct radiation dose. Added to this dose were 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).

Breathing rates for calculating doses from normal operations were those provided in Regulatory Guide 1.109 (NRC, 1977ad), and rates recommended in Regulatory Guide 1.183 (NRC, 2000ag) were employed for accidental releases. Dose conversion factors were based on information presented in EPA (1988aa, 1993aa, 1999aa). As discussed previously, the applicant performed the deterministic dose calculation using the receptor characteristics that bound those of any real member of the public or a worker. Individual parameter values were selected at the 95th percentile level for receptor-related parameters including food consumption rates and periods, and external and inhalation exposure times. The applicant indicated that use of 95th 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, the applicant used model inputs that were representative of Amargosa Valley. Site-specific parameters chosen and assumptions employed were discussed in SAR Section 1.8.1.4.4 and more fully in BSC (2007cm). Site-specific parameters included time typically spent outdoors versus indoors, local weather, soil parameters, use of local feed stock for livestock and poultry, and consideration of human consumption of local foods. Consumption rates incorporated data collected during a 1977 survey of Amargosa Valley residents. Contingent average daily intake values by gender and food group from national data coupled with an estimate of days per year when locally produced food is consumed provided another estimate for site-specific consumption rates of potentially contaminated foods. To assign daily exposure times, the applicant used four population groups: nonworkers, commuters, and local indoor and outdoor workers. The percentages of people assigned to each group were derived from the 2000 census data.

NRC Staff Evaluation: The NRC staff reviewed SAR Sections 1.8.1 and 1.8.2 and supporting documentation using the guidance in YMRP Section 2.1.1.5. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.5.3.2 Source Term Evaluation

To demonstrate compliance with the requirements of 10 CFR 63.21(c)(4), 63.111(a), and 63.111(c), the applicant described the kind, amount, and specifications of the radioactive material to be received and possessed at the GROA. For conducting its PCSA, the applicant assumed that the GROA operations would be carried out at the maximum capacity and rate of receipt of radioactive waste (SAR Sections 1.5, 1.8.1, 1.8.2, and 1.10). Source terms analyzed for normal operations included commercial spent nuclear fuel (CSNF), naval SNF, U.S. Department of Energy (DOE) SNF including a small amount of CSNF in the applicant's possession, and vitrified DOE HLW. The waste stream scenarios for CSNF were assumed to be 5 years old with an upper limit of 25 kW heat load. The applicant indicated that this assumption for the CSNF, when based on the earliest projected fuel receipt date (2017), is conservative when considering the current industry inventory of CSNF that will be available for disposal.

In SAR Section 1.8.1, the applicant discussed the source term released inputs, the material at risk, the damage ratio for fuel releases, the release and respirable fractions, and the leak path factors (LPFs).

The NRC staff review and evaluation of the applicant's source term identification include (i) source term for dose calculation and (ii) cladding damage and leak path factor assumptions, as discussed next.

Source Term for Dose Calculation: In SAR Section 1.8.2, the applicant provided potential releases and direct radiation source terms during normal surface and subsurface operations and Category 1 and Category 2 event sequences that could lead to radiological consequences. In SAR Section 1.10, the applicant discussed gamma and neutron sources for CSNF, naval SNF, DOE SNF, and DOE HLW, including gamma and neutron energy spectra. SAR Section 1.8.2.2 discussed the potential surface and subsurface operations that could lead to radiological doses to the public and radiation workers. The discussion identified the types of exposure that could be expected from the various facilities. A broad range of operational activities were evaluated including potential radiological exposures during cask handling, repackaging of CSNF, receipt and transfer operations, storage of the casks at the aging pads, and storage of the waste packages in the emplacement drifts. 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.

The applicant assumed that DOE SNF (including naval SNF), HLW, and approximately 90 percent of the CSNF are received in sealed canisters inside transportation casks. The remaining 10 percent of the CSNF would be received in either dual-purpose canisters or as bare, intact assemblies in rail or truck transportation casks. The various waste forms are removed from the transportation vehicles and handled in the initial handling facility, canister receipt and closure facility, 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.

The applicant evaluated the potential releases from normal operations using representative pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assembly radionuclide inventories (SAR Section 1.8.1.3). For releases from Category 1 and Category 2 event sequences, the applicant used maximum assembly inventories discussed in SAR Section 1.8.1.3. When developing the source terms, the applicant evaluated the onsite, ongoing work activity to determine the maximum available radioactive material that could contribute to the worker and public dose. For example, exposures from the casks temporarily located at the rail and truck buffer areas considered the maximum number of casks (5 trucks and 25 rails) that would be present at any time. For the aging pad, 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.

The applicant used SCALE/ORIGEN-S to estimate radionuclide inventories and neutron/gamma source terms for various burnups and initial enrichments of CSNF (SAR Section 1.5). In a response to the NRC staff RAI (DOE, 2009ep), the applicant compared the SCALE/ORIGEN-S-calculated concentrations of the dose-significant radionuclides in high-burnup PWR and BWR SNF and in event sequence consequence analyses to the experimental data presented in published papers and NUREG/CR–6798

(Sanders and Gauld, 2003aa). The applicant discussed the effect of the conservatism of the parameters of maximum (bounding) CSNF (SAR Section 1.5) on gamma and neutron sources for shielding analyses.

NRC Staff Evaluation: The NRC staff reviewed the calculations of the source term using the guidance in YMRP Section 2.1.1.5, Review Method 2. (b)(5)

(b)(5)

(b)(5)

(b)(5)

Cladding Damage and Leak Path Factor Assumptions: In assessing dose consequences, the applicant made the following assumptions on cladding damage and LPF: (i) a damage ratio of 1.0 for Category 1 and Category 2 event sequences for CSNF and HLW and Category 2 seismic and fire event sequences; (ii) a damage ratio of 0.01 for normal operations and Category 1 and Category 2 event sequences involving CSNF but not resulting in cladding damage; (iii) an LPF of 0.0 for transportation casks and canisters designed and tested to be leak tight (SAR Section 1.8.1.3.6); (iv) an LPF of 0.1 for waste packages (SAR Section 1.8.1.3.6); (v) an LPF of 1.0 with no credit taken for depletion of particulates released inside the buildings; (vi) an LPF of 0.01 per stage, which resulted in a 10^{-4} two-stage combined HEPA LPF; (vii) a 10-µm [3.9×10^{-5}-in] Aaerodynamic Eequivalent Ddiameter waste form respirable fraction; (viii) an LPF of 1.0 (i.e., no filtration) (SAR Section 1.8.1.3) for HEPA filtration for Category 1 and Category 2 event sequences when HEPA filters are unavailable; and (ix) release fraction and respirable fraction of 0.01 and 1.0, respectively, corresponding to unenclosed filter media, which are higher than values for closed filter media. In addition, the applicant discussed its assumptions on (i) cladding burst release fractions and respirable fractions and oxidation release fractions and respirable fractions from CSNF during normal operations or a Category 1 or Category 2 event sequence for SNF in a dry environment, (ii) fuel fines and volatiles, and (iii) low-burnup and high-burnup SNF.

The applicant provided CSNF in-pool release fractions for drop or impact events in the 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, the applicant discussed 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].

The applicant described the airborne release fraction and respirable fraction for the radioactivity from the combustible portion of the low-level waste facility inventory as the source for release for a fire event (SAR Section 1.8.1.3.5). The applicant used specific release fractions and respirable fractions for the dry active waste in drums, 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, the applicant assumed a respirable fraction of 1.0. The applicant used bounding values of measured respirable fractions and airborne release fractions for uncontained waste.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of the waste form characteristics and evaluation of the potential releases from normal operations and Category 1 and Category 2 event sequences using the guidance in YMRP Section 2.1.1.5, Review Method 2. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.5.3.3 Public Dose Calculation

To demonstrate compliance with the performance objectives and dose requirements in 10 CFR 63.111(a), 63.111(b), 63.204, 20.1101, and 20.1301, the applicant performed calculations for members of the public for both the onsite and offsite area. These calculations included both normal operations and Category 2 event sequences. As stated in SAR Section 1.8.3.2 1.7.5, no Category 1 event sequences were identified that required analysis for public dose. The applicant identified several areas, both onsite and offsite, for determining the public dose.

Public exposure may occur from either direct radiation or from airborne releases. Exposure sources included the release of radioactive gases; volatile species and particulates from surface and subsurface facility operations; and direct exposure from contained radioactive sources within transportation casks, aging overpacks, and surface facilities and buildings. Radiological exposures from background radiation and offsite transportation were not included in the public dose calculations.

The NRC staff review and evaluation of the applicant's public dose calculation includes (i) features limiting onsite public exposures, (ii) onsite members of the public dose calculation, (iii) features limiting offsite public exposures, and (iv) offsite members of the public dose calculation, as presented next.

Features Limiting Onsite Public Exposures: Potential public exposures within the surface facility GROA due to waste handling activities could occur at several locations due to both direct radiation and airborne radiation. To limit the general public's exposure to direct radiation while onsite at the GROA, the applicant established a restricted area within the surface facility GROA where radioactive material is handled and stored. The restricted area includes the fenced protected area that encompasses the truck and train buffer area (Areas 33A and 33B), the waste handling facilities, the aging pad, and the North Portal entrance. Casks arriving onsite are moved into the restricted area and temporarily stored at the rail and truck buffer areas. From there, the casks are moved to the waste handling buildings to be placed in acceptable canister configurations for storage. In some cases, this requires cutting the cask open and repackaging the fuel. The fuel in the waste package is then moved into its assigned emplacement drift for disposal or ~~are~~ is temporarily placed in suitable casks at the aging pad for aging. Throughout this process, exposures will occur that could affect the onsite public; in particular, the construction workers completing work on other portions of the site. These construction workers were assumed to be onsite 2,000 hours/year as opposed to the transient public, such as delivery personnel.

To reduce the exposure to the onsite public while the casks are inside the waste handling facilities, the applicant established a maximum 0.0025 mSv/hour [0.25 mrem/hour] dose limit for the exterior of the buildings at the personnel level as specified in SAR Table 1.10-2. 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. Shielding requirements were established using the point of maximum or peak radiation dose. Therefore, the overall general area radiation levels would be less than this maximum calculated dose. Dose calculations were performed for the onsite handling facilities using the MCNP code.

The applicant's shielding calculations were presented in a number of SAR sections, including Sections 1.10.3 and 1.8.4.1.3. Concrete required for shielding of personnel associated with the waste handling facilities will be designed to American Concrete Institute code requirements and site seismic criteria. The WHF, where fuel assemblies will be transferred into TAD canisters, are designed with an in-ground steel-lined concrete pool. SAR Section 1.7.2.3 discussed

degradation or loss of shielding for several types of failures. In addition to the structural aspects designed into the buildings, should any shielding or protective systems be lost during an event, the applicant's emergency plan includes provisions for warning site personnel and evacuating personnel to safe areas (SAR Section 5.7.2.2.3). Distance attenuation between the waste handling facilities and the various onsite public areas further reduces the dose rates. The applicant indicated that the restricted area fence is more than 200 m [656 ft] from the waste handling facilities resulting in more than 3,000 times reduction in the dose rate due to distance. By providing shielding and establishing a dose rate limit at the exterior of the waste handling facilities, the dose contribution to the onsite public becomes negligible due to work activities underway within these facilities.

NRC Staff Evaluation: The NRC staff reviewed the facility features used to limit onsite public dose using the guidance in YMRP Section 2.1.1.5 (b)(5)
(b)(5)

Onsite Members of the Public Dose Calculation: For the onsite areas, the Yucca Mountain site consists of a restricted area, protected area, GROA, and controlled area. SAR Section 1.1.1.1 described these areas with their visual representation in SAR Figures 1.1-1, 1.1-2, 1.2.1-2, 1.8-2 and 5.8-2.

SAR Tables 1.8-1 and 1.8-36 and Figure 1.8-2 listed radiation dose limits that apply to the public in the areas within and outside the preclosure controlled area. As discussed previously, the onsite members of the public included construction workers, delivery personnel, public visitors, and mining personnel of the Cind-R-Lite mine within the preclosure controlled area. These onsite members of the public are assumed to be present 2,000 hours per year. Radiological exposures to these members of the public would be bounded by the calculated doses to the construction workers in the surface facility GROA.

For direct radiation during normal operations, the applicant used design limits and regulatory limits for the source terms to assess public doses. This included a 0.0025 mSv/hour [0.25 mrem/hour] exterior building design limit, a 0.4 mSv/hour [40 mrem/hour] design limit on the surface of the aging cask, and the 10 CFR 71.47 dose rate limits for transportation casks. On the basis of these design and regulatory limits, the applicant indicated that the direct radiation dose to the onsite public from the waste handling facilities becomes negligible compared to the direct radiation dose from the aging pad and the rail and truck buffer area.

As discussed in SAR Section 1.8.3.1.3, the dose rates for the transportation casks were assumed to be bounded by the limits in 10 CFR 71.47. These limits are 2 mSv/hour [200 mrem/hour] at any point on the cask exterior surface and 0.1 mSv/hour [10 mrem/hour] at 2 m [6.6 ft] from the cask surface. The applicant used cask dose rates based on calculations from cask safety analysis reports submitted to NRC and calculations the applicant performed to develop dose rate values versus distance that would be below the 10 CFR 71.47 limits. The Transnuclear TN-32 cask and the British Nuclear Fuels TS-125 cask with a W21 canister were used as models. The TN-32 cask holds 32 PWR and the TS-125/W21 cask holds 21 PWR fuel assemblies. Dose calculations at various distances were performed 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 to move the SNF into an acceptable storage canister. 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 exposures and the 1.0 mSv/year [100 mrem/year] limits of 10 CFR 20.1301, even when combining the airborne source terms from the handling facilities and the subsurface exhaust shafts.

The direct exposures resulting from the casks at the aging facility 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. 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.

Normal subsurface radiological releases were considered 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.

The highest estimated doses to the public were determined to be at the lower muck yard (Area 780) and the warehouse and nonnuclear RF (Area 230) due to the casks temporarily stored at the rail and truck buffer areas (Areas 33A and 33B). The maximum dose rate limits for each cask were the 10 CFR 71.47 shipping limits. These limits were discussed in SAR Section 1.8.3.1.3. When considering the maximum number of trucks and rail casks that could be present at the buffer area, the applicant indicated that the dose rates at this location become the predominant dose contributor to the onsite public. The maximum calculated dose to the public was determined to be 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, the applicant plans to 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 the applicant'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 YMRP Section 2.1.1.5 (b)(5)

(b)(5)

Features Limiting Offsite Public Exposures: Buildings that may handle fuel assemblies or are involved with the cutting open of the canisters have HEPA filtration systems to reduce radioactive particulate releases. In addition, the applicant incorporated the operational constraint established in 10 CFR 20.1101(d) for air emissions of 0.1 mSv/year [10 mrem/year] to any individual member of the public into its Operational Radiation Protection Program.

The aging facility incorporates features to reduce exposures to workers and the public (SAR Section 1.2.7.6.5), which include installing shield walls on the horizontal storage modules, locating the aging pads away from other facilities, establishing a posted restricted area around the aging pads to warn personnel of radiation, and controlling access onto the aging pads by use of a security fence.

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 can establish permanent residence is approximately 18.5 km [11.5 mi] south. Currently, the nearest real member of the public lives 22.4 km [13.9 mi] from the surface facility GROA boundary. At these distances, the applicant indicated that radiation from normal site operations would not be distinguishable from normal background levels, even with the facilities operating at maximum capacity. For residents in the Amargosa Valley, average annual dose from cosmic, cosmogenic, and terrestrial radiation is 0.96 mSv/year [96 mrem/year], as outlined in DOE 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 YMRP Section 2.1.1.5. (b)(5)
(b)(5)

Offsite Members of the Public Dose Calculation: For the preclosure period, 10 CFR 63.111(a)(2) requires the applicant to determine the dose to a real member of the public. The applicant performed a series of calculations to determine distances from the surface facility GROA and the subsurface exhaust shafts to the site boundary. The closest real 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 real member of the public is 21.5 km [13.3 mi] toward the south-southeast wind sector (BSC, 2007bp).

The applicant did not use the distance to the real member of the public for the dose calculations, but instead conservatively determined the dose to a hypothetical member of the public located at the site boundary, much closer than real members of the public. To determine X/Q values from the surface facility GROA boundary, the applicant 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) the applicant used 11,000 m [6.8 mi]. The calculations for the eight sectors determined that due to the predominant wind patterns for the site, the south-southeast wind direction resulted in the highest X/Q values. Wind patterns for the site were shown in SAR Figures 1.1-14 through 1.1-51. The south-southeast X/Q value for the offsite public in the general environment (SAR Table 1.8-12) was selected for the dose calculations for the maximum exposure to a hypothetical member of the public.

For the distances from the subsurface exhaust shafts to the site boundary within the general environment, the applicant used the closest exhaust shaft to perform the X/Q calculations. The most conservative X/Q value was in the southeast wind direction (from northwest). The subsurface exhaust shaft X/Q values are similar in magnitude to the surface facility GROA values.

For the site boundary in the north and east direction toward the Nevada Test Site and the Nevada Test and Training Range, SAR Table 1.8-10 listed the distances used to perform X/Q calculations. The highest X/Q value was in the southeast wind direction. The southeast wind direction intersects the southernmost corner of the Nevada Test Site and was used for calculating doses to the Nevada Test Site 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 were considered for normal operations included (i) fission product gases, volatile species, and fuel fines and crud particulates released from the waste handling facility, such as during opening and handling of a canister, that are not removed by the HEPA filters, (ii) neutron activation of the air and silica dust inside the emplacement drifts that could become airborne; and (iii) resuspension of radioactive contamination on the canisters contained in the aging overpacks. The calculated doses from these three source terms were added together 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 Nevada Test Site and the Nevada Test and Training Range were calculated as 0.0011 mSv/year [0.11 mrem/year]. The applicant indicated that these values were well below the dose limits of 0.15 and 1.0 mSv/year [15 and 100 mrem/year] that apply to these offsite areas.

The property north and east of the Yucca Mountain site boundary controlled by the Nevada Test Site and the Nevada Test and Training Range was evaluated against the 10 CFR 20.1301 limits for individual members of the public because these are U.S Government-controlled areas that restrict the presence of the general public. The area south and west of the Yucca Mountain site boundary was evaluated against 10 CFR 63.111(a)(2) and 63.204 because these areas are open to the general public. The applicant determined that the direct radiation levels from source terms associated with waste handling operations at the surface facility GROA decreased by a factor of more than 13 orders of magnitude because of large distances from the offsite public to the surface facility GROA, resulting in insignificant offsite public dose from direct radiation and skyshine from normal operations.

For airborne dose calculations and determination of the X/Q values for the eight sectors, the applicant used the minimum distances from the surface facility GROA boundary to the site boundary, as shown in BSC 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, the applicant 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. Potential airborne release doses from inhalation, ingestion, resuspension inhalation, air submersion, and groundshine pathways were evaluated as a continuous release throughout the year. The applicant modeled the airborne releases as ground-level releases. Offsite public dose values presented in SAR Tables 1.8-28 and 1.8-32 included the sum of the releases from the WHF, the aging pads, and the subsurface exhaust shafts. For the offsite public in the general environment where food ingestion doses were evaluated, internal doses were calculated 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. These doses are calculated on the basis of airborne releases. The applicant did not include dose contributions from offsite transportation, because these dose contributions are not required per 40 CFR 191.01(a).

For Category 2 event sequences, only offsite doses to the public are required to be analyzed by 10 CFR 63.111(b)(2). The dose calculations were modeled 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, the applicant found 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, 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 YMRP Section 2.1.1.5 (b)(5)

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(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.5.3.4 Worker Dose Calculation

To demonstrate compliance with 10 CFR 63.111(a)(1) and 63.111(b)(1), the applicant calculated radiological doses to radiation workers as part of PCSA (SAR Section 1.8.4). The radiation worker dose is assessed for (i) compliance determinations with the dose limits of 10 CFR Parts 63 and 20 and (ii) important to safety structures, systems, and components determinations. Per 10 CFR 63.111(a)(1) and 63.111(b)(1), 10 CFR Part 20 dose limits for radiation workers apply to normal operations and Category 1 event sequences. Radiation worker safety assessments are not required for Category 2 event sequences. Because the applicant's PCSA indicated that there are no Category 1 event sequences (SAR Section 1.8.6), radiation worker safety assessments and mitigation of worker doses provided by the structures, systems, and components do not factor into the applicant's important to safety structures, systems, and components determination. The applicant credited the structures, systems, and components to prevent Category 1 event sequences. SAR Table 1.8-36 indicated that annual doses to radiation workers were estimated to be 26 percent or less {0.013 Sv [1.3 rem] or less} of the 0.05-Sv [5-rem] annual limit for the TEDE. 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 TEDE. Compared to other exposure pathways for workers (e.g., inhalation of airborne radioactive materials), worker external dose from emitted radiation

~~are~~ is significant with respect to preclosure safety and the NRC staff evaluation of the applicant's demonstration of compliance.

The applicant based total annual doses to radiation workers on four major sources: (i) direct radiation from normal operations within the facility, (ii) direct radiation from sealed sources located outside the facility, (iii) airborne releases of radioactive material from normal operations at surface and subsurface facilities, and (iv) Category 1 event sequences. The maximum potential total annual dose of 0.013 Sv [1.3 rem] the applicant calculated corresponds to a radiation worker in the RF (SAR Table 1.8-25). This calculated dose was dominated by direct radiation from normal operations in the RF for an annual process rate of 273 casks.

The applicant calculated radiation worker exposure for the following pathways: (i) direct irradiation inside facilities by contained sources therein, (ii) direct irradiation at outside receptor locations by casks in buffer or aging areas, and (iii) inhalation and air submersion at outside receptor locations due to estimated airborne releases from surface facilities, aging pads, and subsurface emplacement drifts. 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 TEDE for radiation workers.

The NRC staff review and evaluation of the applicant's worker dose calculation includes (i) direct radiation calculation, (ii) airborne releases of radioactive material, and (iii) aggregation of worker doses, as presented next.

Direct Radiation Calculation

Radiation worker doses the applicant estimated 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). The applicant's worker dose assessments for individual facilities culminated in the results in SAR Table 1.8-25. Facility throughput (i.e., amount of radioactive waste processed per year), number of work crews, time spent performing operational tasks, and dose rates from the radiation field at different work locations were factored into these external dose calculations (SAR Section 1.8.4.1.3). Estimated dose rates depend on the radiation emission rates from radioactive waste, direct radiation scaling factors the applicant used, credit taken for shielding materials and distances of workers to direct radiation source terms, and flux-to-dose conversion factors. The applicant assumed five work crews will be available to staff the three shifts of operations, as detailed in SAR Equation 1.8-26 and BSC Section 3.2.4 (2008bw).

The NRC staff organized its evaluation of the applicant's direct radiation calculations in the following four sections: (i) radiation emission rates, (ii) direct radiation scaling factors, (iii) credit for shielding materials and worker distances, and (iv) flux-to-dose conversion factors.

Radiation Emission Rates: The applicant performed shielding calculations with the MCNP computer program (SAR Section 1.8.4.1.3). The applicant used a maximum source term for establishing shielding design parameters and selected a design basis source term for calculating worker doses as shown in SER Section 2.1.1.5.3.2 (BSC, 2008cc). The applicant specified PWR fuel with a burnup of 60 GW-day per metric ton and cooling time of 10 years as the design basis source term for its worker dose assessments (SAR Table 1.10-19).

NRC Staff Evaluation: The NRC staff reviewed the information on radiation emission rates using the guidance in YMRP Section 2.1.1.5. (b)(5)

(b)(5)

Direct Radiation Scaling Factors: The applicant applied scaling factors for facility throughput and direct radiation source term to its worker dose calculations, as described in DOE Enclosure 1 (2009eo). The applicant applied the source-term scaling factor to adjust annual worker doses, initially calculated for irradiation by a maximum source term for the entire year, to an annual dose from direct radiation that is more representative of full-scale operations. Because the applicant used a design basis source term instead of a maximum source term for calculating annual worker doses, the applicant applied a dose reduction by a factor of 2.7, as detailed in DOE Enclosure 1, Section 1 (2009eo).

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on the scaling factor for the direct radiation source term, outlined in DOE Enclosure 1 (2009eo), using the guidance in YMRP Section 2.1.1.5 (b)(5)

(b)(5)

Credit for Shielding Materials and Worker Distances: Although additional credit could have been taken for shielding materials in aging overpacks and shielded transfer casks, the applicant used dose rate profiles for a TS125 rail transportation cask in the canister receipt and closure facility, as shown in BSC Table 3 (2007cl). 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). A similar approach was taken 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, the applicant 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). The applicant 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). The applicant's dose calculations for radiation workers depended on annual throughput estimates, which were based on five DOE HLW canisters and nine DOE SNF canisters per transportation cask, as detailed in DOE Enclosure 5, Section 1 (2009ek).

NRC Staff Evaluation: The NRC staff reviewed the technical bases and input data used in the applicant's worker dose analyses using the guidance in YMRP Section 2.1.1.5 (b)(5)
(b)(5)

Flux-to-Dose Conversion Factors: The applicant used the flux-to-dose conversion factors, provided in ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa), for converting neutron and gamma fluxes to dose rates, as described in BSC Section 6.1.2 (2008cc). The SCALE computer code used 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 used the conversion factors in ANSI/ANS-6.1.1–1977 (American Nuclear Society, 1977aa). ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991) includes internationally accepted data the ICRP proposed. As noted in the appendix to ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa), there is no significant difference between the 1991 and 1977 version of ANSI/ANS-6.1.1 in neutron fluence to dose conversion for calibration of detector systems and conversion factors for exposure to gamma rays, except in the case of low-energy bremsstralung or x-ray sources. However, a significant difference between the two versions in calculated values occurs for neutron energies in the range of 10 keV to 1 MeV because the 1991 version adopted the ICRP-defined effective dose equivalent for personnel exposures. The appendix to ANSI/ANS-6.1.1–1991 (American Nuclear Society, 1991aa) notes that for exposure to a highly shielded source having an average energy in the hundreds of keV range, the result obtained using the effective dose equivalent of the 1991 standard is approximately a factor of 2.7 less than the value obtained using the 1977 conversion factors.

NRC Staff Evaluation: The NRC staff reviewed the information on flux-to-dose conversion factors using the guidance in YMRP Section 2.1.1.5. (b)(5)
(b)(5)
(b)(5) In the 10 CFR Part 63 final rule , NRC stated that "... use of external dosimetry methods in existing federal radiation guidance, Federal Guidance Report No. 12 (EPA, 1993aa), in combination with the more current internal dosimetry methods consistent with 40 CFR Part 197, Appendix A, is an acceptable approach for calculating TEDE." EPA (1993aa) is consistent with the ICRP–26 dosimetry. (b)(5)
(b)(5)
(b)(5) The NRC staff

also reviewed the dose conversion factors the applicant used in the dose assessments for airborne releases of radioactive material, as outlined in SAR Section 1.8.1.4.1 and BSC Section 6.1.2 (2008al). (b)(5)

(b)(5)

(b)(5)

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. The applicant 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). The applicant 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. The applicant's dose assessment for workers operating the spent fuel transfer machine accounted for the presence of radioactive particulates in the pool water from damaged fuel assemblies. The applicant used industry data from operational experience to justify 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). The applicant 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, the applicant 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 the applicant's consequence analysis for radiation workers and onsite members of the public, as described in BSC Section 3.2.8 (2008ak). The applicant indicated that such exclusion is justifiable because the licensee would control onsite areas and monitor them for radionuclide contamination so that remedial actions could be taken. However, in BSC Appendix IV (2008ay), the applicant considered ground surface irradiation in the consequence analysis to constitute radiation worker exposure to an off-normal event, liquid low-level waste spill, because it represented a significant pathway in that calculation.

Airborne releases of radionuclides during normal operations in the WHF are expected (SAR Section 1.8.2.2.1). The applicant performed these atmospheric release and dispersion calculations for several outdoor locations (SAR Tables 1.8-13 and 1.8-14). Among the sources of airborne releases during normal operations, SAR Table 1.8-32 indicated that radiation workers located at the WHF would receive the greatest doses. SAR Section 1.8.2.2.1 described the applicant'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, this information provided a basis for consequence analyses from airborne radioactive material that originates inside the WHF and is subsequently released to the atmosphere during normal operations. The applicant indicated that gaseous releases from cooling or flushing a transportation cask or dual-purpose canister would be routed through HEPA filters before discharge to the atmosphere. The applicant 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 YMRP Section 2.1.1.5. (b)(5)

(b)(5)

Aggregation of Worker Doses

In SAR Section 1.8.4.2 and Table 1.8-25, the applicant aggregated the estimated dose contributions for the four major sources discussed previously to individual radiation workers. The applicant calculated doses for each contribution by assuming worker exposure for 2,000 hours/year (100 percent occupancy). Because the applicant's PCSA identified no Category 1 event sequences, contributions for Category 1 event sequences are zero (SAR Table 1.8-25). The applicant also assessed worker doses from off-normal events in BSC Appendices IV and V (2008ay) and found 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 the applicant's dose aggregation.

NRC Staff Evaluation: The NRC staff reviewed the information on aggregation of worker doses using the guidance in YMRP Section 2.1.1.5. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.5.3.5 Dose Compliance

To demonstrate compliance with 10 CFR 63.111(a), 63.111(b), 63.111(c), 20.1301, 20.1101(d), and 63.204, the applicant discussed potential public and worker dose consequences and compliance confirmation (SAR Section 1.8.6).

The NRC staff organized its evaluation of the applicant's dose compliance information in the following five sections: (i) interactions between hazard, event sequence, and consequence analyses; (ii) facility throughput; (iii) aggregation of annual doses; (iv) compliance with dose limits for normal operations and Category 1 event sequences; and (v) compliance with dose limits for Category 2 event sequences.

Interactions Among Hazard, Event Sequence, and Consequence Analyses: Because there are no Category 1 event sequences to aggregate, the applicant evaluated only the doses from normal operations for compliance with the preclosure performance objectives. SAR Section 1.8.6 summarized the applicant's analysis of potential public and worker dose consequences for normal operations and Category 2 event sequences. In SAR Table 1.8-36, the applicant listed the results of the public and worker dose consequences in comparison to the preclosure dose performance objectives of 10 CFR 63.111(b)(1) for offsite public exposure, onsite public exposure, and radiation worker exposure. SAR Tables 1.7-7 to 1.7-18 listed event sequences at various facilities for which consequence analyses were either performed by the applicant or not needed, as the applicant determined. The applicant 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 YMRP Section 2.1.1.5. (b)(5)

(b)(5)

(b)(5)

The NRC staff reviewed the credit taken for particulate filtration in the applicant's consequence analysis of three Category 2 event sequences that involved structural challenges to a transportation cask with uncanistered SNF, a dual-purpose canister, or a TAD canister. (b)(5)

(b)(5)

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. The applicant 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 the applicant used for calculating consequences to radiation workers using the guidance in YMRP Section 2.1.1.5

(b)(5)

Aggregation of Annual Doses: The applicant aggregated doses for members of the public and radiation workers (SAR Section 1.8.1.2) by including four major contributions discussed in SER Section 2.1.1.5.3.4. Because the applicant did not identify any Category 1 event sequences (SAR Section 1.8.6), contributions from Category 1 event sequences to aggregated doses are zero. In addition to doses for radiation workers, the applicant calculated doses from normal operations for different representative members of the public, including individuals such as onsite construction workers, other onsite persons, offsite persons in the general environment, and offsite persons not in the general environment. Compared to offsite persons, SAR Table 1.8-36 showed significantly higher TEDE estimates for onsite persons. For onsite members of the public, direct radiation doses provided the greatest contribution to aggregated annual doses for normal operation (SAR Table 1.8-28). These direct radiation doses could be received at onsite locations outside of the main operational facilities for waste handling and aging. The applicant calculated doses at these onsite locations by assuming exposure duration of 2,000 hours/year for direct radiation (SAR Table 1.8-28) and approximately 2,000 hours/year for exposure to airborne radioactive material releases (SAR Tables 1.8-16 and 1.8-19). For the greater source-to-receptor distances at offsite locations, contributions from direct radiation emitted from sealed containers at the GROA become negligible. In light of no Category 1 event

sequences, 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 the applicant's methodology to aggregate annual doses for normal operations and doses from Category 1 event sequences using the guidance in YMRP Section 2.1.1.5. (b)(5)

(b)(5)

(b)(5)

Compliance with Dose Limits for Normal Operations and Category 1 Event Sequences: The applicant selected the largest aggregated annual dose for demonstrating compliance with the dose limits (SAR Table 1.8-36). Because no Category 1 event sequences were identified in the applicant's PCSA, there were no dose contributions from Category 1 event sequences. Onsite public exposures during normal operations are attributed to direct radiation and skyshine. The applicant assumed an annual exposure duration of 2,000 hours for an onsite member of the public. The applicant determined that the contribution from airborne releases was insignificant. For workers who are members of the public having access to the restricted area, like construction workers, the applicant estimated a maximum dose of 0.098 mSv/year [9.8 mrem/year]. For public located outside the restricted area but still onsite, results were presented in SAR Table 1.8-28. The highest dose rates (SAR Table 1.8-36) were found to be 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, the applicant determined dose rates to be 0.0011 mSv/year [0.11 mrem/year] TEDE. For the offsite public in the general environment, the applicant determined that, due to the long distance to the closest location in the general environment, direct radiation and skyshine contributions to dose were negligible. The applicant discussed airborne exposures that result from normal operations in SAR Section 1.8.3.1.2. The applicant indicated that the estimated TEDE was a factor of 900 below the applicable regulatory limit. Internal doses were calculated on the basis of a 50-year dose commitment period. SAR Table 1.8-29 provided estimates of dose for members of the public in the general environment {0.0005 mSv/year [0.05 mrem/year] TEDE}. SAR Table 1.8-36 showed that the estimated TEDE was a factor of 300 below the regulatory dose limit. SER Section 2.1.1.5.3.3 describes the detailed technical review of the applicant's public dose assessments.

In SAR Table 1.8-36, the applicant presented consequence analysis results for radiation workers of 0.013 Sv/year [1.3 rem/year] TEDE. The applicant reported a maximum TEDE of 0.014 Sv/year [1.4 rem/year] and maximum shallow dose equivalent to the skin of 0.001 Sv/year [0.1 rem/year] for radiation workers in DOE Enclosure 2, Table 4 (2009el). The applicant

committed to update the license application (SAR Tables 1.8-25 and 1.8-36) with changes to the results for radiation workers in DOE Enclosures 1 and 2, Sections 2 and 3 (2009el). The applicant 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 the applicant's demonstration of compliance with the preclosure performance objectives and regulatory dose limits for workers and members of the public from normal operations and Category 1 event sequences using the guidance in YMRP Section 2.1.1.5. This evaluation included 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. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

Compliance with Dose Limits for Category 2 Event Sequences: The applicant performed public consequence analyses for Category 2 event sequences to demonstrate compliance with the preclosure requirements specified in 10 CFR 63.111(b)(2). As required under 10 CFR 63.111(b)(2), the GROA must be designed so that no individual located on, or beyond, any point on the boundary of the site should receive, as a result of the single Category 2 event sequence, the more limiting of the following: (i) a TEDE of 0.05 Sv [5 rem] or (ii) the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent may not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin may not exceed 0.5 Sv [50 rem]. For external exposure, consistent with the revised definition of TEDE in 10 CFR Part 20 (72 FR 68043), it is permissible to use the effective dose equivalent in place of the deep dose equivalent.

Because of the distances involved, direct radiation from a Category 2 event sequence was negligible. Fourteen bounding Category 2 event sequences were analyzed (SAR Tables 1.8-30 and 1.8-31) for airborne releases. The applicant calculated the highest TEDE for the offsite public in the general environment to be 0.0001 Sv [0.01 rem] and showed that organ, lens of the eye, and skin doses were all well below limits. For offsite exposure of the public not in the general environment, the highest TEDE was 0.0003 Sv [0.03 rem] for the organ dose; the applicant also showed that lens of the eye and skin doses were all well below limits. To demonstrate compliance with 10 CFR 63.111(b)(2), the applicant selected the highest calculated doses for TEDE, individual organ, lens of the eye, and skin from the set of Category 2 consequence analyses, as described in BSC Tables 67 and 68 (2008ay).

NRC Staff Evaluation: The NRC staff reviewed the applicant's demonstration of compliance for members of the public from a single Category 2 event sequence and concludes that the Category 2 consequence analyses do not underestimate dose using the guidance in YMRP Section 2.1.1.5. (b)(5)

(b)(5)

Commitment

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.5.4 Evaluation Findings

(b)(5)

(b)(5)

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 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant'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 assurance that high-level radioactive waste (HLW) operations will not exceed the performance requirements of 10 CFR 63.111(b)(1) for Category 1 event sequences; or prevent or mitigate Category 2 event sequences that could result in radiological exposures exceeding the dose values specified at 10 CFR 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site. 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 applicant's responses to NRC staff's requests for additional information (RAIs) (DOE, 2009dk,dq,fc,fm–fr).

To identify the SSCs as ITS, the applicant performed a preclosure safety analysis (PCSA). The applicant provided criteria used in ITS SSCs identification and listed SSCs ITS and associated nuclear safety design bases in SAR Section 1.9.1. The applicant considered requirements stipulated in 10 CFR 63.112(e)(1) through 63.112(e)(13), as detailed in SAR Sections 1.9.1.1 through 1.9.1.13. 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 Regulatory Requirements and Regulatory Guidance

The regulatory requirements for the PCSA to identify SSCs ITS, PSCs, and measures taken to ensure the availability of safety function are defined 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 guidance applicable to the NRC staff's evaluation of the applicant's SSCs ITS, safety controls, and measures to ensure availability of the safety systems is contained in

the Yucca Mountain Review Plan (YMRP) Section 2.1.1.6 (NRC, 2003aa). The acceptance criteria used in the NRC staff's review follow:

- The applicant has provided an adequate 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 the applicant identified to prevent event sequences or mitigate their effects are adequate.

The review guidance related to the risk significance categorization of SSCs ITS in YMRP Section 2.1.1.6 is not applicable, because the applicant 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 Staff Review and Analysis

The applicant stated that to demonstrate compliance with the regulatory requirements of 10 CFR 63.112(e), it used the acceptance criteria in YMRP Section 2.1.1.6. The NRC staff's evaluation focuses on determining whether the applicant's PCSA has adequately addressed the requirements in 10 CFR 63.112(e) to identify the ITS SSCs. The NRC staff's focus was to determine that the ITS SSCs the applicant relied upon to limit or prevent potential event sequences or mitigate their consequences and PSCs have been adequately identified. The NRC staff also reviewed the information provided in the SAR and supporting documents to ensure that the applicant's PCSA adequately 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 evaluation of the applicant's SSCs ITS, safety controls, and measures to ensure availability of the safety systems generally following the YMRP guidelines. The NRC staff's review of criticality control and prevention [10 CFR 63.112(e)(6)] and of the ability of the ITS criticality controls to perform the intended safety function [10 CFR 63.112(e)(8)] is performed together in the Safety Evaluation Report (SER) Section 2.1.1.6.3.2.6 to streamline the review. Compliance with 10 CFR 63.112(e)(9), consideration of fire detection and suppression, is evaluated in the review of ITS fire suppression systems to perform the intended safety functions [10 CFR 63.112(e)(8)] in SER Section 2.1.1.6.3.2.8.4. Compliance with 10 CFR 63.112(e)(11), consideration of timely and reliable emergency power, is evaluated along with the review of the ITS electrical power systems in SER Section 2.1.1.6.3.2.8.5. Compliance with 10 CFR 63.112(e)(12), consideration of redundant systems, is evaluated in subsections of SER 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. Use of these redundant systems enhances the NRC staff confidence that 10 CFR 63 112(e)(8) can be met.

The NRC staff also used the review results from SER Sections 2.1 1.3, 2.1.1.4, and 2.1 1.7, as appropriate, to support the review of and make regulatory decisions for this chapter

2.1.1.6.3.1 List of Structures, Systems, and Components Important to Safety and Safety Controls

To demonstrate compliance with 10 CFR 63.112(e), the applicant 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 SER section, and its review of the applicant's compliance with the specific requirements in 10 CFR 63.112(e)(1) through 63.112(e)(13) is presented in SER Section 2.1.1.6.3.2.

SSCs ITS Identification

The applicant 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.

The applicant used four criteria to identify SSCs ITS If one or more criteria were met, an SSC would be classified as ITS. Following identification criteria, the SSC must (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 YMRP Section 2.1.1.6, Criterion 1(1) to review the applicant's criteria to identify SSCs ITS. (b)(5)

(b)(5)

The NRC staff performed a verification of the selected SSCs ITS the applicant 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 acceptably to identify SSCs ITS. (b)(5)

(b)(5)

The applicant 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. (b)(5)

(b)(5)

Nuclear Safety Design Bases

The applicant developed the nuclear safety design basis requirements 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. The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(1) to review the applicant's nuclear safety design bases for the ITS SSCs. (b)(5)

(b)(5)

(b)(5)

Proposed Conditions of Construction Authorization

(b)(5)

NRC Staff Conclusion

(b)(5)

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 the applicant adequately 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 as required in 10 CFR 63.112(e)(1) through 63.112(e)(13).

Compliance with each regulatory requirement in 10 CFR 63.112(e) is evaluated in individual sections with some exceptions discussed in SER Section 2.1.1.6.3. The NRC staff also used the review results from SER 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

To demonstrate compliance with 10 CFR 63.112(e)(1), the applicant 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. The applicant 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. The applicant 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. The applicant's analyses show that the airborne concentrations from these releases are below regulatory limits.

According to SAR Section 1.9.1.1, radiation-monitoring systems that the applicant plans to 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 SER 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(a) and Regulatory Guide 8.8 (NRC, 1978ab) to review SAR Section 1.9.1.1. (b)(5)

(b)(5)

In addition, the radiation is monitored at the source point, manned operating stations, or other locations in the surface facilities. (b)(5)

(b)(5)

(b)(5) Evaluation of the HVAC systems the applicant proposed using to limit concentration of radioactive material in air is provided in SER Section 2.1.1.6.3.2.8.2.2.

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.2 Limiting Worker Exposure Time When Performing Work

To demonstrate compliance with 10 CFR 63.112(e)(2), the applicant discussed equipment and facility designs that are intended to limit the exposure time when performing work in radiation areas ~~work~~ to assess the potential that such equipment may be designated as ITS. The applicant provided this information in SAR Section 1.9.1.2. The applicant 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 the applicant's as low as is reasonably achievable (ALARA) principles and program. Therefore, these design features are non-ITS. SER Section 2.1.1.8 provides the NRC staff's review of the applicant's ALARA program.

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.6.3, Criterion 1(2)(b) to review SAR Section 1.9.1.2 to determine whether the applicant adequately considered the means to limit the required time to perform work in radiological areas in its PCSA. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.3 Shielding Protection

To demonstrate compliance with 10 CFR 63.112(e)(3), the applicant discussed equipment and facility designs that are intended to provide shielding protection to assess the potential that such equipment may be designated as ITS. The applicant 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. The applicant indicated that it relied on the shielding design for normal operations and

Category 1 and Category 2 event sequences. The applicant designated shielding as non-ITS, where the shielding is used exclusively to comply with 10 CFR Part 20 during normal operations. Although no Category 1 event sequences were identified in the applicant's PCSA, the applicant 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. The ITS shielding features include shield doors and slide gates in the Initial Handling Facility (IHF), CRCF, Receipt Facility (RF), and WHF. The applicant did not include shielding 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 the applicant's information in SAR Section 1.9.1.3 using the guidance in YMRP Section 2.1.1.6.3, Criterion 1(2)(c) (b)(5)
(b)(5) The NRC staff's evaluation was to determine whether the applicant adequately considered the need for shielding to limit worker exposure in its PCSA. (b)(5)
(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.4 Radioactive Contamination Dispersal Monitoring and Control

To determine compliance with 10 CFR 63.112(e)(4), the applicant 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. The applicant provided this information in SAR Section 1.9.1.4.

The applicant described how it plans to 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, the applicant designated radioactive contamination dispersal monitoring and control as non-ITS. The applicant described the capability to monitor radioactive effluents in SAR Section 1.4.2 2.

The applicant'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. The applicant 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

The applicant described the surface facility HVAC systems in SAR Sections 1.2.3.4, 1.2.4.4, 1.2.5.5, and 1.2.6.4 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(d). (b)(5)

(b)(5)

(b)(5) The NRC staff evaluates the adequacy of the HVAC systems controlling the dispersal of radioactive contamination in the surface facilities in SER Section 2.1.1.6.3.2.8.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.5 Access Control to High Radiation Areas and Airborne Radioactivity Areas

To demonstrate compliance with 10 CFR 63.112(e)(5), the applicant 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. The applicant provided this information in SAR Section 1.9.1.5.

The applicant stated that controlling personnel access to normally unoccupied high radiation areas, very high radiation areas, or airborne radioactivity areas is part of normal operations and is not relied upon for prevention or mitigation of Category 1 or Category 2 event sequences. Therefore, the applicant designated these design features as non-ITS. For those areas requiring periodic personnel access for waste handling operations where the radiation levels are subject to change as a result of Category 1 or Category 2 event sequences, the applicant 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, the applicant'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). The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(e) (b)(5) The NRC staff's evaluation was to determine whether the applicant adequately considered access control to high radiation areas and airborne radioactivity areas in its PCSA. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.6 Criticality Control and Prevention and Ability to Perform Safety Functions

To demonstrate compliance with the regulatory requirements of 10 CFR 63.112(e)(6) and 63.112(e)(8) and the dose limits of 63.111(b), the applicant presented its criticality safety program. The goal of the program is to prevent criticality during the preclosure period for normal conditions and Category 1 and Category 2 event sequences. The applicant described its criticality safety program in SAR Section 1.14 and BSC (2008ba).

The applicant provided information in response to NRC staff RAIs (DOE, 2009az) about the organization of the applicant's proposed criticality safety program. The use of the organizational structure to implement the program was described in SAR Section 5.3. ~~The applicant's~~In its response to an NRC staff RAI (DOE, 2009az), the applicant committed to revise SAR Figure 5.3-1 to show~~ed~~ 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).

The applicant stated that it followed the applicable portions of Regulatory Guide 3.71 (NRC, 2005ac), which endorses certain criticality-related industry standards. To demonstrate that the program is robust and is based upon accepted industry standard practices, the applicant listed the American National Standards Institute/American Nuclear Society (ANSI/ANS) standards used in SAR Section 1.14.3.1. For example, the applicant's criticality safety training will be developed in accordance with ANSI/ANS-8.20–1991 (American Nuclear Society, 1991ab). The applicant 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). The applicant also used other endorsed standards related to such things as moderator control, use of neutron absorbers, and validation of criticality safety analysis.

In BSC (2008ba), the applicant presented a detailed preclosure criticality safety evaluation, which is used to demonstrate criticality prevention. The applicant 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, the applicant performed criticality sensitivity calculations for the different fuel types and conditions. The applicant summarized its sensitivity studies in SAR Section 1.14.2.4.1.7.

Using bounding waste form characteristics, such as modeling 5 wt% enriched fresh fuel, and bounding reflection based upon its sensitivity analyses, the applicant determined that moderation is the primary criticality control parameter for commercial SNF. The applicant 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, the applicant screened out all initiating events important to criticality.

The applicant 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 U.S. Department of Energy (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. The applicant 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.

The applicant 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 the applicant's information using the guidance given in YMRP Section 2.1.1.6.3, Criteria 1(2)(f) and 1(2)(h) (b)(5) (b)(5) to determine whether the applicant adequately considered criticality control and prevention in its PCSA [10 CFR 63.112(e)(6)] and the related SSCs ITS could perform their intended safety functions [10 CFR 63.112(e)(8)]. (b)(5)

(b)(5)

(b)(5)

(b)(5) The NRC staff reviewed the applicant's commitment (DOE, 2009az) to determine whether the applicant'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 the applicant proposed to use for its surface

facility operations such as radiation protection. (b)(5)

(b)(5)

(b)(5)

(b)(5) In addition, the NRC staff reviewed the applicant's technical basis for identifying fixed neutron absorbers as non-ITS. (b)(5)

(b)(5)

Commitment

The applicant commits to revise SAR Figure 5.3-1 and text describing the radiation protection and the nuclear criticality safety functions as shown in DOE (2009az)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.7 Radiation Alarm System

To demonstrate compliance with 10 CFR 63.112(e)(7), the applicant discussed the radiation alarm system in SAR Section 1.9.1.7 and the radiation monitoring system in SAR Section 1.4.2.2. The 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, the applicant 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). A uninterruptible power supply (UPS) powers the radiation monitoring system. The applicant 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 YMRP Section 2.1.1.6, Criterion 1(2)(g) (b)(5)
(b)(5)
(b)(5) The NRC staff reviewed the information presented in SAR Sections 1.4.2.2 and 1.9.1.7. (b)(5)
(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.8 Ability of Structures, Systems, and Components Important to Safety to Perform Intended Safety Functions

The applicant indicated that it demonstrated 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 discussed in SAR Section 1.7. To provide additional assurance that the SSCs ITS will perform their safety functions to an appropriate level of reliability, the applicant plans to qualify these SSCs ITS for the range of environmental conditions anticipated at the time of functional demand. Additionally, the applicant plans to implement a monitoring program to detect operation deviations of SSCs ITS to permit appropriate corrective actions. The applicant 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 the applicant identified and listed in SAR Tables 1.9-2 through 1.9-8 and 1.9-10 to determine compliance of the applicant's PCSA with 10 CFR 63.112(e)(8). The conclusions drawn from this limited evaluation will be applied, with reasonable assurance, to other ITS SSCs because the applicant used the same reliability assessment approach described in SAR Section 1.7 to demonstrate 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

To demonstrate compliance with 10 CFR 63.112(e)(8), the applicant is required to demonstrate that ITS surface facility structures can perform their intended safety functions assuming the occurrence of event sequences. The applicant 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. The applicant also addressed the ability of ITS surface facility structures to perform their intended safety functions in SAR Section 1.9.1.8.

The ITS surface facility structures include the IHF, CRCF, WHF, and RF. The applicant 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. The applicant 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. The applicant sets the design basis straight wind at 193 km/hour [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^{-6}. 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, the applicant indicated that the mean annual probability of failure of a building collapse due to a spectrum of seismic events shall 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. The applicant assumed the preclosure period of 100 years and 50 years of operational period for the surface facilities involving handling SNF and HLW. 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, the applicant 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), the applicant evaluated the frequency of failure (per year) as 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). The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h). The NRC staff's assessment of the ability of the facility structures to perform the intended safety functions and meet the required nuclear safety design bases and criteria for wind, ash fall, and seismic events is based on the evaluation of the structural design in SER Section 2.1.1.7 and the structural performance in SER Section 2.1.1.4. In SAR Section 1.6.3.4.4, the applicant estimated the 10^{-6}/year 3-second gust straight wind to be 193 km [120 mph]. The maximum design tornado wind speed for structural facilities is 304 km [189 mph] (SAR Table 1.2.2-1). The applicant indicated that even though the design tornado wind speed exceeds the mean frequency 10^{-6}/year straight wind speed by a large margin, the damage probability will be below the screening probability of 10^{-6}/year. The NRC staff's evaluation of the probability of tornado effect on structural damage is given in SER Section 2.1.1.3.3.1.2.3.

(b)(5)

(b)(5) The applicant's justification for the design ash load is evaluated in SER Section 2.1.1.1.3.6 (b)(5)

(b)(5)

As discussed previously, the NRC staff's evaluation of the ITS facility structural design is provided in SER Section 2.1.1.7.3.1.1 and seismic performance or probability of failure requires to reviewing facility structure design and fragility parameters is provided in SER Section 2.1.1.4.3.3.1.2.2. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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

To comply with 10 CFR 63.112(e)(8), the applicant is required to demonstrate that the ITS mechanical handling systems can perform their intended safety functions if an event sequence occurs. The NRC staff reviewed five major groups of ITS mechanical handling systems [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

The five ITS mechanical handling systems are relied on to prevent or mitigate event sequences related to handling of commercial SNF assemblies, canisters, and casks. In general, all five systems must limit movement speed. The CTM, WPTT, and CHC also must have the ability to prevent spurious movement. The CTM, CHC, and SFTM must not collapse during a seismic event and must prevent a load drop. In addition, the WPTT must be designed to prevent tipover or rocking during a seismic event and rapid tilt down. Furthermore, 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 SER Section 2.1.1.7.3.2. (b)(5)

(b)(5)

The applicant stated that it will to 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 SER 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 acceptable radiation levels and temperature fluctuations the applicant specified~~used~~, provides the ITS mechanical handling systems with substantial design margin to perform their intended safety functions.

In several cases, the applicant 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, the applicant 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. The applicant 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), the applicant will use PSCs for the ITS mechanical handling systems to prevent event sequences or mitigate their effects. The applicant 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, the applicant included a corrective action program to document and evaluate equipment failures (SAR Section 1.13.2.6). The applicant 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, the applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h), the NRC staff reviewed the ability of the five mechanical handling systems to perform their intended safety functions if an event sequence occurs. (b)(5)

(b)(5)

(b)(5) The applicant's plan to develop reliability-centered maintenance programs for ITS SSCs is evaluated in SER Section 2.1.1.6.3.2.10.

(b)(5)

Important to Safety Controls Linking the Mechanical Handling Systems and Shield Doors

The applicant identified 29 key groups of ITS SSCs (SAR Table 1.4.2-1) that 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, the applicant indicated that all ITS controls will be made up of individual hardwired devices, instead of being driven by software or programmable devices. The applicant further indicated that 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.

The applicant plans to 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. (b)(5)

(b)(5)

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 provided in YMRP Section 2.1.1.6.3, Criterion 1(2)(h). (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.8.2.2 Heating, Ventilation, and Air Conditioning Systems Important to Safety

To demonstrate compliance with 10 CFR 63.112(e)(8), the applicant is required to demonstrate that the ITS HVAC SSCs can perform their intended safety functions assuming the occurrence of event sequences. The applicant provided information to demonstrate 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. The applicant 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. The applicant identified the nuclear safety design bases for the ITS HVAC SSCs in SAR Tables 1.9-3 (for the CRCF) and 1.9-4 (for the WHF) and in both of these tables for the ITS HVAC SSCs for the EDGF. The applicant relies on the ITS HVAC SSCs to provide (i) cooling to ITS electrical equipment and battery rooms or (ii) nuclear confinement.

Procedural Safety Controls

The applicant described PSCs in SAR Section 1.9.3, identifying that for HVAC systems, the PSCs may be included as part of normal operating procedures. The applicant identified PSC–7 for the ITS HVAC subsystems in the CRCF and WHF in SAR Table 1.9-10. In PSC–7, the applicant specified that one ITS HVAC train is required to be operating with the other one in standby before waste handling operations begin. The applicant also specified PSC–7 in SAR Sections 1.2.4.4.4 (for the CRCF) and 1.2.5.5.4 (for the WHF). The applicant 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. The applicant 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. The applicant 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, the applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h). (b)(5)
(b)(5)

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.

The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(a). (b)(
(b)(5)

(b)(5)

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. The applicant will minimize the spread of contamination by having filtration zones and by controlling the air flow from areas with low potential for contamination to higher potential for contamination. The applicant described the confinement zoning in SAR Table 1.2.2-13 and defined nonconfinement zones as noncontaminated (i.e., clean) areas, tertiary confinement zones as areas where airborne contamination is not expected during normal operations, and secondary confinement zones as areas with a potential for airborne contamination during normal operations. The applicant stated that this designation is in accordance with DOE-HDBK-1169–2003, as described in DOE Section 2.2.9 (2003ae).

In response to an NRC staff RAI (DOE, 2009fo), the applicant stated that confinement areas are designated as ITS if the area has an identified Category 2 event sequence in~~for~~ which a loaded canister may breach and an ITS HVAC system is used to mitigate the potential release. In addition, the applicant indicated that it does not rely on seals through walls and slabs to maintain confinement and committed in its response (DOE, 2009fo) to update SAR Section 1.9.1.10 to reflect this intent. Furthermore, the applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(d):

(b)(5)

(b)(5) Specifically, the NRC staff reviewed DOE–HDBK–1169–2003 Section 2.2.9 (DOE, 2003ae) (b)(5)

(b)(5)

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, the applicant 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). The applicant 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 the applicant'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, the applicant included redundancy within a train by specifying operating and standby components (e.g., operating and standby HEPA filter plenums and operating and standby exhaust fans). For example, the applicant showed this redundancy in SAR Figure 1.2.4-104 for the subsystem that 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(l). (b)(5)

(b)(5)

Means to Inspect, Test, and Maintain ITS HVAC SSCs

The applicant 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, the applicant identified independent trains and standby (or backup) components within individual trains. For example, in the CRCF, for the ITS HVAC subsystem serving ITS electrical equipment and battery rooms, SAR Figure 1.2.4-104 showed the system 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, the applicant did not show standby air handling units in SAR Figure 1.2.8-26. The applicant 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. The applicant 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. The applicant 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. The applicant 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 provided in YMRP Section 2.1.1.6.3, Criterion 1(2)(m). (b)(5)

(b)(5)

Ability of the ITS HVAC SSCs to Perform Their Intended Safety Functions

The applicant 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.

The nuclear safety design bases (including the safety functions) are evaluated in SER 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. The NRC staff evaluated the controlling parameters for the nuclear safety design bases in SER Section 2.1.1.4.3.3.2.1 to determine whether the applicant's fault tree models support the controlling parameters specified such that the safety function could be accomplished in the mission time the applicant specified.

The applicant identified certain component failures when quantifying the failure probability of a system or subsystem in its fault tree models. However, the applicant 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), the applicant stated that components are designated ITS in drawings, tables, and the text of the SAR but not specifically in fault trees. The applicant 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 this information using the guidance in YMRP Section 2.1.1.6.3, Criterion 1(2)(h) to assess the applicant's compliance with the requirements of 10 CFR 63.112(e)(8) (b)(5)

(b)(5)

Commitment

The applicant commits to update SAR Section 1.9.1.10 to indicate that it does not rely on seals through walls and slabs to maintain confinement.

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.8.3 Transportation Systems Important to Safety

To demonstrate compliance with 10 CFR 63.112(e)(8), the applicant 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.

TEV ITS

The applicant 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. The applicant indicated that these requirements 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, the applicant 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.

The applicant considered a spectrum of seismic events to address the TEV's ability to perform under seismic conditions. The applicant conducted a study to assess risk and quantify the mean frequency of seismic-related event sequences (BSC, 2008bg). 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. The applicant provided an analysis (BSC, 2008co) to support the fragility calculations.

The applicant stated that the TEV maintenance plan would be developed and implemented in accordance with SAR Section 5.6. The applicant further indicated that the maintenance process will be centered on reliability and developed before the receipt and possession of HLW. Furthermore, the applicant 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. The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h) to assess the applicant's compliance with the requirements of 10 CFR 63.112(e)(8). To verify whether the applicant 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). (b)(5)

(b)(5)

(b)(5)

The NRC staff reviewed the applicant's calculations for the estimated fragility of tipover, sliding, and collision used to determine the TEV's ability to withstand the seismic events, as outlined in BSC Attachment H (2008co). (b)(5)

(b)(5)

The NRC staff also reviewed how the applicant applied the component reliability assessment to demonstrate the TEV's overall ability to perform its intended safety functions (BSC, 2008bk) (b)(5

(b)(5)

The NRC staff also reviewed SAR Sections 1.3.3.5.2 and 5.6 that described the applicant's plans to develop the TEV inspection and maintenance plans. (b)(5)

(b)(5)

Surface Transportation Equipment ITS

The applicant 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. The applicant indicated that these requirements will be used to define design specifications for the surface transportation equipment. The applicant 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). The applicant 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).

The applicant considered a spectrum of seismic events to address the ability of the site transporter to perform under seismic conditions. The applicant conducted a study to quantify the mean frequency of seismic-related event sequences (BSC, 2008bg). The study identified two potential site-transporter 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, the applicant estimated the fragility related to tipover failure on the basis of conservative engineering judgments supported by general earthquake experience with railcars and truck trailers. The applicant relied primarily on the transportation cask design to mitigate tipover events and therefore, the applicant concluded that tipover failure was not a major contributor to risk. Accordingly, the applicant did not identify any safety function related to tipover for the cask tractor, cask transfer trailer, or site prime mover

The applicant identified PSCs to ensure the transportation equipment's ability to remain in a safe state under certain conditions. The applicant 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. The applicant 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. The applicant will analyze any significant deviation to determine risk significance.

The applicant 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. The applicant briefly addressed the maintenance for the site transporter and site prime mover, but not for the cask tractor and cask transfer trailer. However, the applicant indicated that the maintenance process will be centered on reliability and developed before the receipt and possession of HLW for the cask tractor and cask transfer trailer (DOE, 2009dk). In addition, the applicant 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. The applicant 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, the applicant 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. The applicant 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). The applicant'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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h) to assess the applicant's compliance with the requirements of 10 CFR 63.112(e)(8). After evaluating the applicant's component reliability assessments to demonstrate the surface transportation equipment's overall availability to perform the intended safety functions (BSC, 2008au), (b)(5)

(b)(5)

The NRC staff reviewed the applicant'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). (b)(5)

(b)(5)

The NRC staff reviewed the information related to PSC-2 (b)(5)

(b)(5)

The NRC staff reviewed the applicant'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. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.8.4 Electrical Components and Emergency Power Systems Important to Safety

To demonstrate compliance with 10 CFR 63.112(e)(8) and 63.112(e)(11), the applicant 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. To demonstrate compliance with 10 CFR 63.112(e)(12) and 63.112(e)(13) t~~T~~he applicant also provided information in SAR Sections 1.9.1.12 and 1.9.1.13 on the redundant systems for the ITS EPS SSCs and the means to maintain, inspect, and test the ITS EPS SSCs and, in particular, ITS diesel generator SSCs, as 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

The applicant 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, the applicant stated in SAR Section 5.5 (Table 5.5-1) that it will perform periodic functional tests [e.g., tests described in Regulatory Guide 1.118 (NRC, 1995aa)] to the electrical distribution systems. Furthermore, the applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(m).

(b)(5)

Reliable and Timely Emergency Power

The applicant estimated that the time required for starting, accelerating, sequencing, and loading of the ITS diesel generators will be less than 3 minutes, but a justifiable estimate will not be available until the design is completed. The applicant 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. The applicant also stated (DOE, 2009fp) that the ITS diesel generators will be designed to start and accept load within the 8-hour time period established by the PCSA. However, the applicant also plans to use the ITS UPS SSCs to feed the ITS instrumentation that is needed to support ITS equipment upon loss of alternating current power and stated that the ITS UPS SSCs are similar to normal power UPS SSCs. The normal power UPS SSCs are designed to provide continuous alternating current power up to 15 minutes. The SAR did not explain the relationship between the nominal or maximum time intervals (between the occurrence of a loss of offsite power and the availability of the ITS power provided by the ITS diesel generators) and the design capacity of the ITS UPS SSCs. More specifically, the SAR did not describe which ITS SSCs will require continuous power from the redundant ITS UPS SSCs that are located in the CRCFs, WHF, and EDGF (e.g., what the load and operating time capacity of the ITS UPS SSCs is designed to be, the impact on ITS SSCs if the ITS UPS SSCs become depleted before the ITS alternating current electrical power is restored. In its response to an NRC staff RAI (DOE, 2009gj), the applicant stated that it will develop the ITS UPS SSC maximum power requirements, allowing for future growth, should needs for ITS UPS SSCs are identified during detailed design.

NRC Staff Evaluation: The NRC staff reviewed this information using the guidance in YMRP Section 2.1.1.6.3, Criterion 1(2)(k). (b)(5)

(b)(5)

System Redundancy

The applicant indicated in SAR Section 1.9.1.12 that redundant, independent, and physically separated systems are available for ITS diesel generators. In addition, the applicant 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, and 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(l). (b)(5)

(b)(5)

(b)(5)

(b)(5)

Ability to Perform Intended Safety Function

The applicant identified the performance requirements for the ITS electrical power distribution systems and diesel generators for its CRCF and WHF operations in SAR Tables 1 9-3 and 1.9-4. SAR Table 1.4.1-1 listed the related design criteria for the ITS electrical power distribution systems and diesel generators. The controlling parameters and values for the ITS power generation and distribution systems were also specified in SAR Table 1 4.1-1. The 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h). (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

(b)(5)

(b)(5)

2.1.1.6.3.2.8.5 Fire Protection Systems Important to Safety

To demonstrate compliance with 10 CFR 63.112(e)(8), 63.112 (e)(9), and 63.112 (e)(13), the applicant 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 the applicant's response to an NRC staff RAI (DOE, 2009fr).

The applicant 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 the applicant 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

The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(i) to evaluate the selection of the DIPA system as a suitable means to provide the designated ITS function. (b)(5)

(b)(5)

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). The applicant 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, the applicant 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. The applicant 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), the applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h) to assess the ability of the DIPA system to perform the intended safety functions. (b)(5)

(b)(5)

Assessment of Continued Functionality and Means to Inspect, Test, and Maintain

The applicant indicated that the DIPA systems will be designed, installed, and maintained in accordance with 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 acceptable 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(m) to assess the functionality of the designed system and the ability of the designed system to be properly inspected, tested, and maintained. (b)(

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.8.6 Transportation, Aging, and Disposal Canisters

To demonstrate compliance with 10 CFR 63.112(e)(8), the applicant provided a general description of its approach to demonstrate 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 the applicant's SAR Sections 1.5.1.1.1.2.1.3 and 1.7.2.3.1 and referenced reports (BSC, 2008ac,cp).

The applicant 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 covered 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). The applicant determined the canister failure probabilities by utilizing a fragility curve for the stainless steel material along with the maximum effective plastic strains obtained from the finite element analyses.

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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h). In addition, the NRC staff utilized the review results from SER Section 2.1.1.4 to assist evaluation.

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.8.7 Waste Packages

To demonstrate compliance with the requirements defined in 10 CFR 63.112(e)(8), the applicant provided information relative to waste package design and performance to demonstrate the waste package capability in performing 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).

The applicant 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

The applicant 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, the applicant calculated the stress intensities in the waste package outer corrosion barrier and invoked the tiered screening criteria method (SAR Table 1.5 2-10) that was based on elastic-plastic analysis methods provided in ASME 2001, Section III, Appendix F (American Society of Mechanical Engineers, 2001aa). For the thermal analysis, the applicant 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, the applicant developed reliability estimates to calculate the probabilities of radionuclide release from waste packages assuming the occurrence of event sequences. Also, the applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(h). (b)(5)

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(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.9 Radioactive Waste and Effluents Control

To demonstrate compliance with 10 CFR 63.112(e)(10), the applicant provided information in SAR Sections 1.9.1.10 and 1.4.5.1.1 regarding radioactive waste and effluents control. The applicant provided information to demonstrate 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.

Liquid Low-Level Waste Management

The applicant stated that it will include a subsystem to collect low-level radioactive waste (LLW) liquids and potentially radioactive waste liquids in the waste handling facilities. Liquid 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). The applicant 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, the applicant 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); the applicant identified these liquid waste facilities as non-ITS.

The applicant described the capacity of the effluent systems to contain the largest credible volume of fire-water discharge from fire suppression systems (DOE, 2009fo). 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 these descriptions using the guidance in YMRP Section 2.1.1.6.3, Criterion 1(2)(j) to ensure that the applicant'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 evaluated the applicant's technical basis for managing liquid LLW discussed in SAR Section 1.4.5.1.1 to ensure that the applicant 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 adequately considered liquid waste management systems to 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 ensure 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.

(b)(5)

Solid Low-Level Waste Management

The applicant 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 the acceptance 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(j). (b)(5) The NRC staff evaluated the applicant's technical basis for managing solid LLW and examined the capacity of the applicant-designed facilities and concludes the LLWF has sufficient capacity to manage, package, and ship these waste streams under normal operating conditions

The NRC staff reviewed the WHF operation for fluidizing the resin bed, transferring the fluidized resin to the mobile processing container, and using berms or diked areas/rooms to contain spillage or system leakage from WHF storage tanks and processing equipment (SAR Section 1.4.5.1.1.1). (b)(5)

(b)(5)

(b)(5)

(b)(5)

Gaseous Low-Level Waste Management

The applicant 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). SER Section 2.1.1.6.3.2.8.2.2 concludes 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. The applicant designed the normal repository operations to control gaseous low-level radioactive effluents to an acceptable 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(j). The NRC staff evaluated the information the applicant provided in SAR Section 1.4.5.1.1.3, which described the potential sources of gaseous LLW and the proposed mechanisms and processes to capture these wastes. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.2.10 Structures, Systems, and Components Important to Safety Inspection, Testing, and Maintenance

To demonstrate compliance with 10 CFR 63.112(e)(13), the applicant provided information in SAR Section 1.9.1.13 on considerations of the means to inspect, test, and maintain SSCs ITS, if the applicant relies on inspection, testing, and maintenance to ensure availability of the SSCs safety functions.

SAR Section 1.9.1.13 stated the applicant will provide license specifications that include the limiting conditions for operation of selected SSCs. According to the applicant, the limiting conditions will include specific surveillance requirements, appropriate functional testing, and other inspections. The applicant stated in SAR Section 1.9.1.13 that SAR Sections 1.2, 1.3, 1.4, 1.5, and 5.6 provided information regarding inspection, testing, and maintenance of SSCs. Also, the applicant stated in SAR Section 5.6.1 that the waste handling manager will write, test, and approve plans and procedures for operations, maintenance, surveillance, and periodic testing of SSCs before receipt of waste.

In response to the NRC staff request to identify the inspection, testing, and maintenance needs for SSCs ITS, the applicant (DOE, 2009dk) stated that the reliability-centered maintenance process will be used to develop plans and procedures for inspection, testing, and maintenance of SSCs ITS. According to the applicant, the inspection, testing, and maintenance needs for each component will be based on manufacturer's recommendations, industry codes and standards, equipment qualification, and reliability requirements from the PCSA. The applicant 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 YMRP Section 2.1.1.6.3, Criterion 1(2)(m). (b)(5)

(b)(5)

(b)(5) the applicant stated in SAR Section 1.9.1.13 that the reliability-centered maintenance process will be used to develop plans and procedures for inspection, testing, and maintenance of SSCs ITS (DOE, 2009dk). (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.3.3 Administrative or Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

To demonstrate compliance with 10 CFR 63.21(c)(5) and 63.112(e), the applicant described procedures that will be developed to prevent event sequences or mitigate their effects in SAR Section 1.9.3. The applicant's description referred to the management controls and procedures that will be implemented to ensure that administrative controls and PSCs will function properly.

The applicant 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 the applicant, 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 YMRP Section 2.1.1.6.3, Criterion 2 (b)(5)

(b)(5)

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 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.6.4 Evaluation Findings

The NRC staff reviewed the applicant's SAR and other supporting information relevant to the requirements for PCSA of the GROA (b)(5)

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(b)(5)

(b)(5)

(b)(5)

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CHAPTER 7

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

2.1.1.7.1 Introduction

This Safety Evaluation Report (SER) 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 applicant adequately 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 the applicant provided in response to the NRC staff's requests for additional information (RAIs) (DOE 2009dk,dl,do,dq,dv,dw,dy.eh,er–ew,ez, fa–fe,fg; 2010ak–an).

The applicant 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, and are required to perform their intended safety functions. 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. The applicant did not identify any SSCs as ITS or SCs for the subsurface facilities in the GROA.

2.1.1.7.2 Regulatory Requirements and Regulatory Guidance

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 the applicant to provide information for codes and standards that the applicant proposes to apply to the GROA design and construction. 10 CFR 63.21(c)(3) requires the applicant 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 a description and discussion of the GROA design, including (i) the relationship between design criteria and the preclosure performance objectives specified at 10 CFR 63.111(a) and 63.111(b) and (ii) the design bases and their relation to the design criteria.

The NRC staff reviewed the applicant's design information using the guidelines provided in the Yucca Mountain Review Plan (YMRP) Final Report Section 2.1.1.7 (NRC, 2003aa). 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 Staff Review and Analysis

The NRC staff focused its review on the applicant's design of the SSCs ITS and SCs to determine whether the design information the applicant provided in its SAR adequately demonstrates that the SSCs ITS and SCs design, construction, and operation will meet the facility performance objectives. Specifically, the NRC staff's evaluation determines whether the design information adequately describes the relationship between the proposed design criteria and the GROA performance requirements, and the relationship between the design bases and the design criteria, in accordance with the requirements of 10 CFR 63.112(f). Furthermore, the NRC staff verifies that applicable codes and standards have been used for design and construction of the SSCs ITS and SCs in the GROA.

In addition, the evaluation determines whether the design information adequately addresses the requirements of 10 CFR 63.21(c)(2) and 63.21(c)(3), which require that the applicant provide 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 determines whether the design methodologies, design analysis, and design are supported by adequate 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 Canister; and Other Canisters, Overpacks, and Casks; and (x) Criticality Prevention and Shielding Systems.

2.1.1.7.3.1 Structural and Civil Facilities

The applicant provided design information on the ITS surface waste-handling facilities, the aging facility, and flood control features. To meet the dose limits of the preclosure performance objectives, 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. To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant 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 the applicant 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 reviews on the description of the surface facilities and the performance of the surface facilities to prevent or mitigate event sequences are provided in SER Sections 2.1.1.2 and 2.1.1.4, respectively.

2.1.1.7.3.1.1 Surface Structural Buildings

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant 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.

Design Bases and Design Criteria

The applicant 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. 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, the applicant specified that ITS surface buildings should be designed to withstand the external loads in SAR Tables 1.2.2-1 through 1.2.2-3, which are summarized next:

Wind and tornado: 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 SER 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 5585 Pa [0.81 psi], and a rate of pressure drop of 2068 Pa/s [0.30 psi/s] (NRC, 1976ab; Ramsdell and Andrews, 1986aa). The applicant treated the effects of tornado missile impacts as live loads with impact. The NRC staff reviews tornado hazard in SER Section 2.1.1.3.3.1.2.3.2.

Explosion: Explosion hazard design values are based on a maximum no-damage overpressure of 6895 Pa [1 psi] (SAR Table 1.2.2-1). The NRC staff evaluates the explosion hazard design values in SER Section 2.1.1.3.3.1.2.6.5.

Volcanic Ash: The roof live load caused by volcanic ash fall is 1005 Pa [21 psf] (SAR Section 1.2.2.1.6.5). The NRC staff reviews ash load in SER 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 SER Section 2.1.1.1.3.3.

Seismic: The ITS surface buildings were designed for a site-specific seismic ground motion level [Design Bases Ground Motion 2 (DBGM-2)], associated with a mean annual probability of exceedence (MAPE) of 5×10^{-4} (SAR Table 1.2.2-3). The horizontal and vertical peak ground accelerations (PGAs) for DBGM-2 seismic events are 0.45g and 0.32g, 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 SER Section 2.1.1.1.3.5.2.

NRC Staff Evaluation: The NRC staff reviewed the applicant's design bases and design criteria for the ITS surface buildings using the guidance in YMRP Section 2.1.1.7.2.1. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

The applicant determined that seismic loading bounds the design of the ITS surface buildings (DOE, 2009es). The applicant's methodology for seismic design of ITS buildings was based on elastic analyses developed in SAP2000 (Computers and Structures, 2005aa). The applicant 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), the applicant 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 the applicant's modeling methodology discussed in the SAR and the response to the NRC staff's RAIs using the guidance in YMRP Section 2.1.1.7.2.2. (b)(5)

(b)(5)

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: The applicant listed the codes and standards used for the structural design of surface buildings in SAR Section 1.2.2.1.8. The applicant 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). The applicant 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, ~~2007ba~~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, the applicant 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 the applicant's proposed use of design codes and standards for surface facilities using the guidance given in YMRP Section 2.1.1.7.2.3.

(b)(5)

Consistency of Materials With Design Methodologies: The applicant described the materials used for the surface facility design in SAR Section 1.2.2.1.7. The applicant'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 the applicant proposed for the design of the reinforced concrete and steel structures and foundations, using guidance in YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Load Combinations: The applicant listed the load combinations to be used in ITS surface building design in SAR Section 1.2.2.1.9.2 and BSC Sections 4.2.11.4.5 and 4.2.11.4.6 (2007av). In response to an NRC staff's RAI (DOE, 2009es), the applicant stated that some of these load combinations are not applicable to the design of specific structures (e.g., 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, the applicant used only one load combination that includes gravity (dead load and 25 percent of live load) and seismic loadings. For the seismic load combination, the applicant 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's RAI (DOE, 2009es), the applicant 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. The applicant 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 the applicant proposed for the design of ITS surface buildings using the guidance in YMRP Section 2.1.1.7.2. (b)(5)

(b)(5)

Seismic Analysis Methodology: For the seismic analyses of ITS surface buildings subjected to DBGM-2 events, as described in BSC Section 5.1 (2007ba), the applicant 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, the applicant 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's RAI (DOE 2009es), the applicant described the generation of site-specific ground motion and design spectra in SAR Section 1.1.5. The applicant 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, the applicant 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, the applicant 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), the applicant 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. The applicant found 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. The applicant 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's RAI (DOE, 2009eu), the applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

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Structural Design Methodologies: The following subsections present the applicant's approach to evaluate the design of the structural components of the ITS buildings. The applicant'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 the applicant 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).

Shear Wall Design: The applicant's methodology for shear wall design, as outlined in BSC Appendix D (2007ba), is based on ACI 349–01 (American Concrete Institute, 2001aa). The applicant determined that the predominant load path of seismic load is through the diaphragms and shear walls to the base of the concrete slab. The applicant 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).

The applicant determined, on the basis of its design calculations, that reinforced concrete shear walls (BSC, 2007ba,cy) have sufficient capacity to withstand the design loads. The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

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(b)(5) In response to an NRC staff's RAI (DOE, 2009eu), the applicant stated that a comparison of the seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that a 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. (b)(5)

(b)(5)

Slab Design: The applicant 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. The applicant 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, the applicant 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)]. The applicant 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, the applicant assumed that the structural steel components provided support for the concrete slabs and superimposed loads for all applicable service and extreme load combinations. The 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 YMRP Section 2.1.1.7.2.3. (b)(5)

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(b)(5) In response to an NRC staff's RAI (DOE, 2009eu), the applicant stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that a 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. (b)(5)

(b)(5)

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's RAI (DOE, 2009eu), the applicant 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. The applicant 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's RAI (DOE, 2009ev), the applicant 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. The applicant indicated that sensitivity analyses of the mesh refinement would be performed to support the detailed design.

To account for SSI, the applicant calculated soil spring constants on the bases of the elastic and shear modulus of alluvium and tuff. The applicant 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). The applicant 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's RAI (DOE, 2009ev), the applicant 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). The applicant stated that the damping ratios obtained from SASSI were greater than 20 percent, but supporting calculations were not provided.

For the foundation seismic analyses, the applicant 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's RAI (DOE, 2009eu), the applicant 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, the applicant estimated an allowable soil bearing pressure of 2394 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 the applicant proposed using the guidance in YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

(b)(5)

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 SER 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, the applicant 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 conditions (BSC, 2008af). The applicant 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. The applicant 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}. The applicant 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, the applicant 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). The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

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 SER section. For the CRCF, the maximum D/C ratios of the shear walls the applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

CRCF and RF Slab Design: The designs of CRCF and RF slabs were based on the general design methodologies evaluated previously in this SER 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), the applicant used ANSI/AISC N690–1994 (American Institute of Steel Construction, 1994aa) and obtained maximum D/C ratios in flexure of 0.85.

The applicant presented the RF design slab results in BSC Section 7 (2008cj) and concluded that a reasonable slab design is achieved for the imposed design loads. The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

CRCF and RF Foundation Analysis and Design: The CRCF and RF foundation design followed the general design methodologies evaluated previously in this SER 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, the applicant 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 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's RAI (DOE, 2009eu), the applicant 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, the applicant estimated that the alluvium thickness varies from about 38 to 44 m [125 to 145 ft] (SAR Figure 1.1-130). The applicant 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, the applicant 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] the applicant proposed. The applicant used soil springs in the foundation that can model linear elastic compression, but no tension forces.

For the CRCF foundation, the applicant computed a safety factor against the structure overturning of 3.1. For the foundation resistance against sliding, however, the computed safety factor was 0.68, which is less than the acceptable limit of 1.1. The applicant 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). The applicant 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, the applicant indicated that any connection that enters the CRCF and RF structures should have a flexibility of at least 10.2 m [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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

(b)(5) In response to an NRC staff's RAI (DOE, 2009eu), the applicant stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that a 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. (b)(5)

(b)(5)

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 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). The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

WHF Shear Wall Design: The WHF shear wall design used the general design methodologies evaluated previously in this SER 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

WHF Slab Design: The WHF slab design used the general design methodologies evaluated previously in this SER section. For the design of the reinforced concrete slabs, the applicant provided the calculation of five representative slabs in BSC Section 7 (2007cw), concluding that a reasonable slab design is achieved for the imposed design loads. The applicant indicated that these results were preliminary and should only be used in the preliminary design phase of the project.

The applicant 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 approximated following the assumptions of 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 YMRP Section 2.1.1.7.2.3 (b)(5)
(b)(5)

WHF Foundation Analysis and Design: The WHF foundation consists of a 1.8-m [6-ft]-thick reinforced concrete slab at the grade level, an 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). The applicant 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, the applicant calculated soil spring constants based on the Young's modulus and shear modulus of alluvium and tuff soil layers.

The applicant 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's RAI (DOE, 2009eu), the applicant 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). The applicant did not provide a comparison in its response.

On the basis of the above structural analyses, the applicant generated moment and shear contour plots that were used to compute the shear and flexural reinforcing in the foundation mat. The applicant proposed a typical rebar pattern, with additional reinforcement in critical regions. For the grade basemat, the applicant computed maximum moment and shear D/C ratios of 0.85 and 0.82, respectively, as shown in BSC 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 2006 kPa [41.9 ksf] for the pool basemat. These bearing pressures are smaller than the allowable bearing capacity of 2394 kPa [50 ksf], as detailed in BSC Section 6.2.3 (2007ba), which is evaluated in SER Section 2.1.1.1.3.5.4. The applicant's nonlinear soil springs in the foundation were linear elastic under compression with zero capacity under tension. (b)(5)
(b)(5)

The applicant 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

the applicant used the reserve energy method (American Society of Civil Engineers, 2005aa), obtaining a sliding displacement of 5 mm [0.2 in]. The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)
(b)(5)

(b)(5) In response to an NRC staff's RAI (DOE, 2009eu), the applicant stated that a comparison seismic analysis using the original data (BSC, 2007ae) and revised data (BSC, 2007bx) demonstrated that a 11-m [35-ft]-thick alluvium depth controls the foundation design because it provides the highest story shears and diaphragm accelerations for the WHF. (b)(5)
(b)(5)

Initial Handling Facility: 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 SER section. The applicant 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, the applicant did not include SSI effects, because the structure supports were modeled as pinned connections at the basemat, as described in BSC Section 3.1.2 (2008am). In response to an NRC staff's RAI (DOE, 2009ev), the applicant 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. The applicant 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, the applicant 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's RAI (DOE, 2009ev), the applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

IHF Steel Frame Design: The design of the IHF steel frame followed the general design methodologies evaluated previously in this SER section. The applicant 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Design of IHF Shear Walls: The design of the IHF shear walls followed the general design methodologies evaluated previously in this SER 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Design of IHF Slabs: The analysis and design of the IHF slabs followed the general design methodologies evaluated previously in this SER 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 indicated in BSC Section 6.6 (2007aq) 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 YMRP Section 2.1.1.7.2.3 (b)(5)

(b)(5)

IHF Foundation Analysis and Design: The design of the IHF foundation followed the general design methodologies evaluated previously in this SER section. The IHF foundation consists of two individual basemats that were modeled using a finite element analysis. Unlike the analysis of the IHF superstructure, the applicant included SSI effects in the foundation analysis. To account for SSI, the applicant calculated soil spring constants on the basis of the elastic and shear modulus of alluvium and tuff.

For the IHF foundation analysis, the applicant 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). The applicant 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. The applicant'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.

The applicant also stated in BSC Section 6.7 (2008aq) that the soil-bearing pressure for the small mat may reach 2107 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 2394 kPa [50 ksf]. This allowable soil-bearing capacity is evaluated in SER 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 YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Commitment

- The applicant commits to use finite element models for the detailed design for construction that will include SSI effects on the structural response of the IHF superstructure (DOE, 2009ev).

- The applicant commits to evaluate, as part of the process in the detailed design for construction, effects of the steep sloping alluvium stratum beneath the foundation on the IHF foundation response (DOE, 2009ev).

- The applicant commits to perform Tier #2 analysis to support the detail design of surface nuclear facilities (BSC, 2007ba, ACN 02).

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.1.2 Aging Facility

To demonstrate compliance with 10 CFR 63.112(f), the applicant 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 transportation, aging, and disposal (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). The applicant 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, the applicant provided the design bases and their relationship to the design criteria of the aging facility. The applicant derived the design bases from site characteristics and the PCSA. SAR Table 1.2.7-1 provided nuclear safety design bases for (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. The applicant 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 YMRP Section 2.1.1.7. The NRC staff integrated the review of the design information presented in this chapter with the NRC staff's evaluations of the site-specific information in SER Section 2.1.1.1 and the PCSA information in SER Section 2.1.1.6.

(b)(5)

Design Methodologies

In SAR Section 1.2.7.6, the applicant described the design methodologies used for the structural design of the aging pads. To design the concrete mats of the aging facility, the applicant performed a finite element static analysis using SAP2000 (Computers and Structures, 2005aa). For the mats supporting vertical casks, the applicant modeled a small representative area 26.5 × 35 m [87 × 114 ft] of the mat supporting 16 vertical aging casks. In response to an NRC staff's RAI (DOE, 2009ew) on the adequacy of the model to simulate the behavior of the actual mats, which are much longer in length {195 to 314 m [640 to 1,030 ft]}, the applicant used repeating arrays of casks on a continuous concrete slab. The applicant did not model the mats supporting horizontal aging modules, because the loadings on the pads for the vertical aging casks and horizontal aging modules are similar.

The applicant modeled the concrete slab using shell elements with a 0.91 × 0.91 m [3 × 3 ft] mesh size. Because the mesh size can affect the level of accuracy of the model in the finite element analysis, the NRC staff issued an RAI requesting a technical basis for the use of this coarse mesh size. In response to the NRC staff's RAI (DOE, 2009ew), the applicant stated that computed design forces (i.e., moments and shear forces) were reasonable and that a finer mesh is not expected to significantly affect the computed forces.

To account for the stiffness properties of the soil underlying the aging pads, the applicant modeled the soil foundation as springs in SAP2000 in two horizontal directions and one vertical direction. 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. In response to an NRC staff's RAI (DOE, 2009ew) about how the applicant considered the potential uncertainties in the alluvium material properties that may affect the design forces of the concrete foundation, the applicant stated that future analyses for detailed design will consider potential varying soil properties for the concrete foundation.

NRC Staff Evaluation: The NRC staff reviewed the applicant's design methodologies for the aging facility using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analyses

The applicant 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, the applicant 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's RAI (DOE, 2009ew), the applicant provided the results of its flood inundation analysis and the planned drainage channels. (b)(5)

(b)(5)

(b)(5) The NRC staff's evaluation of the flood protection barriers is discussed in SER Section 2.1.1.7.3.1.3.

The applicant 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 peak ground acceleration 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, the 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 an assumed coefficient of friction (COF) of 0.35 between the concrete pad and the aging cask. The applicant determined that the wind and tornado loads on the aging casks were less than those from the seismic design basis event. (b)(5)

(b)(5)

For the analysis of the concrete pad under seismic loads, the applicant 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's RAI (DOE, 2009ew), the applicant 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.

In determining the forces on the concrete pad from the aging casks during a seismic event, the applicant assumed that the horizontal forces were limited to those based on a COF of 0.35 between a cask and pad concrete. This approach of limiting the forces on the basis of the COF is reasonable. However, 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 an NRC staff's RAI (DOE, 2009ew), the applicant recognized that a higher COF for steel–concrete interface may be possible. However, the applicant stated it will employ construction or engineering measures to achieve a low COF.

The applicant 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. The applicant 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.

The applicant 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, the applicant 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). The applicant concluded that shear strength of concrete is greater than the demand and the shear reinforcement is not required. However, the applicant 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's RAI (DOE, 2009ew) on the amount and spacing of shear reinforcement of the concrete pad, the applicant 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 SER 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 the applicant's aging facility structural design using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

(b)(5)

Proposed Condition of Construction Authorization

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.1.3 Flood Control Features

To comply with 10 CFR 63.112(f), the applicant 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. The applicant 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). The applicant 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 an NRC staff's RAI (DOE, 2009eh,fh). The applicant also provided PMF and flood inundation analyses for the proposed flood control features (BSC, 2007db).

The applicant'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. However, the applicant's analysis of the PMF water surface elevations at the site with the mitigating flood control features consists of only items (i), (ii), and (iii) (BSC, 2007db). In response to an NRC staff's RAI to clarify these discrepancies (DOE, 2010ak,an), the applicant stated that (i) the two diversion channels were added to the design after the PMF analysis was completed, (ii) the flood flow these channels expect to capture would be relatively minor, and (iii) the final flood analysis and design for the diversion channels will be performed as part of the detailed design. (b)(5)

(b)(5)

Design Basis and Design Criteria

In SAR Table 1.2.3-3, the applicant 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 adequate 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 the applicant proposed using the guidance in YMRP Section 2.1.1.7.2.1. (b)(5)

Design Methodologies

The design methods the applicant used for the flood control features include estimation of the design basis PMP at the GROA site, which produces the PMF runoff. The applicant 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. The applicant 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, the applicant 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's RAI (DOE, 2010an), the applicant stated that the diversion ditches will be sized to transport the PMF provided in BSC (2007db). However, the applicant did not provide a design methodology for the storm water detention ponds and stated that the design of the storm water detention ponds will be performed as part of the detailed design (DOE, 2010ak). (b)(5) (b)(5)

The SAR did not present any information on the geotechnical engineering aspects of the proposed flood control features except typical cross sections in SAR Figure 1.2.2-7. In response to an NRC staff's clarification question (DOE, 2010an), the applicant provided a summary of the geotechnical design aspects of the flood control features and stated that detailed design will address geotechnical engineering aspects of the flood control features. The applicant further committed to following guidance and engineering practices in U.S. Army Corps of Engineers (2000aa, 1994aa), Federal Highway Administration (2005aa), and Regulatory Guide 1.102 (NRC, 1976ac) in the detailed design of dikes (levees) and channels of flood control features at the GROA site (DOE, 2010an). (b)(5) (b)(5)

NRC Staff Evaluation: The NRC staff reviewed the design methodologies for the flood control features the applicant proposed using the guidance in YMRP Section 2.1.1.7.2.2. (b)(5) (b)(5)

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, the applicant's analysis did not consider this. In response to an NRC staff's RAI on the aging facility design described in SAR Sections 1.2.2 and 1.2.7 (DOE, 2009ew), the applicant stated that grading of the existing topography and associated cross sections through the aging facility area is deferred to the detailed design. (b)(5)

(b)(5)

The applicant 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. 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]. However, in response to an NRC staff's RAI, the applicant 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], the applicant 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 bulking factor (BSC, 2007db). However, on the basis of an analysis with more conservative inputs, the applicant 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's RAI to clarify this discrepancy, the applicant 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 10^6-year return period screening criterion for Category 2 event sequences by a margin of approximately 38 percent (DOE, 2010ak).

(b)(5)

Also, for the North Portal pad area, the applicant did not present any design information for the flood detention ponds and the associated inlet and outlet structures. In response to an NRC staff's RAI, the applicant stated that subject to the requirement in the nuclear safety design bases, sizing and placement of the diversion ditches and flood detention ponds will be determined during detailed design (DOE, 2010ak, 2009fe).

(b)(5)

(b)(5) The site plans in BSC

(2008ch,ci) showed riprap protection in plan view, but it is not shown in any cross sections or other documents. 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). (b)(5)

(b)(5)

NRC Staff Evaluation: The NRC staff reviewed the design and analysis for the flood control features the applicant proposed using the guidance in YMRP Section 2.1.1.7.2.3. (b)(5)

(b)(5)

Commitment

The applicant commits to provide a detailed design of the flood control features proposed in the conceptual design (DOE, 2010ak,an).

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.2 Mechanical Handling Transfer Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant 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 SER 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, the applicant 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. The applicant also provided instrumentation and logic diagrams in SAR Figures 1.2.4-51 through 1.2.4-56.

Design Bases and Design Criteria

The applicant 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. The applicant provided specific design criteria to meet each of the required safety functions, along with controlling parameters and bounding values.

The applicant provided several design criteria for the safety functions to (i) protect against a drop of the load and (ii) protect against the drop of a load onto a canister so that the drop energy does not breach the load or canister. These criteria, based on 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, the applicant'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, 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, the applicant 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 SER 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

The applicant used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The applicant 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 adequate capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

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). The applicant 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.

The applicant considered static, dynamic, and environmental loads associated with normal operation as well as Categories 1 and 2 event sequences. The specific design loads that the applicant 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, the applicant 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, the applicant considered dead loads, live loads, and seismic loads of DBGM-2 levels. In the case of extreme wind loads, the applicant considered the dead load and a nonseismic wind load. In the case of a collision, the applicant considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidelines in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

NRC Staff Conclusion

On the basis of the NRC staff's evaluation of information provided in SAR Sections 1.2.2, 1.2.3, 1.2.4, 1.2.5, and 1.2.6, (b)(5)

(b)(5)

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 load-out subsystem of these surface facilities. It operates between the waste package positioning room, waste package closure room, and waste package load-out room. It presents an empty waste package to the CTM when positioned under the waste package port in the vertical position. Before the 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

The applicant 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). The applicant 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 the WPTT 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

The applicant used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The applicant 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 adequate capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

Design and Design Analysis

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). The applicant 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.

The applicant'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, the applicant 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, the applicant 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, the applicant considered dead loads, live loads, and seismic loads of DBGM-2 levels. For extreme wind loads, the applicant considered the dead load and a nonseismic wind load. For a collision, the applicant considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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 1361 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. The applicant 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.

Design Bases and Design Criteria

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, the applicant presented multiple design criteria to meet the probability of failure of each design basis.

The applicant 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 requiring 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 safe limits for workers, the applicant required 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 the applicant's information using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

Design Methodologies

The applicant used ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The applicant 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 adequate capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

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. The applicant 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.

The applicant'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, the applicant 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, the applicant 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 seismic consideration, the applicant included dead loads, live loads, and seismic loads of DBGM-2 levels. For extreme wind loads, the applicant considered the dead load and a nonseismic wind load. In the case of a collision, the applicant considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidance in YRMP Section 2.1.1.7 (b)(5)

(b)(5)

NRC Staff Conclusion

On the basis of the NRC staff's evaluation of information provided in SAR Sections 1.2.2, 1.2.3, 1.2.4, and 1.2.5, (b)(5)

(b)(5)

(b)(5)

2.1.1.7.3.2.4 Cask Transfer Trolley

The main function of the 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]. The applicant described the CTT design features in SAR Sections 1.2.3.2.1, 1.2.4.2.1, 1.2.5.2.1, and 1.2.6.2.1, and the mechanical envelope diagram was shown in SAR Figure 1.2.3-20. The applicant also provided the process and instrumentation diagram in SAR Figure 1.2.4-27.

Design Bases and Design Criteria

The applicant presented the nuclear design bases and their relationship with the design criteria in SAR Tables 1.2.3-3, 1.2.4-4, 1.2.5-3, and 1.2.6-3. To limit the CTT speed, the applicant'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, the applicant proposed several other safety controls (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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

The applicant 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, the applicant 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, the applicant 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 adequate capacity during a seismic event.

NRC Staff Evaluation: The NRC staff reviewed the applicability of the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

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. The applicant 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.

The applicant'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, the applicant 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, the applicant 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, the applicant considered dead loads, live loads, and seismic loads of DBGM-2 levels. In the case of extreme wind loads, the applicant considered the dead load and a nonseismic wind load. For a collision, the applicant considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

NRC Staff Conclusion

On the basis of the NRC staff's evaluation of information provided in SAR Sections 1.2.2, 1.2.3, and 1.2.4, (b)(5)

(b)(5)

2.1.1.7.3.3 Heating, Ventilation, and Air Conditioning System

To demonstrate compliance with 10 CFR 63.21(c) and 63.112(f), the applicant 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 SER Sections 2.1.1.7.3.6 and 2.1.1.7.3.7, respectively. The NRC staff's review of the HVAC system general description is in SER Section 2.1.1.2.3.2.4, and ITS HVAC system performance is in SER Section 2.1.1.4.3.3.2.1.

Design Bases and Design Criteria

The applicant 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. The applicant 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.

The applicant 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, the applicant 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, the applicant 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. The applicant provided HVAC system fault trees to support the controlling parameter values used in the nuclear safety design bases. In response to the NRC staff's RAI (DOE, 2009fd) on design bases and design criteria for the ITS HVAC systems, the applicant provided additional information to address the requirements for overall filtration efficiency and the cooling requirements for ITS electrical equipment. The applicant 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).

Although 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, these tables did not specify hydrogen removal with the safety function. Hydrogen accumulation during battery charging (identified as precluded in SAR Section 1.2.4.4.1) is not linked to fire or explosion in the PCSA, nor is it screened out. In response to an NRC staff's RAI (DOE, 2009fd), the applicant indicated that it will perform 21 air changes per hour; this exceeds the 5 volume changes per hour required in ASHRAE 2007 (American Society of Heating, Refrigerating, and Air Conditioning Engineers, 2007aa) to preclude hydrogen accumulation.

NRC Staff Evaluation: The NRC staff reviewed the applicant's design bases and design criteria for ITS HVAC systems using the guidance in YMRP Section 2.1.1.7.2. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

The applicant 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's RAI (DOE, 2009fd), the applicant 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, the applicant stated that ASDs will be designed in accordance with National Electrical ~~Manufactures~~Manufacturers Association (NEMA) ICS 7–2006 (National Electrical ~~Manufactures~~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, the applicant 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 the applicant's design methodology using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analyses

In addition to the codes and standards described in the previous section, the applicant 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. The applicant also used Regulatory Guide 3.18 (NRC, 1974ab), which provides guidance on design of confinement barriers and systems. The applicant 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, the applicant identified that it used DOE–HDBK–1169–2003 (DOE, 2003ae), in lieu of ERDA 76-21 (Burchsted, et al., 1976aa).

The applicant 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. The applicant further identified in SAR Section 1.2.2.3.7 the use of 304L stainless steel for the ductwork, HEPA filter casings, and HEPA filter housings, referencing ASTM A240/A240M–06c (ASTM International, 2006aa). Additionally, the applicant 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.

The applicant identified the use of independent trains in its design criteria for the ITS HVAC systems. For example, the applicant identified two full-capacity, independent trains 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, the applicant identified an independent train for the rooms associated with each ITS electrical train. Additionally, the applicant described the physical separation of the trains. For example, for the CRCF, the applicant identified Train A HVAC equipment located on the opposite end of the building from the Train B HVAC equipment (BSC, 2008ac). In response to an NRC staff's RAI (DOE, 2009dw) on the independence of trains and the potential for a single point of failure, the applicant stated that individual components such as interlocks and ASDs would not be a single point of failure as they may cause a spurious transfer of the operating train but will not cause system failure by themselves.

The applicant'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, the applicant identified the International Building Code 2000 (International ~~Building~~ Code Council, 2003aa) for the design of HVAC ducts and duct supports for seismic loads. The applicant, however, did not credit the HVAC system for confinement following a seismic event.

The applicant 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. The applicant simulated the thermal behavior of the waste package and the transfer trolley under normal and off-normal conditions. Simulated off-normal conditions included two different scenarios: (i) ventilation provided by ITS exhaust fans only and (ii) 30-day no airflow conditions. The applicant stated that the calculated peak cladding temperature 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, the applicant demonstrated 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 the applicant's design information using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.4 Other Mechanical Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant provided design information for ITS mechanical systems other than the mechanical handling transfer systems evaluated in SER Section 2.1.1.7.3.2. 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'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 are provided in SER Section 2.1.1.2 and the ability of the other mechanical systems to perform their intended safety functions is evaluated in SER 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, the applicant specified the load ratings 2,721 to 2.7×10^5 kg [3 to 300 ton] for these cranes.

Design Bases and Design Criteria

The applicant 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). The applicant presented the design bases and design criteria for the CTM maintenance crane of the RF in SAR Table 1.2.6-3.

The applicant 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, not-important to safety (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, the applicant'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, the applicant imposed a speed limitation of 6.1 m/min [20 ft/min].

To protect against crane collapse onto a waste container, the applicant'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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

For crane systems, the applicant'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. The applicant'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, the applicant 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 the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

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. The applicant 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, the applicant defined the loads for which the cranes are designed. The applicant 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, the applicant defined the design loads for jib cranes. The applicant considered the same design loads for the jib crane.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidelines in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.4.2 Special Lifting Devices

To comply with 10 CFR 63.112(f), the applicant 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

The applicant 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, the applicant provided specific design criteria to meet each of the design bases.

To protect against a cask drop or load drop onto a cask/canister, the applicant'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, the applicant 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, the applicant 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 the applicant's design bases and design criteria using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

The applicant'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, the applicant 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 the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

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, the applicant 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), the applicant 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, the applicant 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, the applicant 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, the applicant considered dead loads, live loads, and seismic loads of DBGM-2 levels. For a collision, the applicant considered dead loads, live loads, and loads associated with a collision event sequence.

NRC Staff Evaluation: The NRC staff reviewed the applicant's design and design analyses using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.4.3 Other Mechanical Structures

To demonstrate compliance with 10 CFR 63.21(c)(3) and 63.112(f), the applicant 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 cask. 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.

Design Bases and Design Criteria

The applicant presented the design bases and design criteria for shield doors, slide gates, and platforms in SAR Tables 1.2.3-3 (IHF), 1.2.4-4 (CRCF), 1.2.5-3 (WHF), and 1.2.6-3 (RF). The design bases and design criteria for the rails and racks 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, the applicant 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) adequate 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 the applicant's design bases and design criteria using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

In SAR Section 1.2.4.2.1.6, the applicant 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, the applicant stated that the design methodology for TEV and WPTT rails is based on ASME NOG–1–2004 (American Society of Mechanical Engineers, 2005aa). The applicant'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 the applicant's design methodologies using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

Design and Design Analysis

Although the applicant did not provide a detailed design analysis, the applicant 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. 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 conservative with an adequate structure to provide personnel safety and seismic protection for canisters that are staged on various racks.

The applicant 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, the applicant 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, the applicant specified the load combinations for racks in accordance with ANSI/AISC N690–1994 Table Q1.5.7.1 (American Institute of Steel Construction, 1994aa). The applicant 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, the applicant 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 the applicant's design and design analyses using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.5 Transportation Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant 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 SER Section 2.1.1.2, and the ability of transportation system to perform its intended safety function is provided in SER Section 2.1.1.6.

2.1.1.7.3.5.1 Transport and Emplacement Vehicle

The applicant plans to use the 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. The

applicant 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 SER 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.

The applicant presented the TEV design features in SAR Sections 1.3.2, 1.3.3, and 1.3.4. The applicant also provided a complete mechanical handling and design report on the TEV that included its detailed design. In addition, the applicant 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 SER Section 2.1.1.2, and the ability of the TEV to perform its intended safety functions is provided in SER Section 2.1.1.6.

Design Bases and Design Criteria

The applicant provided the nuclear safety design bases and their relationship to the design criteria 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, the applicant specified that wheel size, gearbox configuration, and disk brakes be designed into the TEV to achieve runaway prevention requirements.

The applicant 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, the applicant specified interlocks that prevent opening the TEV doors in unrestricted areas between the surface handling facility and the emplacement drift turnouts.

And finally, to protect the waste packages from ejection during a spectrum of seismic events, the design criterion listed in the SAR was to provide locks to the TEV shield doors. The applicant indicated using 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 using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

In designing the TEV, the applicant first considered specific characteristics of the GROA site, such as (i) layout and operations of surface facilities and load-out 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, the applicant utilized a methodology for designing the TEV that included several studies. In one study (BSC, 2008ck), the applicant 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), the applicant (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), the applicant 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, the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analysis

The applicant 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. The applicant 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). The applicant 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. The applicant 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

The applicant 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 in the vertical direction is low {i.e., less than 356 mm [14 in] from top of the rail to bottom of the emplacement pallet}. In addition, the applicant 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 the applicant's information on preventing TEV tipover using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Protection of TEV Against Runaway: To protect the TEV against runaway during operations, the applicant incorporated five design and control elements: (i) eight electric drive motors, (ii) eight integrated disk brakes, (iii) eight gearboxes with splined drive shafts, (iv) appropriately sized drive wheels, and (v) redundant rail brakes.

The applicant 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. The applicant chose the fastest acceptable TEV speed of 2.7 km/hour [1.7 mph] (SAR Section 1.3.3.1.2.2) with a 10 percent variation. To achieve the desired speed limit, the applicant selected a 1,750-rpm motor coupled to a 914 mm [36-in]-wheel.

The applicant considered the interaction of the TEV with other SSCs that could potentially affect the speed control, such as electrical power failure. The applicant 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, the applicant added a second level of runaway-prevention redundancy by adding eight high gear-ratio (100.75:1) gearboxes, one for each drive motor. The final TEV design will use noncoasting gearboxes currently available through commercial vendors (DOE, 2009ez).

The applicant also introduced an additional braking system for parked or off-normal conditions. The applicant selected rail brakes (or "thrusters") that directly couple the TEV to the rail in a wedgelike braking action.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on protection of the TEV against runaway using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

(b)(5)

(b)(5) The applicant stated that rail brakes (thrusters) are commercially available units and are designed to railroad standards.

Prevention of TEV From Derailment: The applicant addressed TEV derailment prevention at the load-out station during a seismic event. The applicant 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.

The applicant also addressed derailment due to wheel climb on the tightest curve {radius of 61 m [200 ft]}. The applicant performed a geometric assessment to ensure that 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 the applicant's information on TEV derailment using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5) The applicant specifically 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. (b)(5)

(b)(5)

(b)(5)

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. The applicant 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.

The applicant also considered the prevention of inadvertent opening of the TEV's front and rear doors in unrestricted areas. The applicant 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. The applicant referenced Doman (1988aa) as the industry guidance for the design of these components. In addition, the applicant indicated that operators in the Control Center will confirm the proper position of the ITS switch before the TEV is allowed to proceed.

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on protection against waste package ejection using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5) A more detailed evaluation of interlocks and ITS controls is provided in SER Section 2.1.1.7.3.7.

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.5.2 Site Transporter

The applicant plans to use the site transporter in intrasite operations to transport loaded and unloaded aging overpacks and unloaded dual-purpose canisters inside shielded transfer casks between surface facilities such as the CRCF, WHF, RF, aging facility, and low-level radioactive waste facility. The applicant provided the design features of the site transporter in SAR Section 1.2.8.4.1.1 and a mechanical envelope diagram in SAR Figure 1.2.8-49. The applicant also indicated that the site transporter 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

The applicant presented the nuclear safety design bases for the site transporter and their relationship to the design criteria in SAR Table 1.2.8-2. The applicant provided specific design criteria to meet each of the nuclear safety design bases.

To protect against spurious movement, the applicant defined procedural controls for site transporter motion to prevent spurious movement. Complementing this requirement, the applicant 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].

The applicant also defined a design basis to preclude fuel tank explosion. The applicant indicated that the site transporter will be equipped with a limited supply of fuel and an onboard fire suppression system. In addition, the applicant specified operational procedures to prevent the use of the internal combustion engine 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 adequate 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

The applicant 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, the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analysis

The design features for satisfying each of the nuclear bases and criteria follow.

Protection Against Runaway: The applicant 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. The applicant acknowledged a deviation of the site transporter speed specification from ASME NOG–1–2004 (Table 5333.1-1). 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, the applicant quantified the robustness of transportation casks that were required to survive a 1016-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 1016-mm [40-in] drop. The applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Protection Against Fuel Tank Explosion: The applicant indicated that the site transporter will be equipped with a fire suppression system. The applicant acknowledged that ASME NOG–1–2004 does not address fuel tank explosion protection guidelines or fire protection for internal-combustion-engine-driven crane systems.

NRC Staff Evaluation: The NRC staff reviewed the protection against fuel tank explosion of the site transporter using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Protection Against Spurious Movement: The applicant 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, the applicant required a PSC to disconnect the electrical power to the site transporter. The applicant also required independent verification of the control. Furthermore, the applicant required the activation of the brakes during loading and unloading of canisters.

NRC Staff Evaluation: The NRC staff reviewed the protection against spurious movement of the site transporter using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Reduction in the Severity of a Drop: The applicant 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, the applicant prescribed testing of the lifting system before operation of the site transporter in the GROA. A dynamic load test over the full range of the lift using a test weight at least equal to 110 percent of the lift weight will be conducted to provide assurance against premature failure of the lifting members. The applicant also referenced an industry standard—ASME NQA–1–2000 Subpart 2.15 (American Society of Mechanical Engineers. 2000aa)—that will be followed and that addresses hoisting, rigging, and transporting of items. In addition, the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Protection Against Sliding Impacts: The applicant credited the site transporter with limiting the frequency of sliding impact of the site transporter into a wall and inducing stresses on the waste package due to seismic events. The applicant defined operating clearances and energy-absorbing features (SAR Table 1.2.8-2). In addition, the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Protection Against Tipover: The applicant credited the site transporter with the safety function of protecting against tipover. The applicant designed the site transporter with a wide base to prevent tipover. 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. The applicant mentioned 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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.

The applicant 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. The applicant 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

The applicant presented the nuclear safety design bases for the CTCTT and their relationship to the design criteria in SAR Table 1.2.8-2. The applicant provided specific design criteria to meet each of the nuclear safety design bases.

The applicant defined a design criterion that limits the maximum speed of the CTCTT, which is intended to protect against runaway. In addition, the applicant specified PSC–2 to the CTCTT in SAR Table 1.9-10 that requires the operator to secure the tractor before any waste handling operations are initiated. The applicant also specified that a fire suppression system onboard the vehicle precludes fuel tank explosions. Furthermore, the applicant specified a criterion for a maximum trailer height for accepting the cask that reduces the severity of a drop. Finally, the applicant indicated that cask puncture is precluded by limiting the hydraulic forces 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

The applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analysis

Protection Against Runaway: The applicant 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. The applicant 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. The applicant also indicated that it will use ANSI/ITSDF B56.9–2006 Section 7.13 (DOE, 2009ez) to design the system.

NRC Staff Evaluation: The NRC staff reviewed protection against runaway using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Preclusion of Fuel Tank Explosion: The tractor design is also credited with the safety function of precluding fuel tank explosion. The applicant 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). The applicant described the fuel tank explosion in its responses to NRC staff's RAIs (DOE, 2009ez,fa).

NRC Staff Evaluation: The NRC staff reviewed the preclusion of fuel tank explosion using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Reduction in Severity of a Drop: The applicant credited the cask trailer with preventing a cask drop from a height of more than 1.8 m [6 ft]. The applicant indicated that the design is based on ANSI/ITSDF B56.8–2006 and ANSI N14.30–1992 (American National Standards Institute, 1992aa).

NRC Staff Evaluation: The NRC staff reviewed reduction in severity of a drop using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Cask Puncture Prevention: The cask transfer trailer precludes transportation cask puncture due to a collision or to hydraulic ram operation. The applicant 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. The applicant further indicated that the inherent toughness of the casks provides the necessary puncture resistance. In addition, the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.5.4 Site Prime Movers

The applicant 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.

The applicant 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. The applicant provided additional design envelope information on the site prime movers (DOE, 2009ez). The applicant indicated that the dimensions of the legal-weight truck or overweight truck cask trailer are 2,591 mm [102 in wide] and 16.2 m [53 ft] long. The 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

The applicant presented the nuclear safety design bases for the prime mover and their relationships to the design criteria in SAR Table 1.2.8-2. The applicant provided specific design criteria to meet each of the nuclear safety design bases, along with controlling parameters and

bounding values. The applicant also provided two design criteria for addressing the safety design bases.

First, to protect against runaway, the applicant defined a speed limit requirement of 14.5 km/hour [9 mph] or less. Secondly, to preclude fuel tank explosion inside a surface facility, the applicant stated that the site prime movers should be restricted to outdoor operations only.

NRC Staff Evaluation: The NRC staff reviewed the design bases and design criteria using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodology

The applicant stated that the site prime mover design is based on NRC regulation and 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).
The applicant indicated 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. The applicant will also utilize other applicable sections of the code, such as 49 CFR 571.106 that provides guidelines for the design of the brake hoses and 49 CFR 571.301 that provides guidance on design for fuel system integrity (DOE, 2009ez).

NRC Staff Evaluation: The NRC staff reviewed the design methodology using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analyses

Runaway Prevention: The applicant specified a design criterion for reducing runaway probability by limiting the maximum vehicle speed to 14.5 km/hour [9 mph] when traveling in the GROA and 4.42 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. The applicant 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, the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Protection Against Fire Explosion: The applicant credited the site prime movers with the safety function of precluding fuel tank explosion. The applicant designed the site prime movers with limited fuel capacity {tank size of 378 l [100 gal] of diesel fuel} and protection against fire and explosions (SAR Section 1.2.8.4.3.1; DOE, 2009fa). The applicant indicated that fuel tanks that preclude explosions 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. The applicant further indicated that application of U.S. Department of Defense (DOD) guidelines ~~in DOD~~ (2006aa) is conservative because it applies to environments more severe than those expected at the Yucca Mountain repository.

The applicant also 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.6 Electrical Power Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant 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 required safety functions. The ITS AC, DC, and UPS subsystems are described in SER Section 2.1.1.2, and the performance of these subsystems is evaluated in SER 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 determines whether the description and discussion of the proposed ITS electrical power system design for both surface and subsurface operations adequately describe the relationship between the proposed design criteria and performance requirements for the ITS electrical power system, and the relationship between the design bases and the design criteria, in accordance with the requirements of 10 CFR 63.112(f). In addition, the NRC staff evaluated the information to determine whether the applicant adequately addressed the requirements of 10 CFR 63.21(c)(2) and 63.21(c)(3), which require that the applicant provide information relative to the codes and standards proposed for design and construction of the ITS electrical power system and the design bases for the ITS electrical power system and their relationship to proposed design criteria.

The ITS electrical power system provides redundant, independent, and physically separate trains (Trains A and B) of power distribution to designated facilities and loads within the GROA. The ITS electrical power system normally derives operating power from commercial offsite power connections. When detecting a loss of offsite power (LOSP), the ITS electrical power system automatically disconnects itself from the offsite power grid source and begins to automatically start and load the onsite ITS diesel generators. This action supplies power to designated electrical facilities and loads that rely on the ITS electrical power system to meet the facility safety requirements and performance objectives. Onsite ITS electrical power system diesel power generation, distribution, and control are provided by SSCs in the EDGF, CRCF, WHF, and interconnecting power transmission facilities. During an LOSP, both ITS diesel generators are started and maintained in a running mode to maintain the redundant features of the ITS electrical power system. Either of the ITS electrical power system trains (A or B) can provide the full power needed to perform the safety functions, thus allowing one of the two trains to be disabled for maintenance while still meeting the required facility performance objectives and required ITS safety functions.

The SAR described that major ITS loads powered by the redundant ITS electrical power system trains, such as ITS HVAC SSCs, are also redundant, independent trains. Each of the redundant ITS HVAC trains receives power from a dedicated ITS electrical power system train. Normally, one of the redundant ITS electrical power system load trains (e.g., one of the ITS HVAC trains) is operational while the alternate ITS load train is in standby mode. When an operational ITS load train fails (e.g., an operating ITS HVAC fan stops running for any reason) or the respective ITS electrical power system train fails to supply power to the operational ITS load train, the standby ITS load train is automatically engaged. This action automatically provides continued operation of the associated safety function.

Certain ITS SSCs require reliable and timely electrical power to perform or contribute to the performance of a mitigative safety function. An example of a mitigative safety function, as referenced in SAR Section 1.4.1.2.1, is the required continued operation of the ITS HVAC SSCs to provide confinement, filtering, and removal of airborne radioactive particles for primary confinement areas within spent fuel handling facilities, thereby mitigating the effects of a radionuclide release event. Another safety function of the ITS HVAC SSCs is to provide cooling to the rooms housing critical ITS electrical equipment, maintaining the reliability of the ITS electrical power system and continually assuring the confinement and filtering role of the HVAC SSCs.

However, many ITS SSCs were to be designed to fail in a "safe condition" in the event of an LOSP; therefore these ITS SSCs and most non-ITS electrical loads within the GROA are to be powered by the non-ITS normal electrical power system rather than the ITS electrical power system. The applicant also stated that there are no ITS SSCs requiring ITS electrical power in the subsurface operations areas of the GROA.

Design Bases and Design Criteria

The applicant listed the nuclear safety design bases and their relationships with design criteria for the ITS electrical power system in SAR Tables 1.9-3 and 1.9-4 and presented the information for the ITS electrical power system in SAR Table 1.4.1-1. The applicant provided specific design criteria to meet the nuclear safety design bases.

The applicant 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 on the IEEE standards for electrical power systems for nuclear facilities as listed by the applicant, 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, the applicant 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.

The applicant 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 the applicant. Battery-operated UPS SSCs are required within the CRCF, WHF, and EDGF to provide continuous AC power for these SSCs.

Adequate 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's RAIs (DOE, 2009fb,fc) on design bases and design criteria and bounding limits for this ITS electrical power system performance requirement, the applicant indicated that the required reliability is the same as for the ITS electrical power system SSCs supplying power to ITS HVAC systems in the CRCF and WHF.

For each of the design criteria the applicant 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 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.

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

The applicant stated that it will design the ITS electrical power system in accordance with industry codes and standards, with exceptions, 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, the applicant 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's RAI (DOE, 2009fc), following Regulatory Guide 1.9 (NRC, 2007ag).

The applicant 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, 2005~~aa,~~ab,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, the applicant 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). The applicant 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).

The applicant classified the ITS electrical power system distribution and transmission cable raceway as non-ITS and described the implementation of the raceway as composed of a number of physical configurations, ranging from open electrical trays and boxes to an underground duct bank system, all to be designed in accordance with applicable criteria within NFPA 70. The applicant stated that it will establish ampacity ratings for ITS electrical power system cable and conductor SSCs in accordance with IEEE 835–1994 (Institute of Electrical and Electronics Engineers, 1994aa) and NEMA WC 51–2003 (National Electrical ~~Manufactures~~Manufacturers Association, 2003aa).

The sizing of ITS electrical power system distribution cabling is to be based on 125 percent of the full anticipated load current at a 100 percent load factor. The applicant described the use of design enhancements for the cable raceway supporting the ITS cables associated with each of the redundant ITS electrical power system trains. Separation between non-ITS and ITS cables and between redundant ITS cables would be provided to prevent adverse interactions from faults or fires. The applicant plans to design the ITS cable raceway enhancements in accordance with IEEE 384–1992 and IEEE 379–2000 (Institute of Electrical and Electronics Engineers, 2001ab, 1998aa). The applicant also described its equipment qualification program for all ITS electrical power system components, including wiring and cabling, in SAR Section 1.13.

The design description in the SAR and supplemental materials presented a design that does not adhere to established appropriate codes and standards for nuclear safety systems. Important design attributes for safety systems such as defense in depth, redundancy, isolation, independence, diversity, single-failure criterion, and component qualification have not been demonstrated for portions of the ITS electrical power system that are relied upon to provide or support identified safety functions.

(b)(5)

(b)(5) In the description of its intended compliance with the terminology used in the identified industry codes and standards, for example, the applicant stated: "In lieu of a single failure criteria, the PCSA identifies the necessary redundancy and diversity to achieve the required reliability of ITS SSCs. A risk-informed PCSA defines both ITS SSCs and the nuclear safety design bases necessary to achieve categorization in compliance with 10 CFR Part 63. As detailed design progresses, specific items that might require application of redundancy and/or diversity to achieve compliance with the nuclear safety design bases will be identified."

(b)(5)

In response to an NRC staff's RAI (DOE, 2009dl), the applicant illustrated numerous proposed alternatives for key requirements of the cited industry codes and standards affecting system reliability. The applicant's design methodology incorporates a partial set of the design criteria reflected by the cited codes and standards. The applicant has applied criteria contained in 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. Predicted component failure rates the applicant used in the PCSA were, in many cases, derived from historical component reliability data for components and applications that were not demonstrated to be applicable to the proposed design. The results of this iterative process influence the outcome of the classification of SSCs as ITS and has shaped the selection and consistency of proposed design implementation and attributes (e.g., component qualification requirements; the ability of the electrical power system to tolerate a single failure within a safety train; the extent of using redundancy to achieve reliability, independence, isolation, diversity, and maintenance/surveillance planning) for the design of ITS electrical power system SSCs. (b)(5)

(b)(5)

NRC Staff Evaluation: The NRC staff reviewed the applicant's design methodology using the guidance in YMRP Section 2.1.1.7 to verify that design criteria adequately consider the PCSA results and confirm that the codes and standards to be used in the design and construction of facility ITS electrical power systems have been identified. (b)(5)

(b)(5)

Design and Design Analyses

The applicant described the power supply requirements 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. The applicant also provided sufficient information for the ITS diesel generators and their associated ITS mechanical support and 13.8-kV distribution systems to adequately 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.

This design configuration is a redundant centralized ITS electrical power system power source (commercial power or diesel generators) and distribution system that provides power through multiple dedicated (Train A electrical power system to Train A HVAC; Train B electrical power system to Train B HVAC) electrical connections and physical power flow paths to respective ITS SSCs (electrical power system and HVAC) in the previously listed facilities. The applicant stated that no means or procedures are planned within the proposed GROA design for providing power to any of the Train B ITS electrical power system or ITS HVAC SSCs from the Train A ITS diesel generators and/or main switchgear. A similar relationship exists for Train B ITS electrical power system and Train A ITS electrical power system and ITS HVAC SSCs.

If one of the ITS electrical power system trains (e.g., Train A) becomes unavailable due to maintenance or for any other reason, all the CRCFs, WHF, EDGF, and (non-ITS) RF must depend on Train B ITS electrical power system and Train B HVAC to fulfill identified confinement and cooling/ventilation functions. In this example, a fault occurring in ITS electrical power system Train B SSCs providing power connections to or residing within (or HVAC Train B SSCs residing within) any of the affected facilities would result in loss of ITS HVAC availability for some or all facilities described previously (depending on the location of the fault) until the Train A ITS electrical power system or the Train B ITS electrical or HVAC fault is corrected.

The design description information provided in the SAR and supplemental materials includes a number of areas where specific design criteria, methods, and/or compliance with (or exceptions to) applicable established industry codes, standards, and practices have not been fully described or justified, or the descriptions contain insufficient or inconsistent information. The design of ITS power distribution/connection beyond main ITS switchgear, ITS load centers, and ITS motor control centers were not described in the SAR. In response to an NRC staff's RAI (DOE, 2009fc), the applicant indicated that specific ITS power flow and control components will be identified as detailed design proceeds 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 the applicant's design and design analysis using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.7 Instrumentation and Controls

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant provided information on ITS I&C equipment in SAR Sections 1.2.3, 1.2.4, 1.2.5, 1.2.6, 1.2.8, 1.3.3, 1.4.2, and 1.9.1 (DOE, 2008ab) to 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 accident sequences. The NRC staff's reviewed the SAR sections mentioned previously to determine whether they adequately describe the design criteria used for ITS I&C equipment and their relationships to the preclosure performance objectives, and the design bases for the ITS I&C equipment and their relationship to the design criteria. In addition, the NRC staff evaluated the information to determine whether the applicant adequately addressed the requirements of 10 CFR 63.21(c)(2) and 63.21(c)(3), which require that the applicant provide information relative to the codes and standards proposed for design and construction of the ITS I&C SSCs and the design bases for the ITS I&C SSCs and their relationship to proposed design criteria.

Design Bases and Design Criteria

In its SAR, the applicant provided its general control philosophy for the GROA, stating that repetitive operations will utilize automation to support the facility operators. To facilitate this, the applicant 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 DCMIS to convey the normal (non-ITS) control and monitoring commands and signals between the local sensors and controllers to the HMI consoles located in each surface facility and the CCC. 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.

The applicant 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 DCMIS cannot override safety functions.

The applicant described the intended normal operations, safety functions, and applicable design criteria associated with these ITS controls within the descriptions of the various electro-mechanical SSCs. In SAR Section 1.4.2, the applicant indicated that all ITS controls 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.

The applicant 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.

The applicant described the ITS I&C equipment functions needed to ensure the SCCs achieve the safety functions and described the design bases and design 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.

The applicant presented the nuclear 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. The applicant 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 as required by 10 CFR 63.112 for inclusion in the PCSA were described in SAR Sections 1.9.1.1 through 1.9.1.13.

To support the evaluation of these nuclear safety design bases, the NRC staff performed a "vertical slice" review of the design and intended CRCF operations to ascertain how potential event sequences are prevented or mitigated such that the GROA performance objectives will be met. Specifically, CRCF event sequences ESD18–DSTD–SEQ2 and ESD18–TAD–SEQ2 were reviewed to help evaluate ITS I&C equipment design to gain an understanding of potential interactions with material handling equipment associated with equipment shield door and cask port slide gate operations. In conjunction with the ITS Controls review needed to support the continued ITS electrical power system operation, which is needed to support the

safety function of HVAC confinement and electrical equipment cooling capability for the CRCF following a radionuclide release, ESD09–HLW–SEQ3–RF and ESD09–TAD–SEQ3–RRF were examined. The NRC staff examined other event sequences to gain an understanding of potential challenges to the ITS I&C functions if they needed to perform during a temporary loss of power condition.

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 the applicant identified serve in applications within the preclosure 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.

In SAR Sections 1.2.4, 1.4.1 and 1.13, the applicant 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).

NRC Staff Evaluation: The NRC staff reviewed the design bases and the relationship between the design bases and design criteria using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

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. The stage of the design presented in the application is conceptual. The applicant identified the principal industry design codes and standards it intends to use when completing the detailed design of the ITS controls. However, the applicant took several exceptions to the application of key design criteria in nuclear safety industry codes and standards. The applicant justified this practice, citing the use of a risk-informed PCSA process to identify where criteria, such as redundancy, diversity, and single-failure proof design, are appropriate.

In response to the NRC staff's RAI (DOE, 2009do) related to ITS controls, the applicant 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. The applicant 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 this response, the applicant 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, the applicant identified several notable exceptions to specific clauses within these safety codes.

In the description of its intended compliance with the terminology used in the identified industry codes and standards, for example, the applicant stated: "In lieu of a single failure criteria, the PCSA identifies the necessary redundancy and diversity to achieve the required reliability of ITS SSCs. A risk-informed PCSA defines both ITS SSCs and the nuclear safety design bases necessary to achieve categorization in compliance with 10 CFR Part 63. As detailed design progresses, specific items that might require application of redundancy and/or diversity to achieve compliance with the nuclear safety design bases will be identified."

For many ITS controls applications, the applicant consistently took exception to the applicability of key design criteria contained in applicable industry codes and standards advocating that a minimum of two safety groups (e.g., redundancy and independence of SCs in the design) be used to provide added reliability commensurate with the potential risks (e.g., inadvertent direct exposure of facility personnel to highly radioactive waste packages). The applicant provided for redundancy where the code explicitly requires it (e.g., 10 CFR 63.112 (e)(12) pertaining to the consideration of means to provide redundant systems necessary to maintain, with adequate capacity, the ability of utility services ITS), but largely did not provide for redundancy and independence as an important design consideration in the application of ITS controls, as depicted in applicable nuclear safety industry codes and standards. Further, the applicant took exception to the use of the single failure criterion identified within applicable nuclear industry safety design standards as an important design consideration to enhance the reliable achievement of important safety functions. In responses to the NRC staff's RAI regarding these design criteria exceptions (DOE, 2009do), the applicant cited the results of calculations of the likelihood of event sequence occurrence within the PCSA to justify this exception.

Using event tree analyses as input to the event categorization process, the applicant 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) based on the 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 the applicant believes it to be an industry standard data source. The applicant 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 event sequence analyses describe potential failures occurring in conjunction with other adverse actions (e.g., undesired operator action), regardless of the type of equipment used (e.g., limit switch, pressure switch, proximity detector, temperature sensor, radiation detectors and amplifiers) for the sensing portion of an interlock.

NRC Staff Evaluation: The NRC staff used the guidance in YMRP Section 2.1.1.7 to review the applicant's proposed methodology for determining appropriate design criteria to be applied to the design of the ITS controls design described in its SAR, (b)(5)

(b)(5)

Design and Design Analyses

The applicant 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 description of the NRC staff's evaluation and due to similarity in design methods used, NTC staff broadly categorizes these 29 groups in this SER into three types of ITS control applications as discussed next. For each group, the NRC staff reviewed the applicant's text descriptions and piping and instrumentation diagrams and logic diagrams within the SAR to determine whether (i) it is apparent that the safety functions would be performed as intended with respect to the I&C SSC and (ii) the design criteria contained within the applicable nuclear industry codes and standards have been appropriately applied.

Doors, Materials Handling Cranes, and WPTT

The cask port slide gate 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 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 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.

- Protect against a load drop by ensuring that power to the CTM hoist motor is shut off if the "no final hoist upper limit" switch or "no rope mis-spool" switch trips.

- Limit drop height by preventing hoist raising/lowering without safety permissives, limit lift heights, and require a "grapple engaged" signal to allow the load to be lifted further.

- 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.

- 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 YMRP Section 2.1.1.7 to review information the applicant provided on the design and design analyses of I&C for doors, cranes, and the WPTT. (b)(5)

(b)(5)

(b)(5)

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.

- <u>Mitigate the Consequences of Radionuclide Release</u>

SAR Section 1.2.4.4.1 stated that the CRCF surface nuclear confinement HVAC is designed to limit the release of radioactive contaminants to protect workers and the public. The specific safety function the applicant identified is 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.

The applicant proposed two independent trains (A and B). ITS interlocks are 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 (Trains A and Trains B).

The specific safety functions the applicant identified follow.

— Failure of Train A initiates the standby fan ASD fan coil unit fan for Train B, ensuring electrical room fan coil unit fans are running for each train (A and B). ITS Train B operates exactly like Train A.

— ASD fan coil unit motors (operating and standby for each Train A and B electrical room) are interlocked with the following components:

  - Fan Coil Unit Discharge Flow Not Low (FSL–1039–A, FSL–1039–B,

FSL–1049–A, and FSL–1049–B)

- o Fan Coil Unit Differential Pressure Low (PDSL–1031–A, PDSL–1031–B, PDSL–1041–A, and PDSL–1041–B)
- o Sequencer Start (YS–1030-A, YS–1030-B, YS–1040-A, and YS–1040-B)
- o Fan Coil Unit Failure or Tripped (Output of ITS Logic)

— The ITS direct expansion fan coil unit provides coding and maintains temperature with remote air-cooled condensing units for each of the Train A or Train B ITS electrical and battery rooms. The fan coil unit is rated sufficient to maintain normal room temperature. The refrigerant compressors in the condensing units are controlled by a signal from the temperature sensor/ transmitter located in the electrical room or the battery room (whichever is higher) as determined by a signal selector.

— Failure of one train initiates the standby ASD for the battery room exhaust fan, ensuring each train (A and B) battery room exhaust fans are running.

— The ASD battery exhaust fan motors (operating and standby) are interlocked with the following components:

- o Fan Discharge Flow Not Low (FSL–1179–A, FSL–1179–B, FSL–1191–A, and FSL–1191–B)
- o Fan Differential Pressure Not Low (PDSL–1175–A, PDSL–1189–A, PDSL–1175–B, and PDSL–1189–B)
- o Sequencer Start (YS–1188–A, YS–1188–B, YS–1178–A, and YS–1178–B)
- o Battery Room Fan Failure or Tripped (Output of ITS Logic)

— The battery room exhaust operates continuously with sufficient volume changes per hour to preclude accumulation of hydrogen. A redundant, explosion-proof exhaust HEPA assembly is provided for each battery room with the standby unit starting automatically when either of the following occurs: (i) low differential pressure across the operating fan coincident with low air flow (fan failure) or (ii) fan trip.

(b)(5)

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on the design and design analyses of I&C for the CRCF ITS HVAC SSCs using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Diesel Generator (ITS Diesel Generator and its Mechanical Support Systems)

The SAR stated that two independent ITS diesel generators (Trains 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.

- Provide ITS Electrical Power

  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.

  The applicant proposed two independent, 100 percent load diesel generators. The applicant responded to the NRC staff's 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's 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.

  In SAR Section 1.4.1.2.1, the applicant described a solid-state-type undervoltage device for sensing a loss of normal 13.8-kV power into the EDGF feeder bus. The applicant further stated that logic is provided so the incoming breaker can trip on undervoltage logic signals. (b)(5)

  (b)(5)

  (b)(5)

  — Each ITS diesel generator is required to have ITS support systems that are electrically and physically independent from the support systems for the other ITS diesel generator.

  The applicant stated that the design provides for physical and electrical separation between the ITS mechanical support systems for the two ITS diesel generators.

  (b)(5)

  (b)(5) There are ITS controls used in the mechanical

support systems, including the ITS Diesel Fuel Oil System, the ITS Diesel Starting Air System, and the ITS Diesel Generator Jacket Water Cooling System. The applicant provided the principal standards and codes for ITS diesel generators and associated supporting systems.

- Each ITS diesel generator fuel oil storage tank is required to be sized for 14 days of continuous operation and be capable of online refueling.

  SAR Section 1.2.8.2.1 stated that there are two independent, underground diesel fuel oil storage tanks, two redundant diesel fuel oil transfer pumps, and a diesel fuel oil day tank for Train A. SAR Figure 1.2.8-17 depicted the engine-mounted fuel oil pump and the connections to the engine. SAR Figure 1.2.8-18 depicted the EDGF ITS diesel fuel oil system Train A piping and instrumentation diagrams. SAR Figure 1.2.8-19 depicted the ITS diesel generator fuel oil transfer pump logic diagram. (b)(5)

  (b)(5)

NRC Staff Evaluation: The NRC staff reviewed the applicant's information on the design and design analyses of I&C of the EDGF ITS I&C and interlock systems using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.8 Fire Protection Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant 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, the applicant 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. The applicant 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 SER Section 2.1.1.2, and the ability of the fire protection system to perform its intended safety functions is reviewed in SER 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.

The applicant 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.

The applicant indicated in its response to an NRC staff's 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. The applicant 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. (b)(5)

(b)(5)

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 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).

(b)(5)

Design Methodologies

The applicant 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 YMRP Section 2.1.1.7 (b)(5)

(b)(5)

Design and Design Analysis

The applicant 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, the applicant provided responses to the NRC staff's RAI in DOE Enclosures 1 and 2 (2009fg) to supplement the design information presented in the SAR.

Although specific design details were not provided in the SAR or the RAI responses, the applicant noted that an approved testing agency will specifically list and label DIPA system equipment 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.9 Canisters and Overpacks

To demonstrate compliance with 10 CFR 63.21(c)(2) and (c)(3), and 63.112(f), the applicant 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 SER Section 2.1.1.2, and their performance is evaluated in SER Section 2.1.1.4.

2.1.1.7.3.9.1 Waste Package

The applicant described the waste package design in SAR Sections 1.5.2, 1.2.1.4.1, 1.2.4.2.3.1.3, 1.3.1.2.5, and 2.3.6.7.4 and Tables 1.5.2-6 and 1.5.2-7. The 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 high level waste and DOE SNF, or a short or long naval canister with naval SNF (see SER Section 2.1.1.2.3.5.1 for more information on waste package configurations). 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

The applicant presented the nuclear safety design bases for the waste packages and their relationship with the design criteria in SAR Table 1.5.2-6. The applicant also provided the specific design criteria for each of the design bases, along with controlling parameters and bounding values.

The applicant provided several design criteria for the safety design bases to (i) provide containment for a sealed waste package for an event sequence resulting from an impact, a drop of a load onto the waste package, or the spectrum of fire and (ii) protect against waste package breach waste package from a rock or vibratory ground motion impacts. These 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 as a mean conditional probability or a mean frequency of breach are imposed on each design basis.

In addition, the design criteria to address design bases for the important to waste isolation waste packages 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

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. The applicant presented the structural design methodologies, including the analyses performed for various load combinations (normal loads and event sequences loads), and applicable acceptance criteria in SAR Tables 1.5.2-8 and 1.5.2-9. The thermal design methodologies for the waste package were addressed by performing parametric studies to analyze waste package response to accidental fires.

The applicant 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. The applicant stated that, for the normal loads, the 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 SER Section 2.2.1.3.1.3.3. (b)(5)

(b)(5)

The applicant used elastic-plastic finite element analyses and analytical methods to evaluate waste package performance for event sequence loads. The applicant 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. The applicant stated that, for event sequence loads, the 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analyses

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"

The applicant presented the waste package fabrication materials and process in SAR Section 1.5.2.7.1 and described the fabrication procedure and the welds for the waste package in SAR Figure 1.5.2-11. In addition, the applicant described the methods for the closure welds in SAR Section 1.2.4.2.3.

The 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 specifically included as a corrosion-resistant component to address postclosure requirements and is not a pressure vessel (BSC, 2007bi).

In SAR Section 1.5.2.7.3, the applicant 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.

Using these methods, the applicant analyzed three waste package configurations (TAD canister, DOE short codisposal canister, and naval long canister). In response to an NRC staff's RAI (DOE, 2009er), the applicant stated that the remaining three waste package configurations (5-DHLW/DOE long codisposal, 2-MCO/2-DHLW codisposal, and naval canistered SNF short waste package) will be subject to similar analyses and the final qualifications of the waste package design to be included in proposed administration controls.

The applicant 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 the applicant'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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

The NRC staff also reviewed the information provided regarding proposed fabrication materials, fabrication processes, and closure methods for the waste package. (b)(5)

(b)(5)

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. (b)(5)

(b)(5)

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. The applicant 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. (b)(5)

(b)(5)

Commitment

The applicant commits to perform structural and thermal analyses for the 5-DHLW/DOE long codisposal, 2-MCO/2-DHLW codisposal, and naval canistered SNF short waste package configurations for the repository operations and permanent disposal using the structural design methodology described in SAR Section 1.5.2.1.1. This commitment should be completed for NRC staff review prior to receiving and possessing waste (DOE, 2009er).

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.9.2 Transportation, Aging, and Disposal Canister

The applicant 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 loaded into a waste package.

The applicant 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. The applicant also indicated that the TAD canister is designed to withstand the natural phenomena listed in SAR Table 1.2.2-1 and horizontal and vertical ground motion shown in SAR Figures 1.2.2-8 to 1.2.2-13.

Design Bases and Design Criteria

The applicant 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). The applicant 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 the applicant 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, the applicant identified the TAD canister design criteria and design bases. The applicant 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, the applicant 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 YMRP Section 2.1.1.7.] (b)(5)
(b)(5)

Design Methodologies

In SAR Section 1.5.1.1.1.2.6, the applicant presented performance specifications and methodologies that should be used to design the TAD canister such that it will meet the performance specifications. The applicant focused on two parameters: leakage rate and fuel cladding temperature. The applicant 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). The applicant 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 methodology for the TAD canister using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

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.

The applicant 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, the applicant 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]. The applicant 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 within the context of 10 CFR Part 72 to address the qualification of final closure welds on austenitic stainless steel canisters. The applicant 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.

The applicant 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). The applicant 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, the applicant 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).

The applicant stated in SAR Section 1.5.1.1.1.2.8 that the materials, design, fabrication, testing, and examination of the TAD canister shall meet the 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 YMRP Section 2.1.1.7. (b)(5)
(b)(5)

The NRC staff reviewed the characteristics and design specifications of the criticality-significant components of the TAD canister. (b)(5)
(b)(5)

The NRC staff reviewed the specification the applicant identified for drying and inerting of the TAD canister. (b)(5)
(b)(5)

The NRC staff reviewed the material specifications and restrictions for TAD canister construction (b)(5)
(b)(5)

The NRC staff also reviewed the design codes and standards to be used for the TAD canister design and construction. (b)(5)
(b)(5)

NRC Staff Conclusion

(b)(5)

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

The applicant 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)].

Design Bases and Design Criteria

The safety function of the DOE standardized canister is to provide containment of radioactive materials. In SAR Section 1.5.1.3.1.2.5 the applicant provided the design criteria and design bases for the DOE standardized canisters, with the nuclear safety design bases given in SAR Table 1.5.1-25.

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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodologies

In SAR Section 1.5.1.3.1.2.6.1, the applicant provided the overall methodology used for DOE standardized canister design. The applicant 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), the applicant 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. The applicant 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 an 457-mm [18-in] diameter standardized canister.

NRC Staff Evaluation: (b)(5)

(b)(5) On the basis of the guidance in YMRP Section 2.1.1.7, (b)(5)

(b)(5)

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. The applicant 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. The applicant 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 the applicant 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 the applicant 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, the applicant used a through-wall strain limit of 48 percent as the failure criteria. The applicant 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, the applicant 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 YMRP Section 2.1.1.7.

(b)(5)

The NRC staff also reviewed the applicant's approach for evaluating the DOE canisters' capacity to withstand possible repository structural challenges. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

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 the applicant provided the design criteria and design bases for the HLW canisters. The nuclear safety design bases and the design criteria for the HLW canisters were given in SAR Table 1.5.1-17.

On the basis of the PCSA results, the canister must provide containment when it is subjected to structural and thermal challenges. The applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

Design Methodologies

The applicant 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 the applicant stated that several full-scale vertical, top, and corner drop tests from a height of 7 m [23 ft] were performed

to evaluate the structural design of these canisters. The applicant stated that for all tests, the HLW canister did not breach.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design methodology for the HLW canister using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analyses

In SAR Table 1.5.1-16, the applicant provided geometric details of the four HLW canisters from Hanford, Idaho National Laboratory, Savannah River, and West Valley. 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). The canister welding procedures will follow the industry-accepted requirements 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 the applicant's information on design of HLW canisters using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.9.3.3 Dual-Purpose Canister

The DPC is designed to store CSNF at the utility sites and could also be used to ship the SNF. The DPC can be placed in an aging overpack for aging; however, the current DPC design is not appropriate for disposal. Therefore, SNF 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 the applicant briefly discussed the DPC; however, no DPC design details were given in the SAR.

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, the applicant 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 YMRP Section 2.1.1.7. (b)(5)
(b)(5)

Design Methodology

In SAR Section 1.5.1.1.1.2.1.2, the applicant 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, the applicant stated that similar structural analyses have been performed on generic canisters (BSC, 2008cp). The applicant 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. The applicant used the results in quantifying an estimate of the passive reliability for a generic canister.

NRC Staff Evaluation: The NRC staff reviewed the information provided regarding design methodology for the DPC using the guidance in YMRP Section 2.1.1.7. The NRC staff reviewed the structural analyses the applicant performed for a generic canister, for which geometrical and material properties were based upon typical DPC and naval canisters. The NRC staff reviewed the applicant's approach for evaluating the generic canister capacity to withstand possible repository structural challenges. (b)(5)
(b)(5)

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. The applicant stated in SAR Section 1.5.1.1.1.2.1.2 that if the selected DPC falls within the design envelope of the 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Commitment

(b)(5)

NRC Staff Conclusion

(b)(5)

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]. The applicant stated that the maximum external dimensions ensure that the naval SNF canisters fit into the waste packages. The maximum design weight of the loaded long or short naval SNF canister is 44,452 kg [98,000 lb]. The 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 the applicant 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.

The applicant 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 has reviewed the relationship between the design bases and design criteria of the naval canister using guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

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. The applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analyses

In SAR Section 1.5.1.4.1.2.8 the applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

(b)(5)

2.1.1.7.3.9.3.5 Aging Overpack

In SAR Section 1.2.7.1, the applicant 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 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

The applicant 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 the applicant 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.

The applicant 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 peak ground acceleration 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. The applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodology

In SAR Section 1.2.7.6.2, the applicant stated that before an aging overpack is used at the GROA, the aging overpack system 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, the applicant 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). The applicant stated that, for DPCs contained in an aging overpack, the maximum canister leakage rate is equal to its design value.

The applicant 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 a spent nuclear fuel canister is presented in BSC (2008cp). The structural analysis focused on impact events including drop onto unyielding ground and slapdown from an upright position. The applicant 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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

The NRC staff also reviewed the applicant's approach for evaluating the aging overpack's structural capacity as given in BSC (2008cp). (b)(5)

(b)(5)

In addition, in SAR Section 1.2.7, the applicant committed to evaluate any specific aging overpack system for normal handling, dead, and thermal loads and loads that result from event sequences prior to use at the Yucca Mountain repository.

Design and Design Analyses

In SAR Section 1.2.7.7 the applicant provided general material specifications for the fabrication of the aging overpack system. The applicant stated that the vertical aging overpack will be constructed of a metal liner, which is surrounded by reinforced concrete sidewalls and a top. The applicant 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).

The applicant 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, the applicant 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, the applicant 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 YMRP Section 2.1.1.7. The NRC staff reviewed the cited design codes and standards to be used for the design and construction of the aging overpack system. (b)(5)
(b)(5)

Commitment

The applicant commits to evaluate any specific aging overpack system for normal handling, dead, and thermal loads and loads that result from event sequences prior to use at the Yucca Mountain repository (SAR Section 1.2.7). This commitment should be completed for NRC staff review prior to receiving and possessing waste.

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.9.3.6 Transportation Cask

The applicant 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 the applicant identified the TAD transportation cask as an ITS component of the TAD canister system. The applicant listed preclosure nuclear safety design bases and criteria for the TAD transportation cask in SAR Table 1.9-2. The applicant provided performance specifications for the TAD transportation cask and the vertical aging overpack for a TAD canister (DOE, 2008ag).

Design Bases and Design Criteria

The applicant detailed 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.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design bases and design criteria for the transportation cask using guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design Methodology

In SAR Section 1.2.8.4.5.6, the applicant stated that the design methodologies used in the design of the transportation casks will be in accordance with codes and standards used to meet the performance requirements of 10 CFR Part 71.

As part of the design methodology, the applicant 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. The approach used for the transportation cask was presented in BSC (2008cp) in which structural analyses were presented for a transportation cask containing a representative SNF canister. The structural analyses focused on different drop/impact conditions. The applicant 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 relationship between the design methodology for the transportation cask using guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

The NRC staff reviewed the applicant's approach for evaluating the transportation cask's structural capacity as given in BSC (2008cp). (b)(5)

(b)(5)

Design and Design Analyses

The applicant has specified that the transportation cask will be certified under the requirements of 10 CFR Part 71 as an approved package for transportation. The codes and standards, materials of construction, and design load combinations will be based on the requirements of 10 CFR Part 71.

NRC Staff Evaluation: The NRC staff reviewed the relationship between the design and design analyses for the transportation cask using guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Commitment

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.3.10 Criticality Prevention and Shielding Systems

This section contains the NRC staff's review of the design of ITS systems to prevent and control criticality and provide shielding. The applicant provided this information in SAR Sections 1.14, 1.2.1 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

To demonstrate compliance with the requirements of 10 CFR 63.112(f), the applicant 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, the applicant 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, the applicant 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. The applicant'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. The applicant's criticality analyses evaluated changes to these parameters to provide input to the PCSA.

The applicant'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.

The applicant used a USL of 0.93 for CSNF and 0.89 for DOE SNF. This included an administrative margin of 0.05 (SAR Section 1.14.2.3.4.1). The applicant stated that it screened out Category 1 and 2 event sequences that could result in criticality. The applicant relied on the use of passive design features (physical barriers against introduction of moderation), engineered design features, and procedural controls to screen out criticality.

The applicant 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). The applicant 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 SER 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 SER 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, the applicant used design criteria where cranes have double retention capability and leaked oil could enter a breached canister (SAR Table 1.2.4-4). (b)(5)

(b)(5)

If a canister is breached, the fire protection system is one of the main sources that could introduce moderator into the canister. Thus the applicant 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 SER Section 2.1.1.7.3.8.

To prevent a canister from being crushed by a closing slide gate or equipment shield door, the applicant 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 SER 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. (b)(5)

(b)(5)

(b)(5) The equipment shield doors are further evaluated in SER Section 2.1.1.7.3.4.3.

Staging racks have design bases and design criteria to prevent criticality as discussed in SER 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 YMRP Section 2.1.1.7 to review the information regarding the design bases and design criteria for criticality prevention. (b)(5)

(b)(5)

Design Methodologies

The applicant described its methodology for meeting the requirements of 10 CFR 63.112(e)(6) by 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. The applicant stated that criticality is prevented through a combination of ITS SSCs and PSCs. The applicant did not analyze whether potential criticality dose consequences would meet the requirements of 10 CFR 63.111(a) and 63.111(b), because criticality was screened out as beyond Category 2.

For dry handling in the surface and subsurface facilities, the applicant relied primarily on moderator control to prevent criticality. The moderator of most concern was water. The applicant 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, the applicant identified several SSCs that are ITS because they were relied on to exclude moderator or prevent a canister breach. (b)(5)

(b)(5)

In the WHF pool, where water cannot be removed, the applicant relied primarily on neutron absorbers to control criticality. For wet handling operations, the applicant 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. The applicant 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, the applicant developed PSC-9, which requires operators to check the boron concentration.

NRC Staff Evaluation: The NRC staff reviewed the applicant's design methodologies for ITS SSC used to prevent criticality as discussed in the SAR and responses to the NRC staff's RAIs using the guidance in YMRP Section 2.1.1.7. (b)(5)

(b)(5)

Design and Design Analysis

Code Validation

The applicant 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, 2003ai, 2002ac) and BSC (2008cf). The applicant 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, the applicant 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, the applicant also provided the values and ranges for the CSNF

models. On the basis of the MCNP results, the applicant 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 the applicant 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), the applicant 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, the applicant 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, the applicant 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 the applicant provided using the guidance in YMRP Section 2.1.1.7 and Regulatory Guide 3.71 (NRC, 2005ac). (b)(5)

(b)(5)

(b)(5)

(b)(5)

Dry Handling

The applicant described the general characteristics of the canisters used in the GROA in SAR Section 1.5.1. Outside the WHF pool, the applicant relied on moderator and interaction control to prevent criticality. The applicant 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.

The applicant 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, the applicant discussed the criticality potential of the DOE standardized SNF canisters. The applicant described the eight combinations of canister, basket, and representative fuel that 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, the applicant concluded that the minimum canister spacing that would ensure subcriticality is 30 cm [12 in]. The applicant used this spacing in the DOE SNF staging rack design, discussed next. The applicant's screening arguments for criticality-initiating events, including interaction, are evaluated in SER Section 2.1.1.3.3.2.1.

The HLW containers are used for the vitrified (glass) waste. The applicant 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).

The applicant 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 SER Section 2.1.1.3.3.2.1. SER 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). The applicant'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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

(b)(5)

(b)(5)

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. The applicant 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). The applicant 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).

The applicant 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. The applicant found that it required 30 percent of the soluble boron concentration to prevent criticality (SAR Section 1.14.2.3.2.2.4).

The applicant 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). The applicant considered event sequences concerning drops and earthquakes during transfer operations into or out of the WHF pool that could modify the system geometry. The applicant stated that criticality could be prevented 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 methodology using the guidance in YMRP Section 2.1.1.7. (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.1.7.3.10.2 Shielding Systems

To demonstrate compliance with 10 CFR 63.21(c)(2), 63.21(c)(3), and 63.112(f), the applicant provided design information of 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. Other non-ITS shielding features that are relied upon to meet 10 CFR Part 20 requirements, such as concrete walls, floors, and ceilings, are evaluated in SER Section 2.1.1.8, in conjunction with the program for personnel access controls to meet 10 CFR Part 20 as low as is reasonably achievable requirements. The objective of the NRC staff's review is to verify the design of ITS shielding features.

Design Bases and Design Criteria

The applicant 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.

The applicant 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 guidelines in YMRP Section 2.1.1.7 with regard to the applicant's presentation of shielding design objectives. (b)(5)

(b)(5)

Design Methodologies

The applicant 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, the applicant 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 SER Section 2.1.1.8.3.3.

NRC Staff Evaluation: The NRC staff reviewed the applicant's methodology using the guidance in YMRP Section 2.1.1.7 (b)(5)

(b)(5)

(b)(5)

Design and Design Analysis

The main safety function of ITS shielding is to protect personnel from direct radiation exposure such that the performance requirements of 10 CFR 63.21(c)(2) and (3) and 63.112(f) are met, thus ensuring the 10 CFR 63.111(a)(1) requirement that the GROA meet the requirements of 10 CFR Part 20. The applicant discussed the analysis and design of the shielding features, including the calculation methodology, computer codes, and radiation sources, in SAR Sections 1.10.3.2, 1.10.3.3, and 1.10.3.4.

The shielding analyses for both ITS and non-ITS use the same data and assumptions. This is evaluated in SER Section 2.1.1.8 and was found to be acceptable. 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. The applicant proposed a two-step approach, using the proposed license conditions listed in SAR Table 5.10-1 that include limiting conditions for operations. The specific, proposed license specification involving ITS SSCs evaluated in this SER section involves the ITS radiation detectors and interlocks. This is intended to ensure that radiation detectors interlocked with ITS shield doors are operable to prevent inadvertent door opening if high radiation conditions from a waste package are present.

The second approach is for ITS SSCs that do not meet the criteria for limiting conditions for operations. The applicant will detail the controls necessary to assure adequate 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 license specifications. The applicant'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 YMRP Section 2.1.1.7. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.7.4 Evaluation Findings

The NRC staff has reviewed and evaluated the applicant's SAR and other information submitted in support of the license application, relevant to design of ITS SSCs and SCs for the GROA, (b)(5)

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(b)(5)

2.1.1.7.5 References

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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 applicant's description of its proposed RPP reflects as low as is reasonably achievable (ALARA) considerations of maintaining the occupational doses to workers, and doses to members of the public, to as far below regulatory limits as is practical, consistent with the purpose for which the licensed activity is undertaken.

The applicant has described the 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. The policy considerations the applicant used include its management commitment to maintain doses ALARA and the implementation of ALARA principles in the design process throughout the repository design and construction so that shielding design and structural loads are part of the design process. The applicant has also described the facility shielding design that is used to meet the ALARA requirements for normal operations and Category 1 event sequences. The applicant's implementation of ALARA principles into repository operations, including administrative controls to maintain doses ALARA and general operational guidelines, will be accomplished through its Operational RPP described in SAR Section 5.11.

2.1.1.8.2 Regulatory Requirements and Regulatory Guidance

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 that the applicant shall submit a description of its program for control and monitoring of radioactive effluents and occupational radiological exposures to maintain these in accordance with the requirements of 10 CFR 63.111. According to 10 CFR 20.1101(a), the applicant is required to develop, document, and implement an RPP commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR Part 20. 10 CFR 20.1101(b) requires the applicant to use, to the extent practical, procedures and engineering controls based upon sound engineering practices to achieve doses to members of the public and occupational doses that are ALARA. In implementing the ALARA requirements of 10 CFR 20.1101(b), the applicant is required by 10 CFR 20.1101(d) to establish a constraint on air emissions of radioactive material, excluding Radon-222 and its daughters, to the environment such that individual members of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent exceeding 0.1 mSv/year [10 mrem/year]. In addition, the applicant is required to control the annual occupational dose to individual adults to the dose limits specified in 10 CFR 20.1201(a).

The NRC staff reviewed the applicant'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 acceptance criteria used in the staff's review are as follows:

- An adequate statement of management commitment to maintain exposures to workers and the public ALARA is provided.

- ALARA principles are adequately considered in geologic repository operations area (GROA) design.

- Proposed operations at the GROA adequately incorporate ALARA principles.

- The 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)

Other supporting NRC guidance documents that the applicant referenced and NRC staff reviewed for applicability are listed in the References section of this chapter.

2.1.1.8.3 Staff Review and Analysis

In SAR Section 1.10, the applicant stated that to demonstrate compliance with the regulatory requirements of 10 CFR 63.111(a)(1), it addressed the acceptance criteria in YMRP Section 2.1.1.8.3 and other applicable sections. The applicant 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. To demonstrate compliance with 10 CFR 63.21(c)(6), the applicant submitted a description of its RPP for control and monitoring of radioactive effluents and occupational radiological exposures to maintain these in accordance with the requirements of 10 CFR 63.111. The Operational RPP was described in SAR Section 5.11. As indicated in the introduction of SAR Section 5.11, the RPP will be available for NRC review in connection with the consideration of an updated application for a license to receive and possess spent nuclear fuel and high-level radioactive waste. A detailed RPP is not necessary for issuance of a construction authorization, because there are no radioactive materials in the applicant's possession. If the applicant, U.S. Department of Energy (DOE), submits an updated license application for a license to receive and possess material, the NRC review will verify that the RPP meets the program description provided in the applicant's license application.

The NRC staff's review of the applicant's program description for implementing ALARA principles, including its RPP, is discussed next.

2.1.1.8.3.1 Management Commitment To Maintain Doses As Low As Is Reasonably Achievable

In SAR Section 1.10.1, the applicant described the management commitment to maintain doses ALARA in accordance with 10 CFR 20.1101(b). As a part of its commitment, the applicant stated it will control worker doses and releases of radioactive materials to the environment below regulatory limits to include constraints on air emissions with the exception of Ra-222 and its daughters as required in 10 CFR 20.1101(d). The applicant's management will support the ALARA policy through direct communication, instruction, inspection, and audit of the workplace. As indicated in the SAR, aspects of this commitment are the development of an ALARA program, implementation of an operational RPP, and personnel training.

During design and construction, the applicant will conduct ALARA-specific reviews to ensure ALARA principles are incorporated in the design. The applicant will conduct and document audits in accordance with the recommendations of NRC Regulatory Guide 8.8 (NRC, 1978ab). The applicant will estimate occupational doses in accordance with the recommendations of NRC Regulatory Guide 8.19 (NRC, 1979aa) and use construction inspections to verify that shielding features are installed as designed.

For operations, the applicant has made a management commitment to implement an operational ALARA program at the repository in accordance with NRC Regulatory Guide 8.8, Position C.1, (NRC, 1978ab). The operational program will implement ALARA principles in policies and procedures, goals, and objectives for planning, design, and construction of modifications to operating facilities, operating activities, maintenance, housekeeping, decontamination, and dismantlement. During the decommissioning and dismantlement of the repository surface and subsurface nuclear facilities (SAR Section 1.10.1.3), the applicant has committed to apply the ALARA principles by (i) reviewing prior radiation surveys to assess radiological conditions and (ii) performing visual inspections and radiation surveys to ensure that there are no unidentified radiation sources that might affect personnel exposures. The applicant committed to develop procedures for implementing ALARA principles in decommissioning and dismantlement activities. SER Section 2.1.3 provides the NRC staff's evaluation of the applicant's plans for permanent closure and decontamination or decontamination and dismantlement of the Yucca Mountain surface facilities (b)(5)

NRC Staff Evaluation: The NRC staff evaluated the applicant's management commitment to maintain doses ALARA, using the review methods and acceptance criteria in YMRP Sections 2.1.1.8.2 and 2.1.1.8.3. (b)(5)
(b)(5)
(b)(5) As described in SAR Section 1.10.1, personnel will be made aware of the applicant's management commitment to ALARA through policy and instruction. Personnel will be instructed on their individual responsibilities related to ALARA implementation to ensure compliance. Supervisors will be instructed to integrate appropriate radiation protection controls into work activities. (b)(5)

(b)(5)

(b)(5)

(b)(5) The applicant has committed to provide radiation protection instruction to individuals who are likely to receive in a year an occupational dose exceeding 1 mSv [100 mrem], as is required by 10 CFR 19.12. The training will be commensurate with the duties and responsibilities of training recipients. The applicant also made a commitment that workers will be periodically retrained in radiation protection procedures and techniques on the basis of job responsibility.

The applicant has committed to providing operational radiation protection requirements through written policies and procedures. In SAR Section 5.11, the applicant described the Operational RPP. The applicant will use program policies and procedures to maintain radiation exposures to workers and members of the public during operations to meet 10 CFR Parts 20 and 63 and maintain exposures ALARA. (b)(5)

(b)(5)

(b)(5) NRC staff's evaluation of the RPP is provided in SER Section 2.1.1.8.3.5 of this chapter.

NRC Staff Conclusion

(b)(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, the applicant discussed the application of the ALARA principle into the design. 10 CFR 20.1101 requires the use, to the extent practicable, of procedures and engineering controls to achieve occupational doses and doses to members of the public that are ALARA. The applicant stated that its ALARA program is conducted through engineering procedures, training for engineering and design personnel, design reviews, cost-benefit analyses, audits, self-assessments of effectiveness, and a policy for its consistent application in the design process. Formal design criteria are used to implement ALARA design requirements. Reviews during the design process considered potential radiation exposure and contamination from normal operations and any Category 1 event sequences. The applicant's assessments considered radiation workers, construction workers during staged operations, and members of the public. The program focused on activities associated with higher potential doses so that greater reductions in worker and public doses could be realized. An annual dose goal of 5 mSv [0.5 rem] was set 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 for both compliance demonstrations on the basis of minimized staffing levels, maximized source terms, facility annual throughput, and nominal conditions based on 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.

The applicant 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, the applicant incorporated design features to control the spread of contamination and facilitate maintenance and decommissioning.

NRC Staff Evaluation: NRC staff evaluated the applicant's consideration of ALARA principles in design and modifications by using the Review Methods and Acceptance Criteria in YMRP Sections 2.1.1.8.2 and 2.1.1.8.3. (b)(5)

(b)(5)

(b)(5)

The applicant committed to locating radioactive material handling and storage facilities sufficiently away from the site boundary and from other onsite work areas to maintain doses ALARA. (b)(5)

(b)(5)

(b)(5) SER Section 2.1.1.5 provides additional details on the NRC evaluation of the applicant's demonstration of compliance with dose requirements. The applicant committed to control access to the restricted area and committed to apply access controls to high and very high radiation areas in accordance with 10 CFR 20.1601 and 10 CFR 20.1602 by using the guidance in NRC Regulatory Guide 8.38 (NRC, 2006ac).

The applicant plans to conduct ALARA design reviews using multidisciplinary teams with experience in radiological safety, operations, and engineering backgrounds. (b)(5)

(b)(5)

(b)(5)

(b)(5) The applicant acknowledged situations when estimated doses did not meet the ALARA design goal and identified options for dose reduction (BSC, 2008bw). The applicant also assessed average worker doses when workers who perform similar tasks (operators, health physics technicians, or security) are rotated to different facilities. (b)(

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.8.3.3 Facility Shielding Design

In SAR Section 1.10.3, the applicant discussed the facility surface and subsurface shielding design objectives, criteria, and evaluation used to implement ALARA requirements for normal operations and Category 1 event sequences. The objective is to design facility shielding to reduce dose rates from radiation sources such that worker doses are within the standards of 10 CFR Part 20 and are ALARA when combined with the program to control personnel access and occupancy of restricted areas. The applicant performed shielding evaluations to ensure that adequate space envelopes and structural loads are identified. Surface facility shielding will include concrete walls, floors, and ceilings; shielded viewing windows; slide gates; and shield doors. The applicant has committed to the concrete design used for shielding in accordance with ANSI/ANS-6.4-2006 (American Nuclear Society, 2006aa).

As part of the design objectives, the applicant provided its shielding design descriptions for individual facilities used in the shielding evaluation. The shielding design 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.

The applicant described its shielding evaluation methodology as follows:

- Radiation sources, summarized in SAR Figure 1.10-18, and bounding terms, described in SAR Section 1.10.3.4, are used to approximate the geometry and physical condition of sources in the various repository facilities.

- Flux-to-dose rate conversion factors taken from ANSI/ANS-6.1.1-1977 (American Nuclear Society, 1977aa) are used to develop dose rates. (b)(5)
(b)(5)

- 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 demonstrate 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. The applicant implemented ALARA principles through the combination of facility shielding design and the RPP.

NRC Staff Evaluation: The NRC staff evaluated the applicant's considerations of ALARA principles in the facility shielding design (b)(5)
(b)(5) The applicant'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). (b)(5)
(b)(5)

(b)(5)

(b)(5) SER Section 2.1.1.5 provides the NRC staff's evaluation of the applicant's source term calculations, direct exposure calculations, and radiation dose calculations to workers and members of public from airborne radionuclides.
(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.8.3.4 Incorporation of As Low As Is Reasonably Achievable Principles Into Proposed Operations at the Geologic Repository Operations Area

In SAR Section 1.10.4, the applicant has described incorporating 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. Radiation protection training and personnel testing will be conducted for radiation workers before those individuals are allowed to begin work activities in restricted areas. Periodic retraining is also included. Individuals with job tasks outside of restricted areas are classified as onsite members of the public and will receive instruction on emergency procedures. Preplanning for significant worker doses and dry-run training for jobs associated with significant collective doses will be applied during operations. 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.

The SAR also indicated that radiological work permits and written procedures will be used as administrative controls for operations and maintenance. Radiation areas will be designated and posted within restricted areas. Access to high and very high radiation areas will be controlled. ALARA reviews will be conducted before design changes and administrative control changes are approved. The ALARA program will also address recovery actions from event sequences during operations and reviews of planned decommissioning and decontamination activities. Although no Category 1 event sequences have been identified for which recovery actions are preplanned, the applicant did consider reduction of worker doses for recovery from potential event sequences that the applicant described as off-normal events. SER Section 2.1.1.4 describes the NRC staff's evaluation of the applicant's categorization of event sequences.

NRC Staff Evaluation: NRC staff evaluated the applicant's description of how it will incorporate the ALARA principles into operations, using the Review Methods and Acceptance Criteria in YMRP Sections 2.1.1.8.2 and 2.1.1.8.3. (b)(5) Because the ALARA requirement applies during operations, SER Section 2.5.6 provides the NRC staff's evaluation of the applicant's plans for conducting normal activities, including

maintenance, surveillance, and testing of structures, systems, and components. (b)(5)

(b)(5)

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.8.3.5 Radiation Protection Program

The applicant's Operational RPP was described 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, the detailed RPP will be available for NRC review in connection with the submittal of the updated application for a license to receive and possess spent nuclear fuel and high-level radioactive waste. The NRC review will verify that the detailed RPP meets the program description provided in the applicant's license application. 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 adequate facilities, equipment, qualified staff, and radiation protection training, as required by 10 CFR 63.21(c)(6). 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

The applicant has described the RPP organization in SAR Section 5.11.1. The applicant committed to have the radiation protection and criticality safety (RPCS) program organization under the RPCS manager. The RPP organization will work independently of the operations and maintenance organizations. The RPCS manager will report directly to the site operations manager and chief nuclear officer, as discussed in SAR Section 5.3.1.2 and revised by the applicant in response to a staff request for additional information (DOE, 2009az). The RPCS manager will be responsible for developing and implementing the RPP as well as the program for nuclear criticality safety. SAR Section 5.3.2.1.7 addressed the qualifications of the RPCS manager. The applicant has committed to using the guidance in ANSI/ANS-3.1-1993 (American Nuclear Society, 1993aa) for its radiation protection staffing requirements.

NRC Staff Evaluation: NRC staff reviewed the applicant's description of the administrative organization of its RPP using the guidance in YMRP Section 2.1.1.8.3. (b)(5)

(b)(5)

(b)(5) In SAR Section 5.3.2.1.7, the applicant provided the qualification requirements for the RPCS manager; minimum qualifications are a bachelor's degree in science, health physics, or engineering with a combined 6 years of experience in the radiological protection aspects of nuclear facility design and operations and 3 years of supervisory or management experience.

(b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.8.3.5.2 Equipment, Instrumentation, and Facilities

As a part of the RPP, the applicant has made a commitment to describe and identify the equipment, instrumentation, and facilities used to support radiological monitoring, personnel protection, and contamination control. The RPP will 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, the applicant described its plans to use instrumentation that is appropriate for the types, levels, and energies of radiation at the GROA and for the expected environmental conditions. Instrumentation will also be periodically calibrated to the National Institute of Standards and Technology standards and routinely tested for operability. The applicant will calibrate instruments and equipment used for quantitative measurements in accordance with NRC Regulatory Guide 8.6 (NRC, 1973ab), ANSI N323A-1997 (American Nuclear Society, 1997aa), and ANSI N323B-2003 (American Nuclear Society, 2003aa), as well as manufacturer recommendations. The applicant also committed to provide a radiation protection organization with adequate facilities to effectively implement its RPP.

NRC Staff Evaluation: NRC staff reviewed the applicant's description of the radiation protection equipment, instrumentation, and facilities using the guidance in YMRP Section 2.1.1.8.3.(b)(5)

(b)(5)

(b)(5)

(b)(5) Personnel dosimeters, as required by 10 CFR 20.1501(c), must 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, the applicant provided in SAR Section 1.4.2 a high-level system description of the process and area monitoring equipment used to monitor effluents from the GROA release points. This system will provide both historical and real-time information and will operate on a continuous basis.

The NRC staff reviewed the applicant's commitment to 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. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.1.8.3.5.3 Policies and Procedures

The applicant described the policies and procedures that will be used to implement the RPP in SAR Section 5.11.3. The applicant will develop the policies and procedures for the following program elements so that its program will meet the 10 CFR Part 20 requirements:

- Radiation Surveys and Radiological Postings—Policies and procedures will be developed to address the survey requirements of 10 CFR 20.1501, 10 CFR 20.1502, 10 CFR 20.1703, 10 CFR 20.1906, and 10 CFR 20.2101. Postings will be in accordance with 10 CFR 20.1901 through 10 CFR 20.1903.

- Radiological Access Control and Onsite Dose—The access control system will be developed to comply with 10 CFR 20.1601 and 10 CFR 20.1602. The applicant has also committed to follow NRC Regulatory Guide 8.38 (NRC, 2006ac). Onsite dose requirements of 10 CFR 20.1201 through 10 CFR 20.1208 and 10 CFR 20.1301 will be met by identifying occupational dose monitoring practices in the RPP. The applicant has also committed to following NRC Regulatory Guide 8.35 (NRC, 1992ac) for planned special exposures.

- Control of Radiological Material and Contamination—Controls will be implemented to minimize the amount of material and equipment brought into areas and to control radioactive materials. Materials will be labeled and marked in accordance with 10 CFR 20.1904 and 10 CFR 20.1905. The applicant committed to meeting the requirements 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—Procedures and polices will be developed to meet requirements of 10 CFR 20.1501(c), 10 CFR 20.1502, and 10 CFR 20.1204. The applicant committed to following NRC Regulatory Guide 8.34 (NRC, 1992ab) for monitoring methods and criteria for occupational doses. The applicant committed to following NRC Regulatory Guide 8.9 (NRC, 1993aa) for internal dose monitoring. The

applicant committed to 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). The applicant also committed to follow ANSI N42.20-2003 (American Nuclear Society, 2003ab) for active personnel dose and dose rate warning system.

- Analysis of Airborne Radioactivity Sampling—Procedures will be developed to meet the 10 CFR Part 20 requirements for survey and measurements, and the applicant committed to follow the guidance in NRC Regulatory Guide 8.25 (NRC, 1992aa).

- Respiratory Protection—To demonstrate compliance with 10 CFR 20.1701 through 10 CFR 20.1705, the applicant committed to follow the guidance in NRC Regulatory Guide 8.15 (NRC, 1999ac).

- Radiation Protection Training—The applicant committed to following 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—The applicant will post notices in accordance with 10 CFR 19.11 and 10 CFR 63.9(e)(1).

- Protection of the Pregnant Worker and Embryo/Fetus—The applicant committed to developing a program to implement requirements of 10 CFR 20.1208 and 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—The applicant's program will address requirements of 10 CFR 20.2101 through 10 CFR 20.2110, 10 CFR 20.2201 through 10 CFR 20.2206, and 10 CFR 19.13. The applicant also committed to following guidance in NRC Regulatory Guide 8.7 (NRC, 2005ab) and ANSI/HPS N13.6-1999 (American Nuclear Society, 1999aa).

- Environmental Radiological Monitoring—The Environmental Monitoring Program will comply with 10 CFR 63.21(c)(6), 10 CFR 63.111(a)(2), 10 CFR 20.1101(d), 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR 20.1501, and 10 CFR 20.2001. The applicant also committed to follow NRC Regulatory Guide 1.21 (NRC, 2009aa).

NRC Staff Evaluation: (b)(5)

(b)(5)

(b)(5)

The RPP also addresses (i) the administrative organization of the RPP; (ii) the descriptions of health physics equipment, facilities, and instruments; (iii) the description of policies and procedures for controlling access to the radiation area, description of procedures for the accountability and storage of radioactive material, and the radiation protection training programs; and (iv) the description of program implementation.

NRC Staff Conclusion

(b)(5)

2.1.1.8.4 Evaluation Findings

(b)(5)

2.1.1.8.5 References

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NRC. 1974aa. Regulatory Guide 1.86, "Termination of Operation Licenses for Nuclear Reactors." Rev 0. Washington, DC: NRC.

NRC. 1973aa. Regulatory Guide 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters." Rev.0. Washington, DC: NRC.

NRC. 1973ab. Regulatory Guide 8.6, "Standard Test Procedure for Geiger-Mueller Counters." Rev. 0. Washington, DC: NRC.

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 applicant [U.S. Department of Energy (DOE)] responses to the NRC staff's requests for additional information (RAIs) (DOE, 2009ba, 2009bb). The objective of the review is to evaluate the feasibility and reasonableness of the applicant's retrieval plan and alternate storage by determining whether the repository design preserves the option of waste retrieval if retrieval becomes necessary.

In SAR Section 1.11, the applicant described its plans for retrieval and alternate storage of the radioactive wastes, including a discussion of compliance with the preclosure performance objectives. The applicant's description of its alternate storage plan identified a proposed alternate storage facility, including the location, size, and storage operations. The applicant also provided a schedule for retrieval operations, should retrieval become necessary.

2.1.2.2 Regulatory Requirements and Regulatory Guidance

The regulatory requirements applicable to this section are specified 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 retrievability preclosure performance objective must be met.

- 10 CFR 63.21(c)(7) requires that the safety analysis 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 through the preclosure period.
- 10 CFR 63.111(a) provides the radiation exposure limits for preclosure period.
- 10 CFR 63.111(b) provides numerical guides for design objectives.

The NRC staff reviewed the applicant's information using the guidance in the Yucca Mountain Review Plan (YMRP), Section 2.1.2 (NRC, 2003aa). For initial licensing, the retrieval plan should contain enough information to demonstrate that retrieval is 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. YMRP Section 2.1.2 states that the review should document whether the plan describes (i) how any or all of the wastes emplaced in the underground drifts can be retrieved and (ii) how the retrieval operation can be accomplished on a reasonable schedule. The description of the retrieval plan should also document why the plan is adequate to meet the preclosure performance objectives of 10 CFR 63.111(a) and (b). The description should be sufficiently detailed and contain (i) identification and location of alternate storage areas within or near the geologic repository operations area, (ii) dimensions and suitability of available land for storage, and (iii) plans for protection of workers and public while wastes are transported to the alternate storage site. Because demonstrating safe design, construction, and operations to meet the performance

objectives of 10 CFR 63.111(a) and (b) for the preclosure period is accomplished through preclosure safety analyses, the applicant may need to update its preclosure safety analyses if retrieval occurs. The results of these analyses and the retrieval operations will be subjected to NRC review if such operations occur. The applicable acceptance criteria are addressed as discussed next.

2.1.2.3 Staff Review and Analysis

In SAR Section 1.11, the applicant stated that to demonstrate compliance with 10 CFR 63.21(c)(7) and 10 CFR 63.111(e), it addressed the acceptance criteria in YMRP Section 2.1.2. The applicant'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 applicant plans to accomplish this objective by (i) designing the ground support system in the access and ventilation mains and emplacement drifts to function for 100 years; (ii) developing a maintenance plan to test, inspect, and repair ground support as necessary to ensure functionality of the underground openings through a 100-year preclosure period; and (iii) designing the subsurface communication and transportation infrastructure to function through the preclosure period to support access for maintenance or equipment replacement as needed. The applicant also stated that if off-normal events such as collapse of an emplacement drift section occurred, specialized procedures and equipment could be developed to restore access to waste packages. The applicant also identified an alternate storage facility location. The applicant did not propose the option of backfilling of emplacement drifts.

The NRC staff reviewed SAR Section 1.11 to determine whether the applicant demonstrated that retrieval operations would comply with the preclosure performance objectives, as required by 10 CFR Part 63. Specifically, the NRC staff reviewed the applicant's waste retrieval plan to determine whether (i) waste packages could be retrieved during the period of potential waste retrieval by reversing the operational procedure for waste emplacement, (ii) the applicant identified a reasonable range of potential problems (off-normal scenarios) during retrieval, and (iii) the applicant described a feasible approach for restoring access to waste packages from potential off-normal conditions without physical damage or overheating of the affected waste packages. NRC staff also reviewed the applicant's retrieval operations schedule and description of alternate waste storage plans to determine whether the applicant demonstrated compliance with the regulatory requirements of 10 CFR Part 63. NRC staff's review follows.

2.1.2.3.1 Waste Retrieval Plan

Retrieval Under Normal Operations

To demonstrate compliance with 10 CFR 63.21(c)(7), the applicant described its waste retrieval plan in SAR Section 1.11.1. In the plan, the applicant described the structures, systems, and components (SSCs) used for retrieval. The applicant would retrieve waste by performing emplacement operations in reverse, using the same SSCs used for emplacement. The SSCs relied upon are the transport and emplacement vehicle (TEV), invert structure and rails, electrical power system, communication system, and drift ventilation system. The applicant's plan includes maintaining access to the emplacement drifts, keeping the SSCs available throughout the preclosure period (DOE, 2009ba, 2009bb).

The applicant 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. The applicant's monitoring plan for accessible openings (such as access mains and the North Ramp) consists of regular visual inspection of the openings by qualified personnel and use of a geotechnical instrumentation program to obtain measurements of drift convergence, ground support loads, and potential overstressed zones (DOE, 2009bb). For the emplacement drifts and turnouts, the applicant will use remote-controlled equipment to inspect the openings to detect any indications of rockfall, drift deterioration, or instability and measure drift convergence at locations selected on the basis of previous inspections (DOE, 2009bb). The applicant stated that every emplacement drift and turnout will be inspected over its entire length, once a year initially after waste emplacement but at a modified frequency subsequently. The applicant will determine subsequent inspection frequencies using results of previous inspections and geologic mapping and intends to determine the extent of documentation of any changes to the monitoring locations or inspection frequency based on satisfying the requirements of 10 CFR 63.44.

NRC Staff Evaluation: The NRC staff used the guidance in the YMRP to determine whether the applicant adequately described the retrieval operations, including the equipment to be used, and how the preclosure performance objectives would be met. (b)(5)

(b)(5)

(b)(5) During emplacement, the TEV is only loaded when descending. The NRC staff reviewed the TEV design to determine whether the TEV can 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. NRC staff's evaluation of the TEV design is documented in SER Section 2.1.1.7.3.1.3.1. (b)(5)

(b)(5)

The NRC staff also evaluated whether the applicant's plan to inspect the emplacement drifts and turnouts using remote-controlled equipment adequately provides for proper maintenance plans to support retrieval if necessary. In response to NRC staff's RAI, the applicant stated that it will inspect the entire length of every emplacement drift and turnout annually. (b)(5)

(b)(5)

DOE's Retrieval Scenarios Under Off-Normal Conditions

The applicant evaluated recovery strategies after an off-normal event to the TEV 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. The applicant 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.

The applicant described the conceptual design of a maintenance and repair vehicle (MRV) for recovering from potential off-normal occurrences. According to the applicant's description, the MRV design will be based on the TEV NRC staff reviews in SER Section 2.1.1.7. The MRV will be a rail-based, remotely operated vehicle with hardware to support recovery operations. The MRV hardware includes (i) lights, cameras, and communication (potentially wireless) for remote, visual operation (teleoperation); (ii) batteries or tethered cables for loss-of-power conditions; and (iii) telescoping boom crane, manipulator arms and various attachments, winch, and rail clamps for remotely clearing rubble and for pulling a TEV or disassembling equipment. The applicant's plan relied on this concept of a multipurpose vehicle for restoring a derailed TEV or a collapsed emplacement drift to normal conditions.

The applicant identified three derailment conditions within emplacement drifts that encompass several severity levels of TEV failures. The minor failure case considered a derailment where the TEV retains full functionality. The applicant's recovery plan consisted of placing commercially available "rerailers" along the rail and using the MRV to drag the TEV back onto the rail. A more severe case considered a derailment that includes damage to the TEV drive system, the front shield doors in an open state, and the base plate fully extended and inoperable and thus providing no shielding protection. The applicant proposed to use winches, rail clamps, and rerailers to remove the base plate and pull the TEV onto the rails. The most critical scenario considered involved the repair of damaged rails near emplaced waste packages. In DOE Enclosure 3 (DOE, 2009ba), the applicant described how a boom crane could be used to construct a temporary shield wall near the waste package such that workers could enter the emplacement drift and install new rails.

The applicant identified (BSC, 2008bt) three waste package failure modes that could result from overheating of a waste package buried in rubble: (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. The applicant 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. The applicant 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. The applicant also concluded, on the basis of its calculations, that low-probability, beyond design bases conditions would not likely interfere with retrieval operations.

The applicant described the installation of a temporary shield wall, the design of the support structure, and the shape of the shield bricks such that direct radiation from the joints is prevented.

NRC Staff Evaluation: Staff used the guidance in YMRP Section 2.1.2 to determine whether the applicant identified retrieval scenarios under degraded drift conditions and adequate 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 the applicant'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 staff reviewed the assumptions for representing a rockfall condition and the analytical approach to determine the thermal effects of rock rubble and finds them to be reasonable and appropriate. The applicant calculated the temperature profile of a waste package partially or completely buried in rubble and analyzed failure modes of concern and the associated temperature limits. 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.

(b)(5)

The NRC staff reviewed the calculations, and their bases, the applicant used to estimate the temperature of the waste package due to drift collapse and loss of ventilation. The applicant concluded that the waste package temperature limits would not be exceeded. This conclusion is based on restoring ventilation and removing rubble from around waste packages within a 30-day period as the applicant defined in SAR Section 1.3.1.2.4. The applicant described the design and development of an MRV that would be needed for rubble removal. According to the applicant'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.

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(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.2.3.2 Compliance With Preclosure Performance Objectives

Preclosure Safety During Retrieval

In SAR Section 1.11.1.3.1, the applicant discussed the approach to meeting the preclosure performance objectives of 10 CFR 63.111(a) and (b) during waste retrieval. The approach is to characterize event sequences, perform consequence analysis, and impose design requirements as explained in SAR Section 1.7.

In SAR Section 1.11.1.3.2 the applicant discussed, in general terms, how as low as is reasonably achievable (ALARA) concepts will be implemented. The applicant did not develop occupational dose limits for retrieval. However, the applicant stated that whatever radiation exposure considerations are applicable to emplacement operations would also apply to retrieval scenarios.

NRC Staff Evaluation: According to SAR Section 1.11.1.3.1, the applicant did not identify any new event sequences for retrieval scenarios. The applicant's approach is based on the assumption that the same equipment and methods would be used for retrieval as in the emplacement operation. (b)(5)

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(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.2.3.3 Proposed Alternate Storage Plans

The applicant indicated in SAR Section 1.11.2 that facilities for handling and storage of retrieved waste packages could be sited in Midway Valley at approximately the location of surface waste handling and aging facilities for waste emplacement (SAR Figure 1.11-1). The applicant estimated that the alternate storage location can be developed to accommodate waste packages containing 70,000 metric tons of heavy metal. As the applicant explained, the facility could be developed to include equipment for unloading waste packages from the retrieval vehicle, transferring the waste packages into shielded long-term storage containers, and transporting the shielded containers to storage pads. The alternate storage location the applicant identified has been characterized for surface waste handling buildings and aging pads, as reviewed in SER Section 2.1.1.1.

NRC Staff Evaluation: NRC staff reviewed the applicant's information using the guidance in the YMRP. (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.2.3.4 Retrieval Operations Schedule

10 CFR 63.111(e)(3) requires that the applicant complete the retrieval operations under a reasonable schedule. The applicant provided a conceptual retrieval timeline in SAR Figure 1.11-2 and BSC Section 4.2 (2008ad).

NRC Staff Evaluation: The NRC staff reviewed the applicant's information using the guidance in the YMRP. (b)(5)

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(b)(5) The applicant 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 ensure that accessibility to waste packages is maintained throughout the preclosure period. (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.2.4 Evaluation Findings

(b)(5)

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 for the 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 the applicant provided in response to the NRC staff's request for additional information (RAI) (DOE, 2009ao)

The applicant included the design considerations in the SAR to facilitate PCDDD and a planning timeline for decontamination and dismantlement.

2.1.3.2 Regulatory Requirements and Regulatory Guidance

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 include information on plans for PCDDD of surface facilities. 10 CFR 63.21(a) specifies that the license application must be as complete as possible in light of information that is reasonably available at the time of docketing.

The NRC staff reviewed the applicant's information using the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.1.3 (NRC, 2003aa). YMRP Section 2.1.3 provides guidance on the review of features of the geologic repository operation area (GROA) design that will facilitate PCDDD and preliminary plans for PCDDD of surface facilities.

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

2.1.3.3 Staff Review and Analysis

In SAR Section 1.12, the applicant stated that to demonstrate compliance with the regulatory requirements of 10 CFR 63.21(c)(8) and 63.21(c)(22)(vi), it addressed the acceptance criteria in YMRP Section 2.1.3.3. The applicant 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). The applicant stated that it will provide the plan to NRC in accordance with the planning timeline shown in SAR Figure 1 12-1. 10 CFR 63.51 specifies that the applicant shall submit an application to amend the license before permanent closure of a geologic repository at the Yucca Mountain site and that this submission must consist of an update of the license application submitted under 10 CFR 63.21. 10 CFR 63.52 specifies that NRC will review this submission before license termination.

According to YMRP Section 2.1.3.2, the applicant is not required to provide a financial assurance plan in support of closure or decommissioning, because the applicant is a government agency.

The staff's review of the design considerations that will facilitate PCDDD and the PCDDD plan ~~is below~~follows.

Comment (b)(5)
(b)(5)

2.1.3.3.1 Design Considerations That Will Facilitate Permanent Closure and Decontamination or for the Decontamination and Dismantlement

To demonstrate compliance with 10 CFR 63.21(c)(8), the applicant described the requirements and criteria that it will use to ensure that the design will facilitate and support PCDDD in SAR Section 1.12.1. The applicant also specified that the requirements and criteria applied as the design progresses will ensure maintaining radiation doses to workers and the public as low as is reasonably achievable (ALARA). Examples of the considerations the applicant will use to facilitate the PCDDD are 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. The applicant 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.

NRC Staff Evaluation: The NRC staff used the guidance in ~~the~~ YMRP Section 2.1.3 ~~(NRC, 2003aa)~~ and NUREG–1757 (NRC, 2006aa) to determine whether the applicant adequately described and provided bases for the GROA design considerations that will facilitate PCDDD. The NRC staff also evaluated the information the applicant provided in its response to the NRC staff's RAI (DOE, 2009ao). (b)(5)
(b)(5)
(b)(5) In its response to the NRC staff's RAI, the applicant identified requirements and criteria that will ensure that design considerations to facilitate PCDDD will be evaluated as the design progresses. Specifically, the applicant will manage requirements that form the bases for design in accordance with its Quality Assurance Requirement and Description (DOE, 2008af) document. The applicant also uses its ALARA review process to ensure the design features that will facilitate decontamination and dismantlement are considered and evaluated. The applicant ensured that information necessary at the time of PCDDD will be available through the applicant record management and document control program.

NRC Staff Conclusion
(b)(5)

2.1.3.3.2 Plans for Permanent Closure and Decontamination or for the Decontamination and Dismantlement

10 CFR 63.21(c)(22)(vi) requires information concerning PCDDD of the surface facilities. To demonstrate compliance with this requirement, the applicant described in SAR Section 1.12.3 the information it will collect and how it will maintain that information during the facility lifetime. SAR Section 1.12 stated that the applicant will submit the final plans for the decontamination and dismantlement of the repository facilities in the GROA before permanent closure for NRC review. The NRC staff's review and determination of whether this information is adequate and will meet the requirements of 10 CFR 63.21(c)(22)(vi) are described ~~below~~next.

2.1.3.3.2.1 Facility History

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant 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: The NRC staff reviewed the description of the information the applicant will collect using the guidance in YMRP Section 2.1.3. (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.2 Facility Description and Dose Modeling

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant described information related to the GROA and its environment that will be used to estimate doses (SAR Section 1.12.3.2,). The applicant also described the objective of dose models in SAR Section 1.12.3.4, which is to demonstrate that the total effective dose equivalent to a critical group of individuals near the preclosure controlled area is consistent with ALARA principles and will not exceed regulatory requirements for PCDDD. The applicant also discussed the type of information it will use for dose modeling such as source term information and physical features used to model exposure pathways

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of the information it will use for its facility dose modeling evaluations at the time of permanent closure and decommissioning using the guidance in YMRP Section 2.1.3 (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.3.3.2.3 Facility Radiological Status

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant described how it will 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). The information to perform 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, the applicant 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 regulatory requirements that guide the program; (vi) a summary of the types and approximate quantities of contaminated materials at each location; and (vii) a scale drawing or map showing the location of contaminated systems and components.

In SAR Section 1.12.3.3.3, the applicant stated that the following information will be available during PCDDD to evaluate the radiological status of 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, the applicant discussed its plans for addressing potential water contamination from process operations. On the basis of the site characterization, the applicant determined that there are no natural surface water bodies at the site (SAR Section 1.1.1.2). The applicant proposed that storm water drainage diversion channels will protect the GROA from runoff from slopes above the facilities and keep storm water from becoming contaminated. The applicant will provide two storm water detention impoundments and analyze the radiation of the water in these impoundments. One impoundment will collect runoff from the North Portal pad operations area, and the other impoundment will collect cooling tower blow down and nonradioactive wastewater.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of information pertaining to the site radiological status using the guidance in YMRP Section 2.1.3 (b)(
(b)(5)

(b)(5) For example, the applicant proposed that storm water drainage diversion channels will protect the GROA from runoff from slopes above the facilities. The applicant will provide two storm water detention impoundments and analyze the radiation of the water in these impoundments. One impoundment will collect runoff from the North Portal pad operations area, and the other impoundment will collect cooling water and nonradioactive wastewater. For structures and buildings, systems and components, and soil contamination, the applicant identified and described the process to collect the information needed to address those radiological issues.
(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.4 Alternatives for Decommissioning

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant stated that it will evaluate alternative decontamination and dismantlement activities (SAR Section 1.12.3.5). The applicant 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 low-level radioactive waste disposal methods that meet regulatory requirements.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of how it will evaluate alternatives for decommissioning using the guidance in YMRP Section 2.1.3 (b)(5

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NRC Staff Conclusion

(b)(5)

2.1.3.3.2.5 As Low As Is Reasonably Achievable Analysis

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant described the scope of its ALARA assessment and the information that will be provided to support the ALARA analyses to facilitate PCDDD (SAR Section 1.12.3.6). The applicant stated that the plan will be provided to NRC in accordance with the planning timeline shown in SAR Figure 1.12-1. Examples of the information that the applicant 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 the applicant's description of its ALARA analyses to support PCDDD using the guidance in YMRP Section 2.1.3 (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.3.3.2.6 Planned Decommissioning Activities

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant described the information it will create and maintain to support its planned decommissioning activities (SAR Section 1.12.3.7). The applicant described this information for contaminated structures, contaminated systems and components, and contaminated soil. The applicant 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. The applicant proposed to build artificial water retention impoundments at the GROA for the purpose of water collection and evaporation. The impoundment will be decommissioned after facility operations support is no longer required, and the water remaining in the impoundment will be appropriately processed.

The applicant also described the scope of the schedule. The applicant stated that it will provide its plan to NRC in accordance with the planning timeline shown in SAR Figure 1.12-1, at which time NRC will review the plan in accordance with 10 CFR 63.52.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of information it will use for its decommissioning activities using the guidance in the YMRP Section 2.1.3. (b)(5)

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NRC Staff Conclusion

(b)(5)

2.1.3.3.2.7 Project Management and Organization

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant described its plan for developing a management organization that is responsible for task management for PCDDD (SAR Section 1.12.3.8). Specifically, the applicant described the information that will be included concerning management organization such as reporting hierarchy and responsibilities and authorities. The applicant 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 the applicant will provide to each employee to comply with 10 CFR Part 19; and documentation that will be maintained to demonstrate that training commitments are met.

NRC Staff Evaluation: The NRC staff reviewed the applicant's description of information it will use to establish its PCDDD project management and organization using the guidance in YMRP Section 2.1.3 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.8 Health and Safety Program During Permanent Closure and Decontamination or for the Decontamination and Dismantlement

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant 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: The NRC staff reviewed the applicant's description of information it will use to establish its PCDDD radiological health and safety program using the guidance in YMRP Section 2.1.3 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.9 Environmental Monitoring and Control Program During Permanent Closure and Decontamination or for the Decontamination and Dismantlement

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant 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. The applicant also stated that its Environmental Radiological Monitoring Program (SAR Section 5.11.3.1) will be evaluated and revised to measure and record potential impacts to the site environment during closure and during decontamination and dismantlement. 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 of the applicant's environmental monitoring program using the guidance in YMRP Section 2.1.3.3 (b)(5)

(b)(5)

(b)(5) The applicant will evaluate and revise its environmental monitoring and control program as necessary to measure impacts to the site environment during PCDDD. The applicant will also ensure that the necessary records will be available through the applicant record management and document control program.

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.10 Radioactive Waste Management Program

To demonstrate compliance with 10 CFR 63.21(c)(22)(vi), the applicant 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 of 3,500 m^3 [equivalent] after treatment, which is expected to be generated during the closure phase of the repository. In the former section, the applicant also listed the information that will be developed for the PCDDD activities. Examples of the information the applicant 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

SAR Section 1.12.3.11.2 listed the information that will 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 the applicant 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 the applicant's radioactive waste management information using the guidance in YMRP Section 2.1.3.3 (b)(5)

(b)(5)

(b)(5) The applicant will update the estimated types and quantities of low-level waste generated during the

development of the final plans for PCDDD activities. The applicant 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. The applicant's information necessary at the time of PCDDD will be available through the applicant's record management and document control program.

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.11 Radiation Surveys

To demonstrate compliance with 10 CFR 63.21(c)(8) and 10 CFR 63.21(c)(22)(vi), the applicant 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. 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 the applicant provided using the guidance in YMRP Section 2.1.3.3 (b)(5)

(b)(5)

NRC Staff Conclusion

(b)(5)

2.1.3.3.2.12 Quality Assurance Program

In SAR Section 1.12.3.12, the applicant committed 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 YMRP Section 2.1.3.3 to determine whether the applicant has adequately described the QA information (i.e., the type of information that will be required to facilitate decommissioning, with respect to QA) that will be used for PCDDD. (b)(5)

(b)(5)

Commitment

The applicant commits to integrate the information required to facilitate PCDDD with respect to quality assurance (QA) with the preclosure QA program (SAR Section 1.12.3.12).

NRC Staff conclusion

(b)(5)

2.1.3.4 Evaluation Findings

(b)(5)

2.1.3.5 References

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The U.S. Nuclear Regulatory Commission staff (NRC) has reviewed the Safety Analysis Report and the other information the U.S. Department of Energy submitted in support of its license application (b)(5)

(b)(5)

Preclosure Safety Analysis [10 CFR 63.111(a)–(c), 63.112, and 63.204]

(b)(5)

Retrieval and Alternate Storage [10 CFR 63.111(e)]

(b)(5)

Permanent Closure and Decontamination [10 CFR 63.21(c)(8) and 63.21(c)(22)(vi)]

(b)(5)

APPENDIX B

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