CONTENTS

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TABLES

Page

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FIGURES

Page

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1.9 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY; NATURAL AND ENGINEERED BARRIERS IMPORTANT TO WASTE ISOLATION; SAFETY CONTROLS; AND MEASURES TO ENSURE AVAILABILITY OF THE SAFETY SYSTEMS

Determining which structures, systems, and components (SSCs) are important to safety (ITS) is an integral part of the iterative preclosure safety analysis (PCSA) and design process. This section describes the process for identifying ITS SSCs, describes the process for identifying design features (and relevant SSCs) that are important to waste isolation (ITWI), and provides information that addresses how the requirements in 10 CFR 63.142 are applied to SSCs classified as ITS or ITWI. This section also provides information that addresses acceptance criteria presented in Sections 2.1.1.6.3 and 2.2.1.1.3 of NUREG-1804. Figure 1.9-1 illustrates the steps that are discussed in this section for classifying SSCs as part of the overall PCSA process. The steps for identifying the barriers that are ITWI and their associated features that are important to barrier capability are presented in Section 2.1.

Preclosure internal and external initiating events that could potentially impact operations are presented in Section 1.6. Event sequences developed from event sequence diagrams, event trees, and fault trees, and their potential frequencies, are presented in Section 1.7. Applicable event sequences are categorized as Category 1 or Category 2; these categorizations establish which performance objectives of 10 CFR 63.111 govern each event sequence. Dose consequences of potential event sequences, as well as normal operations, are presented in Section 1.8.

ITS SSCs are identified following an examination of the function performed by the SSC in an identified event sequence (Section 1.7) and an examination of the dose consequences associated with the inclusion or elimination of the SSC in the event sequence in accordance with the performance objectives established in 10 CFR 63.111 (Section 1.8). Those SSCs identified as necessary to prevent (i.e., reduce the frequency of occurrence of) or mitigate (i.e., limit the consequences of) an event sequence are classified as ITS. ITS SSCs implemented for event sequences provide reasonable assurance that the performance objectives of 10 CFR 63.111 are met. Section 1.9.1 describes the analysis methodology and presents the criteria used for the classification of the repository SSCs. Table 1.9-1 presents the results of the preclosure safety classification of SSCs within the geologic repository operations area (GROA) for the period prior to permanent closure. Tables 1.9-2 to 1.9-7 identify the safety functions and controlling parameters and values (the nuclear safety design bases) for the ITS SSCs in the Initial Handling Facility (IHF), Canister Receipt and Closure Facility (CRCF), Wet Handling Facility (WHF), Receipt Facility (RF), intrasite operations, and subsurface operations, respectively. The ITS SSCs relied upon to perform a criticality prevention function are also identified in these tables (Section 1.14).

Interactions between ITS SSCs and non-ITS SSCs and the methodology used to prevent event sequences due to these interactions are discussed in Section 1.9.1.14.

The process for identifying barriers and natural features and the SSCs that compose each barrier that is ITWI is derived from the development of the total system performance assessment (TSPA) (Section 2.1). The methodology used to develop the performance assessment involves a series of steps from the collection of data and empirical observations through the identification and screening of features, events, and processes (FEPs). Features are the physical components of the total repository system, including the natural system (i.e., geologic setting) and the engineered system (i.e., engineered components such as the waste package). Processes typically act more or less continuously on the features. Events also act on the features, but at discrete times. Examples of events include seismic and volcanic events. Those barriers that prevent or substantially reduce the rate of water or radionuclide movement or prevent or substantially reduce the release rate of radionuclides from the waste are classified as ITWI. Within each barrier, features and SSCs that contribute to the barrier function are identified to determine the parameters to be controlled. In some instances, an SSC may have different preclosure and postclosure performance criteria, depending on its function in each period.

Section 1.9.2 describes the classification of ITWI barriers and identifies the postclosure performance assessment control parameters for monitoring during the preclosure period. Table 1.9-8 provides the waste isolation classification of the three principal barriers, the related features of the geologic setting, and the design features (and relevant SSCs) that are ITWI. The description of each design control parameter is presented in Table 1.9-9. These parameters require controls to ensure that the postclosure performance assessment analytical bases are established during design, construction, procurement, operations, and closure.

Section 1.9.3 describes procedural safety controls presented in Table 1.9-10 to be implemented in facility operations to prevent and mitigate event sequences, including the procedural safety controls placed on non-ITS SSCs to prevent interactions between these SSCs and ITS SSCs. The procedural safety controls also place interface controls on activities outside of the GROA that could potentially lead to an event sequence.

Section 1.9.4 addresses the use of risk significance in the SSCs classified as ITS.

The following table lists each section and the corresponding regulatory requirements and acceptance criteria from NUREG-1804 addressed in that section.

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
1.9.1	Structures, Systems, and Components Classified as Important to Safety	63.21(c)(5) 63.21(c)(18) 63.112(e)	Section 2.1.1.6.3: Acceptance Criterion 1 Acceptance Criterion 3
1.9.2	Identifying Postclosure Performance Assessment Design Control Parameters and Classifying ITWI Structures, Systems, and Components	63.142(c)(1)	Section 2.2.1.1.3: Acceptance Criterion 1
1.9.3	Procedural Safety Controls	63.21(c)(5)	Section 2.1.1.6.3: Acceptance Criterion 2
1.9.4	Risk Significance Categorization	63.21(c)(5) 63.112(e) 63.142(c)(1)	Section 2.1.1.6.3: Acceptance Criterion 3

Additional information related to the classification of SSCs, procedural safety controls, and measures to ensure availability of the safety systems is found in the following references:

- *Canister Receipt and Closure Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008a)
- *Initial Handling Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008b)
- *Receipt Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008c)
- *Wet Handling Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008d)
- *Intra-site Operations and BOP Reliability and Event Sequence Categorization Analysis* (BSC 2008e)
- *Subsurface Operations Reliability and Event Sequence Categorization Analysis* (BSC 2008f)
- *Seismic Event Sequence Quantification and Categorization Analysis* (BSC 2008g)
- *Preclosure Nuclear Safety Design Bases* (BSC 2008h)
- *Preclosure Procedural Safety Controls* (BSC 2008i)
- *Preclosure Criticality Safety Analysis* (BSC 2008j
- *Postclosure Nuclear Safety Design Bases* (SNL 2008)
- *Postclosure Modeling and Analyses Design Parameters* (BSC 2008k).

1.9.1 Structures, Systems, and Components Classified as Important to Safety *[NUREG-1804, Section 2.1.1.6.3: AC 1, AC 3]*

The repository facilities are designed to protect repository workers and the onsite and offsite members of the public. The results of the PCSA are used to define design bases for repository SSCs to prevent, to the extent practical, or mitigate event sequences that could lead to the release of radioactive material or result in radiological exposure of workers or the public. If prevention measures alone do not reduce an event sequence frequency to beyond Category 2, then mitigation measures are developed (e.g., ITS heating, ventilation, and air-conditioning (HVAC)) to reduce the worker and public exposure to radiation to comply with 10 CFR 63.111. This is achieved by performance of the PCSA as an integral part of the design process in a manner consistent with a performance-based, risk-informed philosophy. This integral design approach ensures that the ITS design features and procedural safety controls are selected in a manner that ensures safety. Using this strategy, design rules are developed to provide guidance on the safety classification of SSCs. The following information is developed to implement this strategy:

- Identification of essential safety functions needed to provide worker and public safety
- Identification and classification of SSCs relied upon to perform these essential safety functions
- Establishment of controlling parameters and values that ensure that the essential safety functions will be performed with reliability while retaining a safety margin
- Development of procedural safety controls that, in conjunction with the repository design, ensure that operations are conducted within the limits established in the PCSA.

This approach is depicted in the lower-right portion of Figure 1.9-1, which illustrates the PCSA process.

ITS is defined in 10 CFR 63.2 as follows:

> Important to safety, with reference to structures, systems, and components, means those engineered features of the geologic repository operations area whose function is:
>
> (1) To provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of § 63.111(b)(1) for Category 1 event sequences; or
>
> (2) To prevent or mitigate Category 2 event sequences that could result in radiological exposures exceeding the values specified at § 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site.

ITS SSCs have design bases established as defined in 10 CFR 63.2:

> Design bases means that information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be constraints derived from generally accepted “state-of-the-art” practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals.

When describing the contents of the license application, 10 CFR 63.21(c)(3) states that the safety analysis must include, among other items:

> (ii) The design criteria used and their relationships to the preclosure and postclosure performance objectives specified at § 63.111(b), § 63.113(b), and § 63.113(c); and
>
> (iii) The design bases and their relation to the design criteria.

Design basis requirements are developed from the event sequence analysis for the SSCs that have been classified as ITS for preventing or mitigating an event sequence. The applicable design criteria used to ensure that the design basis requirements are satisfied are set forth in Sections 1.2, 1.3, 1.4, and 1.5 for ITS SSCs.

The results of event sequence and fault tree analyses leading to the identification of event sequences, as well as the consequence analyses of potential radiological releases, are used as the basis for the classification of SSCs. In accordance with 10 CFR 63.112, the PCSA is used to identify ITS SSCs, procedural safety controls, and measures to ensure the availability and reliability of ITS SSCs. According to 10 CFR 63.112(e), these analyses include the consideration of the following (as described in Sections 1.9.1.1 through 1.9.1.13):

- Means to limit concentration of radioactive material in air
- Means to limit the time required to perform work in the vicinity of radioactive materials
- Suitable shielding
- Means to monitor and control the dispersal of radioactive contamination
- Means to control access to high radiation areas or airborne radioactivity areas
- Means to prevent and control criticality
- A radiation alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and increased radioactivity in effluents
- The ability of SSCs to perform their intended safety functions, assuming the occurrence of event sequences
- Explosion and fire detection systems and appropriate suppression systems
- Means to control radioactive waste and radioactive effluents, and to permit prompt termination of operations and evacuation of personnel during an emergency
- Means to provide reliable and timely emergency power to instruments, utility service systems, and ITS operating systems if there is a loss of primary electric power

- Means to provide redundant systems necessary to maintain, with adequate capacity, the ability of ITS utility services

- Means to inspect, test, and maintain ITS SSCs, as necessary, to ensure their continued functioning and readiness.

Classification of SSCs—10 CFR Part 63 is a risk-informed regulation and, accordingly, ITS SSCs have reliability requirements dependent on the safety functions relied upon. As described in Section 1.6, event sequence analyses are systematically developed for the repository surface and subsurface facilities used to receive, handle, package, store, and emplace high-level radioactive waste (HLW). The analyses consider external and internal events and include natural phenomena; military and industrial activities conducted outside the GROA, within, or near the site; and surface and subsurface operations within the GROA. The results of the analyses are used to identify potential event sequences that could potentially lead to the exposure of workers or the public to radioactive materials or radiation (Section 1.7).

An event sequence consists of one or more SSC failures or human errors that could potentially lead to the release of, or exposure to, radioactive materials or radiation. Potential event sequences are analyzed to estimate their frequency of occurrence. Based on the resulting frequencies, the event sequences are categorized as Category 1, Category 2, or beyond Category 2 (Section 1.7). The potential radiological consequences of normal operation and the Category 1 or Category 2 event sequences are estimated (Section 1.8). The evaluation of radiological consequences includes consideration of potential exposures to onsite workers and onsite or offsite members of the public (due to normal operational releases and aggregated releases from Category 1 events) and potential exposures to offsite members of the public (due to releases from Category 2 event sequences) (10 CFR 63.111(b)).

The SSCs that prevent or mitigate event sequences must be identified. The specific functions relied upon to prevent or mitigate event sequences and any necessary procedural safety controls must also be identified.

Implementation of the 10 CFR 63.2 regulatory definition of ITS has produced specific criteria in the PCSA to classify SSCs as ITS or non-ITS. To ensure that the limits of 10 CFR 63.111(b)(1) and 10 CFR 63.111(b)(2) are not exceeded, an SSC is classified as ITS if it appears in an event sequence and at least one of the following criteria apply:

- The SSC is relied upon to reduce the frequency of an event sequence from Category 1 to Category 2.

- The SSC is relied upon to reduce the frequency of an event sequence from Category 2 to beyond Category 2.

- The SSC is relied upon to reduce the aggregated dose of Category 1 event sequences by reducing the event sequence mean frequency.

- The SSC is relied upon to perform a dose mitigation or criticality control function.

An SSC is classified as non-ITS if the preceding criteria do not apply. In addition, an SSC is classified as non-ITS if at least one of the following criteria apply:

- The SSC is relied upon to exclusively perform only the normal operational functions of the repository.
- The SSC performs a defense-in-depth function for which another SSC provides an ITS function.

For non-ITS SSCs that could impair the ability of an ITS SSC to perform its intended safety function, the design attributes relied upon to either preclude failure or reduce the probability of failure to beyond Category 2 may require a reclassification from non-ITS to ITS. A discussion of potential interactions between ITS and non-ITS SSCs is provided in Section 1.9.1.14. A summary of the strategy used to analyze these interactions and the approach used to eliminate the interaction is described.

The specific safety functions of an SSC that are relied upon to prevent the occurrence of an event sequence or mitigate the consequences of an event sequence, as demonstrated in the event sequence analyses, are identified. These specific safety functions and any associated controlling parameters and values are established as part of the design basis requirements for that SSC. If an SSC is relied upon for other functions in addition to the prevention or mitigation of an event sequence, then only the prevention or mitigation functions of the SSC are classified as ITS. The boundary of the SSCs relied upon in the safety analyses (including avoidance of a credible failure mechanism for the SSC) extends to a physical point that includes the SSCs necessary to achieve the reliability of the safety function in the analyses. If a safety function and its controlling parameter and value is sufficient to achieve another safety function, there is no additional requirement necessary. For example, transportation casks compliant with 10 CFR Part 71 are intrinsically robust against lightning strikes, as analyzed in the external events analysis. Therefore, an additional safety function is not required for this ITS component. The methodology used to assess the reliability of repository SSCs, as well as the use of reliability data in event tree and fault tree analysis of the repository SSCs, is discussed in Section 1.7.

Development of Nuclear Safety Design Bases—Design bases are established for the ITS SSCs.

The safety functions and controlling parameters and values (which together are referred to as the nuclear safety design bases in the PCSA) are developed from the Category 1 and Category 2 event sequences for the SSCs that have been classified as ITS. In addition to the values for controlling parameters, reliability, and availability, goals for ITS SSCs are identified based upon (but are not limited to) the following categories:

- Mean probability of SSC failure: It has been demonstrated by analysis that the ITS SSC will have a mean probability of failure of the safety function, with consideration of uncertainties, less than or equal to the stated criterion value.
- Specific design features (e.g, a designed-in speed limit for ITS crane trolleys is required).

- Mean unavailability over a time period: It has been demonstrated by analysis that the ITS SSC or SSCs (e.g., HVAC and ITS electrical power) will have a mean unavailability over a period of a specified number of days, with consideration of uncertainties, of less than or equal to the criterion value.

These controlling parameters and values and reliability and availability goals ensure that the ITS SSCs perform their identified safety functions such that the 10 CFR Part 63 performance objectives are met. Frequencies and probabilities associated with the controlling parameters and values provide a direct link from the design to the 10 CFR Part 63 requirement for categorization of event sequences. The parameters and values in Tables 1.9-2 through 1.9-7 (BSC 2008h) are derived from the event sequence analysis. These values provide reasonable assurance, in some cases with considerable conservatism, that the categorization of event sequences provided in Section 1.7 is realized. In many cases, increasing a probability value will not change an event sequence categorization if a compensating probability value for another ITS SSC in the same event sequence can be achieved. As long as the event sequence categorization is maintained, compliance with 10 CFR Part 63 is ensured.

1.9.1.1 Means to Limit Concentration of Radioactive Material in Air
[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(a)]

Design features are incorporated to minimize and control the flow of airborne contaminants consistent with Regulatory Position C.2.d of Regulatory Guide 8.8. Areas of potential contamination are identified and evaluated. HVAC systems capable of limiting the spread of airborne radioactive contamination are provided in the surface handling facilities. These systems control the flow of air from areas with a low potential for contamination to areas with a higher potential for contamination. A discussion of the surface confinement HVAC systems used in the IHF, CRCF, WHF, and RF is provided in Sections 1.2.3.4, 1.2.4.4, 1.2.5.5, and 1.2.6.4. Radiation is monitored at source points, at manned operating stations, and at other locations in surface facilities to provide early indication of changing conditions. The surface confinement HVAC system exhaust streams from the waste-handling areas are filtered through high-efficiency particulate air (HEPA) filters and monitored before release to the environment.

The portions of the surface confinement HVAC system that exhaust from areas with a potential for a breach of a waste container in the CRCF and WHF are classified as ITS. These systems ensure that radiological material released due to a potential breach of a waste container will pass through two-stage HEPA filters prior to exhaust to the atmosphere, thereby mitigating the consequences of this event sequence. The HVAC systems in the IHF and RF are not relied upon to mitigate the consequences of an event sequence; therefore, these HVAC systems are classified as non-ITS.

Potential radioactive sources in the subsurface include the resuspension of radioactive contamination from the external surfaces of the waste packages and neutron activation of air and dust. Analyses affirm that the potential releases from these sources are below regulatory limits and do not require additional engineered controls, such as HEPA filters, for exhaust air release to the environment (Sections 1.3.5.3.4 and 1.8).

1.9.1.2 Means to Limit Time Required to Perform Work in Radiological Areas *[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(b)]*

Design features implemented to reduce the time required to work in radiological areas during normal operations, consistent with as low as is reasonably achievable principles, are non-ITS. Sections 1.10 and 5.11 provide a description of as low as is reasonably achievable principles implemented through design and operation and through the Operational Radiation Protection Program.

1.9.1.3 Suitable Shielding *[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(c)]*

The shielding design methodology and facility radiation zoning methodology are presented in Section 1.10. The shielding design considers normal operations and Category 1 and Category 2 event sequences. Shielding exclusively used to comply with 10 CFR Part 20 during normal operations is non-ITS; event sequences are not part of normal operations. Offsite doses resulting from direct radiation as a result of normal operations are negligible. No Category 1 event sequences were identified in the PCSA (Section 1.7). However, some shielding features that are credited in the PCSA for reducing the mean frequency of inadvertent exposure of personnel to below the Category 1 event sequence mean frequency are classified as ITS; other aspects of permanent shielding are classified as non-ITS. The ITS shielding features include the shield doors and slide gates in the IHF, CRCF, RF, and WHF, as applicable. Shielding is not included in the consequence evaluation of Category 2 event sequences.

1.9.1.4 Means to Monitor and Control Dispersal of Radioactive Contamination *[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(d)]*

A description of the capability to monitor radioactive effluents is provided in Section 1.4.2.2. Normal release points and occupied areas within handling facilities are routinely monitored. Alarms of high radiation levels are provided locally, in the Central Control Center, and on appropriate consoles in the facility operations rooms. Airborne radioactivity effluent monitors provide information for use in emergency response dose projections. The sampled air is continuously monitored for radioactivity by monitors located in designated release points in surface process facilities. At each of the surface effluent points, airborne radioactivity effluent monitors sample the effluent stream for airborne radioactivity particulates and gases. The monitoring equipment alerts operators to event sequences or off-normal conditions such as radiological releases or extreme radiation. 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 and, therefore, has been determined to be non-ITS.

The surface facility HVAC systems are described in Sections 1.2.3.4, 1.2.4.4, 1.2.5.5, and 1.2.6.4. As described in Section 1.9.1.1, the portions of the surface confinement HVAC system that exhaust from areas with a potential for a breach in the CRCF and WHF are classified as ITS; the HVAC systems for the IHF and RF are classified as non-ITS. The HVAC systems are designed to minimize the spread of radioactive contamination by filtration zones, and ensuring air flows from areas of low potential contamination toward areas of higher potential contamination. The subsurface ventilation system (classified as non-ITS; all event sequences involving a breach of a waste package are classified as beyond Category 2 (Section 1.7)) is designed to have two separate systems for the ventilation of the emplacement drifts and development area. The air pressure differential between the development side and emplacement side of the subsurface repository is maintained to ensure that airflow leakage travels from the development side (supply positive pressure system) to the emplacement side (exhausting negative pressure system) of the subsurface repository. As described in Section 1.4.2.2.1, continuous air monitors are located throughout the subsurface waste emplacement area, including the access main and alcoves. These instruments sample the air and collect airborne contaminants on a filter medium that is periodically removed and tested.

1.9.1.5 Means to Control Access to High Radiation Areas or Airborne Radioactivity Areas

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(e)]

Control of access to high and very high radiation areas is performed consistent with U.S. Nuclear Regulatory Commission Regulatory Guide 8.38. Section 5.11 describes the Operational Radiation Protection Program, including access control to restricted areas and within restricted areas. 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. Design features implemented for control of personnel access to these areas are non-ITS. In 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, procedural safety controls or ITS SSCs provide access controls to these areas.

1.9.1.6 Means to Prevent and Control Criticality

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f)]

Criticality is prevented in the preclosure period and controlled in the postclosure period. Preclosure nuclear criticality safety is discussed in Section 1.14. ITS design features (i.e., SSCs) and procedural safety controls for nuclear criticality safety are listed in Tables 1.9-2 to 1.9-7 and Table 1.9-10. Postclosure nuclear criticality FEPs for the repository are discussed in Section 2.2.1.4.1 and the criticality control measures that are considered to be important to waste isolation are identified in Table 1.9-8.

1.9.1.7 Radiation Alarm System

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(g)]

The radiation/radiological monitoring system discussed in Section 1.4.2.2 provides alarm annunciation to personnel when a threshold radiation level has been reached. The radiation monitoring system is not relied upon to alert the operator to take manual actions in response to an event sequence. In addition, the system does not initiate automatic actions required to prevent or mitigate an event sequence. Therefore, the radiation monitoring system is classified as non-ITS. The radiation detectors interlocked with the shield doors separating the waste package loadout areas in the IHF and CRCF are not part of the radiation monitoring system. These radiation detectors are interlocked with the ITS shield doors to ensure that the shield doors are not inadvertently opened if high radiation conditions (due to the presence of a loaded, sealed waste package) are present. Therefore, the dedicated shield door interlock radiation detectors are ITS.

1.9.1.8 Ability of Structures, Systems, and Components to Perform Their Intended Safety Functions

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(h)]

The PCSA process is described in Section 1.6.1. This process is used to identify the safety functions of ITS SSCs that must be available during the occurrence of an event sequence. The credited safety functions and procedural safety controls for ITS SSCs are identified in Tables 1.9-2 to 1.9-7 and Table 1.9-10. As described in Section 1.7, reliability assessments have been performed as part of the PCSA to demonstrate that the reliability estimations of ITS SSCs and procedural safety controls are achievable. These analyses produce a documented estimate (with uncertainties) for the reliability associated with the safety functions of the analyzed SSCs. The reliability assessment process is applicable to the reliability estimates of ITS SSCs that are relied upon in the PCSA for the prevention or mitigation of event sequences during the repository preclosure period and to the establishment of the frequencies of the analyzed initiating events.

To ensure that SSCs perform their functions to an appropriate level of reliability, they are qualified for the range of environmental conditions under which they are anticipated to operate, as discussed in Section 1.13. In addition, as described in Sections 1.2.1 and 5.6, a program is implemented that monitors the operation of SSCs and detects deviations from proper operations that are indicative of a degraded state of reliability. The program also initiates appropriate corrective action.

The applicable design criteria used to ensure that the nuclear safety design basis requirements are satisfied are set forth in Sections 1.2, 1.3, 1.4, and 1.5 for ITS SSCs. Section 1.13 describes the equipment qualification program that has been established to ensure that ITS SSCs can accomplish their intended function under the environmental conditions present at the time of functional demand.

Section 1.2.2.2 identifies the seismic design criteria for ITS SSCs if their performance is required to prevent or mitigate a Category 1 or Category 2 event sequence caused by a seismic event.

1.9.1.9 Explosion and Fire Detection and Suppression Systems

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(i)]

Event sequence analyses identify the fire and explosion scenarios that might occur in the surface and subsurface facilities of the repository. Based on the results of these analyses to support the PCSA, fire and explosion event sequences resulting in the release of radionuclides to the environment have been categorized as beyond Category 2, except for fire event sequences associated with transportation casks containing uncanistered commercial spent nuclear fuel (SNF). Section 1.4.3 describes the fire protection program and design of the detection and suppression systems. This description includes the codes, standards, and analyses used for the location and installation of detection and suppression components. Interactions with the ventilation systems are addressed in Section 1.4.3. However, no credit is taken in the PCSA event sequence analysis for either fire protection or fire detection. Therefore, no SSCs in the fire detection or fire suppression subsystems are ITS for the detection or prevention of fires. It should be noted, however, that portions of the fire suppression system (solenoid valves, sprinkler heads, etc.) have been classified as ITS for the prevention of inadvertent suppression in moderator controlled areas.

Administrative controls minimize the potential for fires that could initiate an event sequence and ensure that a fire in these areas is not of sufficient intensity, duration, or magnitude to initiate an event sequence. Those controls include limiting the presence of combustibles and flammable material in areas in which SNF or HLW are present. The PCSA includes these controls when analyzing the severity of fires in the fire event sequence analyses. The PCSA takes credit for these controls in the fire event sequence analysis.

1.9.1.10 Means to Control Radioactive Waste and Effluents and to Permit Prompt Termination of Operations and Evacuation of Personnel during an Emergency

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(j)]

The radioactive waste management system description is presented in Section 1.4.5. The radiation/radiological monitoring system is described in Section 1.4.2. The radiation/radiological monitoring system performs monitoring of radioactive effluents and alerts personnel of any need to evacuate specific areas. The digital control and management information system (DCMIS) and the communications system described in Section 1.4.2 provide the mechanisms to facilitate a controlled termination of operations and evacuation of personnel, if required. SSCs in these systems are not relied upon to prevent or mitigate event sequences and, therefore, are non-ITS.

As described in Section 1.2.2.3, the surface facility HVAC systems provide flow control and filtration during normal operation to ensure airflow from areas of low to high potential contamination and minimize contamination in facility effluents.

The surface facilities mitigate the potential release of radioactivity (in the event of an event sequence that includes a release of radionuclides from casks or canisters containing HLW or SNF) with HVAC systems that pass exhaust from the confinement zones through HEPA filters before it is discharged to the atmosphere. Penetrations through walls and slabs are sealed to maintain the confinement zone boundaries. These confinement measures serve to control airborne radioactive waste and effluents in the handling facilities. No radioactive liquid effluents are discharged from the repository to the environment.

The Emergency Plan is not relied upon to prevent or mitigate event sequences. As described in Section 5.7, the Emergency Plan will provide a discussion of each type of accident that could result in the release of radioactive material and a description of the means for detecting initiating events and accident conditions that apply to each accident identified in the Emergency Plan. It also will describe the rationale for the locations and types of detection devices used to detect accidents.

1.9.1.11 Electrical Power

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(k)]

The electrical power system is connected to the offsite grid and is backed up by standby and ITS diesel generators. As described in Section 1.4.1, the electrical power system includes batteries and inverters and uninterruptible power supplies so that instrumentation, controls, and monitoring systems can continue to function in the event of a loss of power. The normal power subsystem (including the standby diesel generators), normal direct current electrical power supply subsystem, and site electrical distribution subsystem (for normal power) are classified as non-ITS. These non-ITS subsystems are not relied upon to prevent or mitigate the consequences of an event sequence. These subsystems are described in Section 1.4.1.

The ITS power subsystem provides power to ITS systems and equipment that require electrical power to perform a safety function in the event that the normal power source is lost, including the ITS HVAC systems. The ITS diesel generators and feeders, up to and including ITS loads that are included in the ITS power subsystem, are classified as ITS. The ITS diesel generators are discussed in Section 1.2.8. The ITS direct current power subsystem and ITS uninterruptible power supply subsystem are classified as ITS. Tables 1.9-3 and 1.9-4 provide the design bases for the ITS power subsystem SSCs for the CRCF and WHF, respectively, based on the need to provide electrical power to the confinement and cooling HVAC system components for these facilities.

A loss of offsite or onsite electrical power is an off-normal operating condition, but such a loss of power does not initiate an event sequence. Upon a loss of power, active components stop in a safe configuration. The SSCs that prevent an event sequence upon loss of power are classified as ITS. Immediate restoration of offsite electrical power is not required to prevent or mitigate an event sequence. However, the reliability of ITS electrical systems is specified as a controlling parameter and value to reduce the probability of loss of ITS HVAC following a sequence of events that causes a radionuclide release.

The reliability requirements of the ITS electrical power system are provided in Tables 1.9-3 and 1.9-4 for the CRCF and WHF, respectively. Upon a loss of electrical power, equipment motion stops and loads are retained. There are no impacts or collisions caused by a loss of power. Shield doors remain in position to protect facility workers from high radiation fields. Through the use of commercial nuclear power plant reliability information, the frequency of occurrence of a loss of offsite and onsite electrical power (blackout condition) with coincident breach of containment barriers has been calculated to be beyond the Category 2 threshold (Section 1.7).

1.9.1.12 Redundant Systems and Inherent Reliability
[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(l)]

SSCs classified as ITS may be designed with internal components that are redundant or diverse, or both, to meet the design basis reliability demands.

As described in Section 1.2.2.2, ITS cranes, trolleys, transporters, and other SNF- and HLW-handling equipment incorporate redundant and diverse design features to ensure a high degree of reliability with low probability of failure. Heavy-lift cranes and canister transfer machines are designed to ASME NOG-1-2004 and, as such, have redundant design features for load-bearing components, braking systems, and travel limit switches. Redundancy is inherent in the design for features such as inadvertent motion interlocks that reduce the probability of event sequences associated with lifting equipment, casks, and canisters. The analysis of redundancy includes dependencies among alternative methods for achieving a function. Dependency among components and functions tends to reduce the effectiveness of redundancy below expectations if the same components and functions were analyzed as independent. As an example of diversity, operational limits for lifts and transfer motions are coded in the digital control system and provide limits to operator controls, while hard-wired limit switches are relied upon to ensure that limits for lifts and transfer motions do not exceed the specified limits. The human reliability analysis, in conjunction with the fault tree analysis described in Section 1.7, includes the dependencies associated with human and equipment malfunctions.

In recognition that common-cause failures apply to identical redundant components operating at the same time, ITS safeguards (such as means to prevent two-blocking on cranes) often use diverse means to preclude common-cause failures. For example, adjustable speed drives that limit lift heights are backed up by electro-mechanical safety limit switches, which in turn are backed up by wire tension attenuation features (such as a slip clutch) to prevent drops associated with lifting loads too high (from a two-block configuration). The surface confinement HVAC ITS subsystems have redundant trains for operation. However, this HVAC system is operated with one train running while the other is in standby, with the trains alternating between these two modes. Since both trains are periodically operated, undetected common-cause failure factors are reduced for this configuration.

The canister transfer machine, cask transfer trolley, and waste package transfer trolleys (all of which are ITS) are designed with inherent safeguards to prevent damage to waste containers due to collisions and to prevent waste container tipovers. For example, motor speeds are incapable of moving the canister transfer machine at speeds greater than 20 ft/min. At this speed, damage to casks or canisters due to a collision is calculated (Section 1.7) to be of such low probability that it may be considered not credible. In the case of the cask transfer trolley, the low aspect ratio of the trolley and the low coefficient of friction of the trolley bottom surface will result in the trolley sliding rather than tipping over. In the case of the waste package transfer trolley, the ITS trolley seismic restraints grip the trolley rails to reduce the probability of tipover.

The ITS power subsystem consists of two independent and physically separated ITS diesel generators, each with an associated 13.8-kV ITS switchgear and distribution (Section 1.4.1.2.2). The ITS diesel generators, the ITS HVAC equipment relied upon to cool the switchgear, as well as the mechanical systems that support the operations of the ITS diesel generators, are located at the Emergency Diesel Generator Facility (Section 1.2.8). The two ITS 13.8-kV trains are independent and redundant; one train is adequate to satisfy safety requirements. The ITS diesel generators are electrically isolated from each other. Physical separation for fire and missile protection is provided between the ITS diesel generators because they are housed in separate rooms of the Emergency Diesel Generator Facility. Power and control cables for the ITS diesel generators and associated switchgear are routed to maintain physical separation. Redundant and independent ITS buses allow maintenance to be performed on the equipment of one load group while the equipment of the other load group is in service. Cables associated with each ITS power supply load group are run in separate conduits, cable trays, or ducts.

The ITS 125-V DC power supply provides control power for tripping and closing the 13.8-kV ITS switchgear circuit breakers (Section 1.4.1.3.1). The supply consists of two redundant and independent 125-V DC battery banks with their associated chargers and distribution panels.

The ITS uninterruptible power supply units are supplied from an ITS 480-V AC source. The system contains independent and redundant uninterruptible power supplies, each supplying an associated bus by battery charger through a static inverter. The components of the ITS uninterruptible power supply units are similar to the uninterruptible power supply units found in the normal power supply subsystem. These ITS uninterruptible power supply units are located in the facilities in which these units are required.

Both ITS diesel generators are maintained on standby status. In this configuration, they are susceptible to common-cause failure to start and run for their full mission time. Common-cause failures are included in the fault tree models using the method described in Section 1.7. ITS and non-ITS circuits are separated and isolated such that failure in non-ITS circuits will not impact ITS circuits (Section 1.4.1). The internal fire analysis, however, includes the potential of fire from ITS or non-ITS sources starting and spreading in any location of the building.

Communication is provided between the non-ITS DCMIS on the surface and the ITS transport and emplacement vehicle (TEV) in the subsurface through the communications system. Signals from operations pass through routers that direct the control information to the proper set of redundant radio frequency transceivers that communicate with the TEV (Section 1.4.2.4.2.1). For redundant instrumentation or equipment, the DCMIS utilizes an input and output partitioning design philosophy. This philosophy ensures that no redundant instrumentation or equipment shares common input or output modules. The system components are fully modular to enable online replacement of defective parts under power. The system has built-in test capabilities to perform diagnostics without affecting the system performance. The DCMIS is powered from an uninterruptible power supply to ensure that it can perform monitoring functions during loss of power, as discussed in Section 1.4.2.1.1. The DCMIS is not relied upon to reduce the frequency or mitigate the consequences of Category 1 or Category 2 event sequences. Missed or inadvertent permissives from the DCMIS to the TEV are included in the fault tree analyses. TEV-related event sequences rely on onboard electrical or mechanical features, independent of the DCMIS, for event sequence frequency reduction. These features are ITS.

The methodology and results of the event sequence quantification described in Section 1.7 include the relevant redundancy, dependency, and inherent features to categorize event sequences.

1.9.1.13 Inspection, Test, and Maintenance Programs
[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(m)]

Section 5.6 contains a description of the plans for conduct of normal activities including information about inspection, test, and maintenance programs. In addition, Sections 1.2, 1.3, 1.4, and 1.5 identify unique inspection, test, or maintenance features associated with a particular SSC.

10 CFR 63.43 requires that the U.S. Nuclear Regulatory Commission include in the license to receive and possess SNF and HLW specific restrictions or controls that are derived from the analyses and evaluations included in the license application. In accordance with 10 CFR 63.21(18), Section 5.10 identifies probable subjects of such restrictions or controls. The proposed subjects of license specifications include limiting conditions for operation of selected SSCs. These limiting conditions for operation will include specific surveillance requirements, including appropriate functional testing and other inspections, required to provide confidence that the SSCs subject to limiting conditions for operation are capable of performing their design functions.

The reliability of an ITS SSC to perform its function is monitored, as applicable, under maintenance programs as a part of management systems to ensure proper reliabilities are met and maintained throughout operations, and to ensure that the classifications of event sequences (as Category 1, Category 2 or beyond Category 2) are not impacted. If the reliability required by the PSCA analysis is not achieved, appropriate corrective actions are taken to restore the reliability to the value used in the PCSA.

1.9.1.14 ITS Structure, System, or Component/Non-ITS Structure, System, or Component Interactions

The event sequences evaluated in the PCSA, and the SSCs classified as ITS in this analysis, are based on direct wasteform-handling operations and confinement operations. Event sequences were not developed for equipment and operations that are not directly involved with the handling of waste containers. All such equipment, therefore, is classified as non-ITS, as described in Section 1.9.1.

Due to the variety of functions performed by both the ITS SSCs and non-ITS SSCs at the repository, instances where ITS and non-ITS SSCs could potentially interact have been identified. These interactions could include the following four potential interaction categories:

- **Functional Dependence**—One component or system depends on another to supply vital functions. An example of this type of dependence is an ITS confinement HVAC system that is dependent on the electrical power system (which is comprised of both ITS and non-ITS SSCs).

- **Environmental Dependence**—System functionality depends on maintaining the environment within designed or qualified limits. An example of environmental dependence is a non-ITS waste package survey system or non-ITS waste package decontamination system ensuring that a waste package is in compliance with the prescribed surface contamination limits.

- **Spatial Dependence**—One system or component fails by virtue of close proximity to another. The failure could potentially be caused by common events such as seismic, fire, flood, and other external events. An example of spatial dependence is an interaction between an ITS cask-handling crane and a non-ITS maintenance crane. Internal hazards such as rotating missiles, overpressurized components, gravitational missiles, and inadvertent fire suppression can also cause failures due to spatial dependencies.

- **Human Dependence**—A system, component, or function fails because of a human activity involved with the process and its associated ITS and/or non-ITS SSCs. Examples of this dependence include any activity involving human interaction with ITS mechanical-handling equipment (e.g., a crane or a spent fuel transfer machine).

Portions, parts, subparts, or subsystems of a non-ITS SSC could potentially fail and adversely interact with an ITS SSC and prevent it from performing its safety function.

Therefore, interactions of the non-ITS SSCs with ITS SSCs are managed using the following SSC interaction criteria:

1. The interaction will be prevented or eliminated through either a redesign or through the use of operating procedures. If not,

2. The probability of the event sequence involving the interaction will be shown to be below the Category 2 range; or

3. The calculated dose consequence for the event sequence involving the interaction will be shown to be in compliance with the performance requirements of 10 CFR 63.111(b).

To implement this strategy, the functions of the ITS and non-ITS SSCs are required to be explicitly identified. A determination is then made as to which of the three criteria for the SSC interaction strategy are applicable.

None of the identified interactions demonstrate a need for a redesign of the ITS or non-ITS SSCs. Where applicable, operating procedures are relied upon to reduce the frequency of interaction between the respective ITS SSCs and non-ITS SSCs (BSC 2008l).

1.9.2 Identifying Postclosure Performance Assessment Design Control Parameters and Classifying ITWI Structures, Systems, and Components
[NUREG-1804, Section 2.2.1.1.3: AC 1]

The postclosure performance assessment analyzes the natural environment and engineered component performance after closure of the repository. The performance assessment presented in Chapter 2 considers the natural and engineered FEPs that are potentially significant to the performance of the repository. The relevant FEPs included in or excluded from the performance assessment are presented in Section 2.2. The model abstractions for the FEPs included in the modeling of the performance assessment are presented in Section 2.3. The analyses of the natural and engineered FEPs, including the long-term process effects and interactions that contribute to the performance of the repository system, are presented in Section 2.4. Finally, the natural features of the natural barriers and the engineered features of the Engineered Barrier System that are considered ITWI as well as the FEPs that are significant in contributing to the capability of these barriers are presented in Section 2.1. ITWI is a determination assigned to a barrier or a barrier's feature/component, based on its capability of preventing or substantially reducing the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment, or preventing the release or substantially reducing the release rate of radionuclides from the waste. In addition, ITWI includes those engineered features/components of the geologic repository whose function is to prevent or mitigate the consequences of potential disruptive events (e.g., criticality), as well as consumable materials to be incorporated into any engineered item important to waste isolation during fabrication of that item (SNL 2008, Section 6.1).

A feature is classified as ITWI if it meets two conditions. The first condition is that the feature is associated with one or more characteristics classified as important to barrier capability. The second condition is that the feature is a significant contributor to the barrier capability relative to the other features of the barrier. The details of the barrier capability analyses are presented in Section 2.1.

The waste isolation classification of the three principal barriers (the Upper Natural Barrier, the Engineered Barrier System, and the Lower Natural Barrier), as well as the related features of the geologic setting and the design features (and relevant SSCs) that are ITWI, are identified in Table 1.9-8. In addition, Table 1.9-8 indicates the barrier function(s) that are fulfilled by each feature or SSC and the relevant design control parameter associated with each ITWI feature or SSC.

These control parameters are quantities or variables that define or support a contribution to barrier capability or a model describing that capability (SNL 2008, Section 6.1). These design control parameters reflect the important aspect of the feature or SSC that must be controlled by either configuration management or procedural safety controls. The description of each design control parameter, whether relevant to the ITWI classification or to the postclosure analyzed basis, is presented in Table 1.9-9. The postclosure design control parameters identified in Table 1.9-9 serve a similar role as the preclosure controlling parameters identified in Tables 1.9-2 to 1.9-7.

Table 1.9-9 contains a summary of the parameters that require controls to ensure the postclosure performance assessment analytical bases are established during design, construction, procurement, operations, and closure. The parameters are grouped into eight engineered subsystem categories (related to features or SSCs): subsurface facilities, emplacement drift configuration, emplacement drift ventilation, drip shield, waste package, waste form and transportation, aging, and disposal canister, pallet, and closure. Each design control parameter is indicated by a numeric identifier (e.g., 01-01) and a corresponding title. Each control parameter then has either a control parameter value or range of values or a constraint relevant to the design, operations, or construction of the SSC. Some of the controls are related to controlled interface parameters. These represent parameters that are controlled through the configuration management system presented in Section 5. Table 1.9-9 also identifies which control parameters are relevant to the ITWI classification and the approach applied to control each parameter.

Table 1.9-9 also indicates whether the control parameter is related to a procedural safety control (indicated by a “1” in the fifth column) or a design configuration (indicated by a “2” in the fifth column). Procedural safety controls apply when there are specific and unique operator, inspector, or verification activities required by the control parameter that are not addressed by standard administrative controls such as those management systems identified in Chapter 5. Design configuration includes general configuration control as well as fabrication, welding, and other specifications for items that are expected to be procured. Both the procedural safety controls and design configuration are controlled by management systems identified in Chapter 5. Management systems are used throughout the life of the repository to control activities and integrate programs to provide assurance that the repository will be constructed and operated within analyzed conditions and that the validity of the design and analytical bases is maintained as modifications occur. The management systems are implemented through procedures governing work processes in accordance with the Quality Assurance Program.

In addition to identifying the parameter to be controlled, Table 1.9-9 indicates where each parameter is described in the repository design. Each section (or corresponding table referenced from that section) presents the design criteria/configuration or procedural safety control associated with that control parameter. Additional information associated with how and where each control parameter is used in the postclosure analyses and models and whether that control parameter is used to support the basis to exclude a FEP or used as a basis to model an included FEP is presented in Table 2.2-3.

Postclosure control parameters will be subject to quality assurance controls applied in accordance with 10 CFR 63.142. Through the application of the controls, the analytical basis of the TSPA will be established during the preclosure.

The Performance Confirmation Program (Section 4) is designed to confirm the performance of the natural and engineered systems that are assumed or designed to operate as barriers after repository closure.

1.9.3 Procedural Safety Controls

[NUREG-1804, Section 2.1.1.6.3: AC 2]

The PCSA and the postclosure analysis rely upon the management systems described in Chapter 5 being in place and functioning as required. In addition to the general requirements of the management systems, specific procedural safety controls have been identified. Procedural safety controls are activities performed by both repository and nonrepository personnel whose actions affect repository activities to ensure that operations are within the analyzed conditions of the PCSA and TSPA.

Procedural safety controls are documented and controlled specific actions, or a series of actions, taken by the operating staff in preparation for, or during the execution of, waste handling and emplacement operations. Procedural safety controls implement human activities that:

- For preclosure, are relied upon to reduce the likelihood of an initiating event or an event sequence

- For preclosure, are relied upon to mitigate the consequences of an event sequence

- For postclosure, are relied upon to ensure that the control parameters in the FEPs and TSPA are satisfied.

Procedural safety controls are derived from:

- The screening analyses of initiating events described in Section 1.6
- The event sequence quantification analyses described in Section 1.7
- The radiological consequence analyses described in Section 1.8
- The criticality control measures described in Section 1.14
- The postclosure analyses described in Section 2.1.

Procedural safety controls are not intended to be used as a substitute for normal operating practices. There are no identified procedural safety controls for manual actions followed to terminate event sequences. Procedural safety controls are derived from the preclosure and postclosure safety analyses. However, procedural safety controls may be implemented as an individual written procedure or they may be subsumed into normal operating procedures (e.g., for alignment of HVAC systems), administrative controls (e.g., the fire protection program), management controls (Chapter 5), or the radiation protection program (Section 5.11).

Procedural safety controls are identified in Tables 1.9-9 and 1.9-10. The description of the procedural safety control includes the applicable facility and SSC, a statement of the control, and the basis for including the procedural safety control.

The use and content of the procedural safety controls is determined by the nature of the event sequence and the existence (or non-existence) of ITS SSCs that can be relied upon to prevent or mitigate the event sequence. Elements of the management systems contribute to the effective implementation of the procedural safety controls, such as procedures, training, maintenance, configuration control, human factor evaluations, and audits and self-assessments.

Procedural safety controls are relied upon in the PCSA to prevent or mitigate event sequences. The procedural safety controls relied upon in the preclosure period are captured in *Preclosure Procedural Safety Controls* (BSC 2008i). Procedural safety controls are relied upon in the postclosure analysis to ensure the controlling parameters of the postclosure analysis are satisfied. The postclosure procedural safety controls are captured in *Postclosure Modeling and Analyses Design Parameters* (BSC 2008k). The postclosure procedural safety controls are included in Table 1.9-9 and are identified by a “1” in the fifth column of the table. The preclosure procedural safety controls are listed in Table 1.9-10.

Emergency operating procedures are not relied upon to terminate an event sequence or to mitigate the radiological consequences in order to meet the performance objectives of 10 CFR 63.111. Emergency operating procedures that may include manual operator actions for recovery or restoration after an event sequence is terminated are separate from procedural safety controls. Emergency preparedness procedures enable further mitigation of consequences from Category 1 or Category 2 event sequences, but such procedures are not relied upon to demonstrate compliance with the 10 CFR 63.111 performance objectives for radiation exposure.

1.9.4 Risk Significance Categorization

[NUREG-1804, Section 2.1.1.6.3: AC 3]

Although 10 CFR 63.142(c)(1) allows a graded approach for the application of the Quality Assurance Program to ITS and ITWI SSCs based on their importance to safety, this approach has not been taken. The quality assurance program is applied uniformly to SSCs that have been identified as ITS or ITWI. SSCs are classified as ITS based on their performance of event sequence preventive or mitigative functions to ensure that radiation doses meet the performance requirements of 10 CFR 63.111(b). Similarly, SSCs are classified as ITWI based on their contribution to barrier performance, as discussed in Section 2.1.

1.9.5 General References

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Table 1.9-1. Preclosure Safety Classification of SSCs

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Aging Facility	Aging Facility	Aging pad	ITS
		Horizontal aging module (170-HAC0-ENCL-00001)	ITS
		Mobile platform (170-HAP0-PLAT-00001/00002)	Non-ITS
		Support structures (including utility buildings, if applicable)	Non-ITS
	Aging Handling/Cask Transfer	Cask tractor (for use with the cask transfer trailer) (170-HAT0-HEQ-00001)	ITS
		Cask transfer trailers (for use with transportation casks and horizontal shielded transfer casks) (PWR DPC: 170-HAT0-TRLY-00001) (BWR DPC: 170-HAT0-TRLY-00002)	ITS
		Mobile cranes (170-HAT0-CRN-00001/00002)	Non-ITS
		Site transporter (170-HAT0-MEQ-00001)	ITS
	Aging Handling/Aging Overpack	Horizontal shielded transfer cask (for use with horizontal aging module) (170-HAC0-HEQ-00001)	ITS
		Aging overpack (TAD: 170-HAC0-ENCL-00003) (Vertical DPC: 170-HAC0-ENCL-00002)	ITS
Balance of Plant Facilities	Balance of plant facilities that include administration, security, utilities, emergency response, offsite, warehouse and nonnuclear receipt; materials and consumables, maintenance and repair, transportation, balance of plant construction, Central Control Center, and infrastructure	Structures	Non-ITS
	Balance of Plant	Roads, Rails for Commercial Railcars	Non-ITS
		Surface Rails for the TEV	Non-ITS
		Flood Control Features	ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Canister Receipt and Closure Facility	Canister Receipt and Closure Facility (CRCF)	Structure	ITS
		Rails for the Commercial Railcars (Inside the Building)	Non-ITS
		Rails for the TEV (Inside the Waste Package Loadout Room)	ITS
		Shield windows	Non-ITS
		Shield Doors (Including Anchorages) and Equipment Confinement Doors	ITS
		ALARA Shielding Features	Non-ITS
		DOE Canister Slide Gates (060-HTC0-HTCH-00005/00006/00007/00008/ 00009)	ITS
		Cask Port Slide Gates (060-HTC0-HTCH-00001/00002)	ITS
		TAD Slide Gates (060-HTC0-HTCH-00010/00011)	ITS
		Waste Package Port Slide Gates (060-HTC0-HTCH-00003/00004)	ITS
		Cask Preparation Platform (060-HMH0-PLAT-00001)	ITS
		Waste Package Transfer Carriage Docking Stations (060-HL00-75-00001/00002)	Non-ITS
		CRCF Loadout Platforms (060-HL00-PLAT-00001/00002/00003)	Non-ITS
Cask/Canister/ Waste Package Process System	Cask Cavity Gas Sampling	Entire (IHF, RF, CRCF, WHF)	Non-ITS
	Cask Cooling	Cask/Dual-Purpose Canister (DPC) Overpressure Protection Features (WHF)	ITS
		System Components Other than Cask/DPC Overpressure Protection Features (WHF)	Non-ITS
	Cask, Canister, and Waste Package Inerting	Entire (IHF, CRCF, WHF)	Non-ITS
	Decontamination Water Treatment	Entire (WHF)	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Cask/Canister/ Waste Package Process System (Continued)	Waste Package Survey	Entire (IHF, CRCF)	Non-ITS
	Waste Package Decontamination	Entire (IHF, CRCF)	Non-ITS
	TAD Canister Drying	Entire (WHF)	Non-ITS
Communications System	Communications	Entire	Non-ITS
Digital Control and Management Information System	Digital Control and Management Information System	Entire	Non-ITS
DOE and Commercial Waste Package System	DOE and Commercial Waste Package	Entire	ITS
	High-Level Waste/DOE SNF Codisposal	DOE Standardized Canister	ITS
		HLW Canister	ITS
	Canistered Spent Nuclear Fuel	DPC	ITS
		TAD Canister	ITS
Electrical Power System	Switchyard and Standby Power	Entire	Non-ITS
	ITS Power	ITS Distribution (Feeders Up to and including ITS Loads, ITS Direct Current Power, ITS Uninterruptible Power Supply Power)	ITS
		ITS Diesel Generators A and B (including ITS diesel generator fuel oil system, ITS diesel generator air start system, ITS diesel generator jacket water cooling system, ITS diesel generator lubricating oil system, ITS diesel generator air intake and exhaust system)	ITS
	Emergency Power (Life Safety)	Entire	Non-ITS
	Normal Power	Entire	Non-ITS
	Normal Direct Current Electrical Power	Entire	Non-ITS
	Normal Uninterruptible Power Supply Power	Entire	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Electrical Power System (Continued)	Site Electrical Distribution (for Normal Power)	Entire	Non-ITS
	Renewable Energy	Entire	Non-ITS
	Standby Diesel Generator	Entire	Non-ITS
Electrical Support System	Lighting	Entire	Non-ITS
	Grounding	Entire	Non-ITS
	Lightning Protection	Entire	Non-ITS
	Cathodic Protection	Entire	Non-ITS
	Heat Tracing	Entire	Non-ITS
	Cable Raceway	Entire	Non-ITS
Emergency Diesel Generator Facility	Emergency Diesel Generator Facility	Structure	Non-ITS
Emplacement and Retrieval/ Drip Shield Installation System	Emplacement and Retrieval /Drip Shield Installation System	TEV	ITS
		Drip Shield Gantry	Non-ITS
		Inspection Gantry	Non-ITS
Environmental/ Meteorological Monitoring System	Environment and Meteorological Monitoring	Entire	Non-ITS
Fire Protection System	Fire Water	Entire	Non-ITS
	Fire Barriers	Entire	Non-ITS
	Explosion Protection	Entire	Non-ITS
	Fire Suppression	Preaction valves, sprinkler heads, and system actuation panels associated with double-interlock preaction suppression systems that protect areas where there is a potential for canister breach (CRCF, WHF)	ITS
		Fire suppression system components other than those associated with double-interlock preaction suppression systems that protect areas where there is a potential for canister breach	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Fire Protection System (Continued)	Fire Detection	Fire Detection System for the ITS preaction valves with associated detectors and control box (CRCF, WHF)	ITS
		Fire Detection System or all other systems except the preaction valve with associated detectors and control box	Non-ITS
	Fire Alarm	Entire	Non-ITS
Initial Handling Facility	Initial Handling Facility	Structure	ITS
		Rails for the Commercial Railcars (Inside the Building)	Non-ITS
		Rails for the TEV (Inside the Waste Package Loadout Room)	ITS
		Shield Doors (Including Anchorages)	ITS
		ALARA Shielding Features	Non-ITS
		Cask Port Slide Gate (51A-HTC0-HTCH-00001)	ITS
		Waste Package Port Slide Gate (51A-HTC0-HTCH-00002)	ITS
		Cask Preparation Platform (51A-HMH0-PLAT-00001)	ITS
		Waste Package Transfer Carriage Docking Station (51A-HL00-75-00001)	Non-ITS
		IHF Loadout Platforms (51A-HL00-PLAT-00001/00002)	Non-ITS
Low-Level Radioactive Waste Management System	Low-Level Radioactive Waste Management	Entire	Non-ITS
Low-Level Waste Facility	Low-Level Waste Facility	Structure	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Mechanical Handling System	Cask Handling	Transportation Cask	ITS
		Site Prime Mover	ITS
		Cask Handling Yoke (CRCF: 060-HM00-BEAM-00001) (IHF: 51A-HM00-BEAM-00001) (RF: 200-HM00-BEAM-00001) (WHF: 050-HM00-BEAM-00001)	ITS
		Pool Cask Handling Yoke (WHF: 050-HM00-BEAM-00002)	ITS
		Platform Shield Plate (RF: 200-HM00-BUF-00001) (CRCF: 060-HM00-BUF-00001/00002) (WHF: 050-HM00-BUF-00001)	Non-ITS
		Cask Handling Crane (IHF: 300-ton; 51A-HM00-CRN-00001) (CRCF: 200-ton; 060-HM00-CRN-00001) (RF: 200-ton; 200-HM00-CRN-00001) (WHF: 200-ton; 050-HM00-CRN-00001)	ITS
		Decontamination Pit Equipment—Spray Nozzle (WHF: 050-HM00-NZL-00001)	Non-ITS
		Decontamination Pit Equipment—Pump Module (WHF: 050-HM00-P-00001)	Non-ITS
		Long Reach Tool Adapter (WHF: 050-HM00-TOOL-00001)	Non-ITS
		Pool Yoke Lift Adapter (WHF: 050-HM00-TOOL-00002)	ITS
		Cask Transfer Trolley and Pedestals Trolleys: (IHF: 51A-HM00-TRLY-00001) (CRCF: 060-HM00-TRLY-00001/00002) (RF: 200-HM00-TRLY-00001) (WHF: 050-HM00-TRLY-00001) Pedestals: (IHF: 51A-HM00-PED-00001/00002) (CRCF: 060-HM00-PED-00001/00002) (RF: 200-HM00-PED-00001) (WHF: 050-HM00-PED-00001/00002/00003/ 00004/00005) Naval Cask Pedestal: (IHF: 51A-HM00-PED-00003)	ITS
		Cask Preparation Crane; 30-ton (IHF: 51A-HM00-CRN-00002)	ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Mechanical Handling System (Continued)	Cask Handling (Continued)	Horizontal Cask Stand (RF: 200-HM00-RK-00001)	Non-ITS
		Mobile Lift (CRCF: 060-HM00-ELEV-00001) (RF: 200-HM00-ELEV-00001) (WHF: 050-HM00-ELEV-00001)	Non-ITS
	Cask Handling/Cask Receipt	Entrance Vestibule Crane; 20-ton (WHF: 050-HMC0-CRN-00001)	ITS
		Cask Tilting Frame (CRCF: 060-HMC0-FRM-00001) (RF: 200-HMC0-FRM-00001) (WHF: 050-HMC0-FRM-00001)	Non-ITS
		Mobile Access Platform (IHF: 51A-HMC0-PLAT-00001) (CRCF: 060-HMC0-PLAT-00001) (RF: 200-HMC0-PLAT-00001) (WHF: 050-HMC0-PLAT-00001)	Non-ITS
		Impact Limiter Lifting Device (IHF: 51A-HMC0-HEQ-00001/00002) (CRCF: 060-HMC0-HEQ-00001/00002/00003/ 00004/00005/00006/00007/00008/00019/00020) (RF: 200-HMC0-HEQ-00001/00003/00005/00007/ 00009/00011/00014) (WHF:050-HMC0-HEQ-00001/00002/00003/ 00004/00005/00006/00007/00008/00009)	Non-ITS
		Personnel Barrier Lifting Device (IHF: 51A-HMC0-HEQ-00003/00004) (CRCF: 060-HMC0-HEQ-00010/00011/00012/ 00013/00014/00015/00016/00017/00021/00022) (RF: 200-HMC0-HEQ-00002/00004/00006/ 00008/00010/00012/00013) (WHF: 050-HMC0-HEQ-00010-18)	Non-ITS
		Lid Bolting Room Crane; 10-ton (RF: 200-HMC0-CRN-00001)	ITS
		Naval Cask Lift Bail (IHF: 51A-HMC0-BEAM-00001)	ITS
		Naval Cask Lift Plate (IHF: 51A-HMC0-HEQ-00005)	ITS
		Horizontal Lifting Beam (RF: 200-HMC0-BEAM-00001)	ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Mechanical Handling System (Continued)	Cask Handling/Cask Preparation	Auxiliary Pool Crane; 10-ton (WHF: 050-HMH0-CRN-00001)	ITS
		Preparation Station Jib Cranes (1 and 2) (WHF: 050-HMH0-CRN-00002/00003)	ITS
		Cask Support Frame (Preparation Station #2) (WHF: 050-HMH0-FRM-00001)	ITS
		Cask Lid Lifting Grapples (CRCF: 060-HMH0-HEQ-00012) (RF: 200-HMH0-HEQ-00008) Lid Lifting Grapples (WHF: 050-HMH0-HEQ-00001/00002/00003/00004/00006) Truck Cask Lid Lifting Grapples (WHF: 050-HMH0-HEQ-00007/00008/00009)	ITS
		Truck Cask Lid Adapters (WHF: 050-HMH0-HEQ-00010/00011) Rail Cask Lid Adapters (WHF: 050-HMH0-HEQ-00012/00013) (CRCF: 060-HMH0-HEQ-00003/00004) (RF: 200-HMH0-HEQ-00002) (IHF: 51A-HMH0-HEQ-00002)	ITS
		Truck Cask Lid Adapters (CRCF: 060-HMH0-HEQ-00001/00002) (IHF: 51A-HMH0-HEQ-00001)	Non-ITS
		Cask Lid Bolt Impact Wrench (RF: 200-HMH0-HEQ-00003)	Non-ITS
		DPC Lid Adapter (CRCF: 060-HMH0-HEQ-00005/00006) (WHF: 050-HMH0-HEQ-00014) (RF: 200-HMH0-HEQ-00001)	ITS
		Cask Shield Ring (WHF: 050-HMH0-HEQ-00015/00016/00017/00018/00019)	Non-ITS
		Long Reach Grapple Adapter (WHF: 050-HMH0-TOOL-00001/00002)	ITS
	Cask Handling/Waste Package Preparation	Waste Package Handling Crane; 100-ton (IHF: 51A-HMP0-CRN-00001) (CRCF: 060-HMP0-CRN-00001)	ITS
		Waste Package Pallet Yoke (IHF: 51A-HMP0-BEAM-00001) (CRCF: 060-HMP0-BEAM-00001)	Non-ITS
	Cask Handling/Cask Restoration	Entire	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Mechanical Handling System (Continued)	Waste Transfer/Fuel Assembly Transfer	Spent Fuel Transfer Machine (WHF: 050-HTF0-FHM-00001)	ITS
		Pressurized Water Reactor Lifting Grapples (WHF: 050-HTF0-HEQ-00001) Boiling Water Reactor Lifting Grapples (WHF: 050-HTF0-HEQ-00002)	ITS
		W74 Upper Basket Lifting Device (WHF: 050-HTF0-HEQ-00003)	Non-ITS
		SNF Staging Racks (WHF: 050-HTF0-RK-00001 [PWR SNF]) (WHF: 050-HTF0-RK-00010 [BWR SNF]) (WHF: 050-HTF0-RK-00011 [DFCA SNF])	ITS
		Truck Cask Handling Frame (WHF: 050-HTF0-RK-00007)	ITS
	Waste Transfer/ Canister Transfer	Canister Transfer Machine Maintenance Crane; 15-tons (IHF: 51A-HTC0-CRN-00001) (CRCF: 060-HTC0-CRN-00001) (WHF: 050-HTC0-CRN-00001)	Non-ITS
		Canister Transfer Machine Maintenance Crane; 15-tons (RF: 200-HTC0-CRN-00001)	ITS
		Canister Transfer Machine (IHF: 51A-HTC0-FHM-00001) (CRCF: 060-HTC0-FHM-00001/00002) (RF: 200-HTC0-FHM-00001) (WHF: 050-HTC0-FHM-00001)	ITS
		Canister Grapples (IHF: 51A-HTC0-HEQ-00003/00004) (CRCF: 060-HTC0-HEQ-00003/00004/00005/ 00006/00007) Canister Transfer Machine Grapples (IHF: 51A-HTC0-HEQ-00001) (CRCF: 060-HTC0-HEQ-00001/00002) (RF: 200-HTC0-HEQ-00001) (WHF: 050-HTC0-HEQ-00001)	ITS
		Naval Canister Lifting Adapter (IHF: 51A-HTC0-HEQ-00005)	ITS
		DOE Waste Package Inner Lid Grapple (IHF: 51A-HTC0-HEQ-00007)	ITS
		Naval Waste Package Inner Lid Grapple (IHF: 51A-HTC0-HEQ-00008)	ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Mechanical Handling System (Continued)	Waste Transfer/ Canister Transfer (Continued)	TAD Canister Staging Racks (and Thermal Barrier) (CRCF: 060-HTC0-RK-00011/00012)	ITS
		DOE Canister Staging Racks (and Thermal Barrier) (CRCF: 060-HTC0-RK-00006/00007/00008/ 00009/00010)	ITS
		Shielded Transfer Cask (TAD: 050-HT00-HEQ-00001) (DPC: 050-HT00-HEQ-00002)	ITS
	Waste Package Closure	Robotic Arms. (IHF: 51A-HWH0-HEQ-00001/00002) (CRCF: 060-HWH0-HEQ-00001/00002)	Non-ITS
		Remote Handling System Bridge Included as part of: (IHF: 51A-HWH0-HEQ-00003) (CRCF: 060-HWH0-HEQ-00003/00005)	ITS
		Portions of Remote Handling System That Do Not Include the Bridge Included as Part of: (IHF: 51A-HWH0-HEQ-00003) (CRCF: 060-HWH0-HEQ-00003/00005)	Non-ITS
		Remote Handling System Manipulator Arm (IHF: 51A-HWH0-HEQ-00004) (CRCF: 060-HWH0-HEQ-00004)	Non-ITS
		Lid Handling Tool (IHF: 51A-HWH0-TOOL-00001) (CRCF: 060-HWH0-TOOL-00001)	Non-ITS
		Waste Package Closure Room Crane; 15-ton (IHF: 51A-HW00-CRN-00001) (CRCF: 060-HW00-CRN-00001)	Non-ITS
		Closure Support Room Cranes; 5-ton (CRCF: 060-HW00-CRN-00002 [north] /00003 [south])	Non-ITS
		Process Opening Cover (IHF: 51A-HW00-HTCH-00001) (CRCF: 060-HW00-HTCH-00001/00002)	Non-ITS
	TAD Closure	TAD Closure Jib Crane (WHF: 050-HC00-CRN-00001)	ITS
		Cask Support Frame (TAD Closure Station) (WHF: 050-HC00-FRM-00001)	ITS
		TAD Canister Welding Machine (WHF: 050-HC00-TOOL-00001)	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Mechanical Handling System (Continued)	Waste Package Loadout	Waste Package Shield Ring Lift Beam (IHF: 51A-HL00-BEAM-00001) (CRCF: 060-HL00-BEAM-00001)	Non-ITS
		Waste Package Shield Rings (IHF: 51A-HL00-HEQ-00001/00002) (CRCF: 060-HL00-HEQ-00001/00002/00003/ 00004/00005/00006)	ITS
		Waste Package Transfer Trolley (including Pedestals, Seismic Rail Restraints, and Rails) Trolleys: (IHF: 51A-HL00-TRLY-00001) (CRCF: 060-HL00-TRLY-00001/00002) Pedestals: (CRCF: 060-HL00-PED-00001/00002/00003/ 00004/00005/00006/00007/00008) (IHF: 51A-HL00-PED-00001/00002/00003/00004)	ITS
		Waste Package Transfer Carriage (IHF: 51A-HL00-TRLY-00002) (CRCF: 060-HL00-TRLY-00004/00005)	Non-ITS
	DPC Cutting	DPC Cutting Machine (WHF: 050-HD00-TOOL-00001)	Non-ITS
		Siphon Tube Shear Tool (WHF: 050-HD00-TOOL-00002)	Non-ITS
		DPC Cutting Jib Crane (WHF: 050-HD00-CRN-00001)	ITS
		Cask Support Frame (DPC Cutting Station) (WHF: 050-HD00-FRM-00001)	ITS
		DPC Lid Receptacle (WHF: 050-HD00-RCP-00001)	Non-ITS
		DPC Adapter Plate Types 1, 2, 3 (WHF: 050-HD00-HEQ-00002/00003/00004)	Non-ITS
		DPC Shield Plug Lift Adapter (WHF: 050-HD00-HEQ-00005)	Non-ITS
Naval SNF Waste Package System	Naval SNF Waste Package	Entire	ITS
	Naval SNF Canister	Entire	ITS
Non-Nuclear Handling System	Non-Nuclear Handling	Entire	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Non-Radiological Waste Management System	Non-Radiological Waste Management	Entire	Non-ITS
Plant Services System	Plant Services	Entire	Non-ITS
Pool Water Treatment and Cooling System	Pool Water Treatment and Cooling	Entire	Non-ITS
		Boron Makeup System	Non-ITS
Radiation/ Radiological Monitoring System	Radiation/ Radiological Monitoring	Entire	Non-ITS
Receipt Facility	Receipt Facility (RF)	Structure	ITS
		Rails for Railcars	Non-ITS
		Shield Doors (Including Anchorages)	ITS
		ALARA Shielding Features	Non-ITS
		Cask Port Slide Gate (200-HTC0-HTCH-00001)	ITS
		AO Port Slide Gate (200-HTC0-HTCH-00002)	ITS
		Cask Preparation Platform (200-HMH0-PLAT-00001)	ITS
		Lid Bolting Room Platform (200-HMC0-PLAT-00003)	ITS
Safeguards and Security System	Safeguards and Security	Entire	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Subsurface Facility	Subsurface Facility	Rails	Non-ITS
		Emplacement Drift Doors	Non-ITS
	Emplacement	Nonemplacement Openings	Non-ITS
		Ground Support for Emplacement Drifts	Non-ITS
		Ground Support for Nonemplacement Openings	Non-ITS
		Emplacement Drifts	Non-ITS
		Emplacement Drift Invert (Steel and Ballast)	Non-ITS
		Waste Package Emplacement Pallet	Non-ITS
		Drip Shield	Non-ITS
		Drip Shield Emplacement Gantry	Non-ITS
	Post-emplacement	Entire	Non-ITS
	Subsurface Development	Excavation	Non-ITS
Subsurface Ventilation System	Subsurface Ventilation	Entire	Non-ITS
Surface Nonconfinement HVAC System	Surface Nonconfinement HVAC	Portions of the surface nonconfinement HVAC system that do not support the cooling of ITS electrical equipment and battery rooms (IHF, CRCF, WHF, RF, Emergency Diesel Generator Facility (EDGF))	Non-ITS
		Portions of the surface nonconfinement HVAC system that support the cooling of ITS electrical equipment and battery rooms (EDGF)	ITS
Surface Nuclear Confinement HVAC System	Surface Nuclear Confinement HVAC	Portions of the surface nuclear confinement HVAC system that exhaust from areas with a potential for a breach (WHF and CRCF)	ITS
		Portions of the surface nuclear confinement HVAC system that support the cooling of ITS electrical equipment and battery rooms (WHF and CRCF)	ITS
		Portions of the surface nuclear confinement HVAC system that do not exhaust from areas with a potential for a breach or do not support the cooling of ITS electrical equipment and battery rooms, including SSCs that supply ITS confinement areas (IHF, CRCF, WHF, RF)	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Wet Handling Facility	Wet Handling Facility (WHF)	Structure	ITS
		Rails for Railcars	Non-ITS
		Shield Doors (Including Anchorages) and Equipment Confinement Doors	ITS
		ALARA Shielding Features	Non-ITS
		Pool Structure	ITS
		Cask Port Slide Gate (050-HTC0-HTCH-00002)	ITS
		Overpack Port Slide Gate (050-HTC0-HTCH-00001)	ITS
		Aging Overpack Access Platform (050-HAC0-PLAT-00001)	ITS
		TAD Closure Station (050-HC00-PLAT-00001)	ITS
		DPC Cutting Station (050-HD00-PLAT-00001)	ITS
		Preparation Station #1 (050-HMH0-PLAT-00001)	ITS
		Preparation Station #2 (050-HMH0-PLAT-00002)	ITS
		DPC Transfer Station (050-HTF0-RK-00002)	Non-ITS
		Staging Shelf Transfer Station (050-HTF0-RK-00008)	Non-ITS
		Staging Shelf Dual Transfer Station (050-HTF0-RK-00009)	Non-ITS
		DPC Unloading Bay Gate (050-WH00-DR-00002)	Non-ITS
		Deep Remediation Station (050-HR00-RK-00001)	Non-ITS
		Rail Cask Transfer Station (050-HTF0-RK-00004)	Non-ITS
		Shielded Transfer Cask/TAD Transfer Station (050-HTF0-RK-00003)	Non-ITS

Table 1.9-1. Preclosure Safety Classification of SSCs (Continued)

System or Facility	Subsystem or Function (as Applicable)	Component	Preclosure Safety Classification
Wet Handling Facility (Continued)	Wet Handling Facility (Continued)	Truck Cask Transfer Station (050-HTF0-RK-00005)	Non-ITS
		(Pool) Crush Pads (050-HM00-ABS-00001/00002/00003/00004/ 00005)	Non-ITS
		Decontamination Pit; Decontamination Pit Seismic Restraints (050-HM00-BRAC-00001)	ITS
		Decontamination Pit Cover (050-HM00-HTCH-00001)	Non-ITS
		Decontamination Pit Platform (050-HM00-PLAT-00002)	Non-ITS

NOTE: The numbers appearing in parentheses are component numbers. ALARA shielding features for the CRCF, IHF, and RF include the shielding function of the platforms. ALARA shielding features for the WHF include the shielding function of the platforms, the decontamination pit, and the cask preparation area equipment confinement door.
ALARA = as low as is reasonably achievable; BWR = boiling water reactor; DPC = dual-purpose canister; EDGF = Emergency Diesel Generator Facility; PWR = pressurized water reactor; TAD = transportation, aging, and disposal.

Source: BSC 2008h.

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS)	DOE and commercial waste package	Entire	Provide containment	DS.IH.01. The mean conditional probability of breach of a sealed waste package resulting from a side impact shall be less than or equal to 1×10^{-8} per impact.	IHF-ESD-11-HLW (Seq. 4-6)	Table 1.5.2-6
				DS.IH.02. The mean conditional probability of breach of a sealed waste package resulting from a drop of a load onto the waste package shall be less than or equal to 1×10^{-5} per drop.	IHF-ESD-11-HLW (Seq. 3-6)	Table 1.5.2-6
				DS.IH.03. The mean conditional probability of breach of a sealed waste package resulting from an end-on impact or collision shall be less than or equal to 1×10^{-5} per impact.	IHF-ESD-11-HLW (Seq. 5-6)	Table 1.5.2-6
	HLW	HLW Canister	Provide containment	DS.IH.04. The mean conditional probability of breach of an HLW canister resulting from a drop of the canister shall be less than or equal to 3×10^{-2} per drop.	IHF-ESD-07-HLW (Seq. 4-5)	Table 1.5.1-17
				DS.IH.05. The mean conditional probability of breach of an HLW canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	IHF-ESD-07-HLW (Seq. 7-5)	Table 1.5.1-17
				DS.IH.06. The mean conditional probability of breach of an HLW canister contained within a waste package resulting from the spectrum of fires shall be less than or equal to 3×10^{-4} per fire event.	IHF-ESD-13-HLW (Seq. 5-5)	Table 1.5.1-17
				DS.IH.07. The mean conditional probability of breach of an HLW canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	IHF-ESD-13-HLW-WP (Seq. 5-5)	Table 1.5.1-17

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
DOE and Commercial Waste Package System (DS) (Continued)	HLW (Continued)	HLW Canister (Continued)	Provide containment (Continued)	DS.IH.08. The mean conditional probability of breach of an HLW canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	IHF-ESD-13-HLW-WP (Seq. 5-5)	Table 1.5.1-17
				DS.IH.09. The mean conditional probability of breach of an HLW canister, given the drop of another HLW canister onto the first canister, shall be less than or equal to 3×10^{-2} per drop.	IHF-ESD-07-HLW (Seq. 2-5)	Table 1.5.1-17
Emplacement and Retrieval and Drip Shield Installation (HE)	Emplacement and Retrieval and Drip Shield Installation	TEV	Protect against derailment of a TEV during loading of a waste package	HE.IH.01. The mean frequency of derailment of the TEV at the loadout station due to the spectrum of seismic events shall be less than or equal to 1×10^{-4} per year.	IHF-S-IE-NVL (Seq. 16-2)	Table 1.3.3-5
			Protect against a tipover of the TEV	HE.IH.02. The mean frequency of tipover of the TEV due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	IHF-S-IE-NVL (Seq. 16-2)	Table 1.3.3-5
			Protect against ejection of the waste package from the shielded enclosure of the TEV	HE.IH.03. The mean frequency of ejection of a waste package from the TEV due to the spectrum of seismic events shall be less than or equal to 2×10^{-4} per year.	IHF-S-IE-NVL (Seq. 16-2)	Table 1.3.3-5
Initial Handling Facility (IH)	Initial Handling Facility	Structure	Maintain building structural integrity to protect ITS SSCs inside the building from external events	IH.01. The mean frequency of building collapse due to winds less than or equal to 120 mph shall not exceed 1×10^{-6} per year. IH.02. The mean frequency of building collapse due to volcanic ashfall less than or equal to a roof load of 21 lb/ft^2 shall not exceed 1×10^{-6} per year. IH.03. The IHF shall be located such that there is a distance of at least one-half mile between the IHF and the repository heliport.	Initiating event does not require further analysis	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Initial Handling Facility (IH) (Continued)	Initial Handling Facility (Continued)	Structure (Continued)	Protect against building collapse onto waste containers	IH.04. The mean frequency of collapse of the IHF structure due to the spectrum of seismic events shall be less than or equal to 2 x 10^{-6} per year.	IHF-S-IE-HLW (Seq. 03)	Table 1.2.3-3
		Rails for the TEV (inside the Waste Package Loadout Room)	Protect against derailment of the TEV during loading of a waste package	IH.05. The mean frequency of TEV derailment due to failure of the TEV rail system (at the loadout station) due to the spectrum of seismic events shall be less than or equal to 1 × 10^{-4} per year.	IHF-S-IE-NVL (Seq. 16-2)	Table 1.2.3-3
		Shield Doors (Including Anchorages)	Protect against direct exposure of personnel	IH.06. Equipment and personnel shield doors shall have a mean probability of inadvertent opening of less than or equal to 1 × 10^{-6} per transfer.	IHF-ESD-12A-HLW (Seq. 2)	Table 1.2.3-3
			Preclude collapse onto waste containers	IH.07. An equipment shield door falling onto a waste container as a result of an impact from a conveyance shall be precluded.	Initiating event does not require further analysis	Table 1.2.3-3
			Protect against equipment shield door collapse onto a waste container	IH.08. The mean frequency of collapse of equipment shield doors (including attachment of door to wall and frame anchorages) due to the spectrum of seismic events shall be less than or equal to 6 × 10^{-6} per year.	IHF-S-IE-HLW (Seq. 11-6)	Table 1.2.3-3
		Cask Port Slide Gate (51A-HTC0-HTCH-00001)	Protect against dropping a canister due to spurious closure of the slide gate	IH.HTC.01. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2 × 10^{-6} per transfer.	IHF-ESD-07-HLW (Seq. 4-5)	Table 1.2.3-3
			Protect against direct exposure to personnel	IH.HTC.02. The mean probability of inadvertent opening of a slide gate shall be less than or equal to 1 × 10^{-9} per transfer.	IHF-ESD-12A-HLW (Seq. 2)	Table 1.2.3-3
			Preclude canister breach	IH.HTC.03. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Initial Handling Facility (IH) (Continued)	Initial Handling Facility (Continued)	Waste Package Port Slide Gate (51A-HTC0-HTCH-00002)	Protect against dropping a canister due to a spurious closure of the slide gate	IH.HTC.04. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	IHF-ESD-12A-HLW (Seq. 2)	Table 1.2.3-3
			Protect against direct exposure to personnel	IH.HTC.05. The mean probability of inadvertent opening of a slide gate shall be less than or equal to 2 × 10^{-6} per transfer.	IHF-ESD-12A-HLW (Seq. 2)	Table 1.2.3-3
			Preclude canister breach	IH.HTC.06. Closure of the slide gate shall be incapable of breaching a canister	Initiating event does not require further analysis	Table 1.2.3-3
			Preclude canister drop onto floor	IH.HTC.07. The waste package port slide gate shall be incapable of opening without a waste package transfer trolley with waste package in position to receive a canister.	Initiating event does not require further analysis	Table 1.2.3-3
		Cask Preparation Platform (51A-HMH0-PLAT-00001)	Protect against platform collapse	IH.HMH.01. The mean frequency of collapse of the cask preparation platform due to the spectrum of seismic events shall be less than or equal to 9 × 10^{-4} per year.	IHF-S-IE-NVL (Seq. 09-6)	Table 1.2.3-3
Mechanical Handling System (H)	Cask handling	Transportation Cask (Analyzed as a Representative Cask)	Provide containment	H.IH.01. The mean conditional probability of breach of a canister in a sealed cask resulting from a cask drop shall be less than or equal to 1 × 10^{-5} per drop.	IHF-ESD-01-NVL (Seq. 3-6)	Table 1.2.8-2
				H.IH.02. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop of a load onto the cask shall be less than or equal to 1 × 10^{-5} per drop.	IHF-ESD-01-NVL (Seq. 2-6)	Table 1.2.8-2
				H.IH.03. The mean conditional probability of breach of a canister contained within a sealed cask resulting from a side impact or collision shall be less than or equal to 1 × 10^{-8} per impact.	IHF-ESD-04-NVL (Seq. 7-5)	Table 1.2.8-2
			Preclude lid contact with canisters	H.IH.04. The geometry of the casks that carry HLW canisters shall preclude lid contact with canisters following a drop of a cask lid.	IHF-ESD-07-HLW (Seq. 2)	Table 1.2.8-2

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Cask handling (Continued)	Transportation Cask (Analyzed as a Representative Cask) (Continued)	Protect against direct exposure to personnel	H.IH.05. The mean conditional probability of loss of cask gamma shielding resulting from a drop of a cask shall be less than or equal to 1×10^{-5} per drop.	IHF-ESD-02-HLW (Seq. 2-3)	Table 1.2.8-2
				H.IH.06. The mean conditional probability of loss of cask gamma shielding resulting from a collision or side impact to a cask shall be less than or equal to 1×10^{-8} per impact.	IHF-ESD-02-HLW (Seq. 5-3)	Table 1.2.8-2
				H.IH.07. The mean conditional probability of loss of cask gamma shielding resulting from drop of a load onto a cask shall be less than or equal to 1×10^{-5} per impact.	IHF-ESD-02-HLW (Seq. 6-3)	Table 1.2.8-2
		Site Prime Mover	Limit speed	H.IH.08. The speed of the site prime mover shall be limited to 9 mph.	IHF-ESD-01-HLW (Seq. 4-6)	Table 1.2.8-2
			Preclude fuel tank explosion	H.IH.09. The fuel tank of a site prime mover that enters the facility shall preclude fuel tank explosions.	Initiating event does not require further analysis	Table 1.2.8-2
		Cask Handling Yoke (51A-HM00-BEAM-00001)	Protect against drop	H.IH.HM.01. The cask handling yoke is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.3-3
		Cask Handling Crane; 300-ton (51A-HM00-CRN-00001)	Protect against drop	H.IH.HM.02. The mean probability of dropping a loaded cask from less than the two-block height resulting from the failure of a piece of equipment in the load-bearing path shall be less than or equal to 3×10^{-5} per transfer.	IHF-ESD-02-HLW (Seq. 2-6)	Table 1.2.3-3
				H.IH.HM.03. The mean probability of dropping a loaded cask from the two-block height resulting from the failure of a piece of equipment in the load-bearing path shall be less than or equal to 4×10^{-7} per transfer.	IHF-ESD-02-HLW (Seq. 3-6)	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask handling (Continued)	Cask Handling Crane; 300-ton (51A-HM00-CRN-00001) (Continued)	Limit drop height	H.IH.HM.04. The two-block drop height shall not exceed 40 ft from the bottom of the shortest cask to the floor.	IHF-ESD-02-HLW (Seq. 3-6)	Table 1.2.3-3
			Protect against drop of a load onto a cask	H.IH.HM.05. The mean probability of dropping a load onto a loaded cask or its contents shall be less than or equal to 3×10^{-5} per cask handled.	IHF-ESD-02-HLW (Seq. 6-6)	Table 1.2.3-3
			Limit speed	H.IH.HM.06. The speed of the trolley and bridge shall be limited to 20 ft/min.	IHF-ESD-02-HLW (Seq. 5-6)	Table 1.2.3-3
			Protect against crane collapse onto a waste container	H.IH.HM.07. The mean frequency of collapse of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	IHF-S-IE-NVL (Seq. 07-6)	Table 1.2.3-3
			Protect against a cask or heavy object drop from the crane	H.IH.HM.08. The mean frequency of a hoist system failure of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	IHF-S-IE-NVL (Seq. 07-6)	Table 1.2.3-3
		Cask Transfer Trolley and Pedestals (Trolley: 51A-HM00-TRLY-00001) (Cask Pedestals: 51A-HM00-PED-00001-2) (Naval Cask Pedestal: 51A-HM00-PED-00003)	Limit speed	H.IH.HM.09. The speed of the cask transfer trolley shall be limited to 2.5 mph.	IHF-ESD-05-HLW (Seq. 3-5)	Table 1.2.3-3
			Protect against spurious movement	H.IH.HM.10. The mean probability of spurious movement of the cask transfer trolley while a canister is being lifted by the canister transfer machine shall be less than or equal to 1×10^{-9} per transfer.	IHF-ESD-07-NVL (Seq. 3-5)	Table 1.2.3-3
			Protect against impact and inducing stresses on the waste container or on the facility structure	H.IH.HM.11. The mean frequency of sliding impact of the cask transfer trolley into a wall or structural column and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	IHF-S-IE-HLW (Seq. 10-6)	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask handling (Continued)	Cask Transfer Trolley and Pedestals (Trolley: 51A-HM00-TRLY-00001) (Cask Pedestals: 51A-HM00-PED-00001-2) (Naval Cask Pedestal: 51A-HM00-PED-00003) (Continued)	Protect against impact and inducing stresses on the waste container	H.IH.HM.12. The mean frequency of rocking impact of the cask transfer trolley into a wall or structural column and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	IHF-S-IE-HLW (Seq. 10-6)	Table 1.2.3-3
		Cask Preparation Crane; 30-ton (51A-HM00-CRN-00002)	Protect against drop	H.IH.HM.13. The mean probability of a drop of a load onto a loaded cask shall be less than or equal to 3×10^{-5} per transfer.	IHF-ESD-04-NVL (Seq. 4-5)	Table 1.2.3-3
			Protect against collapse of the cask preparation crane	H.IH.HM.14. The mean frequency of collapse of the cask preparation crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	IHF-S-IE-HLW (Seq. 08-6)	Table 1.2.3-3
	Cask Handling/Cask Receipt	Naval Cask Lift Bail (IHF: 51A-HMC0-BEAM-00001)	Protect against drop	H.IH.HMC.01. The naval cask lift bail is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.3-3
		Naval Cask Lift Plate (IHF: 51A-HMC0-HEQ-00005)	Protect against drop	H.IH.HMC.02. The naval cask lift plate is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.3-3
	Cask Handling/Cask Preparation	Rail Cask Lid Adapters (51A-HMH0-HEQ-00002)	Protect against drop	H.IH.HMH.01. The rail cask lid adapter is integral to the load-bearing path for the HLW rail cask lid. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.3-3
	Cask Handling/ Waste Package Preparation	Waste Package Handling Crane (51A-HMP0-CRN-00001)	Protect against collapse of the waste package handling crane	H.IH.HMP.01. The mean frequency of collapse of the waste package handling crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	IHF-S-IE-NVL (Seq. 15-6)	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer	Canister Transfer Machine (51A-HTC0-FHM-00001)	Protect against drop	H.IH.HTC.01. The mean probability of drop of a canister from below the two-block height due to the failure of a piece of equipment in the load-bearing path shall be less than or equal to 2×10^{-4} per transfer.	IHF-ESD-07-HLW (Seq. 4-5)	Table 1.2.3-3
			Protect against drop	H.IH.HTC.02. The mean probability of drop of a canister from the two-block height due to the failure of a piece of equipment in the load-bearing path shall be less than or equal to 3×10^{-8} per transfer.	IHF-ESD-07-HLW (Seq. 5-5)	Table 1.2.3-3
			Limit drop height	H.IH.HTC.03. The two-block drop height shall not exceed 40 ft from the bottom of a canister to the cavity floor of the transportation cask or waste package.	IHF-ESD-07-HLW (Seq. 5-5)	Table 1.2.3-3
			Protect against drop of a load onto a canister	H.IH.HTC.04. The mean probability of dropping a load onto a canister shall be less than or equal to 1×10^{-5} per transfer.	IHF-ESD-07-HLW (Seq. 2-5)	Table 1.2.3-3
			Protect against spurious movement	H.IH.HTC.05. The mean probability of spurious movement of the canister transfer machine while a canister is being lifted or lowered shall be less than or equal to 7×10^{-9} per transfer.	IHF-ESD-07-HLW (Seq. 3-5)	Table 1.2.3-3
			Preclude canister breach	H.IH.HTC.06. Closure of the canister transfer machine slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis	Table 1.2.3-3
			Preclude non-flat-bottom drop of a naval SNF canister	H.IH.HTC.07. The canister transfer machine shall preclude non-flat-bottom drops of naval canisters.	Initiating event does not require further analysis	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	Canister Transfer Machine (51A-HTC0-FHM-00001) (Continued)	Protect against direct exposure of personnel	H.IH.HTC.08. The mean probability of inadvertent radiation streaming due to the inadvertent opening of the canister transfer machine slide gate, the inadvertent raising of the canister transfer machine shield skirt, or an inadvertent motion of the canister transfer machine away from an open port shall be less than or equal to 1 × 10^{-4} per transfer.	IHF-ESD-12B-HLW (Seq. 2)	Table 1.2.3-3
			Limit speed	H.IH.HTC.09. The speed of the canister transfer machine trolley and bridge shall be limited to 20 ft/min.	IHF-ESD-07-HLW (Seq. 7-5)	Table 1.2.3-3
			Protect against drop	H.IH.HTC.10. The mean frequency of drop by the canister transfer machine of the naval SNF canister resulting in breach of the canister shall be less than or equal to 2 × 10^{-5} over the preclosure period.	IHF-ESD-07-NVL (Seq. 4-5)	Table 1.2.3-3
			Protect against collapse of the canister transfer machine	H.IH.HTC.11. The mean frequency of collapse of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 1 × 10^{-5} per year.	IHF-S-IE-HLW (Seq. 12-5)	Table 1.2.3-3
			Protect against a canister or heavy object drop from the canister transfer machine	H.IH.HTC.12. The mean frequency of a hoist system failure of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	IHF-S-IE-HLW (Seq. 12-5)	Table 1.2.3-3
		Canister Transfer Machine Grapple (51A-HTC0-HEQ-00001)	Protect against drop	H.IH.HTC.13. Grapples are an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.3-3
		Canister Grapples (51A-HTC0-HEQ-00003-4)	Protect against drop of a load onto a canister	H.IH.HTC.14. The grapples are an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.3-3
		Naval Canister Lifting Adapter (51A-HTC0-HEQ-00005)	Protect against drop of a canister	H.IH.HTC.15. The naval canister lifting adapter is an integral part of the load-bearing path of the canister transfer machine. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	DOE Waste Package Inner Lid Grapple (51A-HTC0-HEQ-00007)	Protect against the drop of a load onto a canister	H.IH.HTC.16. The lid grapple is an integral part of the load-bearing path of the canister transfer machine. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.3-3
		Naval Waste Package Inner Lid Grapple (51A-HTC0-HEQ-00008)	Protect against the drop of a load onto a canister	H.IH.HTC.17. The lid grapple is an integral part of the load-bearing path of the canister transfer machine. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.3-3
	Waste Package Closure	Remote Handling System Bridge (51A-HWH0-HEQ-00003)	Protect against collapse of the remote handling system bridge	H.IH.HWH.01. The mean frequency of collapse of the remote handling system bridge due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	IHF-S-IE-HLW (Seq. 13-6)	Table 1.2.3-3
	Waste Package Loadout	Waste Package Shield Rings (51A-HL00-HEQ-00001-2)	Provide lateral and vertical stability to the waste package in the waste package transfer trolley.	H.IH.HL.01. The mean frequency of the shield ring becoming displaced from the waste package transfer trolley due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	IHF-S-IE-HLW (Seq. 14-6)	Table 1.2.3-3
		Waste Package Transfer Trolley (including Pedestals, Seismic Rail Restraints, and Rails) (Trolley: 51A-HL00-TRLY-00001) (Pedestals: 51A-HL00-PED-00001-4)	Preclude rapid tilt-down	H.IH.HL.02. The waste package transfer trolley shall be incapable of rapid tilt-down.	Initiating event does not require further analysis	Table 1.2.3-3
			Limit speed	H.IH.HL.03. The speed of the waste package transfer trolley shall be limited to 2.5 mph.	IHF-ESD-08-NVL (Seq. 2-5)	Table 1.2.3-3
			Protect against spurious movement	H.IH.HL.04. The mean probability of spurious movement of the waste package transfer trolley while a canister is being lowered by the canister transfer machine shall be less than or equal to 1×10^{-9} per transfer.	IHF-ESD-07-NVL (Seq. 3-5)	Table 1.2.3-3

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Waste Package Loadout (Continued)	Waste Package Transfer Trolley (including Pedestals, Seismic Rail Restraints, and Rails) (Trolley: 51A-HL00-TRLY-00001) (Pedestals: 51A-HL00-PED-00001-4) (Continued)	Protect against a tipover of the waste package transfer trolley holding a loaded waste package	H.IH.HL.05. The mean frequency of tipover of the waste package transfer trolley due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	IHF-S-IE-HLW (Seq. 14-6)	Table 1.2.3-3
			Protect against rocking (which induces an impact into a wall) of a waste package transfer trolley holding a loaded waste package	H.IH.HL.06. The mean frequency of the rocking impact of the waste package transfer trolley into a wall or column due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	IHF-S-IE-HLW (Seq. 14-6)	Table 1.2.3-3
Naval SNF Waste Package System (DN)	Naval SNF Waste Package	Entire	Provide containment	DN.IH.01. The mean conditional probability of breach of a sealed waste package resulting from a side impact shall be less than or equal to 1×10^{-8} per impact.	IHF-ESD-11-NVL (Seq. 4-6)	Table 1.5.2-6
				DN.IH.02. The mean conditional probability of breach of a sealed waste package resulting from a drop of a load onto the waste package shall be less than or equal to 1×10^{-5} per drop.	IHF-ESD-11-NVL (Seq. 3-6)	Table 1.5.2-6
				DN.IH.03. The mean conditional probability of breach of a sealed waste package resulting from an end-on impact or collision shall be less than or equal to 1×10^{-5} per impact.	IHF-ESD-11-NVL (Seq. 5-6)	Table 1.5.2-6
	Naval SNF Canister	Naval SNF Canister (Analyzed as a Representative Canister)	Provide containment	DN.IH.04. The mean frequency of drop by the canister transfer machine of the naval SNF canister resulting in breach of the canister shall be less than or equal to 2×10^{-5} over the preclosure period.	IHF-ESD-07-NVL (Seq. 4-5)	Table 1.5.1-30

Table 1.9-2. Preclosure Nuclear Safety Design Bases for IHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Naval SNF Waste Package System (DN) (Continued)	Naval SNF Canister (Continued)	Naval SNF Canister (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DN.IH.05. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	IHF-ESD-07-NVL (Seq. 2-5)	Table 1.5.1-30
				DN.IH.06. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	IHF-ESD-07-NVL (Seq. 6-5)	Table 1.5.1-30
				DN.IH.07. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	IHF-ESD-13-NVL (Seq. 8-6)	Table 1.5.1-30
				DN.IH.08. The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	IHF-ESD-13-NVL (Seq. 8-6)	Table 1.5.1-30
				DN.IH.09. The mean conditional probability of breach of a canister contained within a waste package resulting from the spectrum of fires shall be less than or equal to 1×10^{-4}.per fire event.	IHF-ESD-13-NVL (Seq. 8-6)	Table 1.5.1-30

NOTE: “Protect against' in this table means either “reduce the probability of” or “reduce the frequency of.”
Seismic control values shown represent the integration of the probability distribution of SSC failure (i.e., the loss of safety function) with the seismic hazard curve.
Where stated in column 6 that the initiating event does not require further analysis, the design requirement is applied to reduce the frequency of any event sequence that could result in damage to a waste container to the beyond Category 2 frequency range.
The numbers appearing in parentheses in the third column are component numbers.
The event sequence identifier is that of an event sequence before grouping for the purpose of categorization. Therefore, there is no direct correspondence with the event sequence identifier used for the categorized event sequences in Tables 1.7-7 through 1.7-18. Refer to *Initial Handling Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008b) for the description of these event sequences and *Seismic Event Quantification and Categorization Analysis* (BSC 2008g) for a description of seismic event sequences.
Facility Codes: IH: Initial Handling Facility.
System Codes: DN: Naval Spent Nuclear Fuel Waste Package; DS: DOE and Commercial Waste Package; H: Mechanical Handling; HE: Emplacement and Retrieval/Drip Shield Installation; HL: Waste Package Loadout; HM: Cask Handling.
Subsystem Codes: HMC: Cask Receipt; HMH: Cask Preparation; HMP: Waste Package Preparation; HTC: Canister Transfer; HWH: Material Handling.

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Aging (AP)	Aging Handling/ Cask Transfer	Site Transporter (170-HAT0-MEQ-00001)	Protect against spurious movement	AP.CR.HAT.01. The mean probability of spurious movement of the site transporter while the canister is being lifted or lowered shall be less than or equal to 1 × 10^{-9} per transfer.	CRCF-ESD09-TAD (Seq. 4-3)	Table 1.2.8-2
			Limit speed	AP.CR.HAT.02. The speed of the site transporter shall be limited to 2.5 mph.	CRCF-ESD06-TAD (Seq. 4-4)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.CR.HAT.03. The site transporter fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
			Reduce severity of a drop	AP.CR.HAT.04. The site transporter shall preclude a drop of an aging overpack from a height greater than 3 ft measured from the equipment base.	CRCD-ESD16-DPC (Seq. 3-3)	Table 1.2.8-2
			Protect against sliding impact and inducing stresses on the waste container	AP.CR.HAT.05. The mean frequency of a sliding impact of the site transporter into a wall and inducing stresses on the waste container due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	CRCF-S-IE-TWP (Seq. 06-5)	Table 1.2.8-2
			Protect against tipover of a site transporter	AP.CR.HAT.06. The mean frequency of a tipover of the site transporter due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-6} per year.	CRCF-S-IE-TWP (Seq. 06-5)	Table 1.2.8-2
	Aging Handling/Aging Overpack	Aging Overpack (TAD: 170-HAC0-ENCL-00003) Vertical DPC: 170-HAC0-ENCL-00002)	Protect against direct exposure to personnel	AP.CR.HAC.01. The mean conditional probability of loss of shielding of the aging overpack resulting from an impact or collision shall be less than or equal to 1 × 10^{-5} per impact.	CRCF-ESD02-TAD (Seq. 2-2)	Table 1.2.7-1
				AP.CR.HAC.02. The mean conditional probability of loss of shielding of the aging overpack resulting from a drop shall be less than or equal to 5 × 10^{-6} per drop.	CRCF-ESD16-TAD (Seq. 3-2)	Table 1.2.7-1

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Canister Receipt and Closure Facility (CR)	Canister Receipt and Closure Facility (CRCF)	Structure	Maintain building structural integrity to protect ITS SSCs inside the building from external events	CR.01. The mean frequency of building collapse due to winds less than or equal to 120 mph shall not exceed 1×10^{-6} per year. CR.02. The mean frequency of building collapse due to volcanic ash fall less than or equal to a roof load of 21 lb/ft^2 shall not exceed 1×10^{-6} per year. CR.03. The CRCF shall be located such that there is a distance of at least one-half mile between the CRCF and the repository heliport.	Initiating event does not require further analysis.	Table 1.2.4-4
			Protect against building collapse onto waste containers	CR.04. The mean frequency of collapse of CRCF structure due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	CRCF-S-IE-TWP (Seq. 03)	Table 1.2.4-4
		Rails for the TEV (Inside Waste Package Loadout Room)	Protect against derailment of the TEV during loading of a waste package	CR.05. The mean frequency of the TEV derailment due to failure of the TEV rail system (at the loadout station) due to the spectrum of seismic events shall be less than or equal to 1×10^{-4} per year.	CRCF-S-IE-TWP (Seq. 12-2, 12-7)	Table 1.2.4-4
		Shield Doors (Including Anchorages) and Equipment Confinement Doors	Protect against direct exposure of personnel	CR.06. Equipment and personnel shield doors shall have a mean probability of inadvertent opening of less than or equal to 1×10^{-7} per waste container handled.	CRCF-ESD18-TAD (Seq. 2)	Table 1.2.4-4
			Preclude collapse onto waste containers	CR.07. An equipment shield door falling onto a waste container as a result of impact from a conveyance shall be precluded.	Initiating event does not require further analysis.	Table 1.2.4-4
			Mitigate the consequences of radionuclide release	CR.08. The mean probability that the HVAC system in the CRCF confinement areas becomes unavailable (during a 30-day mission time following a radionuclide release) due to the simultaneous opening of an equipment confinement door and a cask unloading room shield door shall be less than or equal to 3×10^{-7}.	CRCF-ESD09-TAD (Seq. 3-5)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
Canister Receipt and Closure Facility (CR) (Continued)	Canister Receipt and Closure Facility (CRCF) (Continued)	Shield Doors (Including Anchorages) and Equipment Confinement Doors (Continued)	Protect against equipment shield door collapse onto a waste container	CR.09. The mean frequency of collapse of equipment shield doors (including attachment of door to wall and frame anchorages) due to the spectrum of seismic event shall be less than or equal to 6 x 10^{-6} per year.	CRCF-S-IE-TAD-AO (Seq. 10-6)	Table 1.2.4-4
		DOE Canister Slide Gates (060-HTC0-HTCH-00005, 6, 7, 8, 9)	Protect against dropping a canister due to a spurious closure of the slide gate	CR.HTC.01. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2 × 10^{-6} per transfer.	CRCF-ESD09-DSTD (Seq. 3-3)	Table 1.2.4-4
			Protect against direct exposure to personnel	CR.HTC.02. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	CRCF-ESD18-DSTD (Seq. 2)	Table 1.2.4-4
			Preclude canister breach	CR.HTC.03. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.4-4
		Cask Port Slide Gates (060-HTC0-HTCH-00001, 2)	Protect against dropping a canister due to a spurious closure of the slide gate	CR.HTC.04. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2 × 10^{-6} per transfer.	CRCF-ESD09-HLW (Seq. 3-3)	Table 1.2.4-4
			Protect against direct exposure to personnel	CR.HTC.05. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	CRCF-ESD18-TAD (Seq. 2)	Table 1.2.4-4
			Preclude canister breach	CR.HTC.06. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Canister Receipt and Closure Facility (CR) (Continued)	Canister Receipt and Closure Facility (CRCF) (Continued)	TAD Slide Gates (060-HTC0-HTCH-00010, 11)	Protect against dropping a canister due to a spurious closure of the slide gate	CR.HTC.07. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2×10^{-6} per transfer.	CRCF-ESD09-TAD (Seq. 3-3)	Table 1.2.4-4
			Protect against direct exposure to personnel	CR.HTC.08. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4×10^{-9} per transfer.	CRCF-ESD18-TAD (Seq. 2)	Table 1.2.4-4
			Preclude canister breach	CR.HTC.09. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.4-4
		Waste Package Port Slide Gates (060-HTC0-HTCH-00003, 4)	Protect against dropping a canister due to a spurious closure of the slide gate	CR.HTC.10. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2×10^{-6} per transfer.	CRCF-ESD09-HLW (Seq. 3-3)	Table 1.2.4-4
			Protect against direct exposure to personnel	CR.HTC.11. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4×10^{-9} per transfer.	CRCF-ESD18-TAD (Seq. 2)	Table 1.2.4-4
			Preclude canister breach	CR.HTC.12. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.4-4
			Preclude canister drop onto floor	CR.HTC.13. The waste package port slide gate shall be incapable of opening without a waste package transfer trolley with waste package in position to receive a canister.	Initiating event does not require further analysis.	Table 1.2.4-4
		Cask Preparation Platform (060-HMH0-PLAT-00 001)	Protect against platform collapse	CR.HMH.01. The mean frequency of collapse of the cask preparation platform due to the spectrum of seismic events shall be less than or equal to 3×10^{-6} per year.	CRCF-S-IE-TAD-AO (Seq. 08-6)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Canister Receipt and Closure Facility (CR) (Continued)	Canister Receipt and Closure Facility (CRCF) (Continued)	Cask Preparation Platform (060-HMH0-PLAT-00001) (Continued)	Protect against platform collapse or waste container breach due to an impact from the cask transfer trolley or site transporter	CR.HMH.02. The mean frequency of platform collapse or waste container breach from the impact of the cask transfer trolley or site transporter into the platform due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	CRCF-S-IE-TAD-AO (Seq. 12-5)	Table 1.2.4-4
DOE and Commercial Waste Package System (DS)	DOE and Commercial Waste Package	Entire	Provide containment	DS.CR.01. The mean conditional probability of breach of a sealed waste package resulting from a side impact shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD13-WP-TAD (Seq. 2-4)	Table 1.5.2-6
				DS.CR.02. The mean conditional probability of breach of a sealed waste package resulting from a drop of a load onto the waste package shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD15-WP-TAD (Seq. 3-4)	Table 1.5.2-6
				DS.CR.03. The mean conditional probability of breach of a sealed waste package resulting from an end-on impact or collision shall be less than or equal to 1×10^{-5} per impact.	CRCF-ESD15-WP-TAD (Seq. 3-4)	Table 1.5.2-6
	High-Level Waste/DOE SNF Codisposal	DOE Standardized Canister	Provide containment	DS.CR.04. The mean conditional probability of breach of a DOE standardized canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-DSTD (Seq. 3-3)	Table 1.5.1-25
				DS.CR.05. The mean conditional probability of breach of a DOE standardized canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-DSTD (Seq. 6-3)	Table 1.5.1-25
				DS.CR.06. The mean conditional probability of breach of a DOE standardized canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD10-WP-H&D (Seq. 3-3)	Table 1.5.1-25

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	High-Level Waste/DOE SNF Codisposal (Continued)	DOE Standardized Canister (Continued)	Provide containment (Continued)	DS.CR.07. The mean conditional probability of breach of a DOE standardized canister contained within a waste package resulting from the spectrum of fires shall be less than or equal to 3×10^{-4} per fire event.	CRCF-ESD20-WP-H&D (Seq. 3-3)	Table 1.5.1-25
				DS.CR.08. The mean conditional probability of breach of a DOE standardized canister contained within a cask or staging area resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	CRCF-ESD20-DSTD (Seq. 5-5)	Table 1.5.1-25
				DS.CR.09. The mean conditional probability of breach of a DOE standardized canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	CRCF-ESD20-DSTD (Seq. 5-5)	Table 1.5.1-25
				DS.CR.10. The mean conditional probability of breach of a DOE standardized canister, given the drop of an HLW canister onto the DOE standardized canister, shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-DSTD (Seq. 6-3)	Table 1.5.1-25
				DS.CR.11. The mean conditional probability of breach of a DOE standardized canister, given the drop of another DOE standardized canister onto the first canister, shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-DSTD (Seq. 6-3)	Table 1.5.1-25
		HLW Canister	Provide containment	DS.CR.12. The mean conditional probability of breach of an HLW canister resulting from a drop of the canister shall be less than or equal to 3×10^{-2} per drop.	CRCF-ESD09-HLW (Seq. 3-3)	Table 1.5.1-17
				DS.CR.13. The mean conditional probability of breach of a HLW canister resulting from a drop of a load onto the canister shall be less than or equal to 3×10^{-2} per drop.	CRCF-ESD09-HLW (Seq. 6-3)	Table 1.5.1-17

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	High-Level Waste/DOE SNF Codisposal (Continued)	HLW Canister (Continued)	Provide containment (Continued)	DS.CR.14. The mean conditional probability of breach of an HLW canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD10-WP-H&D (Seq. 3-3)	Table 1.5.1-17
				DS.CR.15. The mean conditional probability of breach of an HLW canister contained within a waste package resulting from the spectrum of fires shall be less than or equal to 3×10^{-4} per fire event.	CRCF-ESD20-WP-H&D (Seq. 3-3)	Table 1.5.1-17
				DS.CR.16. The mean conditional probability of breach of an HLW canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	CRCF-ESD20-HLW (Seq. 5-5)	Table 1.5.1-17
				DS.CR.17. The mean conditional probability of breach of an HLW canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	CRCF-ESD20-HLW (Seq. 5-5)	Table 1.5.1-17
				DS.CR.18. The mean conditional probability of breach of an HLW canister, given the drop of a DOE standardized canister onto the HLW canister, shall be less than or equal to 3×10^{-2} per drop.	CRCF-ESD09-HLW (Seq. 6-3)	Table 1.5.1-17
				DS.CR.19. The mean conditional probability of breach of an HLW canister, given the drop of another HLW canister onto the first canister, shall be less than or equal to 3×10^{-2} per drop.	CRCF-ESD09-HLW (Seq. 6-3)	Table 1.5.1-17

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel	DPC (Analyzed as a Representative Canister)	Provide containment	DS.CR.20. The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-DPC (Seq. 3-3)	Table 1.5.1-9
				DS.CR.21. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-DPC (Seq. 6-3)	Table 1.5.1-9
				DS.CR.22. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD14-DPC (Seq. 2-3)	Table 1.5.1-9
				DS.CR.23. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	CRCF-ESD20-DPC (Seq. 5-5)	Table 1.5.1-9
				DS.CR.24. The mean conditional probability of breach of a canister contained within an aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	CRCF-ESD20-DPC (Seq. 5-5)	Table 1.5.1-9
				DS.CR.25. The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	CRCF-ESD20-DPC (Seq. 4-3)	Table 1.5.1-9
		TAD Canister (Analyzed as a Representative Canister)	Provide containment	DS.CR.26. The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD09-TAD (Seq. 3-3)	Table 1.5.1-7
				DS.CR.27. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD11-WP-TAD (Seq. 3-3)	Table 1.5.1-7

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel (Continued)	TAD Canister (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DS.CR.28. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD06-TAD (Seq. 4-3)	Table 1.5.1-7
				DS.CR.29. The mean conditional probability of breach of a canister contained within a waste package resulting from the spectrum of fires shall be less than or equal to 3×10^{-4} per fire event.	CRCF-ESD20-TAD (Seq. 7-5)	Table 1.5.1-7
				DS.CR.30. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	CRCF-ESD20-TAD (Seq. 7-5)	Table 1.5.1-7
				DS.CR.31. The mean conditional probability of breach of a canister located within the aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	CRCF-ESD20-TAD (Seq. 7-5)	Table 1.5.1-7
				DS.CR.32. The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	CRCF-ESD20-TAD (Seq. 7-5)	Table 1.5.1-7
Electrical Power System (EE)	ITS Power	ITS Distribution (Feeders Up to and including ITS Loads, ITS Direct Current Power, ITS Uninterruptible Power Supply Power)	Provide electrical power to ITS Surface Nuclear Confinement HVAC Systems	EE.CR.01. The mean conditional probability for ITS electrical power distribution failure, given the loss of offsite power, shall be less than or equal to 7×10^{-3} over a period of 720 hours following a radionuclide release.	CRCF-ESD11-WP-H&M (Seq. 3-3)	Table 1.4.1-1

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Electrical Power System (EE) (Continued)	ITS Power (Continued)	ITS diesel generators (including ITS diesel generator fuel oil system, ITS diesel generator air start system, ITS diesel generator jacket water cooling system, ITS diesel generator lubricating oil system, ITS diesel generator air intake and exhaust system.)	Provide electrical power to ITS Surface Nuclear Confinement HVAC Systems	EE.CR.02. The mean conditional probability for ITS electrical power failure, given the loss of offsite power, shall be less than or equal to 3×10^{-1} over a period of 720 hours following a radionuclide release.	CRCF-ESD11-WP-H&M (Seq. 3-3)	Table 1.4.1-1
Emplacement and Retrieval and Drip Shield Installation (HE)	Emplacement and Retrieval and Drip Shield Installation	TEV	Protect against derailment of a TEV during loading of a waste package	HE.CR.01. The mean frequency of derailment of the TEV at the loadout station due to the spectrum of seismic events shall be less than or equal to 1×10^{-4} per year.	CRCF-S-IE-TWP (Seq. 12-7)	Table 1.3.3-5
			Protect against a tipover of the TEV	HE.CR.02. The mean frequency of tipover of the TEV due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	CRCF-S-IE-TWP (Seq. 12-7)	Table 1.3.3-5
			Protect against ejection of the waste package from the shielded enclosure of the TEV	HE.CR.03. The mean frequency of ejection of a waste package from the TEV due to the spectrum of seismic events shall be less than or equal to 2×10^{-4} per year.	CRCF-S-IE-TWP (Seq. 12-2)	Table 1.3.3-5

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Fire Protection System (FP)	Fire Suppression	Preaction valve, sprinkler heads, and system actuation panels associated with double-interlock preaction suppression systems that protect areas where there is a potential for canister breach	Maintain moderator control	FP.CR.01. The mean probability of inadvertent introduction of fire suppression water into a canister shall be less than or equal to 1×10^{-6} over a 720-hour period following a radionuclide release.	CRCF-ESD09-TAD (Seq. 3-4)	Table 1.4.3-2
	Fire Detection	Fire Detection System for the ITS preaction valves with associated detectors and control box	Maintain moderator control	FP.CR.02. The mean probability of inadvertent introduction of fire suppression water into a canister shall be less than or equal to 1×10^{-6} over a 720-hour period following a radionuclide release.	CRCF-ESD09-TAD (Seq. 3-4)	Table 1.4.3-2
Mechanical Handling System (H)	Cask Handling	Transportation Cask (Analyzed as a Representative Cask)	Provide containment	H.CR.01. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD04-TAD (Seq. 3-4)	Table 1.2.8-2
				H.CR.02. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop of a load onto the cask shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD03-TAD (Seq. 6-4)	Table 1.2.8-2
				H.CR.03. The mean conditional probability of breach of a canister in a sealed cask resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD04-TAD (Seq. 4-4)	Table 1.2.8-2
			Protect against direct exposure to personnel	H.CR.04. The mean conditional probability of loss of cask gamma shielding resulting from a drop of a cask shall be less than or equal to 1×10^{-5} per drop.	CRCF-ESD04-TAD (Seq. 3-2)	Table 1.2.8-2

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Transportation Cask (Analyzed as a Representative Cask) (Continued)	Protect against direct exposure to personnel (Continued)	H.CR.05. The mean conditional probability of loss of cask gamma shielding resulting from a collision or side impact to a cask shall be less than or equal to 1×10^{-8} per impact.	CRCF-ESD04-TAD (Seq. 4-2)	Table 1.2.8-2
				H.CR.06. The mean conditional probability of loss of cask gamma shielding resulting from a drop of a load onto a cask shall be less than or equal to 1×10^{-5} per impact.	CRCF-ESD03-TAD (Seq. 6-2)	Table 1.2.8-2
			Preclude lid contact with canister	H.CR.07. The geometry of the casks that carry DOE standardized canisters or HLW canisters shall preclude a lid contact with canisters following a drop of a cask lid.	Initiating event does not require further analysis.	Table 1.2.8-2
		Site Prime Mover	Limit speed	H.CR.08. The speed of the site prime mover shall be limited to 9 mph.	CRCF-ESD01-HLW (Seq. 4-4)	Table 1.2.8-2
			Preclude fuel tank explosion	H.CR.09. The fuel tank of a site prime mover shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
		Cask Handling Yoke (060-HM00-BEAM-00001)	Protect against drop	H.CR.HM.01. The cask handling yoke is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.4-4
		Cask Handling Crane; 200-ton (060-HM00-CRN-00001)	Protect against drop	H.CR.HM.02. The mean probability of dropping a loaded cask from less than the two-block height resulting from the failure of a piece of equipment within the load path shall be less than or equal to 3×10^{-5} per transfer with the cask yoke or 1×10^{-4} per transfer with a sling.	CRCF-ESD03-TAD (Seq. 2-4) (yoke) CRCF-ESD03-DPC (Seq. 2-4) (sling)	Table 1.2.4-4
			Protect against drop	H.CR.HM.03. The mean probability of dropping a loaded cask from a two-block height resulting from the failure of a piece of equipment within the load-bearing path shall be less than or equal to 4×10^{-7} per transfer.	CRCF-ESD03-TAD (Seq. 7-4)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Cask Handling Crane; 200-ton (060-HM00-CRN-00001) (Continued)	Limit drop height	H.CR.HM.04. The two-block drop height shall not exceed 30 feet from bottom of shortest cask to the floor.	CRCF-ESD03-TAD (Seq. 7-4)	Table 1.2.4-4
			Protect against drop of a load onto a cask	H.CR.HM.05. The mean probability of dropping a load onto a loaded cask or its contents shall be less than or equal to 4×10^{-5} per cask handled.	CRCF-ESD03-TAD (Seq. 6-4)	Table 1.2.4-4
			Maintain moderator control	H.CR.HM.06. The mean probability of inadvertent introduction of an oil moderator into a canister shall be less than or equal to 9×10^{-5} over a 720-hour period following a radioactive release.	CRCF-ESD09-TAD (Seq. 3-4)	Table 1.2.4-4
			Limit speed	H.CR.HM.07. The speed of the trolley and bridge shall be limited to 20 ft/min.	CRCF-ESD09-TAD (Seq. 3-4)	Table 1.2.4-4
			Protect against crane collapse onto a waste container	H.CR.HM.08. The mean frequency of collapse of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	CRCF-S-IE-TAD-AO (Seq. 7-6)	Table 1.2.4-4
			Protect against a cask or heavy object drop from the crane	H.CR.HM.09. The mean frequency of a hoist system failure of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	CRCF-S-IE-TAD-AO (Seq. 7-6)	Table 1.2.4-4
		Cask Transfer Trolley and Pedestals Trolleys: (060-HM00-TRLY-00001-2) Pedestals: (060-HM00-PED-00001-2)	Limit speed	H.CR.HM.10. The speed of the cask transfer trolley shall be limited to 2.5 mph.	CRCF-ESD06-TAD (Seq. 3-3)	Table 1.2.4-4
			Protect against spurious movement	H.CR.HM.11. The mean probability of spurious movement of the cask transfer trolley while a canister is being lifted by the canister transfer machine shall be less than or equal to 1×10^{-9} per transfer.	CRCF-ESD09-HLW (Seq. 4-3)	Table 1.2.4-4
			Protect against impact and inducing stresses on the waste container	H.CR.HM.12. The mean frequency of the sliding of the cask transfer trolley into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	CRCF-S-IE-HLW (Seq. 9-6)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Cask Transfer Trolley and Pedestals Trolleys: (060-HM00-TRLY-00001-2) Pedestals: (060-HM00-PED-00001-2) (Continued)	Protect against impact and inducing stresses on the waste container (Continued)	H.CR.HM.13. The mean frequency of a rocking impact of the cask transfer trolley into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	CRCF-S-IE-HLW (Seq. 9-6)	Table 1.2.4-4
	Cask Handling/Cask Receipt	Horizontal Lifting Beam (200-HMC0-BEAM-00001) (shared with RF)	Protect against drop	H.CR.HMC.01 The horizontal lifting beam is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.4-4
	Cask Handling/Cask Preparation	Cask Lid Lifting Grapples (060-HMH0-HEQ-00012)	Protect against drop of a load onto a canister	H.CR.HMH.01. The cask lid lifting grapple is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.4-4
		DPC Lid Adapter (060-HMH0-HEQ-00005-6)	Protect against drop of a DPC	H.CR.HMH.02. The DPC lid adapter is an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements.	Table 1.2.4-4
		Rail Cask Lid Adapters (060-HMH0-HEQ-00003-4)	Protect against drop	H.CR.HMH.03. The rail cask lid adapter is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.4-4
	Cask Handling/ Waste Package Preparation	Waste Package Handling Crane (060-HMP0-CRN-00001)	Protect against collapse of the waste package handling crane	H.CR.HMP.01. The mean frequency of collapse of the waste package handling crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	CRCF-S-IE-TWP (Seq. 11-6)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer	Canister Transfer Machine (060-HTC0-FHM-00001-2)	Protect against drop	H.CR.HTC.01. The mean probability of dropping a canister from below the two-block height due to the failure of a piece of equipment within the load-bearing path shall be less than or equal to 1 × 10^{-5} per transfer for each canister transfer machine.	CRCF-ESD09-HLW (Seq. 3-3)	Table 1.2.4-4
				H.CR.HTC.02. The mean probability of drop of a canister from the two-block height due to the failure of a piece of equipment within the load-bearing path shall be less than or equal to 3 × 10^{-8} per transfer.	CRCF-ESD09-HLW (Seq. 8-3)	Table 1.2.4-4
			Limit drop height	H.CR.HTC.03. The two-block height shall not exceed 45 ft from the bottom of a canister to the cavity floor of the cask, aging overpack, or waste package.	CRCF-ESD09-HLW (Seq. 8-3)	Table 1.2.4-4
			Protect against drop of a load onto a canister	H.CR.HTC.04. The mean probability of dropping a load onto a canister shall be less than or equal to 1 × 10^{-5} per transfer.	CRCF-ESD09-HLW (Seq. 6-3)	Table 1.2.4-4
			Protect against spurious movement	H.CR.HTC.05. The mean probability of a spurious movement of the canister transfer machine while a canister is being lifted or lowered shall be less than or equal to 7 × 10^{-9} per transfer for each canister transfer machine.	CRCF-ESD09-TAD (Seq. 4-3)	Table 1.2.4-4
			Limit Speed	H.CR.HTC.06. The speed of the canister transfer machine trolley and bridge shall be limited to 20 ft/min.	CRCF-ESD09-HLW (Seq. 5-3)	Table 1.2.4-4
			Preclude non-flat bottom drop of a DPC or TAD	H.CR.HTC.07. The canister transfer machine shall preclude non-flat-bottom drops of DPCs and TADs.	Initiating event does not require further analysis.	Table 1.2.4-4
			Protect against direct exposure to personnel	H.CR.HTC.08. The mean probability of inadvertent radiation streaming resulting from the inadvertent opening of the canister transfer machine slide gate, the inadvertent raising of the canister transfer machine shield skirt, or an inadvertent motion of the canister transfer machine away from an open port shall be less than or equal to 9 × 10^{-6} per transfer.	CRCF-ESD18-TAD (Seq. 2)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	Canister Transfer Machine (060-HTC0-FHM-00001-2) (Continued)	Maintain moderator control	H.CR.HTC.09. The mean probability of inadvertent introduction of an oil moderator into a canister shall be less than or equal to 9 × 10^{-5} over a 720-hour period following the breach of a canister.	CRCF-ESD09-TAD (Seq. 3-4)	Table 1.2.4-4
			Preclude canister breach	H.CR.HTC.10. Closure of the canister transfer machine slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.4-4
			Maintain DOE SNF canister separation	H.CR.HTC.11. The conditional probability of inadvertent placement of more than 4 DOE standardized canisters in a TAD waste package, TAD staging rack, or aging overpack shall be less than or equal to 3 × 10^{-6}.	Initiating event does not require further analysis.	Table 1.2.4-4
			Protect against collapse of the canister transfer machine	H.CR.HTC.12. The mean frequency of collapse of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 1 × 10^{-5} per year.	CRCF-S-IE-TWP (Seq. 8-5)	Table 1.2.4-4
			Protect against a canister or heavy object drop from the canister transfer machine	H.CR.HTC.13. The mean frequency of a hoist system failure of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	CRCF-S-IE-TWP (Seq. 8-5)	Table 1.2.4-4
		Canister Grapples (060-HTC0-HEQ-00003-7) Canister Transfer Machine Grapples (060-HTC0-HEQ-00001-2)	Protect against drop	H.CR.HTC.14. The canister grapple is an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements.	Table 1.2.4-4
			Protect against drop of a load onto a canister	H.CR.HTC.15. The canister grapple is an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements.	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	TAD Staging Racks (and Thermal Barrier) (060-HTC0-RK-00011-12)	Protect against a tipover/impact of a canister	H.CR.HTC.16. The mean frequency of collapse of the TAD staging racks due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-6} per year.	CRCF-S-IE-DOE-SNF (Seq. 12-5)	Table 1.2.4-4
			Protect against canister breach	H.CR.HTC.17. The mean conditional probability of breach of a TAD canister contained within a staging rack resulting from the spectrum of fires shall be less than or equal to 2 × 10^{-6} per fire event.	CRCF-ESD20-DSTD (Seq. 5-3)	Table 1.2.4-4 Table 1.5.1-7
		DOE Canister Staging Racks (and Thermal Barrier) (060-HTC0-RK-00006-10)	Protect against a tipover/impact of a canister	H.CR.HTC.18. The mean frequency of collapse of the DOE canister staging racks (such that the spacing between the surface of adjacent DOE standardized canisters in a staging rack is less than 30 cm) due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-6} per year.	CRCF-S-IE-DOE-SNF (Seq. 12-5)	Table 1.2.4-4
			Protect against canister breach	H.CR.HTC.19. The mean conditional probability of breach of a DOE standardized canister contained within a staging rack resulting from the spectrum of fires shall be less than or equal to 2 × 10^{-6} per fire event.	CRCF-ESD20-DSTD (Seq. 5-3)	Table 1.2.4-4 Table 1.5.1-25
	Waste Package Closure	Remote Handling System Bridge (060-HWH0-HEQ-00003)	Protect against collapse of the remote handling system bridge	H.CR.HWH.01. The mean frequency of collapse of the remote handling system bridge due to the spectrum of seismic events shall be less than or equal to 8 × 10^{-6} per year.	CRCF-S-IE-DOE-SNF (Seq. 13-6)	Table 1.2.4-4
	Waste Package Loadout	Waste Package Shield Rings (060-HL00-HEQ-00001-6)	Provide lateral and vertical stability to the waste package in the waste package transfer trolley	H.CR.HL.01. The mean frequency of the shield ring becoming displaced from the waste package transfer trolley due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	CRCF-S-IE-TWP (Seq. 10-6)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Waste Package Loadout (Continued)	Waste Package Transfer Trolley (including Pedestals, Seismic Rail Restraints, and Rails) (Trolleys: 060-HL00-TRLY-00001-2) (Pedestals: 060-HL00-PED-00001-8)	Preclude rapid tilt-down	H.CR.HL.02. The waste package transfer trolley shall be incapable of rapid tilt-down.	Initiating event does not require further analysis.	Table 1.2.4-4
			Limit speed	H.CR.HL.03. The speed of the waste package transfer trolley shall be limited to 2.5 mph.	CRCF-ESD13-WP-H&D (Seq. 3-3)	Table 1.2.4-4
			Protect against spurious movement	H.CR.HL.04. The mean probability of spurious movement of the waste package transfer trolley while a canister is being lowered by the canister transfer machine shall be less than or equal to 1×10^{-9} per transfer.	CRCF-ESD09-HLW (Seq. 4-3)	Table 1.2.4-4
			Protect against tipover of the waste package transfer trolley holding a loaded waste package	H.CR.HL.05. The mean frequency of tipover of the waste package transfer trolley due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	CRCF-S-IE-TWP (Seq. 10-6)	Table 1.2.4-4
			Protect against rocking (which induces an impact into a wall) of a waste package transfer trolley holding a loaded waste package	H.CR.HL.06. The mean frequency of rocking impact of the waste package transfer trolley into a wall due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	CRCF-S-IE-TWP (Seq. 10-6)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Surface Nuclear Confinement HVAC System (VC)	Surface Nuclear Confinement HVAC	Portions of the surface nuclear confinement HVAC system that exhaust from areas with a potential for a breach	Mitigate the consequences of radionuclide release	VC.CR.01. The mean probability that the HVAC system (including HEPA filtration of exhaust air from the CRCF confinement areas) becomes unavailable during a 30-day mission time following a radionuclide release shall be less than or equal to 4×10^{-2}. This parameter does not apply in the case of large fires, which may disable the HVAC system.	CRCF-ESD09-TAD (Seq. 3-5)	Table 1.2.4-4
		Portions of the surface nuclear confinement HVAC system that support the cooling of ITS electrical equipment and battery rooms	Support ITS electrical function	VC.CR.02. The mean conditional probability of failure of the portions of the surface nuclear confinement HVAC system that support the cooling of ITS electrical equipment and battery rooms in the CRCF shall be less than or equal to 2×10^{-2} per ITS electrical train over a period of 720 hours following a radionuclide release.	CRCF-ESD09-TAD (Seq. 3-5)	Table 1.2.4-4

Table 1.9-3. Preclosure Nuclear Safety Design Bases for CRCF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Surface Nonconfinement HVAC System (VN)	Surface Nonconfinement HVAC	Portions of the surface nonconfinement HVAC system that support the cooling of ITS electrical equipment and battery rooms (EDGF)	Support ITS electrical function	VN.CR.01. The mean conditional probability of failure of the portions of the surface nonconfinement HVAC system that support the cooling of ITS electrical equipment and battery rooms in the EDGF shall be less than or equal to 2×10^{-2} per ITS electrical train over a period of 720 hours following a radionuclide release.	CRCF-ESD09-TAD (Seq. 3-5)	Table 1.4.1-1

NOTE: "Protect against" in this table means either "reduce the probability of" or "reduce the frequency of."

Seismic control values shown represent the integration of the probability distribution of SSC failure (i.e., the loss of safety function) with the seismic hazard curve.

Where stated in column 6 that the initiating event does not require further analysis, the design requirement is applied to reduce the frequency of any event sequence that could result in damage to a waste container to the beyond Category 2 frequency range.

The numbers appearing in parentheses in the third column are component numbers.

The event sequence identifier is that of an event sequence before grouping for the purpose of categorization. Therefore, there is no direct correspondence with the event sequence identifier used for the categorized event sequences in Tables 1.7-7 through 1.7-18. Refer to *Canister Receipt and Closure Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008a) for the description of these event sequences and *Seismic Event Quantification and Categorization Analysis* (BSC 2008g) for a description of seismic event sequences.

Facility Codes: AP: Aging Facility; CR: Canister Receipt and Closure Facility.

System Codes: DS: DOE and Commercial Waste Package; H: Mechanical Handling; HE: Emplacement and Retrieval/Drip Shield Installation.

Infrastructure System Codes: EE: Electrical Power; FP: Fire Protection; H= Mechanical Handling; VC: Surface Nuclear Confinement HVAC; VN: Surface Nonconfinement HVAC.

Subsystem Codes: HAC: Aging Overpack; HAT: Cask Transfer; HL: Waste Package Loadout; HM: Cask Handling; HMC: Cask Receipt; HMH: Cask Preparation; HMP: Waste Package Preparation; HTC: Canister Transfer; HWH: Material Handling.

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Aging (AP)	Aging Handling/Cask Transfer	Site Transporter (170-HAT0-MEQ- 00001)	Protect against spurious movement	AP.WH.HAT.01. The mean probability of spurious movement of the site transporter while the canister is being lifted or lowered shall be less than or equal to 1×10^{-9} per transfer.	WHF-ESD13-TAD (Seq. 6-3)	Table 1.2.8-2
			Limit speed	AP.WH.HAT.02. The speed of the site transporter shall be limited to 2.5 mph.	WHF-ESD03-AODPC (Seq. 3-5)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.WH.HAT.03. The site transporter fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis	Table 1.2.8-2
			Reduce severity of a drop	AP.WH.HAT.04. The site transporter shall preclude a vertical dropping of an aging overpack from a height greater than 3 ft measured from the equipment base.	WHF-ESD03-AODPC (Seq. 2-3)	Table 1.2.8-2
			Protect against sliding impact and inducing stress on the waste container	AP.WH.HAT.05. The mean frequency of sliding impact of the site transporter into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 14-5)	Table 1.2.8-2
			Protect against tipover of the site transporter	AP.WH.HAT.06. The mean frequency of tipover of the site transporter due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 14-5)	Table 1.2.8-2
		Cask Tractor (for use with the Cask Transfer Trailer) (170-HAT0- HEQ-00001)	Limit speed	AP.WH.HAT.07. The speed of the cask tractor shall be limited to 2.5 mph.	WHF-ESD04-DPC (Seq. 3-4)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.WH.HAT.08. The cask tractor fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis	Table 1.2.8-2

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Aging (AP) (Continued)	Aging Handling/Cask Transfer (Continued)	Cask Transfer Trailer (for use with Transportation Casks and Horizontal Shielded Transfer Casks (PWR DPC: 170-HAT0-TRLY-00001) (BWR DPC: 170-HAT0-TRLY-00002)	Preclude fuel tank explosion	AP.WH.HAT.09. The cask transfer trailer fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis	Table 1.2.8-2
			Reduce severity of a drop	AP.WH.HAT.10. The cask transfer trailer shall preclude dropping a cask from a height greater than 6 ft measured from the equipment base.	WHF-ESD04-DPC (Seq. 2-4)	Table 1.2.8-2
			Preclude puncture of a cask	AP.WH.HAT.11. The cask transfer trailer shall preclude puncture of a cask due to collision.	Initiating event does not require further analysis[b]	Table 1.2.8-2
				AP.WH.HAT.12. The cask transfer trailer shall be designed to preclude puncture of a cask due to the spectrum of seismic events.	Initiating event does not require further analysis	Table 1.2.8-2
	Aging Handling/ Aging Overpack	Aging Overpack (TAD: 170-HAC0-ENCL-00003) (Vertical DPC: 170-HAC0-ENCL-00002)	Protect against direct exposure to personnel	AP.WH.HAC.01. The mean conditional probability of loss of shielding of the aging overpack resulting from an impact or collision shall be less than or equal to 1×10^{-5} per impact.	WHF-ESD03-AODPC (Seq. 3-2)	Table 1.2.7-1
				AP.WH.HAC.02. The mean conditional probability of loss of shielding of the aging overpack resulting from a drop shall be less than or equal to 5×10^{-6} per drop.	WHF-ESD03-AODPC (Seq. 2-2)	Table 1.2.7-1
Cask/ Canister/ Waste Package Process System (MR)	Cask Cooling	Cask/DPC Overpressure Protection Features	Protect against cask failure due to overpressure	MR.WH.01. The mean probability of an overpressure of a cask or cooling system line during the cask cooling operation shall be less than or equal to 8×10^{-6} per cask.	WHF-ESD16-CSNF (Seq. 4-1)	Table 1.2.5-3
DOE and Commercial Waste Package System (DS)	Canistered Spent Nuclear Fuel	DPC (Analyzed as a Representative Canister)	Provide containment	DS.WH.01. The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	WHF-ESD13-DPC (Seq. 2-3)	Table 1.5.1-9
				DS.WH.02. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	WHF-ESD13-DPC (Seq. 5-3)	Table 1.5.1-9

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel (Continued)	DPC (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DS.WH.03. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	WHF-ESD13-DPC (Seq. 4-3)	Table 1.5.1-9
				DS.WH.04. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	WHF-ESD31-DPC (Seq. 7-3)	Table 1.5.1-9
				DS.WH.05. The mean conditional probability of breach of a canister contained within an aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	WHF-ESD31-TAD (Seq. 4-3)	Table 1.5.1-9
				DS.WH.06. The mean conditional probability of breach of a canister contained within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	WHF-ESD31-DPC (Seq. 5-3)	Table 1.5.1-9
		TAD Canister (Analyzed as a Representative Canister)	Provide containment	DS.WH.07. The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	WHF-ESD13-TAD (Seq. 2-3)	Table 1.5.1-7
				DS.WH.08. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	WHF-ESD13-TAD (Seq. 5-3)	Table 1.5.1-7
				DS.WH.09. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	WHF-ESD13-TAD (Seq. 4-3)	Table 1.5.1-7

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel (Continued)	TAD Canister (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DS.WH.10. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	WHF-ESD31-TAD (Seq. 6-3)	Table 1.5.1-7
				DS.WH.11. The mean conditional probability of breach of a canister located within the aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	WHF-ESD31-TAD (Seq. 4-3)	Table 1.5.1-7
				DS.WH.12. The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	WHF-ESD31-TAD (Seq. 5-3)	Table 1.5.1-7
Electrical Power System (EE)	ITS Power	ITS Distribution (Feeders Up to and including ITS Loads, ITS Direct Current Power, ITS Uninterruptible Power Supply Power)	Provide electrical power to ITS surface nuclear confinement HVAC systems	EE.WH.01. The mean conditional probability for ITS electrical power distribution failure shall be less than or equal to 8×10^{-3} over a period of 720 hours following a radionuclide release.	WHF-ESD13-TAD (Seq. 2-5)	Table 1.4.1-1
				EE.WH.02. The mean conditional probability for ITS electrical power distribution failure shall be less than or equal to 5×10^{-4} over a period of 24 hours following a cask overpressure or a cooling system line break.	WHF-ESD16-CSNF (Seq. 4-3)	Table 1.4.1-1
		ITS diesel generators (including ITS diesel generator fuel oil system, ITS diesel generator air start system, ITS diesel generator jacket water cooling system, ITS diesel generator lubricating oil system, ITS diesel generator air intake and exhaust system.)	Provide electrical power to ITS surface nuclear confinement HVAC systems	EE.WH.03. The mean conditional probability for ITS electrical power failure, given the loss of offsite power, shall be less than or equal to 3×10^{-1} over a period of 720 hours following a radionuclide release.	WHF-ESD13-TAD (Seq. 2-5)	Table 1.4.1-1

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Fire Protection System (FP)	Fire Suppression	Preaction valves, sprinkler heads, and system actuation panels associated with double-interlock preaction suppression systems that protect areas where there is a potential for canister breach	Maintain moderator control	FP.WH.01 The mean probability of inadvertent introduction of fire suppression water into a canister shall be less than or equal to 6 × 10^{-7} over a 720-hour period following a radionuclide release.	WHF-ESD02-DPC (Seq. 3-5)	Table 1.4.3-2
	Fire Detection	Fire Detection System for the ITS preaction valve with associated detectors and control box	Maintain moderator control	FP.WH.02. The mean probability of inadvertent introduction of fire suppression water into a canister shall be less than or equal to 6 × 10^{-7} over a 720-hour period following a radionuclide release.	WHF-ESD02-DPC (Seq. 3-5)	Table 1.4.3-2
Mechanical Handling System (H)	Cask Handling	Transportation Cask (Analyzed as a Representative Cask) Shielded Transfer Cask (Analyzed as a Representative Cask) (TAD: 050-HT00-HEQ-00001) (DPC: 050-HT00-HEQ-00002) NOTE: Only transportation casks may contain uncanistered SNF; STCs and transportation casks may contain canistered SNF.	Provide containment	H.WH.01. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop shall be less than or equal to 1 × 10^{-5} per drop.	WHF-ESD07-DPC (Seq. 5-4)	Table 1.2.5-3 Table 1.2.8-2
				H.WH.02. The mean conditional probability of breach of a sealed cask containing uncanistered SNF resulting from a drop shall be less than or equal to 1 × 10^{-5} per drop.	WHF-ESD05-CSNF (Seq. 5-3)	Table 1.2.8-2
				H.WH.03. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop of a load onto the cask shall be less than or equal to 1 × 10^{-5} per drop.	WHF-ESD07-DPC (Seq. 4-4)	Table 1.2.5-3 Table 1.2.8-2
				H.WH.04. The mean conditional probability of breach of a sealed cask containing uncanistered SNF resulting from a drop of a load onto the cask shall be less than or equal to 1 × 10^{-5} per drop.	WHF-ESD05-CSNF (Seq. 4-3)	Table 1.2.8-2
				H.WH.05. The mean conditional probability of breach of a canister in a sealed cask resulting from a side impact or collision shall be less than or equal to 1 × 10^{-8} per impact.	WHF-ESD07-DPC (Seq. 3-4)	Table 1.2.5-3 Table 1.2.8-2

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Transportation Cask (Analyzed as a Representative Cask) Shielded Transfer Cask (Analyzed as a Representative Cask) (TAD: 050-HT00-HEQ-00001) (DPC: 050-HT00-HEQ-00002) (Continued) NOTE: Only transportation casks may contain uncanistered SNF; STCs and transportation casks may contain canistered SNF.	Provide containment (Continued)	H.WH.06. The mean conditional probability of breach of a sealed cask containing uncanistered SNF resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	WHF-ESD05-CSNF (Seq. 3-3)	Table 1.2.8-2
				H.WH.07. The mean conditional probability of breach of a sealed cask containing uncanistered SNF resulting from the spectrum of fires shall be less than or equal to 5×10^{-2} per fire event.	WHF-ESD31-CSNF (Seq. 4-5)	Table 1.2.8-2
			Protect against direct exposure to personnel	H.WH.08. The mean conditional probability of loss of cask gamma shielding resulting from a drop of a cask shall be less than or equal to 1×10^{-5} per drop.	WHF-ESD05-CSNF (Seq. 5-2)	Table 1.2.5-3 Table 1.2.8-2
				H.WH.09. The mean conditional probability of loss of cask gamma shielding resulting from a collision or side impact to a cask shall be less than or equal to 1×10^{-8} per impact.	WHF-ESD05-CSNF (Seq. 3-2)	Table 1.2.5-3 Table 1.2.8-2
				H.WH.10 The mean conditional probability of loss of cask gamma shielding resulting from a drop of a load onto a cask shall be less than or equal to 1×10^{-8} per impact.	WHF-ESD05-CSNF (Seq. 4-2)	Table 1.2.5-3 Table 1.2.8-2
		Site Prime Mover	Limit speed	H.WH.11. The speed of the site prime mover shall be limited to 9 mph.	WHF-ESD01-CSNF (Seq. 3-3)	Table 1.2.8-2
			Preclude fuel tank explosion	H.WH.12. The fuel tank of a site prime mover shall preclude fuel tank explosions.	Initiating event does not require further analysis	Table 1.2.8-2
		Cask Handling Yoke (050-HM00-BEAM-00001)	Protect against drop	H.WH.HM.01. The cask handling yoke is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.5-3
		Pool Cask Handling Yoke (050-HM00-BEAM-00002)	Protect against drop	H.WH.HM.02. The pool cask handling yoke is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Cask Handling Crane; 200-ton (050-HM00-CRN- 00001)	Protect against drop	H.WH.HM.03. The mean probability of dropping a loaded cask from less than the two-block height resulting from the failure of a piece of equipment within the load path supporting the cask shall be less than or equal to 3 × 10^{-5} per transfer with the cask yoke or 1 × 10^{-4} per transfer with a sling.	WHF-ESD20-CSNF (Seq. 08-3) WHF-ESD06-TTC (Seq. 2-4)	Table 1.2.5-3
				H.WH.HM.04. The mean probability of dropping a loaded cask from a two-block height resulting from the failure of a piece of equipment within the load-bearing path shall be less than or equal to 4 × 10^{-7} per transfer.	WHF-ESD20-CSNF (Seq. 09-3)	Table 1.2.5-3
			Limit drop height	H.WH.HM.05. The two-block drop height shall not exceed 30 feet from bottom of shortest cask to the floor.	WHF-ESD20-CSNF (Seq. 09-3)	Table 1.2.5-3
			Protect against drop of a load onto a cask	H.WH.HM.06. The mean probability of dropping a load onto a loaded cask or its contents shall be less than or equal to 3 × 10^{-5} per cask handled.	WHF-ESD05-CSNF (Seq. 4-3)	Table 1.2.5-3
			Maintain moderator control	H.WH.HM.07. The mean probability of inadvertent introduction of an oil moderator into a canister shall be less than or equal to 9 × 10^{-5} over a 720-hour period following a radionuclide release.	WHF-ESD05-CSNF (Seq. 4-4)	Table 1.2.5-3
			Limit speed	H.WH.HM.08. The speed of the trolley and bridge shall be limited to 20 ft/min.	WHF-ESD05-CSNF (Seq. 3-3)	Table 1.2.5-3
			Protect against crane collapse onto a waste container	H.WH.HM.09. The mean frequency of collapse of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 8 × 10^{-6} per year.	WHF-S-IE-BARE (Seq. 05-6)	Table 1.2.5-3
			Protect against a cask or heavy object drop from the crane	H.WH.HM.10. The mean frequency of a hoist system failure of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	WHF-S-IE-BARE (Seq. 05-6)	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Pool Yoke Lift Adapter (050-HM00-TOOL-00002)	Protect against drop of a cask	H.WH.HM.11. The pool yoke lift adapter is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.5-3
		Cask Transfer Trolley and Pedestals (Trolley: 050-HM00-TRLY-00001) (Pedestals: 050-HM00-PED-00001-5)	Limit speed	H.WH.HM.12. The cask transfer trolley shall be limited to 2.5 mph.	WHF-ESD12-DPC (Seq. 10)	Table 1.2.5-3
			Protect against spurious movement	H.WH.HM.13. The mean probability of spurious movement of the cask transfer trolley while a canister is being lifted by the canister transfer machine shall be less than or equal to 1×10^{-9} per transfer.	WHF-ESD13-TAD (Seq. 6-3)	Table 1.2.5-3
			Protect against impact and inducing stresses on the waste container	H.WH.HM.14. The mean frequency of the sliding of the cask transfer trolley into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	WHF-S-IE-BARE (Seq. 09-6)	Table 1.2.5-3
				H.WH.HM.15. The mean frequency of a rocking impact of the cask transfer trolley into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	WHF-S-IE-BARE (Seq. 09-6)	Table 1.2.5-3
	Cask Handling/Cask Receipt	Entrance Vestibule Crane (050-HMC0- CRN-00001)	Protect against collapse	H.WH.HMC.01. The mean frequency of collapse of the entrance vestibule crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	WHF-S-IE-BARE (Seq. 07-6)	Table 1.2.5-3
		Horizontal Lifting Beam (200-HMC0-BEAM-00001)	Protect against drop	H.WH.HMC.02 The horizontal lifting beam is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.5-3
	Cask Handling/Cask Preparation	Truck Cask Lid Adapters (050-HMH0-HEQ-00010-11) Rail Cask Lid Adapters (050-HMH0-HEQ-00012-13)	Protect against drop	H.WH.HMH.01. The truck and rail cask lid adapters are an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Cask Handling/Cask Preparation (Continued)	Auxiliary Pool Crane; 10-ton (050-HMH0-CRN-00001)	Protect against a drop of a load onto canister	H.WH.HMH.02. The mean probability of drop of a load onto a canister shall be less than or equal to 3×10^{-5} per lift.	WHF-ESD21-CSNF (Seq. 5-2)	Table 1.2.5-3
			Protect against collapse of the auxiliary pool crane	H.WH.HMH.03. The mean frequency of collapse of the auxiliary pool crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	WHF-S-IE-BARE (Seq. 14-5)	Table 1.2.5-3
			Protect against a heavy object drop from the auxiliary pool crane	H.WH.HMH.04. The mean frequency of a hoist system failure of the auxiliary pool crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-BARE (Seq. 14-5)	Table 1.2.5-3
		Preparation Station Jib Cranes (1 and 2) (050-HMH0-CRN-00002, 3)	Protect against a drop of a load onto canister	H.WH.HMH.05. The mean probability of drop of a load onto a canister shall be less than or equal to 3×10^{-5} per lift.	WHF-ESD08-CSNF (Seq. 3-3)	Table 1.2.5-3
			Protect against collapse of the jib crane	H.WH.HMH.06. The mean frequency of collapse of the jib crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 09)	Table 1.2.5-3
			Protect against a heavy object drop from the jib crane	H.WH.HMH.07. The mean frequency of a hoist system failure of the jib crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 09)	Table 1.2.5-3
		Cask Support Frame (Preparation Station #2) (WHF: 050-HMH0-FRM-00001)	Protect against tipover of a cask	H.WH.HMH.08. The mean frequency of failure of the cask support frame and anchorage due to the spectrum of seismic events shall be less than or equal to 1×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 08)	Table 1.2.5-3
		Lid Lifting Grapples (050-HMH0-HEQ-00001-4, 6) Truck Cask Lid Lifting Grapples (050-HMH0-HEQ-00007-9)	Protect against drop of a load onto a canister	H.WH.HMH.09. The lid lift grapple is an integral part of the load-bearing path. See preparation station jib crane and auxiliary pool crane requirements.	See preparation station jib cranes and auxiliary pool crane requirements	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling/Cask Preparation (Continued)	DPC Lid Adapter (050-HMH0-HEQ- 00014)	Protect against drop of a DPC	H.WH.HMH.10. The DPC lid adapter is an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.5-3
		Long Reach Grapple Adapter (050-HMH0-TOOL-00001-2)	Protect against drop of a load	H.WH.HMH.11. The long reach grapple adapter is an integral part of the load-bearing path. See auxiliary pool crane requirements.	See auxiliary pool crane requirements	Table 1.2.5-3
	Waste Transfer/Fuel Assembly Transfer	Spent Fuel Transfer Machine (050-HTF0-FHM- 00001)	Protect against drop of an SNF assembly	H.WH.HTF.01. The mean probability of dropping an SNF assembly due to a failure of a piece of equipment within the load path shall be less than or equal to 5×10^{-6} per assembly transfer.	WHF-ESD22-FUEL (Seq. 3-1)	Table 1.2.5-3
			Protect against lifting an SNF assembly above the safe limit for workers	H.WH.HTF.02. The mean probability of lifting an SNF assembly such that 10 CFR 63.111(a) limits are exceeded shall be less than or equal to 7×10^{-7} per assembly transfer.	WHF-ESD30-FUEL (Seq. 2)	Table 1.2.5-3
			Protect against collapse of the spent fuel transfer machine	H.WH.HTF.03. The mean frequency of collapse of the spent fuel transfer machine due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-BARE (Seq. 15-5)	Table 1.2.5-3
			Protect against an SNF assembly or heavy object drop from the spent fuel transfer machine	H.WH.HTF.04. The mean frequency of a hoist system failure of the spent fuel transfer machine due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-BARE (Seq. 15-5)	Table 1.2.5-3
		PWR Lifting Grapples (050-HTF0-HEQ-00001) BWR Lifting Grapples (050-HTF0-HEQ-00002)	Protect against drop of an SNF assembly	H.WH.HTF.05. The PWR/BWR grapples are an integral part of the load-bearing path. See spent fuel transfer machine requirements.	See spent fuel transfer machine requirements	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Waste Transfer/Fuel Assembly Transfer (Continued)	SNF Staging Rack (PWR SNF: 050-HTF0-RK-00001) (BWR SNF: 050-HTF0-RK-00010) (DFCA SNF: 050-HTF0-RK-00011)	Protect against tipover of SNF	H.WH.HTF.06. The mean frequency of collapse of the SNF staging racks (sufficient to cause loss of confinement of the fuel assemblies within the staging rack fuel compartments) due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	WHF-S-IE-SNF-XFER (Seq. 09)	Table 1.2.5-3
		Truck Cask Handling Frame (050-HTF0-RK-00007)	Protect against cask drop from a crane	H.WH.HTF.07. The mean frequency of a cask drop due to a failure of the truck cask handling frame due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-BARE (Seq. 05-6)	Table 1.2.5-3
	Waste Transfer/ Canister Transfer	Canister Transfer Machine (050-HTC0-FHM-00001)	Protect against drop	H.WH.HTC.01. The mean probability of dropping a canister from below the two-block height due to the failure of a piece of equipment within the load-bearing path shall be less than or equal to 1×10^{-5} per transfer	WHF-ESD13-TAD (Seq. 2-3)	Table 1.2.5-3
				H.WH.HTC.02. The mean probability of drop of a canister from the two-block height due to the failure of a piece of equipment within the load-bearing path shall be less than or equal to 3×10^{-8} per transfer.	WHF-ESD13-TAD (Seq. 3-3)	Table 1.2.5-3
			Limit drop height	H.WH.HTC.03. The two-block drop height shall not exceed 45 ft from the bottom of a canister to the cavity floor of the cask or aging overpack.	WHF-ESD13-TAD (Seq. 2-3)	Table 1.2.5-3
			Protect against drop of a load onto a canister	H.WH.HTC.04. The mean probability of dropping a load onto a canister shall be less than or equal to 1×10^{-5} per transfer.	WHF-ESD13-TAD (Seq. 5-3)	Table 1.2.5-3
			Protect against spurious movement	H.WH.HTC.05. The mean probability of a spurious movement of the canister transfer machine while a canister is being lifted or lowered shall be less than or equal to 7×10^{-9} per transfer.	WHF-ESD13-TAD (Seq. 6-3)	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	Canister Transfer Machine (050-HTC0-FHM-00001) (Continued)	Limit Speed	H.WH.HTC.06. The speed of the canister transfer machine trolley and bridge shall be limited to 20 ft/min.	WHF-ESD13-TAD (Seq. 4-3)	Table 1.2.5-3
			Preclude non-flat bottom drop of a DPC or TAD	H.WH.HTC.07. The canister transfer machine shall preclude non-flat-bottom drops of DPCs and TADs.	Initiating event does not require further analysis	Table 1.2.5-3
			Protect against direct exposure to personnel	H.WH.HTC.08. The mean probability of inadvertent radiation streaming resulting from the inadvertent opening of the canister transfer machine slide gate, the inadvertent raising of the canister transfer machine shield skirt, or an inadvertent motion of the canister transfer machine away from an open port shall be less than or equal to 9×10^{-6} per transfer.	WHF-ESD29-TAD (Seq. 3)	Table 1.2.5-3
			Maintain moderator control	H.WH.HTC.09. The mean probability of inadvertent introduction of an oil moderator into a canister shall be less than or equal to 9×10^{-5} over a 720-hour period following the breach of a canister.	WHF-ESD13-TAD (Seq. 2-4)	Table 1.2.5-3
			Preclude canister breach	H.WH.HTC.10. Closure of the canister transfer machine slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis[b]	Table 1.2.5-3
			Protect against collapse of the canister transfer machine	H.WH.HTC.11. The mean frequency of collapse of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 1×10^{-5} per year.	WHF-S-IE-DPC (Seq. 14-5)	Table 1.2.5-3
			Protect against a canister or heavy object drop from the canister transfer machine	H.WH.HTC.12. The mean frequency of a hoist system failure of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-DPC (Seq. 14-5)	Table 1.2.5-3
		Canister Transfer Machine Grapples (050-HTC0-HEQ- 00001)	Protect against drop	H.WH.HTC.13. The canister transfer machine grapple is an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	TAD Closure	TAD Closure Jib Crane(050-HC00-CRN-00001)	Protect against drop of a load	H.WH.HC.01. The mean probability of a drop of a load onto a cask containing a TAD shall be less than or equal to 3×10^{-5} per lift.	WHF-ESD25-TAD (Seq. 2-2)	Table 1.2.5-3
			Protect against collapse of the TAD closure jib crane	H.WH.HC.02. The mean frequency of collapse of the TAD closure jib crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 09)	Table 1.2.5-3
			Protect against a heavy object drop from the TAD closure jib crane	H.WH.HC.03. The mean frequency of a hoist system failure of the TAD closure jib crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 09)	Table 1.2.5-3
		Cask Support Frame (TAD Closure Station) (050-HC00-FRM-00001)	Protect against tipover of a cask	H.WH.HC.04. The mean frequency of failure of the cask support frame and anchorage due to the spectrum of seismic events shall be less than or equal to 6×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 08)	Table 1.2.5-3
	DPC Cutting	DPC Cutting Jib Crane (050-HD00- CRN-00001)	Protect against drop of a load	H.WH.HD.01. The mean probability of a drop of a load onto a cask containing a DPC shall be less than or equal to 3×10^{-5} per lift.	WHF-ESD18-DPC Seq. 2-2)	Table 1.2.5-3
			Protect against collapse of the DPC cutting jib crane	H.WH.HD.02. The mean frequency of collapse of the DPC cutting jib crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 09)	Table 1.2.5-3
			Protect against a heavy object drop from the DPC cutting jib crane	H.WH.HD.03. The mean frequency of a hoist system failure of the DPC cutting jib crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 09)	Table 1.2.5-3
		Cask Support Frame (DPC Cutting Station) (050-HD00-FRM- 00001)	Protect against tipover of a cask	H.WH.HD.04. The mean frequency of failure of the cask support frame and anchorage due to the spectrum of seismic events shall be less than or equal to 6×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 08)	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Surface Nuclear Confinement HVAC System (VC)	Surface Nuclear Confinement HVAC	Portions of the surface nuclear confinement HVAC system that exhaust from areas with a potential for a breach	Mitigate the consequences of radionuclide release	VC.WH.01. The mean probability that the HVAC system (including HEPA filtration of exhaust air from the WHF confinement areas) becomes unavailable during a 30-day mission time following a radionuclide release shall be less than or equal to 4×10^{-2}. This parameter does not apply in the case of large fires, which may disable the HVAC system.	WHF-ESD13-TAD (Seq. 2-5)	Table 1.2.5-3
				VC.WH.02. The mean probability that the HVAC system (including HEPA filtration of exhaust air from the WHF confinement areas) becomes unavailable during a 1-day mission time following a radionuclide release from the cask sampling and cooling process shall be less than or equal to 1×10^{-3}.	WHF-ESD16-CSNF (Seq. 4-3)	Table 1.2.5-3
		Portions of the surface nuclear confinement HVAC system that support the cooling of ITS electrical equipment and battery rooms	Support ITS electrical function	VC.WH.03. The mean conditional probability of failure of the portions of the surface nuclear confinement HVAC system that support the cooling of ITS electrical equipment and battery rooms in the WHF shall be less than or equal to 2×10^{-2} per ITS electrical train over a period of 720 hours following a radionuclide release.	WHF-ESD13-TAD (Seq. 2-5)	Table 1.2.5-3
				VC.WH.04. The mean conditional probability of failure of the portions of the surface nuclear confinement HVAC system that support the cooling of ITS electrical equipment and battery rooms in the WHF shall be less than or equal to 5×10^{-4} per ITS electrical train over a period of 24 hours following a cask overpressure or a cooling system line break.	WHF-ESD16-CSNF (Seq. 4-3)	Table 1.2.5-3
Surface Nonconfinement HVAC System (VN)	Surface Nonconfinement HVAC	Portions of the surface nonconfinement HVAC system that support the cooling of ITS electrical equipment and battery rooms (EDGF)	Support ITS electrical function	VN.WH.01. The mean conditional probability of failure of the portions of the surface nonconfinement HVAC system that support the cooling of ITS electrical equipment and battery rooms in the EDGF shall be less than or equal to 2×10^{-2} per ITS electrical train over a period of 720 hours following a radionuclide release.	WHF-ESD13-TAD (Seq. 2-5)	Table 1.4.1-1

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Wet Handling Facility (WH)	Wet Handling Facility (WHF)	Structure	Maintain building structural integrity to protect ITS SSCs inside the building from external events	WH.01. The mean frequency of building collapse due to winds less than or equal to 120 mph shall not exceed 1×10^{-6} per year.	Initiating event does not require further analysis	Table 1.2.5-3
				WH.02. The mean frequency of building collapse due to volcanic ashfall less than or equal to a roof live load of 21 lb/ft^2 shall not exceed 1×10^{-6} per year.	Initiating event does not require further analysis	Table 1.2.5-3
				WH.03. The WHF shall be located such that there is a distance of at least one-half mile between the WHF and the repository heliport.	Initiating event does not require further analysis	Table 1.2.5-3
			Protect against building collapse onto waste containers	WH.04. The mean frequency of collapse of the WHF structure due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	WHF-S-IE-SNF-XFER (Seq. 07)	Table 1.2.5-3
		Shield Doors (Including Anchorages)	Protect against direct exposure of personnel	WH.05. Equipment shield doors shall have a mean probability of inadvertent opening of less than or equal to 1×10^{-7} per waste container handled.	WHF-ESD29-TAD (Seq. 3)	Table 1.2.5-3
			Preclude collapse onto waste containers	WH.06. An equipment shield door falling onto a waste container as a result of impact from a conveyance shall be precluded.	Initiating event does not require further analysis[b]	Table 1.2.5-3
			Protect against equipment shield door collapse onto a waste container	WH.07. The mean frequency of collapse of equipment shield doors (including attachment of door to wall and frame anchorages) due to the spectrum of seismic events shall be less than or equal to 6×10^{-6} per year.	WHF-S-IE-BARE (Seq. 12-6)	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Wet Handling Facility (WH) (Continued)	Wet Handling Facility (WHF) (Continued)	Pool Structure	Maintain pool integrity to protect against collapse onto waste containers and to maintain pool water retention capability	WH.08. The mean frequency of collapse of, or water loss from, the WHF pool due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-6} per year.	WHF-S-IE-SNF-XFER (Seq. 08)	Table 1.2.5-3
		Cask Port Slide Gate (050-HTC0-HTCH-00002)	Protect against dropping a canister due to a spurious closure of the slide gate	WH.HTC.01. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2 × 10^{-6} per transfer.	WHF-ESD13-TAD (Seq. 2-3)	Table 1.2.5-3
			Protect against direct exposure to personnel	WH.HTC.02. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	WHF-ESD29-TAD (Seq. 3)	Table 1.2.5-3
			Preclude canister breach	WH.HTC.03. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis	Table 1.2.5-3
		Overpack Port Slide Gate (050-HTC0-HTCH-00001)	Protect against dropping a canister	WH.HTC.04. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 2 × 10^{-6} per transfer.	WHF-ESD13-TAD (Seq. 2-3)	Table 1.2.5-3
			Protect against direct exposure to personnel	WH.HTC.05. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	WHF-ESD29-TAD (Seq. 3)	Table 1.2.5-3
			Preclude canister breach	WH.HTC.06. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis	Table 1.2.5-3
		Aging Overpack Access Platform (050-HAC0-PLAT- 00001)	Protect against platform collapse	WH.HAC.01. The mean frequency of collapse of the aging overpack access platform due to the spectrum of seismic events shall be less than or equal to 3 × 10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 15-5)	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
Wet Handling Facility (WH) (Continued)	Wet Handling Facility (WHF) (Continued)	Aging Overpack Access Platform (050-HAC0-PLAT- 00001) (Continued)	Protect against platform collapse or waste container breach due to an impact from the site transporter	WH.HAC.02. The mean frequency of platform collapse or waste container breach from the impact of the site transporter onto the platform due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-TAD-AO (Seq. 15-5)	Table 1.2.5-3
		TAD Closure Station (050-HC00-PLAT- 00001)	Protect against platform collapse	WH.HC.01. The mean frequency of collapse of the TAD closure station platform due to the spectrum of seismic events shall be less than or equal to 3×10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 08)	Table 1.2.5-3
		DPC Cutting Station (050-HD00-PLAT- 00001)	Protect against platform collapse	WH.HD.01. The mean frequency of collapse of the DPC cutting station platform due to the spectrum of seismic events shall be less than or equal to 3×10^{-6} per year.	WHF-S-IE-DPC (Seq. 15)	Table 1.2.5-3
		Preparation Station #1 (050-HMH0-PLAT-00001)	Protect against platform collapse	WH.HMH.01 The mean frequency of collapse of the preparation station platform due to the spectrum of seismic events shall be less than or equal to 3×10^{-6} per year.	WHF-S-IE-BARE (Seq. 10-6)	Table 1.2.5-3
			Protect against platform collapse or waste container breach due to an impact of the cask transfer trolley	WH.HMH.02. The mean frequency of platform collapse or waste container breach from the impact of the cask transfer trolley onto the platform due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	WHF-S-IE-BARE (Seq. 10-6)	Table 1.2.5-3
		Preparation Station #2 Platform (050-HMH0-PLAT-00002)	Protect against platform collapse	WH.HMH.03. The mean frequency of collapse of the preparation station platform due to the spectrum of seismic events shall be less than or equal to 3×10^{-6} per year.	WHF-S-IE-TAD-AO (Seq. 08)	Table 1.2.5-3

Table 1.9-4. Preclosure Nuclear Safety Design Bases for WHF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Wet Handling Facility (WH) (Continued)	Wet Handling Facility (WHF (Continued)	Decontamination Pit; Decontamination Pit Seismic Restraints (050-HM00-BRAC-00001)	Provide lateral stability to the cask in the decontamination pit	WH.HM.01. The mean frequency of the failure of the seismic restraints in the decontamination pit due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	WHF-S-IE-TAD-AO	Table 1.2.5-3

NOTE: “Protect against' in this table means either “reduce the probability of” or “reduce the frequency of.”
Seismic control values shown represent the integration of the probability distribution of SSC failure (i.e., the loss of safety function) with the seismic hazard curve.
Where stated in column 6 that the initiating event does not require further analysis, the design requirement is applied to reduce the frequency of any event sequence that could result in damage to a waste container to the beyond Category 2 frequency range.
The numbers appearing in parentheses in the third column are component numbers.
The event sequence identifier is that of an event sequence before grouping for the purpose of categorization. Therefore, there is no direct correspondence with the event sequence identifier used for the categorized event sequences in Tables 1.7-7 through 1.7-18. Refer to *Wet Handling Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008d) for the description of these event sequences and *Seismic Event Quantification and Categorization Analysis* (BSC 2008g) for a description of seismic event sequences.
Facility Codes: AP: Aging Facility; WH: Wet Handling Facility.
System Codes: DS: DOE and Commercial Waste Package; H: Mechanical Handling; MR: Cask/Canister/Waste Package Process System.
Infrastructure System Codes: EE: Electrical Power; FP: Fire Protection; VC: Surface Nuclear Confinement HVAC; VN: Surface Nonconfinement HVAC.
Subsystem Codes: HAC: Aging Overpack; HAT: Cask Transfer; HC: Transport, Aging, and Disposal Canister Closure; HD: Dual Purpose Canister Cutting; HM: Cask Handling; HMH: Cask Preparation; HTC: Canister Transfer; HTF: Fuel Assembly Transfer.
BWR = boiling water reactor; PWR = pressurized water reactor.

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
Aging (AP)	Aging Handling/ Cask Transfer	Site Transporter (170-HAT0-MEQ-00001)	Protect against spurious movement	AP.RF.HAT.01. The mean probability of spurious movement of the site transporter while the canister is being lifted or lowered shall be less than or equal to 1×10^{-9} per transfer.	RF-ESD06-TAD (Seq. 5-4)	Table 1.2.8-2
			Limit speed	AP.RF.HAT.02. The speed of the site transporter shall be limited to 2.5 mph.	RF-ESD07-TAD (Seq. 3-3)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.RF.HAT.03. The site transporter fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
			Reduce severity of a drop	AP.RF.HAT.04.The site transporter shall be incapable of dropping an aging overpack from a height greater than 3 ft measured from the equipment base.	RF-ESD07-TAD (Seq. 3-3)	Table 1.2.8-2
			Protect against sliding impact and inducing stresses that can breach a waste container	AP.RF.HAT.05. The mean frequency of a sliding impact of the site transporter into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	RF-S-IE-TAD-AO (Seq. 13-5)	Table 1.2.8-2
			Protect against tipover of a site transporter	AP.RF.HAT.06. The mean frequency of a tipover of the site transporter due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 13-5)	Table 1.2.8-2
		Cask Tractor (for use with the Cask Transfer Trailer) (170-HAT0-HEQ-00001)	Limit speed	AP.RF.HAT.07. The speed of the cask tractor shall be limited to 2.5 mph.	RF-ESD09 (Seq. 3-3)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.RF.HAT.08. The cask tractor fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Aging (AP) (Continued)	Aging Handling/ Cask Transfer (Continued)	Cask Transfer Trailer (for use with Transportation Casks and Horizontal Shielded Transfer Casks) (PWR DPC: 170-HAT0-TRLY-00001) (BWR DPC: 170-HAT0-TRLY-00002)	Preclude fuel tank explosion	AP.RF.HAT.09. The cask transfer trailer fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
			Reduce severity of a drop	AP.RF.HAT.10 The cask transfer trailer shall preclude dropping a cask from a height greater than 6 ft measured from the equipment base.	RF-ESD09 (Seq. 2-4)	Table 1.2.8-2
			Preclude puncture of a cask due to impact	AP.RF.HAT.11. The cask transfer trailer shall preclude puncture of a cask due to collision.	Initiating event does not require further analysis.	Table 1.2.8-2
			Preclude puncture of a cask	AP.RF.HAT.12. The cask transfer trailer shall be designed to preclude puncture of casks due to the spectrum of seismic events.	Initiating event does not require further analysis.	Table 1.2.8-2
	Aging Handling/ Aging Overpack	Aging Overpack (TAD: 170-HAC0-ENCL-00003) (Vertical DPC: 170-HAC0-ENCL-00002)	Protect against direct exposure to personnel	AP.RF.HAC.01. The mean conditional probability of loss of shielding of the aging overpack resulting from an impact or collision shall be less than or equal to 1×10^{-5} per impact.	RF-EDS07-TAD (Seq. 3-2)	Table 1.2.7-1
				AP.RF.HAC.02. The mean conditional probability of loss of shielding of the aging overpack resulting from a drop shall be less than or equal to 1×10^{-5} per drop.	RF-ESD08-TAD (Seq. 4-2)	Table 1.2.7-1
DOE and Commercial Waste Package System (DS)	Canistered Spent Nuclear Fuel	DPC (Analyzed as a Representative Canister)	Provide containment	DS.RF.01. The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	RF-ESD06-DPC (Seq. 3-3)	Table 1.5.1-9
				DS.RF.02. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	RF-ESD07-DPC (Seq. 2-3)	Table 1.5.1-9

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel (Continued)	DPC (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DS.RF.03. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	RF-ESD01-DPC (Seq. 3-4)	Table 1.5.1-9
				DS.RF.04. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	RF-ESD12-DPC (Seq. 5-3)	Table 1.5.1-9
				DS.RF.05. The mean conditional probability of breach of a canister contained within an aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	RF-ESD12-DPC (Seq. 2-3)	Table 1.5.1-9
				DS.RF.06. The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	RF-ESD12-DPC (Seq. 9-3)	Table 1.5.1-9
		TAD Canister (Analyzed as a Representative Canister)	Provide containment	DS.RF.07. The mean conditional probability of breach of a canister resulting from a drop of the canister shall be less than or equal to 1×10^{-5} per drop.	RF-ESD06-TAD (Seq. 3-3)	Table 1.5.1-7
				DS.RF.08. The mean conditional probability of breach of a canister resulting from a drop of a load onto the canister shall be less than or equal to 1×10^{-5} per drop.	RF-ESD06-TAD (Seq. 6-3)	Table 1.5.1-7
				DS.RF.09. The mean conditional probability of breach of a canister resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	RF-ESD01-TAD (Seq. 3-4)	Table 1.5.1-7

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel (Continued)	TAD Canister (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DS.RF.10. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	RF-ESD12-TAD (Seq. 4-3)	Table 1.5.1-7
				DS.RF.11. The mean conditional probability of breach of a canister located within the aging overpack resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	RF-ESD12-TAD (Seq. 2-3)	Table 1.5.1-7
				DS.RF.12. The mean conditional probability of breach of a canister located within the canister transfer machine shield bell resulting from the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	RF-ESD12-TAD (Seq. 9-3)	Table 1.5.1-7
Mechanical Handling System (H)	Cask Handling	Transportation Cask (Analyzed as a Representative Cask)	Provide containment	H.RF.01. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop shall be less than or equal to 1×10^{-5} per drop.	RF-ESD06-TAD (Seq. 3-3)	Table 1.2.8-2
				H.RF.02. The mean conditional probability of breach of a canister in a sealed cask resulting from a drop of a load onto the cask shall be less than or equal to 1×10^{-5} per drop.	RF-ESD06-TAD (Seq. 6-3)	Table 1.2.8-2
				H.RF.03. The mean conditional probability of breach of a canister in a sealed cask resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	RF-ESD06-TAD (Seq. 5-3)	Table 1.2.8-2

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Transportation Cask (Analyzed as a Representative Cask) (Continued)	Protect against direct exposure to personnel	H.RF.04. The mean conditional probability of loss of cask gamma shielding resulting from a drop of a cask shall be less than or equal to 1 × 10^{-8} per drop.	RF-ESD02-TAD (Seq. 3-2)	Table 1.2.8-2
				H.RF.05. The mean conditional probability of loss of cask gamma shielding resulting from a collision or side impact to a cask shall be less than or equal to 1 × 10^{-8} per impact.	RF-ESD04-TAD (Seq. 3-2)	Table 1.2.8-2
				H.RF.06. The mean conditional probability of loss of cask gamma shielding resulting from a drop of a load onto a cask shall be less than or equal to 1 × 10^{-5} per impact.	RF-ESD-03-TAD (Seq. 5-2)	Table 1.2.8-2
		SIte Prime Mover	Limit speed	H.RF.07. The speed of the site prime mover shall be limited to 9 mph.	RF-ESD01-TAD (Seq. 3-4)	Table 1.2.8-2
			Preclude fuel tank explosion	H.RF.08. The fuel tank of a site prime mover that enters the facility shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
		Cask Handling Yoke (200-HM00-BEAM-00001)	Protect against drop	H.RF.HM.01. The cask handling yoke is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.6-3
		Cask Handling Crane; 200-ton (200-HM00-CRN-00001)	Protect against drop	H.RF.HM.02. The mean probability of dropping a loaded cask from less than the two-block height resulting from the failure of a piece of equipment within the load-bearing path shall be less than or equal to 3 × 10^{-5} per transfer with the cask yoke or 1 × 10^{-4} per transfer with a sling.	RF-ESD02-TAD (Seq. 2-4) (yoke) RF-ESD02-DPC (Seq. 2-4) (sling)	Table 1.2.6-3
			Protect against drop	H.RF.HM.03. The mean probability of dropping a loaded cask from a two-block height resulting from the failure of a piece of equipment within the load-bearing path shall be less than or equal to 4 × 10^{-7} per transfer.	RF-ESD02-TAD (Seq. 7-4) (yoke) RF-ESD02-DPC (Seq. 7-4) (sling)	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Cask Handling Crane; 200-ton (200-HM00-CRN-00001) (Continued)	Limit drop height	H.RF.HM.04. The two-block drop height shall not exceed 30 ft from bottom of shortest cask to the floor.	RF-ESD02-TAD (Seq. 7)	Table 1.2.6-3
			Protect against drop of a load onto a cask	H.RF.HM.05. The mean probability of dropping a load onto a loaded cask or its contents shall be less than or equal to 9×10^{-5} per cask handled.	RF-ESD02-TAD (Seq. 6-4)	Table 1.2.6-3
			Limit speed	H.RF.HM.06. The speed of the trolley and bridge shall be limited to 20 ft/min.	RF-ESD02-TAD (Seq. 6)	Table 1.2.6-3
			Protect against crane collapse onto a waste container	H.RF.HM.07. The mean frequency of collapse of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 7-6)	Table 1.2.6-3
			Protect against a cask or heavy object drop from the crane	H.RF.HM.08. The mean frequency of a hoist system failure of the cask handling crane due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	RF-S-IE-TAD-AO (Seq. 7-6)	Table 1.2.6-3
		Cask Transfer Trolley and Pedestal (Trolley: 200-HM00-TRLY-00001) (Pedestal: 200-HM00-PED-00001)	Limit speed	H.RF.HM.09. The speed of the cask transfer trolley shall be limited to 2.5 mph.	RF-ESD04-TAD (Seq. 3-4)	Table 1.2.6-3
			Protect against spurious movement	H.RF.HM.10. The mean probability of spurious movement of the cask transfer trolley while a canister is being lifted by the canister transfer machine shall be less than or equal to 1×10^{-9} per transfer.	RF-ESD06-TAD (Seq. 4-3)	Table 1.2.6-3
			Protect against impact and inducing stresses on the waste container	H.RF.HM.11. The mean frequency of sliding the cask transfer trolley into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 9-6)	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Cask Transfer Trolley and Pedestal (Trolley: 200-HM00-TRLY-00001) (Pedestal: 200-HM00-PED-00001) (Continued)	Protect against impact and inducing stresses on the waste container (Continued)	H.RF.HM.12. The mean frequency of a rocking impact of the cask transfer trolley into a wall and inducing stresses that can breach the waste container due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 9-6)	Table 1.2.6-3
		Lid Bolting Room Crane (200-HMC0-CRN-00001)	Protect against collapse of the lid bolting room crane	H.RF.HMC.01. The mean frequency of collapse of the lid bolting room crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 15-5)	Table 1.2.6-3
		Horizontal Lifting Beam (200-HMC0-BEAM-00001)	Protect against drop	H.RF.HMC.02. The horizontal lifting beam is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.6-3
		Cask Lid Lifting Grapples (DPC) (200-HMH0-HEQ-00008)	Protect against drop of a load onto a DPC	H.RF.HMH.01. The cask lid lifting grapple is an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.6-3
	Cask Handling/Cask Preparation	Rail Cask Lid Adapters (200-HMH0-HEQ-00002)	Protect against drop	H.RF.HMH.02. The rail cask lid adapters are an integral part of the load-bearing path. See cask handling crane requirements.	See cask handling crane requirements.	Table 1.2.6-3
		DPC Lid Adapter (200-HMH0-HEQ-00001)	Protect against drop of a DPC	H.RF.HMH.03 The DPC lid adapter is an integral part of the load-bearing path. See canister transfer machine requirements.	See canister transfer machine requirements.	Table 1.2.6-3
	Waste Transfer/ Canister Transfer	Canister Transfer Machine Maintenance Crane (200-HTC0-CRN-00001)	Protect against collapse of the canister transfer machine maintenance crane	H.RF.HTC.01. The mean frequency of collapse of the canister transfer machine maintenance crane due to the spectrum of seismic events shall be less than or equal to 8×10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 12-5)	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	Canister Transfer Machine (200- HTC0-FHM-00001)	Protect against drop	H.RF.HTC.02. The mean probability of dropping a canister from below the two-block height due to the failure of a piece of equipment within the load-bearing path shall be less than or equal to 1×10^{-5} per transfer.	RF-ESD06-TAD (Seq. 3-3)	Table 1.2.6-3
			Protect against drop	H.RF.HTC.03. The mean probability of drop of a canister from the two-block height due to the failure of a piece of equipment within the load-bearing path shall be less than or equal to 3×10^{-8} per transfer.	RF-ESD06-TAD (Seq. 8-3)	Table 1.2.6-3
			Limit drop height	H.RF.HTC.04. The two-block drop height shall not exceed 45 ft from the bottom of a canister to the cavity floor of the cask or aging overpack.	RF-ESD06-TAD (Seq. 8-3)	Table 1.2.6-3
			Protect against drop of a load onto a canister	H.RF.HTC.05. The mean probability of dropping a load onto a canister shall be less than or equal to 1×10^{-5} per transfer.	RF-ESD06-TAD (Seq. 6-3)	Table 1.2.6-3
			Protect against spurious movement	H.RF.HTC.06. The mean probability of a spurious movement of the canister transfer machine while a canister is being lifted or lowered shall be less than or equal to 5×10^{-9} per transfer.	RF-ESD06-TAD (Seq. 4-3)	Table 1.2.6-3
			Limit speed	H.RF.HTC.07. The speed of the canister transfer machine trolley and bridge shall be limited to 20 ft/min.	RF-ESD06-TAD (Seq. 5-4)	Table 1.2.6-3
			Preclude non-flat bottom drop of a DPC or TAD	H.RF.HTC.08. The canister transfer machine shall preclude non-flat-bottom drops of DPCs and TADs.	Initiating event does not require further analysis.	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Mechanical Handling System (H) (Continued)	Waste Transfer/ Canister Transfer (Continued)	Canister Transfer Machine (200- HTC0-FHM-00001) (Continued)	Protect against direct exposure to personnel	H.RF.HTC.09. The mean probability of inadvertent radiation streaming resulting from the inadvertent opening of the canister transfer machine slide gate, the inadvertent raising of the canister transfer machine shield skirt, or an inadvertent motion of the canister transfer machine away from an open port shall be less than or equal to 1 × 10^{-6} per transfer.	RF-ESD06-TAD (Seq. 4-2)	Table 1.2.6-3
			Preclude canister breach	H.RF.HTC.10. Closure of the canister transfer machine slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.6-3
			Protect against collapse of the canister transfer machine	H.RF.HTC.11. The mean frequency of collapse of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 1 × 10^{-5} per year.	RF-S-IE-TAD-AO (Seq. 11-5)	Table 1.2.6-3
			Protect against a canister or heavy object drop from the canister transfer machine	H.RF.HTC.12. The mean frequency of a hoist system failure of the canister transfer machine due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	RF-S-IE-TAD-AO (Seq. 11-5)	Table 1.2.6-3
		Canister Transfer Machine Grapples (200-HTC0-HEQ-00001)	Protect against canister drop	H.RF.HTC.13. The canister transfer machine grapple is an integral part of the load-bearing path of the canister transfer machine. See canister transfer machine requirements.	See canister transfer machine requirements.	Table 1.2.6-3
Receipt Facility (RF)	Receipt Facility (RF)	Structure	Maintain building structural integrity to protect ITS SSCs inside the building from external events	RF.01. The mean frequency of building collapse due to winds less than or equal to 120 mph shall not exceed 1 × 10^{-6} per year. RF.02. The mean frequency of building collapse due to volcanic ash fall less than or equal to a roof load of 21 lb/ft^2 shall not exceed 1 × 10^{-6} per year. RF.03. The RF shall be located such that there is a distance of at least one-half mile between the RF and the repository heliport.	Initiating event does not require further analysis.	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Receipt Facility (RF) (Continued)	Receipt Facility (RF) (Continued)	Structure (Continued)	Protect against building collapse onto waste containers	RF.04. The mean frequency of collapse of the RF structure due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-6}/ per year.	RF-S-IE-TAD-AO (Seq. 3)	Table 1.2.6-3
		Shield Doors (Including Anchorages)	Protect against direct exposure of personnel	RF.05. Equipment shield doors shall have a mean probability of inadvertent opening of less than or equal to 1 × 10^{-7} per waste container handled.	RF-ESD011 (Seq. 2)	Table 1.2.6-3
			Preclude collapse onto waste containers	RF.06. An equipment shield door falling onto a waste container as a result of impact from a conveyance shall be precluded.	Initiating event does not require further analysis.	Table 1.2.6-3
			Protect against equipment shield door collapse onto a waste container	RF.07. The mean frequency of collapse of equipment shield doors (including attachment of door to wall and frame anchorages) due to the spectrum of seismic events shall be less than or equal to 6 × 10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 10-6)	Table 1.2.6-3
		Cask Port Slide Gate (200-HTC0-HTCH-00001)	Protect against dropping a canister due to a spurious closure of the slide gate	RF.HTC.01. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 5 × 10^{-6} per transfer.	RF-ESD06-TAD (Seq. 3-3)	Table 1.2.6-3
			Protect against direct exposure to personnel	RF.HTC.02. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	RF-ESD11 (Seq. 2)	Table 1.2.6-3
			Preclude canister breach	RF.HTC.03. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.6-3
		Aging Overpack Port Slide Gate (200- HTC0-HTCH-00002)	Protect against dropping a canister due to a spurious closure of the slide gate	RF.HTC.04. The mean probability of a canister drop resulting from a spurious closure of the slide gate shall be less than or equal to 5 × 10^{-6} per transfer.	RF-ESD06-TAD (Seq. 3-3)	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
Receipt Facility (RF) (Continued)	Receipt Facility (RF) (Continued)	Aging Overpack Port Slide Gate (200- HTC0-HTCH-00002) (Continued)	Protect against direct exposure to personnel	RF.HTC.05. The mean probability of occurrence of an inadvertent opening of a slide gate shall be less than or equal to 4 × 10^{-9} per transfer.	RF-ESD11 (Seq. 2)	Table 1.2.6-3
			Preclude canister breach	RF.HTC.06. Closure of the slide gate shall be incapable of breaching a canister.	Initiating event does not require further analysis.	Table 1.2.6-3
		Cask Preparation Platform (200- HMH0-PLAT-00001)	Protect against collapse	RF.HMH.01. The mean frequency of collapse of the cask preparation platform due to the spectrum of seismic events shall be less than or equal to 3 × 10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 8-6)	Table 1.2.6-3
			Protect against platform collapse or waste container breach due to an impact from the cask transfer trolley	RF.HMH.02. The mean frequency of platform collapse or waste container breach from the impact of the cask transfer trolley into the platform due to the spectrum of seismic events shall be less than or equal to 2 × 10^{-5} per year.	RF-S-IE-TAD-AO (Seq. 9-6)	Table 1.2.6-3
		Lid Bolting Room Platform (200-HMC0-PLAT-00003)	Protect against platform collapse	RF.HMC.01. The mean frequency of collapse of the lid bolting room platform due to the spectrum of seismic events shall be less than or equal to 3 × 10^{-6} per year.	RF-S-IE-TAD-AO (Seq. 14-5)	Table 1.2.6-3

Table 1.9-5. Preclosure Nuclear Safety Design Bases for RF ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Receipt Facility (RF) (Continued)	Receipt Facility (RF) (Continued)	Lid Bolting Room Platform (200-HMC0-PLAT-00003) (Continued)	Protect against collapse or waste container breach due to an impact from the site transporter	RF.HMC.02. The mean frequency of platform collapse or waste container breach from the impact of the site transporter into the platform due to the spectrum of seismic events shall be less than or equal to 2×10^{-5} per year.	RF-S-IE-TAD-AO (Seq. 14-5)	Table 1.2.6-3

NOTE: “Protect against' in this table means either “reduce the probability of” or “reduce the frequency of.”
Seismic control values shown represent the integration of the probability distribution of SSC failure (i.e., the loss of safety function) with the seismic hazard curve.
Where stated in column 6 that the initiating event does not require further analysis, the design requirement is applied to reduce the frequency of any event sequence that could result in damage to a waste container to the beyond Category 2 frequency range. The numbers appearing in parentheses in the third column are component numbers.
The event sequence identifier is that of an event sequence before grouping for the purpose of categorization. Therefore, there is no direct correspondence with the event sequence identifier used for the categorized event sequences in Tables 1.7-7 through 1.7-18. Refer to *Receipt Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008c) for the description of these event sequences and *Seismic Event Quantification and Categorization Analysis* (BSC 2008g) for a description of seismic event sequences.
Facility Codes: AP: Aging Facility; RF: Receipt Facility.
System Codes: DS: DOE and Commercial Waste Package; H: Mechanical Handling.
Subsystem Codes: HAC: Aging Overpack; HAT: Cask Transfer; HM: Cask Handling; HMC: Cask Receipt; HMH: Cask Preparation; HTC: Canister Transfer.
BWR= boiling water reactor; PWR = pressurized water reactor.

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Aging Facility (AP)	Aging Handling/ Aging Pad	Aging Pad	Protect ITS SSCs from external events	AP.SB.01. The aging pads shall be located such that there is a distance of at least one-half mile between the aging pads and the repository heliport.	Initiating event does not require further analysis.	Table 1.2.7-1
			Protect against aging overpack tipover	AP.SB.02. The mean frequency of aging pad structure failure causing aging overpack tipover due to the spectrum of seismic events shall be less than or equal to 1×10^{-5} per year.	ISO-IE-S-MAIN (Seq. 03)	Table 1.2.7-1
	Aging Handling/Cask Transfer	Cask Tractor (for use with the Cask Transfer Trailer) (170-HAT0-HEQ-00001)	Limit speed	AP.SB.HAT.01. The speed of the cask tractor shall be limited to 2.5 mph.	ISO-ESD03-HDPC (Seq. 2-4)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.SB.HAT.02. The cask tractor fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
		Cask Transfer Trailer (for use with Transportation Casks and Horizontal Shielded Transfer Casks) (PWR DPC: 170-HAT0-TRLY-00001] (BWR DPC: 170-HAT0-TRLY-00002)	Preclude fuel tank explosion	AP.SB.HAT.03. The cask transfer trailer fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
			Reduce severity of a drop	AP.SB.HAT.04. The cask transfer trailer shall preclude dropping a cask from a height greater than 6 ft measured from the equipment base.	ISO-ESD03-HDPC (Seq. 3-4)	Table 1.2.8-2
			Preclude puncture of a cask	AP.SB.HAT.05. The cask transfer trailer shall preclude puncture of a cask due to collision.	Initiating event does not require further analysis.	Table 1.2.8-2
			Preclude puncture of a canister	AP.SB.HAT.06. The cask transfer trailer shall preclude puncture of the canister by the hydraulic ram.	Initiating event does not require further analysis.	Table 1.2.8-2

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Aging Facility (AP) (Continued)	Aging Handling/Cask Transfer (Continued)	Cask Transfer Trailer (for use with Transportation Casks and Horizontal Shielded Transfer Casks) (PWR DPC: 170-HAT0-TRLY-00001] (BWR DPC: 170-HAT0-TRLY-00002) (Continued)	Limit speed	AP.SB.HAT.07. The speed of the cask transfer trailer shall be limited to 2.5 mph.	ISO-ESD03-HDPC (Seq. 2-4)	Table 1.2.8-2
			Preclude puncture of a cask	AP.SB.HAT.08. The cask transfer trailer shall be designed to preclude puncture of a cask due to the spectrum of seismic events.	Initiating event does not require further analysis.	Table 1.2.8-2
		Site Transporter (170-HAT0-MEQ-00001)	Limit speed	AP.SB.HAT.09. The speed of the site transporter shall be limited to 2.5 mph.	ISO-ESD02-TAD (Seq. 2-3)	Table 1.2.8-2
			Preclude fuel tank explosion	AP.SB.HAT.10. The site transporter fuel tank shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2
			Reduce severity of a drop	AP.SB.HAT.11. The site transporter shall preclude a vertical drop of an aging overpack from a height greater than 3 ft measured from the equipment base.	ISO-ESD02-TAD (Seq. 3-3)	Table 1.2.8-2
			Protect against tipover of the site transporter	AP.SB.HAT.12. The mean frequency of tipover of the site transporter due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	ISO-IE-S-MAIN (Seq. 04)	Table 1.2.8-2
	Aging Handling/ Aging Overpack	Horizontal Aging Module (170-HAC0-ENCL-00001)	Protect against direct exposure to personnel	AP.SB.HAC.01. The mean conditional probability of loss of horizontal aging module gamma shielding due to an impact or collision shall be less than or equal to 1×10^{-5} per impact.	ISO-ESD04-HDPC (Seq. 3-2)	Table 1.2.7-1
			Protect against structural collapse onto a waste container	AP.SB.HAC.02. The mean frequency of collapse of the horizontal aging module structure due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	ISO-IE-S-MAIN (Seq. 07)	Table 1.2.7-1

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Aging Facility (AP) (Continued)	Aging Handling/ Aging Overpack (Continued)	Horizontal Shielded Transfer Cask (170-HAC0-HEQ-00001)	Provide containment	AP.SB.HAC.03. The mean conditional probability of breach of a canister in a sealed horizontal shielded transfer cask on a cask transfer trailer resulting from a drop shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD03-HDPC (Seq. 3-4)	Table 1.2.5-3
				AP.SB.HAC.04. The mean probability of breach of a canister in an horizontal shielded transfer cask on a cask transfer trailer resulting from a drop of a load onto the horizontal shielded transfer cask shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD04-HDPC (Seq. 2-3)	Table 1.2.5-3
				AP.SB.HAC.05. The mean conditional probability of breach of a canister in a sealed horizontal shielded transfer cask on a cask transfer trailer resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per impact.	ISO-ESD03-HDPC (Seq. 2-4)	Table 1.2.5-3
		Aging Overpack (TAD: 170-HAC0-ENCL-00003) (Vertical DPC: 170-HAC0-ENCL-00002)	Protect against direct exposure to personnel	AP.SB.HAC.06. The mean conditional probability of loss of shielding of the aging overpack resulting from an impact or collision shall be less than or equal to 1×10^{-5} per impact.	ISO-ESD02-TAD (Seq. 2-2)	Table 1.2.7-1
				AP.SB.HAC.07. The mean conditional probability of loss of shielding of the aging overpack resulting from a drop shall be less than or equal to 5×10^{-6} per drop.	ISO-ESD02-TAD (Seq. 3-2)	Table 1.2.7-1
			Protect against sliding of an aging overpack	AP.SB.HAC.08. The mean frequency of sliding of an aging overpack (with a waste container) into another aging overpack on the aging pad due to the spectrum of seismic events shall be less than or equal to 5×10^{-6} per year.	ISO-IE-S-MAIN (Seq. 06)	Table 1.2.7-1
			Protect against tipover of an aging overpack	AP.SB.HAC.09. The mean frequency of tipover of the aging overpack on the aging pad due to the spectrum of seismic events shall be less than or equal to 5×10^{-8} per year.	ISO-IE-S-MAIN (Seq. 06)	Table 1.2.7-1

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Balance of Plant (SB)	Flood Protection	Flood Control Features	Protect ITS SSCs from external flooding events	SB.01. The site flood control features will be designed to the probable maximum flood.	Initiating event does not require further analysis.	Table 1.2.3-3 Table 1.2.4-4 Table 1.2.5-3 Table 1.2.6-3 Table 1.2.7-1
DOE and Commercial Waste Package System (DS)	High-Level Waste/DOE SNF Codisposal	DOE Standardized Canister	Provide containment	DS.SB.01. The mean conditional probability of breach of a DOE standardized canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	ISO-ESD09-DSTD (Seq. 2-4)	Table 1.5.1-25
		HLW Canister	Provide containment	DS.SB.02. The mean conditional probability of breach of a HLW canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	ISO-ESD09-HLW (Seq. 2-4)	Table 1.5.1-17
	Canistered Spent Nuclear Fuel	DPC and TAD Canister (Both Analyzed as a Representative Canister)	Provide containment	DS.SB.03. The mean conditional probability of breach of a canister within an aging overpack following a drop shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD02-TAD (Seq. 3-3)	Table 1.2.7-1 Table 1.5.1-7 Table 1.5.1-9
				DS.SB.04. The mean conditional probability of breach of a canister within an aging overpack resulting from a side impact or collision shall be less than or equal to 1×10^{-8} per event	ISO-ESD02-TAD (Seq. 2-3)	Table 1.2.7-1 Table 1.5.1-7 Table 1.5.1-9
				DS.SB.05. The mean conditional probability of breach of a canister in a horizontal aging module resulting from a collision or side impact shall be less than or equal to 1×10^{-8} per event.	ISO-ESD04-HDPC (Seq. 3-3)	Table 1.2.7-1 Table 1.5.1-7 Table 1.5.1-9
				DS.SB.06. The mean conditional probability of breach of a canister resulting from a drop of a load onto a horizontal aging module shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD04-HDPC (Seq. 2-3)	Table 1.2.7-1 Table 1.5.1-7 Table 1.5.1-9

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
DOE and Commercial Waste Package System (DS) (Continued)	Canistered Spent Nuclear Fuel (Continued)	DPC and TAD Canister (Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	DS.SB.07. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 2 × 10^{-6} per fire event.	ISO-ESD09-TAD (Seq. 2-4)	Table 1.5.1-7 Table 1.5.1-9
				DS.SB.08. The mean conditional probability of breach of a canister located within a horizontal aging module resulting from the spectrum of fires shall be less than or equal to 1 × 10^{-6} per fire event.	ISO-ESD09-HDPC (Seq. 5-4)	Table 1.5.1-7 Table 1.5.1-9
				DS.SB.09. The mean conditional probability of breach of a canister contained within an aging overpack resulting from the spectrum of fires shall be less than or equal to 1 × 10^{-6} per fire event.	ISO-ESD09-HDPC (Seq. 5-4)	Table 1.5.1-7 Table 1.5.1-9
Mechanical Handling System (H)	Cask Handling	Transportation Cask (Analyzed as a Representative Cask)	Provide containment	H.SB.01. The mean conditional probability of breach of a canister in a sealed cask on a railcar, truck trailer, or cask transfer trailer resulting from a drop shall be less than or equal to 1 × 10^{-5} per drop.	ISO-ESD01-TAD (Seq. 2-4)	Table 1.2.8-2
				H.SB.02. The mean probability of breach of a canister in a sealed cask on a railcar, truck trailer, or cask transfer trailer resulting from a drop of a load onto the cask shall be less than or equal to 1 × 10^{-5} per drop.	ISO-ESD01-TAD (Seq. 5-4)	Table 1.2.8-2
				H.SB.03. The mean conditional probability of breach of a canister in a sealed cask on a railcar, truck trailer, or cask transfer trailer resulting from a side impact or collision shall be less than or equal to 1 × 10^{-8} per impact.	ISO-ESD01-TAD (Seq. 3-4)	Table 1.2.8-2

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Mechanical Handling System (H) (Continued)	Cask Handling (Continued)	Transportation Cask (Analyzed as a Representative Cask) (Continued)	Provide containment (Continued)	H.SB.04. The mean conditional probability of breach of a sealed cask containing uncanistered commercial spent nuclear fuel on a truck trailer resulting from a collision followed by a rollover/drop shall be less than or equal to 1×10^{-8} per drop.	ISO-ESD01-UCSNF (Seq. 4-4)	Table 1.2.8-2
				H.SB.05. The mean conditional probability of breach of a sealed cask containing uncanistered commercial spent nuclear fuel resulting from a drop of a load onto the cask shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD01-UCSNF (Seq. 5-4)	Table 1.2.8-2
			Protect against direct exposure to personnel	H.SB.06. The mean conditional probability of loss of gamma shielding of a cask resulting from a drop shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD01-TAD (Seq. 2-2)	Table 1.2.8-2
				H.SB.07. The mean conditional probability of loss of gamma shielding of a cask resulting from a collision or side impact shall be less than or equal to 1×10^{-8} per impact.	ISO-ESD01-TAD (Seq. 3-2)	Table 1.2.8-2
				H.SB.08. The mean conditional probability of loss of gamma shielding of a cask resulting from a drop of a load onto it shall be less than or equal to 1×10^{-5} per drop.	ISO-ESD01-TAD (Seq. 5-2)	Table 1.2.8-2
		Site Prime Mover	Limit speed	H.SB.09. The speed of the site prime mover shall be limited to 9 mph.	ISO-ESD01-TAD (Seq. 2-4)	Table 1.2.8-2
			Preclude fuel tank explosion	H.SB.10. The fuel tank of a site prime mover that enters a facility shall preclude fuel tank explosions.	Initiating event does not require further analysis.	Table 1.2.8-2

Table 1.9-6. Preclosure Nuclear Safety Design Bases for Intrasite Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	Controlling Parameters and Values		
Naval SNF Waste Package System (DN)	Naval SNF	Naval SNF Canister (Analyzed as a Representative Canister)	Provide containment	DN.SB.01. The mean conditional probability of breach of a canister contained within a cask resulting from the spectrum of fires shall be less than or equal to 1×10^{-6} per fire event.	ISO-ESD09-NAV (Seq. 2-4)	Table 1.5.1-30

NOTE: “Protect against' in this table means either “reduce the probability of” or “reduce the frequency of.”
Seismic control values shown represent the integration of the probability distribution of SSC failure (i.e., the loss of safety function) with the seismic hazard curve.
Where stated in column 6 that the initiating event does not require further analysis, the design requirement is applied to reduce the frequency of any event sequence that could result in damage to a waste container to the beyond Category 2 frequency range.
The numbers appearing in parentheses in the third column are component numbers.
The event sequence identifier is that of an event sequence before grouping for the purpose of categorization. Therefore, there is no direct correspondence with the event sequence identifier used for the categorized event sequences in Tables 1.7-7 through 1.7-18. Refer to *Intra-site Operations and BOP Reliability and Event Sequence Categorization Analysis* (BSC 2008e) for the description of these event sequences and *Seismic Event Quantification and Categorization Analysis* (BSC 2008g) for a description of seismic event sequences.
Facility Codes: AP: Aging Facility; SB: Balance of Plant.
System Codes: DN: Naval Spent Nuclear Fuel Waste Package; DS: DOE and Commercial Waste Package; H: Mechanical Handling.
Subsystem Codes: HAC: Aging Overpack; HAT: Cask Transfer.

Table 1.9-7. Preclosure Nuclear Safety Design Bases for the Subsurface Operations ITS SSCs

System or Facility (System Code)	Subsystem or Function (As Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
DOE and Commercial Waste Package System (DS)	DOE and Commercial Waste Package	Entire	Provide containment	DS.SS.01. The mean conditional probability of breach of a sealed waste package resulting from a side impact shall be less than or equal to 1×10^{-8} per impact.	SSO-ESD03-WP (Seq. 6-3)	Table 1.5.2-6
				DS.SS.02. The mean conditional probability of breach of a sealed waste package resulting from a drop of a load onto the waste package shall be less than or equal to 1×10^{-5} per drop.	SSO-ESD01-WP (Seq. 6-4)	Table 1.5.2-6
				DS.SS.03 The mean conditional probability of breach of a sealed waste package inside the TEV resulting from an end-on impact or collision shall be less than or equal to 1×10^{-8} per impact.	SSO-ESD03-WP (Seq. 2-3)	Table 1.5.2-6
				DS.SS.04. The mean conditional probability of breach of a canister inside a sealed waste package as a result of the spectrum of fires shall be less than or equal to 3×10^{-4} per fire event.	SSO-ESD05-WP (Seq. 3-4)	Table 1.5.2-6
			Protect against a rockfall breaching a waste package	DS.SS.05. The mean frequency of breach of the waste package from a rockfall due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	Based on screening analysis of physical and thermal impacts from rockfalls	Table 1.5.2-6
			Protect against a waste package breach due to seismic vibratory motion in an emplacement drift	DS.SS.06. The mean frequency of breach of the waste package from vibratory motion impacts in an emplacement drift due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	Based on screening analysis of waste package impacts in the emplacement drift	Table 1.5.2-6

Table 1.9-7. Preclosure Nuclear Safety Design Bases for the Subsurface Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (As Applicable)	Component	Nuclear Safety Design Bases		Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
			Safety Function	**Controlling Parameters and Values**		
Emplacement and Retrieval/Drip Shield Installation System (HE)	Emplacement and Retrieval/Drip Shield Installation	TEV	Protect against TEV runaway	HE.SS.01. The probability of runaway of a TEV that can result in a potential breach of a waste package shall be less than or equal to 2×10^{-9} per transport.	Initiating event does not require further analysis	Table 1.3.3-5
			Protect against direct exposure of personnel	HE.SS.02. The mean probability of inadvertent TEV door opening shall be less than or equal to 1×10^{-7} per transport.	SSO-ESD-04-WP (Seq. 4-2)	Table 1.3.3-5
			Protect against a tipover of the TEV	HE.SS.03. The mean frequency of tipover of the TEV due to the spectrum of seismic events shall be less than or equal to 2×10^{-6} per year.	SSO-IE-S-MAIN (Seq. 03)	Table 1.3.3-5
			Protect against ejection of the waste package from the shielded enclosure of the TEV	HE.SS.04. The mean frequency of ejection of a waste package from the TEV due to the spectrum of seismic events shall be less than or equal to 2×10^{-4} per year.	SSO-IE-S-MAIN (Seq. 03)	Table 1.3.3-5
Naval SNF Waste Package System (DN)	Naval SNF Waste Package	Entire	Provide containment	DN.SS.01. The mean conditional probability of breach of a sealed waste package resulting from a side impact shall be less than or equal to 1×10^{-8} per drop.	SSO-ESD03-WP (Seq. 6-3)	Table 1.5.2-6
				DN.SS.02. The mean conditional probability of breach of a sealed waste package resulting from a drop of a load onto the waste package shall be less than or equal to 1×10^{-5} per drop.	SSO-ESD01-WP (Seq. 6-4)	Table 1.5.2-6
				DN.SS.03 The mean conditional probability of breach of a sealed waste package in the TEV resulting from an end-on impact or collision shall be less than or equal to 1×10^{-8} per impact.	SSO-ESD03-WP (Seq. 2-3)	Table 1.5.2-6

Table 1.9-7. Preclosure Nuclear Safety Design Bases for the Subsurface Operations ITS SSCs (Continued)

System or Facility (System Code)	Subsystem or Function (As Applicable)	Component	Nuclear Safety Design Bases: Safety Function	Nuclear Safety Design Bases: Controlling Parameters and Values	Representative Event Sequence (Sequence Number)	LA Section Presenting Design Criteria
Naval SNF Waste Package System (DN) (Continued)	Naval SNF Waste Package (Continued)	Entire (Continued)	Provide containment (Continued)	DN.SS.04. The mean conditional probability of breach of a canister inside a sealed waste package as a result of the spectrum of fires shall be less than or equal to 1×10^{-4} per fire event.	SSO-ESD05-WP (Seq. 3-4)	Table 1.5.2-6
			Protect against a rockfall breaching a waste package	DN.SS.05. The mean frequency of breach of the waste package from a rockfall due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	Based on screening analysis of physical and thermal impacts from rockfalls	Table 1.5.2-6
			Protect against a waste package breach	DN.SS.06. The mean frequency of breach of the waste package from vibratory motion impacts in an emplacement drift due to the spectrum of seismic events shall be less than or equal to 1×10^{-6} per year.	Based on screening analysis of waste package impacts in the emplacement drift	Table 1.5.2-6

NOTE: “Protect against' in this table means either “reduce the probability of” or “reduce the frequency of.”
Seismic control values shown represent the integration of the probability distribution of SSC failure (i.e., the loss of safety function) with the seismic hazard curve.
Where stated in column 6 that the initiating event does not require further analysis, the design requirement is applied to reduce the frequency of any event sequence that could result in damage to a waste container to the beyond Category 2 frequency range.
The numbers appearing in parentheses in the third column are component numbers.
The event sequence identifier is that of an event sequence before grouping for the purpose of categorization. Therefore, there is no direct correspondence with the event sequence identifier used for the categorized event sequences in Tables 1.7-7 through 1.7-18. Refer to *Subsurface Operations Reliability and Event Sequence Categorization Analysis* (BSC 2008f) for the description of these event sequences and *Seismic Event Quantification and Categorization Analysis* (BSC 2008g) for a description of seismic event sequences.
Facility Codes: SS: Subsurface Facility.
System Codes: DN: Naval Spent Nuclear Fuel Waste Package; DS: DOE and Commercial Waste Package; HE: Emplacement and Retrieval/Drip Shield Installation.

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
UNB	Topography and surficial soils	NA	ITWI	Prevents or substantially reduces the rate of movement of water	09-04 Reclamation of Lands Disturbed by Repository
UNB	Unsaturated zone above the repository	NA	ITWI	Prevents or substantially reduces the rate of movement of water	01-20 Repository Standoff from Paintbrush Nonwelded Hydrogeologic Unit 01-21 Minimum Thickness of the Paintbrush Nonwelded Hydrogeologic Unit above the Repository
EBS	Emplacement drift—Nonemplacement openings	Subsurface Facilities —Nonemplacement openings	Non ITWI	None	NA
		Subsurface Facilities—Ground support for nonemplacement openings			
EBS	Emplacement drift—Closure	Borehole closure	Non ITWI	None	NA
		Ramp and shaft closure			
EBS	Emplacement drift	Emplacement drift	ITWI	Prevents or substantially reduces the rate of movement of water Prevents or substantially reduces the rate of movement of radionuclides	01-06 Repository Elevation—Overburden Thickness 01-10 Emplacement Drift Configuration 02-03 Committed Materials

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers (Continued)

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
EBS	Emplacement drift	Ground support for emplacement drift	Non ITWI	None	NA
		Subsurface ventilation system			
EBS	Drip shield	Drip shield	ITWI	Prevents or substantially reduces the rate of movement of water Prevents or substantially reduces the rate of movement of radionuclides	07-02 Drip Shield Design and Installation 07-04 Drip Shield Materials and Thicknesses 07-07 EBS Drip Shield/Emplacement Drift Invert Materials Interactions 07-09 Drip Shield Fabrication 07-10 Drip Shield Fabrication Weld Inspections 07-11 Drip Shield Fabrication Welding Flaws 07-12 Drip Shield Fabrication Weld Materials 07-13 Drip Shield Heat Treatment 07-14 Drip Shield Handling

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers (Continued)

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
EBS	Waste package	Waste package outer corrosion barrier	ITWI	Prevents or substantially reduces the rate of movement of water Prevents or substantially reduces the release rate of radionuclides from the waste Prevents or substantially reduces the rate of movement of radionuclides	03-03 Waste Package Outer Barrier 03-12 Waste Package Fabrication 03-13 Waste Package Fabrication Weld Inspections 03-14 Waste Package Welding Materials 03-15 Waste Package Fabrication Welding Flaws 03-16 Waste Package Annealing 03-17 Waste Package Closure 03-18 Waste Package Surface Marring Prior to Emplacement 03-19 Waste Package Outer Barrier Material Specifications 03-21 Waste Package Handling 03-23 Waste Package Surface Finish 03-24 Waste Package Surface Damage Prior to Closure 03-26 Waste Package Moisture Removal and Inerting 05-03 Waste Package Thermal Limits 06-03 Waste Package Temperature Limit

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers (Continued)

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
EBS	Waste packages	Waste package inner vessel	ITWI	Prevents or substantially reduces the release rate of radionuclides from the waste Prevents or substantially reduces the rate of movement of radionuclides	03-14 Waste Package Welding Materials 03-15 Waste Package Fabrication Welding Flaws
EBS	Waste form and waste package internals—TAD canister	TAD canister	ITWI	Prevents or substantially reduces the release rate of radionuclides from the waste Prevents or substantially reduces the rate of movement of radionuclides	See the *Transportation, Aging, and Disposal Canister System Performance Specification* (DOE 2008a, Sections 3.1.1(1) and (2) and 3.1.8(1))
	Waste Form and Waste Package Internals —Naval SNF Canister	Naval SNF Canister	ITWI	Prevents or substantially reduces the release rate of radionuclides from the waste Prevents or substantially reduces the rate of movement of radionuclides Reduces the probability of criticality	See the *Waste Acceptance System Requirements Document* (DOE 2008b, Section 4.4.1(b)) and the *Integrated Interface Control Document* (DOE 2008c Sections 10.3.2.1[d] and 10.3.2.3 and Figure C-6)
EBS	Waste Form and Waste Package Internals—DOE SNF Canister and HLW Canister	DOE SNF canister	Non ITWI	None	NA
		HLW canister			

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers (Continued)

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
EBS	Waste Form and Waste Package Internals—Naval SNF Canister System Components	Naval SNF baskets	ITWI	Reduces the probability of criticality	See the *Waste Acceptance System Requirements Document* (DOE 2008b Section 4.4.8(b)).
		Naval SNF basket spacers			
		Naval neutron absorber assemblies (includes retention hardware)			
		Naval Control Rods (includes retention hardware)			
		Corrosion-resistant cans			
EBS	Waste Form and Waste Package Internals—Codisposal Waste Package Internals	Codisposal packages internals	Non ITWI	None	NA
		Baskets, spacers			
EBS	Waste Form and Waste Package Internals—TAD Canister Internals	Neutron absorbers	ITWI	Reduces the probability of criticality	See the *Transportation, Aging, and Disposal Canister System Performance Specification* (DOE 2008a, Section 3.1.5(2))

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers (Continued)

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
EBS	Waste Form and Waste Package Internals—DOE SNF Canister Internals	Neutron Absorbers	ITWI	Reduces the probability of criticality	See the *Waste Acceptance System Requirements Document* (DOE 2008b, Section 4.3.8 (b))
EBS	Waste Form and Waste Package Internals—Commercial Spent Nuclear Fuel and High Level Glass	CSNF	ITWI	Prevents or substantially reduces the release rate of radionuclides from the waste Prevents or substantially reduces the rate of movement of radionuclides	04-04 Waste Form Moisture Removal and Inerting (applies to CSNF only) 04-07 Waste Package Capacities 04-09 Waste Package and TAD Canister Excluded Materials
		HLW			
EBS	Waste Form and Waste Package Internals—Naval Spent Nuclear Fuel	Naval SNF Structure (includes cladding)	ITWI	Prevents or substantially reduces the release rate of radionuclides from the waste Prevents or substantially reduces the rate of movement of radionuclides	See the Naval Nuclear Propulsion Program Technical Support Document Section 2.3.7.
EBS	Waste Form and Waste Package Internals—DOE Spent Nuclear Fuel	DOE SNF	Non ITWI	None	NA
EBS	Cladding—CSNF/DOE SNF	Cladding—CSNF/DOE SNF	Non ITWI	None	NA
EBS	Waste Package Pallet	Pallet	Non ITWI	None	NA
EBS	Invert	Emplacement Drift Ballast	Non ITWI	None	NA
		Invert Structure			

Table 1.9-8. ITWI Classification of Features that Support the Three Barriers (Continued)

Barrier	Feature[a]	SSC	Safety Classification[b]	Barrier Function	Relevant Design Control Parameter[c]
LNB	Unsaturated zone below the repository	NA	ITWI	Prevents or substantially reduces the rate of movement of radionuclides	01-04 Repository Elevation—Standoff from Water Table
LNB	Saturated zone	NA	ITWI	Prevents or substantially reduces the rate of movement of radionuclides	01-04 Repository Elevation—Standoff from Water Table

NOTE: [a]Some features in this column are further divided into additional groupings signified by text after a dash so that those subparts of that feature as analyzed in Section 6.2 and Appendix Tables A-1, A-2, and A-3 (SNL 2008) could be properly classified as ITWI or non-ITWI.

[b]ITWI classification applies to barriers. The barriers are comprised of features and SSCs that support the function of the barrier. A feature is classified as ITWI if it meets two conditions: (a) the feature is associated with one or more characteristics classified as important to barrier capability); and (b) the feature is a significant contributor to the barrier capability relative to the other features of the barrier. In addition, a feature may be classified as ITWI even if it does not have ITBC control parameters if it is one of the engineered features/components of the geologic repository whose function is to prevent or mitigate the consequences of potential disruptive events (e.g., criticality).

[c]The control parameters are either design configuration controls or procedural safety controls that are relevant to the barrier function of the feature or SSC. The control parameters were selected if they meet two requirements: (a) They are directly associated with the barrier function of the feature or SSC, and (b) They are a relevant control necessary for that feature or SSC to perform its function and contribute significantly to its ITWI status.

[d]There is a specific criterion for naval waste packages that requires an 8.2-ft (2.5-m) minimum emplacement standoff distance from mapped faults with vertical displacements greater than 6.5 ft (2 m) (DOE 2008c, Section 10.3.2.1).

CSNF = commercial SNF; EBS = Engineered Barrier System; ITWI = important to waste isolation (classification applies to barriers; SSCs support barrier function); LNB = Lower Natural Barrier; NA = not applicable; UNB = Upper Natural Barrier.

Source: SNL 2008, Section 7.

Table 1.9-9. Postclosure Analyses Control Parameters

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Subsurface Facilities	01-01 Repository geographic and geologic location (controlled interface parameter)	The location of the subsurface facilities of the repository within the footprint of the emplacement area boundary and the repository host horizon within the lithostratigraphic detail shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	01-02 Repository layout (controlled interface parameter)	The general layout and configuration of the subsurface facilities, including shafts, portals, ramps, mains, emplacement drifts, observation drifts, and other subsurface features; and waste package nominal endpoint coordinates, elevations, and available drift lengths shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	01-03 Repository geologic location (controlled interface parameter)	The repository areas, emplacement area by geologic unit, fault intersection coordinates, and borehole locations shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	01-04 Repository elevation—standoff from water table	The base of the emplacement drifts shall be located at least 120 m above the maximum elevation of the present-day water table. Note: Based on its current location, the maximum elevation of the present-day water table beneath the emplacement area is approximately 850 m above sea level. Thus, the minimum elevation of the base of the emplacement drifts shall be 970 m above sea level.	Yes	2	See Table 1.3.4-5
	01-05 Repository standoff from Quaternary fault	The emplacement drifts shall be located a minimum of 60 m from a Quaternary fault with potential for significant displacement.	No	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Subsurface Facilities (Continued)	01-06 Repository elevation—overburden thickness	The overburden thickness (i.e., the distance from the top of each emplacement drift to the topographic surface) shall be a minimum of 200 m.	Yes	2	See Table 1.3.4-5
	01-07 Repository standoff from perched water	The emplacement drifts shall be located a minimum of 30 m from the top of the Tptpv2 (Topopah Spring Tuff crystal-poor vitric zone), because perched water may occur at the base of the Topopah Spring Tuff unit.	No	2	See Table 1.3.4-5
	01-08 Orientation of emplacement drifts	The emplacement drifts will be nominally parallel. The design azimuth shall be the same for all emplacement drifts and shall be within a range of 70° to 80°.	No	2	See Table 1.3.4-5
	01-09 Excavation methods	The repository ramps, access mains, exhaust mains, and emplacement drifts shall be constructed by tunnel boring machines. The starter tunnel to support each unique tunnel boring machine advance shall be excavated by blasting or mechanical excavation methods.	No	2	See Table 1.3.3-8
	01-10 Emplacement drift configuration	The emplacement drift excavations shall be circular in cross section with a nominal diameter of 5.5 m.	Yes	2	See Table 1.3.4-5
	01-11 Emplacement drift gradient	The grade of the emplacement drifts shall be nominally horizontal so that overall water drainage is directly into the rock to prevent water accumulation.	No	2	See Table 1.3.4-5
	01-12 Nonemplacement opening gradient	The repository nonemplacement opening shall provide a repository grade so overall water drainage and accumulation is away from emplacement areas.	No	2	See Table 1.3.3-8

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Subsurface Facilities (Continued)	01-13 Emplacement drift spacing	The subsurface facility shall be designed to locate the emplacement drifts nominally 81 m apart to prevent thermal interaction between adjacent drifts and to allow drainage of thermally mobilized water within the rock pillars to percolate past the drifts.	No	2	See Table 1.3.4-5
	01-14 Verification of design rock properties	The emplacement openings shall provide for postexcavation investigations of each drift that will be conducted under the Performance Confirmation Program. The objective of postexcavation investigations is to verify that host rock properties are bounded by the rock properties described within the in situ observations and model assumptions used in postclosure analyses. Postexcavation investigations will include geologic mapping to confirm that fracture geometric variability and initial rock properties are within the model input parameter range used in rockfall calculations.	No	2	See Table 1.3.4-5
	01-15 Design of ground support system (controlled interface parameter)	The design and materials used for ground support shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	01-16 Air circulation through ground support	The permanent ground support shall be perforated to allow air circulation between the host rock and the in-drift environment.	No	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Subsurface Facilities (Continued)	01-17 Emplacement drift ground support	(a)The emplacement drift ground support system shall prevent raveling or rockfall in the emplacement drifts that could induce residual tensile stresses in the waste package above 257 MPa. (b) In the event the ground support system fails, the waste packages that come into contact with fallen rock or ground support materials shall be inspected for surface damage and remediated as required prior to closure.	No	(a)2 (b)1	See Table 1.3.4-5
	01-18 Unheated drift length	As boundary conditions for the thermal-hydrologic model in the postclosure, in the event that access main and exhaust main drifts are backfilled, areas at both ends of the emplaced waste will be free of backfill. The two areas will be a minimum of 15 m long and their combined length will total a minimum of 75 m. Note: Emplacement areas will not be backfilled (see parameter 05-04).	No	2	See Table 1.3.4-5
	01-19 Flood protection	The portal and shaft collar locations shall be situated such that they can be protected from water inflow as a result of the probable maximum flood.	No	2	See Table 1.3.3-8
	01-20 Repository standoff from Paintbrush Nonwelded Hydrogeologic Unit	The minimum distance between the top of each emplacement drift and the base of the Paintbrush nonwelded hydrogeologic unit shall be 100 m.	Yes	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Subsurface Facilities (Continued)	01-21 Minimum thickness of the Paintbrush nonwelded hydrogeologic unit above the repository	The minimum thickness of the Paintbrush nonwelded hydrogeologic unit above the repository shall be 10 m.	Yes	2	See Table 1.3.4-5
	01-22 Repository standoff from Calico Hills nonwelded hydrogeologic unit	The minimum distance between the base of each emplacement drift and the top of the Calico Hills nonwelded hydrogeologic unit shall be 60 m.	No	2	See Table 1.3.4-5
Emplacement Drift Configuration	02-01 As-emplaced waste configuration (controlled interface parameter)	The configuration for the emplaced waste packages shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	02-02 As-emplaced waste package–drip shield configuration (controlled interface parameter)	The minimum distance from the top of the waste package to the interior height of the drip shield shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Emplacement Drift Configuration (Continued)	02-03 Committed materials (controlled interface parameter—item (e) only)	During construction of the emplacement drifts, and operation and closure of the repository, administrative controls will be imposed to prevent impact on waste isolation from materials used, lost, or left in the repository. These controls will be supported by technical evaluation. The following constraints will be imposed on the administrative control of tracers, fluids, and materials, construction materials, and committed materials: (a) Material not technically evaluated and determined acceptable prior to the permanent closure of the repository will be removed from subsurface facilities prior to permanent closure. (b) Committed materials that are proposed to remain in the underground repository following permanent closure will be technically evaluated and determined acceptable prior to use. (c) Administrative controls will include accounting and inspection, as appropriate to confirm that controls on the approved tracers, fluids, and material quantities and compositions are met. (d) Concrete dust generation shall be kept to a minimum through the use of surface coatings and/or the use of dust suppression and ventilation control during concrete installation and/or removal. (e) Tracers, fluids, and materials that may be used during construction, operation, or closure shall be controlled through the configuration management system (Section 5).	Yes	(a) and (d) – 2 (b), (c), (d), and (e) – 1	(a), (b), (c), and (e) see Table 1.3.6-3; (d) see Table 1.3.5-4

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Emplacement Drift Configuration (Continued)	02-04 Invert and EBS components in situ stress and thermal response	The invert and EBS components shall be designed to accommodate at least a 10-mm displacement to account for potential in situ stress and thermal response.	No	2	See Table 1.3.4-5
	02-05 EBS in-drift materials interactions	EBS materials shall be inert relative to each other so that physical contact between EBS materials minimizes dissimilar material interaction mechanisms. The waste package outer corrosion barrier shall not contact EBS components other than the Alloy 22 (UNS N06022) support surfaces of the pallet.	No	2	See Table 1.3.4-5
	02-06 EBS material interactions—copper	(a) For the as-emplaced configuration, the drip shields and waste packages shall not contact any copper that may be present in other EBS components such as parts of the emplacement vehicle rail system. (b) The total mass of elemental copper per meter of emplacement drift shall be less than 5.0 kg/m.	No	(a) 1 (b) 2	See Table 1.3.4-5
	02-07 Emplacement drift invert function	The emplacement drift invert (ballast) shall provide a nominally level surface that supports the drip shield, waste package, and waste package emplacement pallet for static loads and that limits degradation associated with ground motion (but excluding faulting displacements) after closure of the repository.	No	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Emplacement Drift Configuration (Continued)	02-08 Invert materials (controlled interface parameter—item (a) only)	(a) The components and materials used in the invert and for the gradation and placement of the invert ballast material shall be controlled through the configuration management system (Section 5). (b) The invert material will be carbon steel and crushed tuff. The crushed tuff shall have properties consistent with the repository host rock excavated by mechanical means.	No	2	See Table 1.3.4-5
	02-09 (Not used)	—	—	—	—
	02-10 Emplacement drift invert configuration (controlled interface parameter)	The general configuration, plan, and details of the emplacement drift invert shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	05-01 Waste package handling and emplacement	Waste package handling and emplacement activities shall be monitored through appropriate equipment. An operator and an independent inspector shall verify proper waste package installation.	No	1	See Table 1.3.4-5
	05-02 Waste package spacing	Adjacent waste packages in a given emplacement drift shall be emplaced 0.1 m (nominal) apart, from the top surface of the upper sleeve of one waste package to the bottom surface of the lower sleeve of the adjacent waste package.	No	1	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Emplacement Drift Configuration (Continued)	05-03 Waste package thermal limits	The waste package emplacement shall be within an envelope such that the emplacement of waste packages does not exceed the other relevant thermal limits of midpillar temperature, drift-wall temperature, waste package temperature, and cladding temperature. In addition, the local-average line-load (over any seven-waste-package segment) in the emplaced repository will not exceed 2.0 kW/m, and no emplaced waste package shall exceed thermal output of 18 kW. Finally, the calculated thermal energy density of any seven adjacent as-emplaced waste packages shall be controlled to ensure that the temperature shall not exceed 96°C at the midpillar, calculated using mean host-rock thermal properties and representative saturation levels for wet and dry conditions. In addition, the thermal loading limits for the naval SNF waste packages are lower than the thermal limits for commercial SNF. These limits are (BSC 2008m, Section 8.2.1.5): • Maximum emplacement thermal load of 11.8 kW for naval waste packages • Naval waste packages shall not be emplaced in a seven-waste-package segment which contains another waste package in excess of 11.8 kW • Maximum emplacement thermal line load limit of 1.45 kW/m for any seven-waste-package segment containing a naval SNF waste package.	Yes	1	See Table 1.3.1-4
	05-04 No backfill in emplacement drifts	Engineered backfill shall not be present in the space between the drip shield and the drift wall.	No	2	See Table 1.3.6-3

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Emplacement Drift Ventilation	06-01 Duration of ventilation period	The duration of the ventilation period shall be a minimum of 50 years after final emplacement.	No	1	See Table 1.3.5-4
	06-02 Drift-wall temperature	The maximum preclosure emplacement drift-wall temperature shall not exceed 200°C to avoid possible adverse conditions (e.g., mineralogical transitions, rock weakening).	No	2	See Table 1.3.5-4
	06-03 Waste package temperature limit	The waste package surface temperature shall be kept below 300°C for the first 500 years after repository closure and below 200°C for the next 9,500 years to eliminate postclosure issues (i.e., phase stability). Note: Compliance with this constraint after repository closure is demonstrated in postclosure analyses only. Parameters 05-03, 06-01, and 06-06 support compliance with this constraint during both the preclosure and postclosure periods.	Yes	2	See Table 1.3.5-4
	06-04 Cladding temperature limit—ventilation	The maximum temperature for the commercial SNF cladding after emplacement shall not exceed 350°C (to prevent damage from creep or hydride reorientation).	No	NA See Table 2.2-3	NA
	06-05 Maximum temperature of HLW glass canisters—ventilation	The maximum HLW glass temperature shall be less than 400°C.	No	2	See Table 1.3.5-4

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Emplacement Drift Ventilation (Continued)	06-06 Average airflow rate for preclosure ventilation of emplacement drifts	During the preclosure phase, the nominal inlet airflow rate per emplacement drift shall be 15 m^3/sec. The range of airflow rate in a given drift shall be 15 m^3/sec ± 2 m^3/sec, based on integrated ventilation efficiency and drift length.	No	2	See Table 1.3.5-4
Drip Shield	07-01 Drip shield design (controlled interface parameter)	The drip shield dimensions and characteristics shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	07-02 Drip shield design and installation	(a) The drip shield shall be designed to interlock and overlap in a manner that prevents a liquid drip path from above the drip shield to the waste package. (b) The drip shield handling and emplacement activities shall be monitored through appropriate equipment. An operator and an independent inspector shall verify proper drip shield installation.	Yes	(a) 2 and (b) 1	See Table 1.3.4-5 and 1.3.6-3
	07-03 (Not used)	—	—	—	—
	07-04 Drip shield materials and thicknesses	The drip shield plates shall be constructed of Titanium Grade 7, with a minimum thickness of 15 mm. The drip shield structural supports shall be manufactured of Titanium Grade 29.	Yes	2	See Table 1.3.4-5
	07-05 (Not used)	—	—	—	—

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Drip Shield (Continued)	07-06 (Not used)	—	—	—	—
	07-07 EBS drip shield/ emplacement drift invert materials interactions	Alloy 22 bases shall be attached to the drip shield to preclude titanium alloy contact with the invert (including transport equipment rails).	Yes	2	See Table 1.3.4-5
	07-08 Drip shield seismic performance (controlled interface parameter)	The drip shield design shall be controlled such that during a seismic event it resists separation through failure of the connector guides, the drip shield connector left/right support beams, and the left/right support beam connectors. Note: Compliance with the postclosure performance aspects of the drip shield within this constraint is demonstrated in postclosure analyses only.	No	2	See Table 1.3.4-5
	07-09 Drip shield fabrication	The drip shield shall be fabricated in accordance with standard nuclear industry practices, including material control, welding, weld flaw detection, and repair and heat treatment.	Yes	2	See Table 1.3.4-5
	07-10 Drip shield fabrication weld inspections	The drip shield full-penetration fabrication welds shall be nondestructively examined by visual, liquid penetrant, and ultrasonic testing for flaws. Fillet welds shall be inspected by means of liquid penetrant and visual testing for flaws. All flaws larger than code standards shall be repaired.	Yes	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Drip Shield (Continued)	07-11 Drip shield fabrication welding flaws	The welding techniques for the fabrication welds shall be constrained to gas metal arc welding, except for short-circuiting mode, and automated gas tungsten arc welding. Welding flaws will be repaired in accordance with written procedures that have been accepted by the design organization prior to their usage.	Yes	2	See Table 1.3.4-5
	07-12 Drip shield fabrication weld materials	(a) Drip shield welding shall be conducted in accordance with standard nuclear industry practices. (b) For Titanium Grade 7 to Titanium Grade 7 welds, Titanium Grade 7 weld filler material shall be used. For Titanium Grade 29 to Titanium Grade 29 welds, Titanium Grade 29 shall be used. For Titanium Grade 7 to Titanium Grade 29 welds, Titanium Grade 28 weld filler shall be used.	Yes	2	See Table 1.3.4-5
	07-13 Drip shield heat treatment	After fabrication, the drip shield assembly and lifting feature assemblies shall be stress-relieved. After completion of required fabrication work except for the final machining, the drip shield assembly and lifting feature assemblies shall be treated for stress-relief. The drip shield assembly and lifting feature assemblies shall be furnace-heated for stress relief at 1,100°F +/-50°F for a minimum of 2 hours. To prevent pickup of hydrogen, a slightly oxidizing atmosphere shall be used; air-cooling is allowed.	Yes	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Drip Shield (Continued)	07-14 Drip shield handling	a) The drip shield shall be handled in accordance with controls to minimize damage, surface contamination, exposure to adverse substances, and impacts. b) Drip shield installation shall be controlled and monitored through appropriate equipment to minimize possible waste package/drip shield damage and/or misinstallation. Installation shall include the use of equipment with an alarm, an operator, and an independent checker.	Yes	1	See Table 1.3.4-5
	07-15 Drip shield thermal expansion constraint	To account for volume increase of corrosion products, the drip shield shall not be constrained laterally or longitudinally, or rigidly mounted to the invert. Drip shield connectors will be designed to allow thermal expansion without binding to 300°C.	No	2	See Table 1.3.4-5
	07-16 As-emplaced waste configuration—waste package/ drip shield clearance (controlled interface parameter)	The minimum distance from the top of the waste package to the interior height of the drip shield shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	07-17 (Not used)	—	—	—	—

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Package	03-01 Waste package dimensions and component masses (controlled interface parameter)	The waste package dimensions and component masses shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.5.2-7
	03-02 Waste package quantities (controlled interface parameter)	The waste packages in the license application design inventory, including quantities, dimensions, materials, and characteristics, shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.5.2-7
	03-03 Waste package outer barrier material and thickness	The waste package outer barrier shall be comprised of Alloy 22 with a minimum thickness of 25 mm for codisposal, naval, and TAD canister waste packages. Note: See parameter 03-19, waste package outer barrier material specifications, for Alloy 22 material composition.	Yes	2	See Table 1.5.2-7
	03-04 Waste package radial gap	The difference between the waste package inner vessel outer diameter and the outer corrosion barrier inner diameter shall be a minimum of 2 mm and a maximum of 10 mm for the as-fabricated package.	No	2	See Table 1.5.2-7
	03-05 Waste package longitudinal gap	The difference between the 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, shall be a minimum of 30 mm.	No	2	See Table 1.5.2-7

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Package (Continued)	03-06 Waste package internal pressurization	The waste package shall be designed to accommodate internal pressurization of the waste package, including effects of a high temperature of 350°C and fuel rod gas release.	No	2	See Table 1.5.2-7
	03-07 Waste package corrosion allowance (information only)	For postclosure mechanical calculations and analysis, a corrosion allowance of at least 2 mm per side shall be accounted for on exposed waste package surfaces. Calculations will be performed using mechanical properties at 150°C or greater.	No	NA See Table 2.2-3	NA
	03-08 Seismic design of waste package	The seismic design spectra, time histories, and ground accelerations for the subsurface facilities shall be controlled.	No	2	See Table 1.5.2-7
	03-09 Waste package worst-case dose rate	The waste package containing the TAD canister with 21-pressurized water reactor fuel assemblies shall represent the worst-case dose rate (80 GWd/MTU burnup, 5% ^{235}U enrichment, and 5 years decay).	No	1	See Table 1.5.1-8
	03-10 Waste package design basis bounding dose rate (controlled interface parameter)	The design basis bounding dose rate calculations for waste packages and representative neutron flux shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.5.2-7
	03-11 Waste package decay heat (controlled interface parameter)	The postclosure design basis waste package decay heat shall be controlled through the configuration management system (Section 5).	No	1	See Table 1.3.1-4

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Package (Continued)	03-12 Waste package fabrication	The waste package outer corrosion barrier cylinder shall be fabricated from no more than 3 sections with longitudinal welds offset. The waste package will be inspected and evaluated per applicable criteria (e.g., parameter 03-18), at the fabricator location and upon receipt at the repository location.	Yes	2	See Table 1.5.2-7
	03-13 Waste package fabrication weld inspections	The waste package outer corrosion barrier fabrication welds shall be nondestructively examined by means of radiographic examination and ultrasonic testing for flaws equal to or greater than 1/16 in. Outer corrosion barrier fabrication welds shall also be examined using liquid penetrant per the applicable specification.	Yes	2	See Table 1.5.2-7
	03-14 Waste package welding materials	The waste package fabrication welds shall be conducted in accordance with standard nuclear industry requirements.	Yes	2	See Table 1.5.2-7
	03-15 Waste package fabrication welding flaws	The welding techniques for the fabrication welds shall be constrained to gas metal arc welding, except for short-circuiting mode, and automated gas tungsten arc welding for Alloy 22 material, limited to less than 45 kJ/in. Welding flaws 1/16 in. and greater will be repaired for the outer corrosion barrier in accordance with written procedures that have been accepted by the design organization prior to their usage.	Yes	2	See Table 1.5.2-7

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Package (Continued)	03-16 Waste package annealing	(a) After fabrication and before inserting the inner vessel, the waste package outer corrosion barrier shall be solution-annealed and quenched. (b) The minimum time for solution annealing will be 20 minutes at 2,050°F (1,121°C) + 50°F (28°C) / –0°F (0°C). (c) The waste package outer corrosion barrier shall be quenched at a rate greater than 275°F (153°C) per minute to below 700°F (371°C). (d) The annealing-induced oxide film shall be removed by means of electrochemical polishing or grit blasting. (e) After solution annealing and quenching, the waste package surface temperature will be kept below 300°C to eliminate postclosure issues (i.e., phase stability), except for short-term exposure (closure-weld).	Yes	2	See Table 1.5.2-7
	03-17 Waste package closure	(a) The Alloy 22 outer lid will be sealed utilizing the gas tungsten arc weld process, limited to less than 45 kJ/in. The weld mass shall be less than 0.104 lb/in. (18.5 g/cm) of weld. (b) The Alloy 22 outer lid weld will be nondestructively examined using visual, eddy current examination/testing, and ultrasonic testing. Flaws greater than 1/16 in. (1.6 mm) shall be repaired. (c) The Alloy 22 outer lid weld will be stress-mitigated using low-plasticity burnishing to induce compressive hoop stresses to a depth of at least 3 mm. (d) Process control to ensure there has been adequate stress mitigation on the welds will be performed. Following the stress mitigation, the final closure weld will be reexamined using visual, eddy current examination/testing, and ultrasonic testing.	Yes	1	See Table 1.5.2-7

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Package (Continued)	03-18 Waste package surface marring prior to emplacement	The waste package shall be certified as suitable for emplacement by process control and/or inspection to ensure surface marring is acceptable per derived internal constraint. The surface marring constraints are (1) the damage to the waste package outer corrosion barrier that displaces material (i.e., scratches) shall be limited to 1/16 in. (1.6 mm) in depth; and (2) modifications to the waste package outer corrosion barrier that deform the surface, but do not remove material (i.e., dents), shall not leave residual tensile stresses greater than 257 MPa.	Yes	1	See Table 1.5.2-7
	03-19 Waste package outer corrosion barrier material specifications	The waste package Alloy 22 material will be manufactured to ASTM B 575-99a specifications with the additional, more restrictive, elemental and chemical composition allowable specifications: (a) chromium = 20.0% to 21.4%, (b) molybdenum = 12.5% to 13.5%, (c) tungsten = 2.5% to 3.0%, and (d) iron = 2.0% to 4.5%.	Yes	2	See Table 1.5.2-7
	03-20 Materials contacting the waste package	After fabrication final cleaning, the waste package shall be prepared for shipment. Materials or objects contacting the waste package outer surfaces during transportation, loading, and emplacement will be evaluated to ensure that any physical degradation and contamination are within allowable limits.	No	1 and 2	See Tables 1.3.3-8 and 1.2.1-2
	03-21 Waste package handling	The waste package shall be handled in a controlled manner during fabrication, handling, transport, storage, emplacement, installation, operation, and closure activities to minimize damage; surface contamination; and exposure to adverse substances.	Yes	1	See Tables 1.3.3-8 and 1.2.1-2

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Package (Continued)	03-22 Waste package handling and emplacement	Waste package handling and emplacement activities shall be monitored through equipment with resolution capable of detecting waste package damage. An operator and an independent checker shall perform the operations.	No	1	See Tables 1.3.3-8 and 1.2.1-2
	03-23 Waste package surface finish	The waste package surface finish shall be specified to be 125 roughness or better as defined in ASME B46.1-2002.	Yes	2	See Table 1.5.2-7
	03-24 Waste package surface damage prior to closure	The emplacement drift ground support system shall be inspected prior to drip shield installation. Waste packages that have come in contact with fallen rock or ground support materials will be inspected to ensure the damage to the waste package outer corrosion barrier that displaces material (i.e., scratches) shall be limited to 1/16 in. (1.6 mm) in depth. Modifications to the waste package outer corrosion barrier that deform the surface, but do not remove material (i.e., dents), shall not leave residual tensile stresses greater than 257 MPa.	Yes	1	See Table 1.3.6-3
	03-25 (Not used)	—	—	—	—
	03-26 Waste package moisture removal and inerting	Waste packages shall be vacuum dried and backfilled with helium in a manner consistent with that described in NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997, Section 8.V.1).	Yes	1	See Table 1.2.1-2

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Form and TAD Canister	04-01 Loading of waste forms	To minimize waste form damage, waste package and TAD canister-loading activities shall be performed and monitored in accordance with industry standard practices, including an operator and an independent checker.	No	1	See Table 1.5.1-8
	04-02 Handling of uncanistered SNF	Uncanistered SNF shall be handled in a standard industry fashion to limit damage and prevent unzipping of fuel rod cladding.	No	1	See Table 1.5.1-8
	04-03 Waste form commercial SNF fuel rod maximum burnup limit	The commercial SNF fuel rod or assembly maximum burnup shall be less than 80 GWd/MTU (this is bounded by the pressurized water reactor burnup).	No	1	See Table 1.5.1-8
	04-04 Waste form moisture removal and inerting	TAD canisters shall be vacuum dried and backfilled with helium in a manner consistent with that described in NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997, Section 8.V.1).	Yes	1	See Table 1.5.1-8
	04-05 Cladding temperature limit—waste form	The maximum temperature for the commercial SNF cladding upon emplacement shall not exceed 350°C (to prevent damage from creep or hydride reorientation).	No	NA See Table 2.2-3	NA
	04-06 Maximum temperature of HLW glass canisters—waste form	The maximum HLW glass temperature shall be less than 400°C.	No	2	See Table 1.2.1-2

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Waste Form and TAD Canister (Continued)	04-07 Waste package capacities	Waste package capacities shall be as follows: (a) TAD-bearing waste package: one commercial SNF TAD canister. (b) Naval waste packages: one naval SNF canister. (c) 2-multicanister overpack/2 high-level radioactive waste package: two multicanister overpacks and two HLW glass canisters (short loading allowed). (d) 5-HLW/DOE SNF codisposal waste packages: Either five HLW glass canisters (including no more than one lanthanide borosilicate glass canister) and one DOE SNF canister in the center position (short loading allowed), or one 24-in. DOE SNF canister and four HLW canisters (center position empty and no lanthanide borosilicate glass canisters) (short loading allowed).	Yes	2 and (d) 1	See Tables 1.5.2-7 and 1.2.1-2
	04-08 Handling of waste forms	Waste form and loaded canister handling operations shall be performed in a standard industry fashion to limit damage. An operator and an independent checker shall perform the operations.	No	1	See Tables 1.5.1-8 and 1.2.1-2
	04-09 Waste package and TAD canister excluded materials	Materials that have not been previously analyzed shall not be placed in the waste package, nor in the TAD canister that will be placed into the waste package.	Yes	1	See Tables 1.5.1-8 and 1.2.1-2

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Pallet	08-01 Emplacement pallet design (controlled interface parameter)	The emplacement pallet dimensions and characteristics shall be controlled through the configuration management system (Section 5).	No	2	See Table 1.3.4-5
	08-02 Emplacement pallet function	For the design static load, the emplacement pallet shall maintain the waste package emplacement nominal position for at least 300 years and shall maintain a nominally horizontal waste package emplacement for 10,000 years.	No	2	See Table 1.3.4-5
	08-03 Emplacement pallet fabrication and corrosion allowance (controlled interface parameter—item (a) only; information only—items (d), (e) and (f) only)	(a) The emplacement pallet material properties shall be controlled through the configuration management system (Section 5). (b) The emplacement pallet shall be fabricated of Alloy 22 plates and square stainless steel tubes. (c) The contacts between the waste package and emplacement pallet shall be Alloy 22. (d) The corrosion allowance for the Alloy 22 components shall be at least 2 mm. (e) The corrosion allowance for the stainless steel components shall be at least 2 mm. (f) The mechanical properties at 150°C or higher shall be used for postclosure analysis.	No	2	See Table 1.3.4-5
	08-04 EBS materials interactions—emplacement pallet function	EBS materials shall be inert relative to each other so that physical contact between EBS materials minimizes dissimilar material interaction mechanisms. The emplacement pallet shall be designed such that, for the nominal scenario (e.g., not seismic or igneous), the waste package outer corrosion barrier shall not contact EBS components other than the Alloy 22 support surfaces of the pallet.	No	2	See Table 1.3.4-5

Table 1.9-9. Postclosure Analyses Control Parameters (Continued)

Feature or Structure, System, or Component	Design Control Parameter	Control Parameter Values, Ranges of Values, or Constraints	Relevant to ITWI Classification (Table 1.9-8)	Approach to Control (1 = Procedural Safety Control 2 = Design Configuration Control)	Design Safety Analysis Report Section Using Control Parameter
Pallet (Continued)	08-05 Waste package and emplacement pallet static stresses	The tensile stresses imposed on the Alloy 22 components of both the waste package and the emplacement pallet shall be less than 257 MPa (the approximate stress corrosion cracking model threshold for Alloy 22).	No	2	See Table 1.3.4-5
Closure	09-01 Closure of shafts and ramps	Closure of the shafts and ramps shall include backfilling for the entire depth of the opening. Closure of ramps shall include backfilling along the entire length of the opening.	No	2	See Table 1.3.6-3
	09-02 (Not used)	—	—	—	—
	09-03 Closure of boreholes	Site investigation boreholes within or near the footprint of the repository block will be backfilled with material compatible with the host rock and plugged.	No	1	See Table 1.3.6-3
	09-04 Reclamation of lands disturbed by repository	Lands disturbed by the repository shall be reclaimed to ensure that there are no preclosure disturbances that will impact postclosure performance.	Yes	1	See Table 1.3.6-3

NOTE: ASME = American Society of Mechanical Engineers; ASTM = American Society for Testing and Materials; DOE = U.S. Department of Energy; EBS = Engineered Barrier System; NA = not applicable; TAD = transportation, aging, and disposal.

Source: *Postclosure Modeling and Analyses Design Parameters* (BSC 2008k).

Table 1.9-10. Preclosure Procedural Safety Controls

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-1	IHF, CRCF, RF, WHF	Cask transfer trolley	The cask transfer trolley is deflated during loading of cask onto trolley, cask preparation activities, and during canister unloading or loading activities.	This control limits the probability of spurious movement of the cask transfer trolley and resulting canister impact.	1.2.3.2.1.4 1.2.4.2.1.4 1.2.5.2.1.4 1.2.6.2.1.4
PSC-2	IHF, CRCF, RF, WHF	Site transporter Site prime mover Cask tractor	The site transporter is turned off during aging overpack bolting and unbolting and canister unloading or loading activities. The site prime mover and cask tractor are disconnected or secured to prevent motion before waste handling operations begin.	This control limits the probability of spurious movement of the site transporter, site prime mover, or cask tractor and resulting collision or tipover.	1.2.8.4.1.4 1.2.8.4.2.4 1.2.8.4.3.4
PSC-3	IHF, CRCF	Waste package transfer trolley	Personnel are verified to be outside of the waste package positioning room and the waste package loadout room prior to movement of a loaded waste package into the waste package positioning room or the waste package loadout room.	This control limits the probability of operators receiving a direct exposure during the loading of a waste package into the TEV.	1.2.3.2.4.4 1.2.4.2.4.4
PSC-4	IHF	Canister transfer machine Naval SNF Canister	Verify that the naval canister lifting adapter is fully detached from the naval SNF canister before using the canister transfer machine to remove the naval canister lifting adapter and shield ring.	Human reliability analysis quantification is based on this PSC being in place. This control protects the canister from a drop by the canister transfer machine during the removal of the naval canister lifting adapter and shield ring.	1.2.3.2.2.4
PSC-5	WHF	TAD canister/DPC shield ring	Prior to commencing operations that rely upon the TAD canister/DPC shield ring, the operating crew is to verify that the shield ring is installed.	This control limits the probability of operators receiving a direct exposure due to miscommunication between the operator and the crew regarding status of the shield ring. The crew that depends on the shield ring for their own safety will ensure its placement.	1.2.5.2.3.4 1.2.5.2.4.4

Table 1.9-10. Preclosure Procedural Safety Controls (Continued)

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-6	WHF	Transportation cask Shielded transfer cask	Whenever a TAD canister or DPC is being moved in a shielded transfer cask or uncanistered SNF is being moved in a transportation cask, the shielded transfer cask or transportation cask will have a lid held in place with a minimum number of installed fasteners such that the stress on the fasteners is less than yield strength for a drop.	This control limits the probability that a drop or tipover of the shielded transfer cask or transportation cask during movement will result in radiological release or criticality.	1.2.5.1.4
PSC-7	CRCF, WHF	Surface nuclear confinement HVAC ITS exhaust subsystem serving ITS confinement areas ITS subsystems serving ITS electrical and battery rooms	One train of HVAC is required to be operating and the second train is required to be in standby before commencing waste handling operations.	HVAC analysis uses this configuration. This control limits the probability that the HVAC system will fail to start when relied upon to mitigate the consequences of an event sequence.	1.2.4.4.4 1.2.5.5.4
PSC-8	EDGF	ITS diesel generators	Before commencing waste handling operations, two ITS diesel generators are aligned to start on detection of undervoltage. Following the start of the diesel generators, the operator manages the operation of the ITS diesel generators to ensure continuous operation of a train of the surface nuclear confinement HVAC, ITS exhaust subsystem serving ITS confinement areas and ITS subsystems serving ITS electrical and battery rooms in each of the waste handling facilities.	The PCSA models that both ITS diesel generators start and run 720 hours to support the operation of the surface nuclear confinement HVAC system.	1.4.1.2.4

Table 1.9-10. Preclosure Procedural Safety Controls (Continued)

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-9	WHF	Spent fuel pool	With SNF in the pool, the concentration of soluble boron in the WHF pool and transportation cask/DPC fill water is maintained at a minimum of 2,500 mg/l, with the soluble boron enriched to a minimum of 90 wt % in the ^{10}B isotope.	This control provides the appropriate initial conditions in the WHF pool to ensure that a critical configuration cannot be created in the pool. For wet operations, the minimum required concentration of 2,500 mg/L of soluble boron (enriched to 90 wt % ^{10}B) in the WHF pool is sufficient to compensate for the complete omission of fixed neutron absorbers in the analyzed designs.	1.2.5.1.4
PSC-10	IHF, CRCF, RF, WHF, Subsurface, Intra-site	ITS SSCs	The amount of time that a waste form spends in each process area or in a given process operation, including total residence time in a facility, is periodically compared against the average exposure times used in the PCSA. Additionally, component failures per demand and component failures per time period are compared against the PCSA. Significant deviations will be analyzed for risk significance.	PCSA uses residence times and reliability data to calculate the probability of an initiating event. This control ensures that the average exposure times and reliability data are maintained consistent with those analyzed in the PCSA.	Table 1.2.1-3 1.3.3.5.2
PSC-11	IHF, CRCF, RF, WHF	Cask cranes	When transferring casks, the crane will remain connected to the cask until the proper seismic restraints are established.	The cask transfer trolley has built-in seismic restraints that prevent seismic interactions between the trolley frame and a cask. When so restrained, the cask is prevented from tipping by the cask transfer trolley design. During cask transfer, however, the crane must provide seismic stability until the cask transfer trolley seismic restraints are engaged.	1.2.3.2.1.4 1.2.4.2.1.4 1.2.5.2.1.4 1.2.6.2.1.4
PSC-12	IHF, CRCF, RF, WHF	Cask preparation platform	Transportation cask lid bolts are independently verified to have been removed prior to moving the cask from the cask preparation area to the unloading room or pool.	This control prevents the canister transfer machine from attempting to remove the cask lid with bolts still in place resulting in failure of the bolts and possible drop of the lid or cask.	1.2.3.2.1.4 1.2.4.2.1.4 1.2.5.2.1.4 1.2.6.2.1.4

Table 1.9-10. Preclosure Procedural Safety Controls (Continued)

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-13	IHF, CRCF, RF, WHF	Canister transfer machine Port slide gates	At completion of a canister transfer operation, the port slide gates are verified to be closed.	While the canister transfer machine is being used to perform transfer operations, the Operational Radiation Protection Program provides the necessary controls to ensure that workers are not present with the slide gates open. This control limits the probability of workers receiving a direct exposure by entering the transfer room with the canister transfer machine away from a port with a waste form present and the slide gate open.	1.2.3.2.2.4 1.2.4.2.2.4 1.2.5.2.5.4 1.2.6.2.2.4
PSC-14	IHF, CRCF, RF, WHF	Canister transfer machine	Prior to lifting or lowering a DPC, TAD canister, or naval canister, the canister transfer machine guide sleeve is to be verified to have been lowered.	This control limits the probability that a DPC, TAD canister, or naval canister is not in a vertical orientation during transfer such that any potential drops would be flat bottom drops.	1.2.3.2.2.4 1.2.4.2.2.4 1.2.5.2.5.4 1.2.6.2.2.4
PSC-15	IHF, CRCF, RF, WHF, Subsurface, Intra-site	Structure	Flights by fixed-wing aircraft in Nevada Test Site or Nevada Test and Training Range airspace within 4.9 nautical mi (5.6 statute mi) of the North Portal and below 14,000 ft mean sea level are prohibited.	External event screening applied the results of the aircraft crash analysis, which assumes a flight restricted airspace around the North Portal.	5.8.3
PSC-16	IHF, CRCF, RF, WHF, Subsurface, Intra-site	Structure	The number of overflights by fixed-wing aircraft at altitudes greater than 14,000 ft mean sea level within the flight-restricted airspace (i.e., within 4.9 nautical mi (5.6 statute mi) of the North Portal) is limited to 1,000 per year, and the overflights are limited to straight and level flights (i.e., maneuvering is not permitted).	External event screening applied the results of the aircraft crash analysis, which assumes these operational controls are in place.	5.8.3
PSC-17	IHF, CRCF, RF, WHF, Subsurface, Intra-site	Structure	Carrying ordnance or engaging in electronic jamming activities over the flight-restricted airspace (i.e., within 4.9 nautical mi (5.6 statute mi) of the North Portal) is prohibited.	External event screening applied the results of the aircraft crash analysis, which assumes these operational controls are in place.	5.8.3

Table 1.9-10. Preclosure Procedural Safety Controls (Continued)

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-18	IHF, CRCF, RF, WHF, Subsurface, Intra-site	Structure	Helicopter flights within 0.5 mi of surface facilities that process, stage, or age nuclear waste forms are prohibited.	External event screening applied the results of the aircraft crash analysis, which assumes that this flight restriction is in place.	5.8.3
PSC-19	Aging facility	TAD canisters and DPCs	The surface contamination on TAD canisters and DPCs sent to the Aging Facility is less than 1.0×10^{-4} µCi/cm^2 for beta-gamma emitters and low-toxicity alpha emitters and 1.0×10^{-5} µCi/cm^2 for all other alpha emitters.	This control ensures that the dose consequences from airborne releases of contamination from the canisters on the aging pads are within the calculated values presented in Tables 1.8-25, 1.8-28, 1.8-29, 1.8-32, and 1.8-36.	Table 1.2.1-3
PSC-20	CRCF, RF, WHF, Subsurface, Intra-site	Commercial SNF	Characteristics of commercial SNF received at the repository are verified to be within the following parameters: • The maximum burnup for commercial SNF is limited to 80 GWd/MTU for PWRs and 75 GWd/MTU for BWRs. • The maximum initial enrichment for commercial SNF is limited to 5% ^{235}U. • The minimum decay time of commercial SNF prior to shipment to the repository is 5 years.	This control ensures that the dose consequences from Category 2 event sequences involving these waste forms are within the values presented in Tables 1.8-30 and 1.8-31.	1.5.1.1.1.2.4
PSC-21	IHF, CRCF, Subsurface, Intra-site	HLW	The individual radionuclide inventories per HLW canister are limited to the values presented in the Section 1.8 consequence analysis.	Table 1.8-5 provides the radionuclide inventories. This control ensures that the dose consequences from Category 2 event sequences involving HLW are within the values presented in Tables 1.8-30 and 1.8-31.	1.5.1.2.1.4
PSC-22	WHF	WHF pool	The height of water above the top of the active portions of commercial SNF assemblies in the WHF pool staging rack(s) and open TAD canisters, DPCs, and casks is maintained at or greater than 23 ft.	This control ensures that the pool leak path factors presented in Table 1.8-9 are maintained. Additionally, the water level is credited for preservation of shielding for workers.	1.2.5.1.4

Table 1.9-10. Preclosure Procedural Safety Controls (Continued)

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-23	LLWF		Dose rate measurements and associated conversions are performed to confirm that the following conditions are maintained in the LLWF: • Total radionuclide inventory on WHF pool resins and pool filters is at or below 2.3×10^3 Ci. • Total radionuclide inventory on the WHF stage 1 ITS HEPA filters is at or below 6,600 Ci. • Radionuclide concentration in the low-level liquid waste tanks is limited to dose equivalents of 1×10^{-3} Ci/m^3 of ^{60}Co and 1.5×10^{-3} Ci/m^3 of ^{137}Cs.	This control ensures that the dose consequences from Category 2 event sequences involving these waste forms are within the values presented in Tables 1.8-30 and 1.8-31. Tables 1.8-6 and 1.8-7 provide the bases for the numerical values.	Table 1.2.1-3
PSC-24	IHF, CRCF, RF, WHF, Subsurface, Intra-site	Cranes and handling equipment	When not in use, cranes, mobile platforms, and handling equipment are maintained in a location such that they cannot fall on a waste form.	The seismic analysis credits the exposure time of components over waste forms. This control ensures that the exposure time is limited to the time necessary to complete the waste handling operations.	Table 1.2.1-3
PSC-25	Subsurface	Waste package, emplacement drift	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 the preclosure period, or a standoff distance from the fault is to be established.	This control limits the potential for fault displacement (or related rockfall hazard) from a seismic event to induce a breach of the waste package at rest in an emplacement drift during the preclosure period.	1.3.4.8.2.5
PSC-26	CRCF	Cask preparation room equipment confinement doors	The cask preparation room equipment confinement doors are to be closed when conducting operations with a potential for a drop involving a loaded cask.	This control ensures that the confinement boundary is intact when there is a potential for an event sequence that could result in radiological releases.	1.2.4.2.1.4

Table 1.9-10. Preclosure Procedural Safety Controls (Continued)

Item	Facility/ Operations Area	SSC	Procedural Safety Controls	Basis	LA Section Describing Implementation
PSC-27	WHF	Cask preparation area equipment confinement door	The cask preparation area equipment confinement door is to be closed when waste handling operations are being conducted with a potential for a drop or collision involving a loaded cask or canister outside the WHF pool.	This control ensures that the confinement boundary is intact when there is a potential for an event sequence that could result in radiological releases outside the WHF pool.	1.2.5.2.1.4

NOTE: BWR = boiling water reactor; DPC = dual-purpose canister; EDGF = Emergency Diesel Generator Facility; LLWF = Low-Level Waste Facility; PSC = procedural safety control; PWR = pressurized water reactor; TAD = transportation, aging, and disposal.

Source: BSC 2008i, Table 1.

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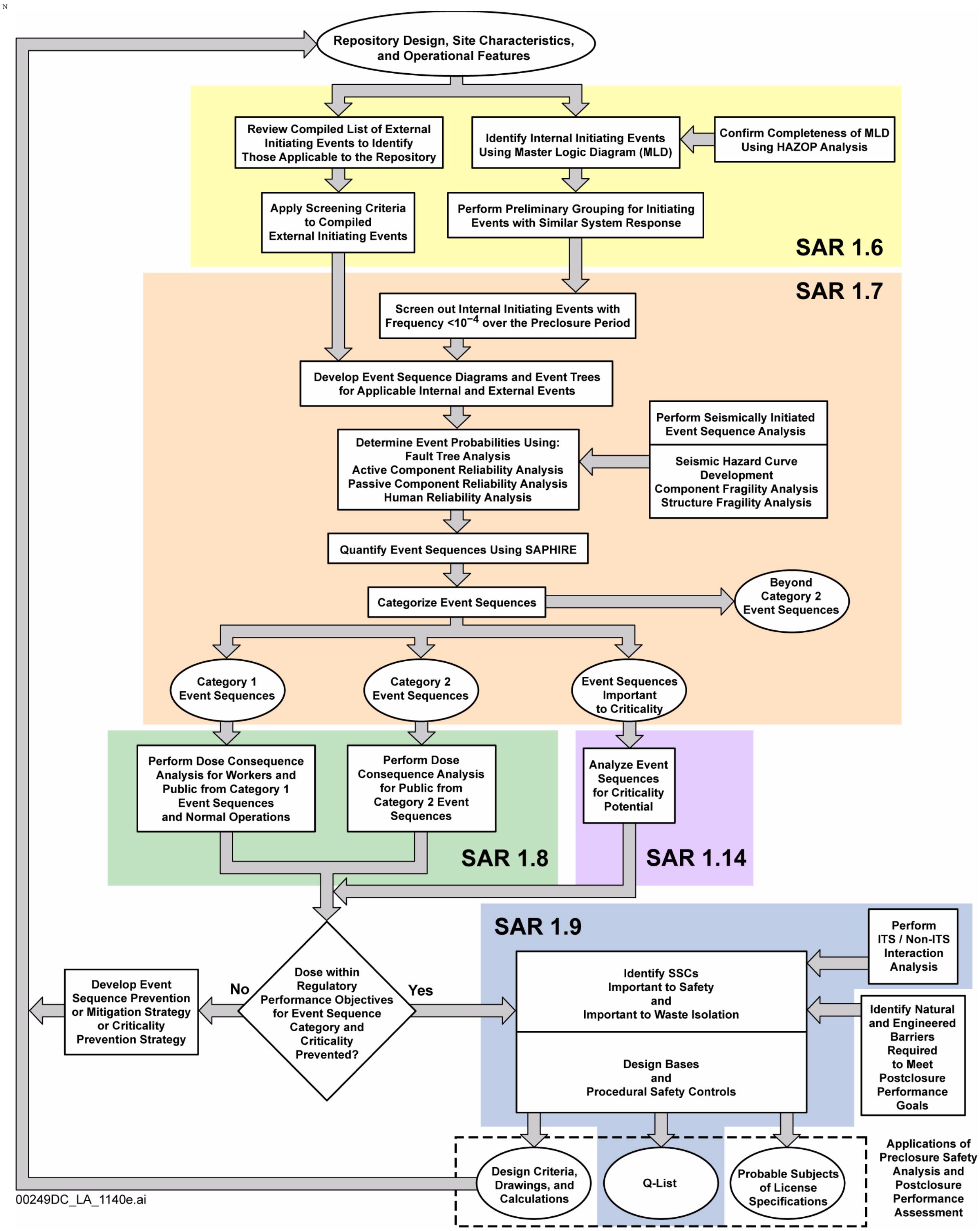

Figure 1.9-1. Preclosure Safety Analysis Process

NOTE: HAZOP = hazard and operability; MLD = master logic diagram

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CONTENTS

Page

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TABLES

TABLES (Continued)

FIGURES

Page

FIGURES (Continued)

1.10 MEETING THE AS LOW AS IS REASONABLY ACHIEVABLE REQUIREMENTS FOR NORMAL OPERATIONS AND CATEGORY 1 EVENT SEQUENCES

The information presented in this section addresses the implementation of as low as is reasonably achievable (ALARA) principles and control of personnel exposures in the design, construction, and operations of the Yucca Mountain repository. The following table lists the information provided in this section, the corresponding regulatory requirements, and the applicable acceptance criteria from NUREG-1804. The information provided in this section also addresses ALARA-related recommendations contained in HLWRS-ISG-03 (NRC 2007), including its revision to Acceptance Criterion 4 for NUREG-1804, Section 2.1.1.8.3.

SAR Section	Information Category	10 CFR Reference	NUREG-1804 Reference (and Changes to NUREG-1804 from HLWRS ISGs)
1.10.1	Management Commitment to Maintain Doses As Low As Is Reasonably Achievable	Part 20 63.111(a)(1)	Section 2.1.1.8.3: Acceptance Criterion 1
1.10.2	As Low As Is Reasonably Achievable Principles in Design	Part 20 63.21(c)(8) 63.111(a)(1) 63.112(e)(2) 63.112(e)(5)	Section 2.1.1.5.1.3: Acceptance Criterion 3(4) Section 2.1.1.6.3: Acceptance Criterion 1(2)(b) Acceptance Criterion 1(2)(e) Section 2.1.1.8.3: Acceptance Criterion 2
1.10.3	Surface and Subsurface Shielding Design	Part 20 63.111(a)(1) 63.112(e)(3)	Section 2.1.1.5.1.3: Acceptance Criterion 2(4) Acceptance Criterion 2(5) Section 2.1.1.5.2.3: Acceptance Criterion 2(5) Section 2.1.1.6.3: Acceptance Criterion 1(2)(c) Section 2.1.1.7.3.1 Acceptance Criterion 1(6) Section 2.1.1.7.3.3(I): Acceptance Criterion 1(1) Section 2.1.1.7.3.3(III): Acceptance Criterion 1(7)
1.10.4	As Low As Is Reasonably Achievable Principles in Operations	Part 20 63.111(a)(1)	Section 2.1.1.8.3: Acceptance Criterion 3 HLWRS-ISG-03 Section 2.1.1.8.3: Acceptance Criterion 4

The U.S. Nuclear Regulatory Commission (NRC) regulations on protection against radiation require licensees to apply a radiation protection program based on sound radiation protection principles that, to the extent practicable, reduces exposures as a result of operation. The objective of the ALARA program is to keep doses to repository workers and the public not only within regulatory limits, but also ALARA.

In developing an overall repository radiation protection program, the repository draws on experience in nuclear facility design and operations, including historical nuclear power plant experience and from the U.S. Department of Energy (DOE) Corporate Operating Experience Program (Section 5.6). ALARA principles are incorporated into the facility design in accordance with Regulatory Guide 8.8. ALARA principles will also be incorporated into operations, maintenance, decommissioning, and dismantling activities.

The overall repository radiation protection program is implemented through procedures and engineering process controls so that activities related to the radiological aspects of design, construction, operation, maintenance, decontamination, dismantlement of surface nuclear facilities, and closure of the repository are conducted in a manner that results in worker and public doses that are consistent with ALARA principles (Ghanooni and Carl 2002, Section 2).

Section 1.10.1 describes the DOE management commitment to ALARA principles. Section 1.10.2 discusses how ALARA principles are incorporated into the design. Section 1.10.3 gives an overview of facility shielding design. Section 1.10.4 discusses the inclusion of ALARA principles in operations, including future design changes. The Operational Radiation Protection Program is described in Section 5.11.

1.10.1 Management Commitment to Maintain Doses As Low As Is Reasonably Achievable

[NUREG-1804, Section 2.1.1.8.3: AC 1]

It is the policy of the DOE (DOE 2004) that occupational doses and doses to members of the public are ALARA in accordance with 10 CFR 20.1101(b). The most critical aspect of successfully achieving ALARA in the radiation protection program is the commitment of management for occupational and public doses to be ALARA. Administrative programs and procedures, in conjunction with facility design, will implement the DOE commitment to ALARA. In achieving this commitment, the DOE will ensure that radiation doses and releases of radioactivity to the environment are maintained below applicable regulatory limits, including 10 CFR 20.1101(d).

This commitment to control occupational doses, public doses, and radioactive releases holds line management responsible for adhering to this policy and operating philosophy. Senior and line management demonstrate their support of ALARA through direct communication, instruction, inspection, and audit of the workplace. Personnel are made aware of management's commitment and instructed on their individual responsibilities to ensure compliance.

Personnel are instructed in the DOE commitment to implement ALARA, what ALARA means, why ALARA is important, and how to implement ALARA on their jobs.

Major elements of the DOE commitment are:

- The DOE has established an ALARA program including requirements, procedures, goals, and expectations. Audits verify implementation through reviews of implementing procedures and records, occupational exposure and dose trends, and ALARA program goals performance.

- The DOE is implementing ALARA principles in design and construction, and will implement ALARA principles in operations.

- Personnel are trained to reduce occupational exposures and perform activities so that worker and public doses are ALARA.

- The DOE will implement a robust and comprehensive Operational Radiation Protection Program with well-defined roles and responsibilities.

- The DOE will ensure that during operations personnel receive training in accordance with 10 CFR 19.12, which provides requirements for the instruction of personnel on radiation protection.

1.10.1.1 Design and Construction

DOE management commitment to ALARA during design and construction is demonstrated by the early development of an ALARA program (Ghanooni and Carl 2002), along with the development of ALARA-related guidance documents, coupled with personnel training. ALARA-specific reviews are conducted to ensure that ALARA principles are incorporated in the design and addressed as the design progresses. ALARA reviews are performed by engineers and radiation protection personnel with experience in nuclear facility design and operations. These reviews are conducted and documented in accordance with the recommendations of Regulatory Guide 8.8. A description of the ALARA reviews is provided in Sections 1.10.2 and 1.10.4. The ALARA reviews confirm that the facility and equipment design features recommended in Regulatory Guide 8.8 are considered at the design stage. In addition, estimates of occupational doses are performed in accordance with the recommendations of Regulatory Guide 8.19. Design features are considered for potential exposure, and changes are recommended to reduce doses.

The engineering design process ensures that structures, systems, and components (SSCs) expected to contain radiation sources are shielded with sources located, and personnel access appropriately controlled, to maintain doses ALARA.

Construction inspections ensure that shield walls and other shielding features are placed or installed as designed. Prior to repository operation, inspections will verify that shielding features can perform their intended functions.

Design features are implemented to ensure that doses are ALARA for maintenance during operations as well as during decommissioning and dismantlement activities at the end of operations. Design features prevent or reduce radioactive contamination of facilities, areas, and SSCs.

1.10.1.2 Operation

DOE management commitment to ALARA principles for operations will be demonstrated by developing and implementing an operational ALARA program at the repository in accordance with Regulatory Guide 8.8, Position C.1, that:

- Includes ALARA principles in operating policies and procedures, including considerations of phased facility development.
- Includes ALARA principles in the review, approval, and implementation of design changes through appropriate operating policies and procedures.
- Provides for qualified personnel and sufficient resources to implement ALARA principles.
- Provides appropriate and sufficient training on ALARA principles and radiation safety principles and requirements. Training will communicate an expectation that minimizing radiation exposures, and doses consistent with ALARA principles, is each individual's responsibility.
- Provides the radiation protection staff and their management with authority to ensure safe operations, including stop work authority.
- Supports continuous improvement in implementing ALARA principles.
- Incorporates radiation safety lessons learned.
- Provides an environment where workers are encouraged to provide dose reduction suggestions.
- Provides for oversight and periodic assessment of implementation of the program, consistent with ALARA principles.
- Provides for assigned personnel reporting to the Radiation Protection Manager with ALARA implementation responsibilities.
- Monitors performance against ALARA goals.

To achieve the goal of minimizing occupational and public doses during operations, the program will implement ALARA principles in policies, goals, and objectives for planning, design, and construction of modifications to operating facilities, operating activities, maintenance, housekeeping, decontamination, and dismantlement (Ghanooni and Carl 2002).

1.10.1.3 Decommissioning

ALARA principles will be applied during the decommissioning and dismantlement of the repository surface and subsurface nuclear facilities (Ghanooni and Carl 2002). Decommissioning activities (Section 1.12) will be reviewed by engineers and radiation protection specialists with experience in ALARA reviews. Records of prior radiation surveys will be reviewed to assist in the determination of radiological conditions. Visual inspections and radiation surveys will be performed to ensure that there are no unidentified radiation sources that might affect personnel exposures. Decommissioning and dismantlement activities will be proceduralized and will follow ALARA principles. Incorporation of design features discussed in Section 1.10.2 below will result in facilities for which contamination has been minimized and equipment that can be mothballed or removed without unnecessary personnel exposure.

1.10.2 As Low As Is Reasonably Achievable Principles in Design

[NUREG-1804, Section 2.1.1.5.1.3: AC 3(4); Section 2.1.1.6.3: AC 1(2)(b) and (e); Section 2.1.1.8.3: AC 2]

This section discusses the methods and features by which the management commitment to ALARA principles of Section 1.10.1 are applied in the design. Design features for maintaining occupational doses ALARA are presented. It is important to design facilities using ALARA principles from the start because their use is most efficient and effective at this point in the design process.

ALARA design requirements are established and implemented through formal project design criteria and ALARA-related supporting documents, which are reinforced through ALARA training. Reviews of the facility and equipment design are performed to ensure that ALARA practices are implemented. ALARA design reviews form the basis for a consistent, systematic implementation of ALARA concepts. Design alterations are made when opportunities are identified to further reduce doses, as practicable, consistent with design criteria and supported, if necessary, by cost–benefit analyses (Section 1.10.2.1.3.4).

The ALARA principles, as applied to design, can be expressed as:

- Design of SSCs to ensure increased reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components

- Design of SSCs to reduce radiation and contamination levels to ensure that operations and maintenance, including inspection activities, can be performed in lower radiation environments

- Design of SSCs to reduce the time spent in radiation environments during operations and maintenance

- Design of SSCs to accommodate remote and semi-remote operations and maintenance.

Guidance to implement ALARA principles is found in the hierarchy of project documents from an ALARA plan to desktop information. The guidance is structured to provide consistent application of ALARA principles throughout the design process. ALARA reviews are implemented during the design process and consider potential radiation exposure, as well as the potential for contamination, from normal operations and any Category 1 event sequences that are identified. This approach enables design attention to be focused on higher potential dose activities where greater reductions in occupational and public doses can be achieved.

Implementation of ALARA principles begins early in the design and continues as an iterative process through detailed design for construction. While the design process assures that operations and maintenance will be conducted within regulatory dose limits, the application of ALARA principles in the design process further assures that both individual and collective annual doses for occupational workers are reduced during operations and maintenance.

The ALARA design process is applied throughout the repository design and construction. Even though the ALARA process involves a routine consideration of alternatives, experience has shown that the greatest potential for significant dose savings at the lowest cost is achieved at the earliest stages of design. Therefore, emphasis in the ALARA process is to identify dose-reducing considerations early in the design cycle. ALARA design criteria are established and applied throughout the design of a facility. The following are ALARA guidelines for the design process:

- The design process has the greatest potential to lower the total dose by implementing ALARA principles. ALARA principles are applied to both collective dose and to individual doses.
- Design alternatives that provide reasonably lower doses (i.e., decreased source term and/or exposure time) or lower dose rates are identified, evaluated, and implemented as appropriate.
- Radiation dose scenarios are evaluated during the design process and alternatives are considered.

Dose reduction alternatives that may be considered include the following:

- Reduce manual operations in radiological work areas
- Increase the reliability of processes and equipment
- Increase shielding
- Design systems and components to reduce crevices and areas that could become physical traps for radioactive material
- Select materials of construction that reduce capture and retention of radioactive material
- Improve access and egress to work areas, within the restricted area, that have a potential for significant radiation exposures.

More than one alternative applied to a dose situation may provide equivalent ALARA benefit. In these cases, operational experience from existing facilities is taken into account whenever it is reasonable to do so. Application of this experience contributes to the estimation of dose and also may indicate areas where dose reduction consistent with ALARA principles has been achieved previously. The creation of an additional hazard does not necessarily eliminate selection of an alternative under consideration. Risk from the resulting hazard could be mitigated to the point of no consequence. However, whenever a significant risk from competing alternatives exists, the final decision is based on minimizing overall risk (Ghanooni and Carl 2002).

As discussed in Section 1.10.2.1.3.5, an important aspect of the DOE ALARA program is verification of effectiveness. This verification is achieved through design process audits and surveillances and ALARA self-assessments.

Another important aspect of the ALARA process is to provide controls to confirm that ALARA principles are properly implemented, including review of procurement documents to ensure that ALARA-related specifications established in the design process are flowed down to vendors and fabricators.

1.10.2.1 General Considerations

Implementation of ALARA principles in the design process is in accordance with Regulatory Guide 8.8, Position C2, and the recommendations of HLWRS-ISG-03 (NRC 2007). The following ALARA design considerations eliminate or reduce potential doses. Priority is given to those features that are most effective. Prevention is preferable to mitigation, and passive features are preferable to active.

1.10.2.1.1 Design Objectives

ALARA objectives considered as part of the repository design process include:

- Minimizing the number of individual radiation workers that have the potential of receiving a total effective dose equivalent of more than 500 mrem/yr.
- Minimizing radiation levels in routinely occupied areas
- Minimizing the time workers must stay in radiological areas
- Minimizing worker dose through remote operations
- Placing and handling of equipment and shielding by remote handling
- Minimizing the potential for accumulation of radioactive materials or surface contaminants
- Locating radioactive material handling, receipt, and holding facilities away from the geologic repository operations area (GROA) boundary
- Establishing specific transient source movement corridors
- Establishing access control barriers especially to high and very high radiation areas.

1.10.2.1.2 Design Considerations

The following ALARA design considerations are applied in the repository design:

- The primary methods used to minimize radiation exposures are physical design features (e.g., confinement, ventilation, shielding, remote handling, and the transportation, aging, and disposal (TAD) canister approach for commercial spent nuclear fuel (SNF)).

- Confinement and ventilation, including filtration, designed to control the spread and release of airborne radioactive material.

- Air handling systems employ the cascade principle with the direction of airflow from areas of no or low contamination to areas of higher potential contamination.

- The facility design does not require routine work activities in areas requiring respiratory protection.

- The facility layout provides for anticipated equipment maintenance and operation needs. Specific provisions include direct access and removal paths, sufficient space for ease of operation, designing equipment for ease of repair and maintenance, and utilizing remote maintenance features, where appropriate.

- Systems containing or handling radioactive materials provide suitable isolation provisions to prevent diffusion, backflow, or other methods of leakage to other areas, especially areas that are not normally contaminated.

1.10.2.1.3 Implementation

Implementation of ALARA goals and the ALARA program in design is accomplished through a combination of procedural requirements, personnel training, ALARA-specific design reviews, cost–benefit analysis of alternatives, and verification of program effectiveness, as discussed below.

1.10.2.1.3.1 Policy and Procedures

ALARA principles are implemented through an ALARA-specific desktop instruction and engineering procedures. The design process is structured to provide consistent application of ALARA principles throughout the design.

1.10.2.1.3.2 Training

Engineering and design personnel, based on their job functions, are required to attend formal training classes on the application of ALARA design fundamentals. The training focuses on applying ALARA design concepts including:

- Generic and discipline-specific applications of the fundamental approaches to reduce radiation doses, releases, and spread of radioactive material contamination
- The process for implementing ALARA principles during design
- Typical applications of ALARA design and performance of cost–benefit analysis.

The training also provides information on design, operational or procedural changes, improvements, and lessons learned.

1.10.2.1.3.3 Design Reviews

Designs are reviewed for incorporation of ALARA principles such that worker doses that are incurred during maintenance, inspections, routine operations, processing radioactive wastes, decontamination, and decommissioning are ALARA. The ALARA design review process ensures that ALARA criteria and considerations are applied and documented as the design develops. ALARA design reviews may be performed either concurrently with other engineering design reviews or as a separate activity, as appropriate.

Early ALARA reviews focus on general arrangement, major component design, process design and operations, traffic patterns, bulk shielding, personnel access, radiation zoning, ventilation and confinement, and contamination control. Design reviews conducted prior to construction will focus on detailed SSC design, detailed bulk and penetration shielding design, pipe routing, detailed contamination and airborne radioactivity control, and operations.

The reviews examine the design to verify implementation of ALARA principles and to record key ALARA decisions. The ALARA reviews may also record dose reduction achieved by the use of good engineering practices (e.g., where a design feature or modification selection has been made to improve the design and there is a resulting significant dose savings). Documentation records the ALARA decisions made in each design phase.

Multidisciplinary teams composed of personnel with radiological safety, operations, and engineering backgrounds, and others as appropriate, review and evaluate the incorporation of ALARA design objectives and ALARA design considerations.

1.10.2.1.3.4 Cost–Benefit Analysis

A cost–benefit analysis may be employed in the selection of processes, design of facilities, setting of operating parameters, and development or revision of procedures. When an activity is being evaluated a base case is established. This base case is the set of radiation protection features that are designed to permit operation. This base case is used as the basis for comparison of the cost-effectiveness of alternatives. However, some of the alternatives may not be practical design candidates due to factors that may be judged to be undesirable or unacceptable.

1.10.2.1.3.5 Self-Assessments

An important aspect of the DOE ALARA program is verification of effectiveness. This verification is achieved through design process audits and surveillances and ALARA process self-assessments. ALARA process self-assessments examine and verify implementation of ALARA policies, procedures, and detailed design criteria. Self-assessments are also conducted to evaluate and verify the effectiveness of ALARA training to ensure both that appropriate personnel are being trained and that the training program content is appropriate.

1.10.2.2 Facility Layout Considerations

This section describes the facility layout design features utilized for maintaining personnel doses ALARA. These features are employed in conjunction with the equipment ALARA features described in Section 1.10.2.3 and include the features discussed in the following paragraphs.

1.10.2.2.1 Design Objectives

ALARA objectives considered for facilities layout include:

- Segregating SNF and high-level radioactive waste (HLW) buffer, staging, aging, and transfer areas from areas normally occupied by personnel to maximize the effects of shielding and increasing distances to personnel

- Locating support personnel away from radiation sources associated with waste receipt, surface transportation, buffer, aging, low-level radioactive waste collection and storage, and staging operations

- Locating SNF and HLW handling facilities and movement corridors away from locations accessible to members of the public

- Shielding SNF and HLW during surface transportation (e.g., in transportation casks, aging overpacks, shielded transfer casks, and transport and emplacement vehicles (TEV))

- Designing the subsurface emplacement drifts with turnouts to reduce worker dose during operations

- Incorporating remote operations in waste handling and emplacement processes

- Minimize the size and number of contamination and radiation areas

- Locating the Aging Facility away from the other surface facilities and their supporting administrative areas.

1.10.2.2.2 Design Considerations

In order to maintain doses ALARA, the following design considerations are applied to the extent practical during facility layout:

- Personnel access, routing of piping, and location of components minimize personnel radiation exposure during operations and maintenance including access control and traffic patterns.
- Access to a lower radiation area does not require personnel to pass through a higher radiation area.
- Personnel escape routes are provided to allow personnel to rapidly exit an area in which the radiation exposure level has unexpectedly increased.
- Radioactive components are located to minimize radioactive piping runs.
- Low-level radioactive wastes are collected in shielded rooms and enclosures separated from normally accessible areas.
- Sample stations are isolated, to the extent practical, from other radioactive equipment, and exposed sample piping is minimized.
- Permanent shielding is provided for radiation sources.
- Support facilities and administrative offices are located away from radiation sources.

1.10.2.2.3 Facility ALARA Features

Facility layout features are directed toward reducing radiation levels in access areas and in the vicinity of equipment. These ALARA design features support operations, as well as maintenance and inspection activities. Facility layout features include, as appropriate:

- Locating equipment, instruments, and sampling stations for ease of access and minimum occupancy time
- Establishing dedicated laydown areas or service bays for equipment maintenance and inspection
- Providing for movement of equipment or components requiring service to lower radiation areas
- Separating radiation sources from occupied areas (e.g., pipes or ducts containing potentially radioactive fluids are not routed through occupied areas)
- Locating redundant components in separate compartments to allow maintenance of one component while the other component is in operation

- Separating radioactive equipment, such as Wet Handling Facility (WHF) demineralizers and filters, by shielding from nonradioactive equipment
- Providing access around filters to accommodate filter changeout
- Providing labyrinth entrances
- Providing space to allow prestaging of necessary items, such as tools, instruments, or shielded containers
- Providing features to control contamination and to facilitate decontamination of contaminated areas, such as floor drains and curbs.

In addition, the following considerations are given to the spent resin handling system:

- Ball, plug, or diaphragm valves are used, depending upon the function. Items such as strainers, check valves, or Y-pattern valves are not utilized.
- Orifices are not utilized.
- Butt welds are employed, where practical, regardless of size. Large radius elbows, or large diameter bends, are also used.
- Ninety-degree tees are not used except to introduce clean services, such as service air or water, into such lines. Dead legs are avoided and flushing connections are taken off above the horizontal centerline of the resin piping.
- Lines are sized to achieve turbulent flow to minimize resin deposits and subsequent buildup.
- Provisions are made for ion exchangers, as well as for spent resin lines, to be pressurized with service air or water to clear plugged lines. Service air or water is introduced at a tee where the leg of the tee is above the resin line to avoid clogging of the clean service inlet line.

Additional layout considerations directed toward reducing radiation levels for common design areas include:

- **Valve Operation**—Valves located in highly radioactive system piping that require periodic use are provided with remote operators or reach rods to minimize operator exposure.
- **Piping**—Pipes carrying radioactive materials are not field routed. Equipment compartments are used as pipeways only for those pipes associated with equipment in the compartment.

Radioactive and nonradioactive piping are separated to minimize personnel exposure. Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated.

- **Penetrations**—To minimize radiation streaming, penetrations are typically designed with an offset between the source and accessible areas, are located as far as possible above the floor elevation to reduce exposure to personnel, or have additional shielding.

- **Contamination Control**—Access control and traffic patterns are considered in a facility layout to minimize the spread of contamination. Decontamination of potentially contaminated areas and equipment is facilitated by the application of decontaminable paints and suitable smooth-surface coatings to the concrete floors and walls. Floor drains with properly sloping floors are provided in potentially contaminated areas of the facilities, except where air pallets are used to move heavy loads.

- **Equipment Layout**—In those systems where process equipment is a major radiation source, such as the WHF pool water treatment system, pumps, valves, and instruments are separated from other process components.

- **Sample Stations**—Sample stations for routine sampling of casks and dual-purpose canisters (DPCs) are shown on equipment location and general arrangement drawings, to ensure that proper shielding and ventilation are provided to maintain radiation zoning in areas and minimize personnel exposure during sampling.

- **Clean Services**—Whenever possible, clean services and equipment such as compressed air piping, clean water piping, nonradioactive ventilation ducts, and cable trays are not routed through higher radiation or contamination areas.

1.10.2.3 Equipment Design Considerations

This section describes equipment design features utilized for maintaining personnel doses ALARA. These features are employed in conjunction with the facility layout ALARA features described in Section 1.10.2.2.

1.10.2.3.1 Design Objectives

ALARA objectives considered for equipment design include:

- Reliability, durability, construction, and materials to reduce or eliminate the need for repair or preventive maintenance

- Incorporation of integrated shielding, or other features, that reduce personnel exposure

- Servicing features for maintenance or repair, including ease of disassembly

- Incorporation of features that minimize the potential for generation or spread of contamination during maintenance and repair, including disassembly

- Provisions, where practical, to remotely operate, repair, service, monitor, replace, or inspect equipment

- Redundancy of equipment or components to reduce the need for immediate repair.

1.10.2.3.2 Design Considerations

Equipment design considerations directed toward reducing radiation and contamination levels in proximity to equipment or components requiring personnel attention include:

- Naval SNF and TAD canisters incorporate integral top shielding to facilitate waste package closure operations while reducing radiation exposure.

- Waste package designs for HLW and DOE SNF canisters (which do not contain integral shielding) have top shielding added to facilitate waste package closure operations while reducing radiation exposure.

- Filters, demineralizers, vacuums, and skimmers are provided for the WHF pool water to reduce pool radionuclide concentrations.

- Heat exchangers are designed with corrosion-resistant tubes.

- Heat exchanges are designed with differential pressures across tubes to prevent the spread of radioactive materials into clean systems.

- Pumps in radioactive systems are purchased with seals designed to reduce servicing time. Additionally, smaller pumps are provided with flanged connections for ease of removal. Pump casings are provided with drain connections for maintenance.

- Filters are designed to allow cartridge replacement with remote or robotic tools where appropriate.

- Ion exchangers are designed to facilitate removal of spent resins via remotely operated sluicing systems.

- Leakage of radioactive material is minimized by use of appropriate valve gaskets and valve packing or bellows-sealed valves.

- Radiation-tolerant materials are used in equipment based on their radioactive service.

- Fuel handling machine is designed such that components not intended for submergence do not contact pool water.

1.10.2.3.3 Features for Maintenance and Inspection

Equipment features directed toward reducing radiation and contamination levels proximate to equipment or components to facilitate maintenance and inspection include:

- Provisions for draining, flushing, or, if necessary, remote cleaning of equipment containing radioactive material
- Design of equipment, piping, connections, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps
- Valves designed to minimize leakage of radioactive materials
- Provisions for minimizing the spread of contamination into equipment service areas, including direct drain connections
- Provisions for isolating equipment from radioactive process fluids
- Incorporation of appropriate access for tools and instruments, especially where test plugs or ports are part of a pressure boundary
- Incorporation of features (e.g., root valves or quick disconnects) to facilitate process access while minimizing personnel stay times and potential leakage of radioactive materials.

1.10.2.3.4 Features of Shield Doors, Slide Gates, and Viewing Windows

This section discusses the features (other than passive structural elements that function as shielding) of equipment specifically provided for personnel radiation protection. This equipment includes shield doors, slide gates, and shielded viewing windows.

Shield doors and shielded viewing windows are designed to reduce dose rates to similar levels as the reduction provided by the shield walls in which they are installed. Unauthorized access to high and very high radiation areas is prevented by interlocks or other positive controls. Remotely controlled and interlocked shielding features (shield doors and slide gates discussed in Section 1.2.4) are provided where the failure, or the inadvertent movement of the shielding feature, has the potential to cause significant worker exposure.

Area radiation instruments monitor dose rates in normally occupied areas and provide local and remote, audible and visible alarms if a shield door, or other feature, is inadvertently opened when high radiation dose rates are present on the other side of the door or other feature. The operability of these doors or other features, their interlocks, and the radiation monitors will be verified during initial startup testing and will be periodically verified during operations to ensure proper function.

1.10.2.4 Access Control Considerations

Access to the restricted area is controlled. Section 1.10.4.1.5 provides a definition of the restricted area.

Access controls to high and very high radiation areas are provided in accordance with 10 CFR 20.1601 and 20.1602 and with Regulatory Guide 8.38. To prevent inadvertent entry by personnel into high and very high radiation areas, access control is maintained, including locked or barricaded doors, interlocks, and a system of local and remote alarms. However, access control barriers (e.g., locked gates or doors) can be opened from the inside to allow personnel egress.

Section 5.11 provides information on access controls for radiological areas. During operations, personnel access controls may be revised based on radiation surveys. Facility layout incorporates restrictions on, and control of, access to radiation and contamination areas. During an off-normal occurrence, access to radiological areas may be temporarily affected and appropriate access controls would be implemented to minimize personnel exposure.

1.10.2.5 Radiation Zones

Five radiation zone categories and their descriptions are described in Table 1.10-1. Radiation zones for normal operations are presented in Figures 1.10-1 to 1.10-17. The radiation zones provide dose rates used to identify the need for design features, such as bulk shielding, local shielding, and access control barriers, to ensure that doses are ALARA.

The dose rates used in establishing the radiation zones for repository facilities are estimated by evaluating expected sources of radiation within the area being evaluated and considering contributions from sources outside the specific area being evaluated.

1.10.2.6 Contamination Control

In accordance with Regulatory Guide 8.8, Position C.2, the potential for the spread of contamination is controlled. Contamination could be encountered during, or result from, routine waste handling operations. Routine waste handling activities are conducted in areas designed specifically to contain potential contamination such as dedicated crane maintenance areas and dedicated laydown areas for equipment maintenance and repair.

Potential sources of contamination are based on the characteristics of the waste material handled at a facility along with the design of systems and equipment and the potential for the release of radioactive material. Other potential sources of contamination could result from inspection or rework or repair of nonconforming items containing SNF or HLW. Rework and repair activities are performed at locations prepared to facilitate mitigation of potential radiological hazards. Most waste handling activities involve sealed canisters that are not expected to present a contamination potential. Areas where contamination is expected during normal operations are designed for ease of maintenance, contamination control, and decontamination operations. Specific design requirements for surface finishes or surface treatment for potential contamination areas minimize contamination buildup and enhance removal. Such design features also support subsequent facility dismantlement by reducing the amount and extent of contaminated equipment. Examples include:

polished stainless steel surfaces, epoxy coatings for concrete floors and walls, and painted metal surfaces.

1.10.2.7 Ventilation Considerations

The surface and subsurface heating, ventilation, and air-conditioning (HVAC) systems are designed to provide a suitable environment for personnel and equipment during normal operation. Certain portions of the surface nuclear HVAC systems perform a confinement function to control contamination in potentially contaminated portions of surface nuclear facilities. The subsurface ventilation system directs the flow from the access mains through the emplacement drifts and out the ventilation exhaust.

Design of important-to-safety (ITS) and non-ITS surface ventilation systems is discussed in Section 1.2.2.3. Waste handling facilities are described in Sections 1.2.3 through 1.2.6. Design of high-efficiency particulate air (HEPA) filtration units is in accordance with applicable portions of Regulatory Guide 1.52 and the *Nuclear Air Cleaning Handbook* (DOE 2003) with regard to the design, inspection, and testing of air filtration.

1.10.2.7.1 Design Objectives

Confinement portions of HVAC systems are designed to ensure that personnel doses are ALARA by reducing exposure to airborne radioactivity in waste handling areas. ALARA goals for workers are also facilitated by incorporating component design features that allow maintenance and testing of HEPA filters to proceed with minimal exposure to personnel.

1.10.2.7.2 Design Considerations

ALARA principles considered for HVAC system design include:

- The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.
- In surface facilities areas with a potential for contamination, the exhaust is designed for greater volumetric flow than is directly supplied to that area. This design feature minimizes the amount of uncontrolled exfiltration from the area.
- Consideration is given to the possible disruption of normal airflow patterns by maintenance and provisions are made in the design to prevent adverse airflow.
- Surface facility air discharged from potentially contaminated areas in waste handling facilities is passed through HEPA filters to remove particulates.

1.10.2.7.3 HVAC Equipment

HVAC maintenance and inspection activities are also governed by ALARA principles and can represent a substantial source for worker exposure if not addressed during design. Ventilation system components in potentially contaminated areas are designed and located to minimize operator exposure during maintenance, inspection, and testing. Provisions specifically included to minimize personnel exposures and to facilitate maintenance or in-place testing operations are as follows:

- Ventilation equipment rooms for outside air supply units and building exhaust system components, including exhaust filtration units, of waste handling facilities are located in lower radiation areas to allow operator accessibility.

- Work space is provided around each filter exhaust unit for maintenance, testing, and inspection. For example: (1) the clear space for bag-in and bag-out replacement of HEPA filters is a minimum of 4 ft; and (2) the service clearance in front of an access door in an exhaust HEPA filter plenum is 4 ft.

- Exhaust ductwork serving confinement areas is of all-welded stainless steel construction to minimize the buildup of radioactive contamination within the ducts.

- Only air from clean areas is recirculated without HEPA filtration. Recirculated air from potentially contaminated areas is filtered through HEPA filters prior to redistribution. Air from potentially contaminated confinement zones is exhausted only after passing through HEPA filtration to remove airborne contaminants.

- HEPA filters are monitored for radioactivity on a regular basis. Filter elements are replaced before the radioactivity level exceeds administrative limits.

1.10.2.8 Radiation and Airborne Radioactivity Monitoring Instrumentation

In accordance with the guidance of Regulatory Guide 8.8, Position C.2.g, an area radiation monitoring system and an airborne radioactivity monitoring system are provided in appropriate areas of the surface and subsurface facilities.

The radiation/radiological monitoring system consists of area radiation monitors, continuous air monitors, and airborne radioactivity effluent monitors that provide both local display and alarms. Additional information on the design of the radiation/radiological monitoring system can be found in Section 1.4.2.2. Sampling capabilities for HVAC flows, process liquids, and the atmosphere contained inside a transportation cask or DPC are also provided.

1.10.2.8.1 Design Objectives

Design objectives for the radiation/radiological monitoring system in support of ALARA are:

- To warn of uncontrolled or inadvertent movement of radioactive material

- To provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where substantial change in radiation levels might be of immediate importance to personnel frequenting the area

- To annunciate and warn of possible equipment malfunctions involving liquid or airborne releases
- To furnish information for radiation surveys.

By meeting the above objectives, the radiation/radiological monitoring system aids radiation protection staff in maintaining doses ALARA.

1.10.2.9 Event Sequence Considerations

As shown in Section 1.7, there are no Category 1 event sequences. However, design features and facility layout assist in reducing worker doses during recovery from event sequences.

The design provides ITS filtered ventilation for waste handling areas potentially subject to event sequences. The control of airflow in waste handling facilities combined with HEPA filtration of exhausts reduce worker exposures inside the affected facility, as well as reduce releases that could impact other onsite personnel and offsite members of the public. Other features include bulk shielding, local shielding, access control, barriers, laydown areas, etc. that would also reduce worker doses during recovery operations.

1.10.2.10 Decommissioning

Areas that contain radioactive piping or equipment, or where contamination could occur during normal operations, incorporate design features to facilitate maintenance, contamination control, and decontamination operations. These design features include specific design requirements for surface finishes or surface treatment for potential contamination areas to minimize contamination buildup. Such design features also support subsequent facility dismantlement by reducing the amount and extent of contaminated facility areas and equipment.

Operational Radiation Protection Program limits on contamination in equipment areas and components also serve to limit radiation doses during decommissioning.

1.10.2.11 Dose Assessment Considerations

Dose estimates for radiation workers are developed in accordance with Regulatory Guide 8.19. The design process assesses the potential for worker exposure when performing necessary actions and incorporates design features that will reduce personnel exposures. The dose assessment process identifies actions or operations which could potentially lead to significant levels of exposure and estimates the frequency and anticipated doses for these actions. Dose estimates combine direct radiation exposures, including airborne exposures, to establish annual doses for individual workers and annual collective doses for work groups.

Radiation doses come from direct radiation from components and equipment containing radioactive materials and from the presence of airborne radionuclides. The methodology to determine radiation doses due to direct radiation and airborne radioactivity at locations, onsite and offsite, is discussed in Section 1.8.

1.10.2.11.1 Radiation Workers

Annual collective doses from direct radiation during the performance of routine functions, such as waste handling activities, inspection, and maintenance, have been estimated for each nuclear handling facility based on the following:

- Expected occupancy times for various worker activities
- Anticipated number of workers in a workgroup.

To demonstrate regulatory compliance, an initial conservative estimate of worker doses was developed based on minimizing workgroup staffing levels while maximizing both source terms and individual handling facility annual throughputs. Neither staff rotation between tasks nor between facilities was considered in these analyses as a means to reduce individual doses. These initial dose estimates confirm that operation of GROA facilities will result in occupational exposures less than the 10 CFR 20.1201 regulatory limit of 5 rem/yr, in compliance with 10 CFR 63.111 (BSC 2008).

Nominal worker doses were also estimated, using more realistic assumptions. This estimate establishes that an annual average worker dose less than 500 mrem/yr is achievable for waste handling operations (BSC 2008). Continued reduction in worker doses will be accomplished through refinements in design, as well as through application of operational ALARA considerations in handling activities.

The ALARA design objective is to pursue, through an iterative process, continuous reduction in estimated individual and collective worker doses. Worker dose estimates presently incorporate a large degree of conservatism. Several factors that will further reduce estimated worker doses are:

- **Source Term Refinement**—10 CFR Part 961 gives preference to an oldest first approach for commercial SNF. However, nuclear utilities will likely ship a combination of older and newer SNF. Because of uncertainties in the commercial SNF waste stream, simplifying assumptions are made that every cask contains a maximum loading of SNF assemblies at the design basis source term.

- **Design Refinement**—Because of uncertainties in the final configuration of facilities and equipment, including TAD canisters, simplifying assumptions are made for equipment and facility layout. These assumptions will be revised to realistically reflect actual equipment configurations, hardware tooling, and facility layout.

- **Analysis Refinement**—Because of uncertainties in the final configuration of facilities and equipment, simplifying assumptions are made in the physical modeling of radiation sources and worker exposure pathways. These assumptions will be revised to more realistically reflect expected source terms, shielding design, and layout.

- **Task Refinement**—Estimates of worker doses are based on a set of worker activities and assumed unit doses to workers. Initial estimates assume that the maximum dose for an activity applies to each worker in a workgroup. As the design progresses, individual worker doses and annual collective doses will be reduced due to a more realistic representation of operations and worker activities.

- **Operational Considerations**—The Operational Radiation Protection Program will take measures to control and further reduce worker exposures through a process of worker dose assessment, workgroup sizing, and rotation of functional tasking within workgroups, as well as rotation of workgroup members from higher exposure activities to lower exposure activities.

The present focus is to ensure that ALARA principles are incorporated into the design, including equipment specification and procurement. During operations, repository management's commitment to ALARA will result in further reduction of worker doses through continued application of experienced-based improvements in handling operations as part of a continuous improvement program.

1.10.2.11.2 Construction Workers

The repository is constructed in four phases as described in GI Section 2.1.

Phased construction includes a restricted area boundary used to physically separate SNF and HLW handling operations from ongoing construction activities in the GROA. The emplacement drifts will be developed as needed, rather than all drifts being completed prior to the start of operations. This development sequence means that surface and subsurface construction activities continue after the repository is operational.

The construction of the surface facilities has been planned to allow the GROA to be developed in phases to include facilities as they are completed and turned over to operations (Figure 1.1-3 and Section 1.2.1.5). The restricted area is identified by barriers and postings that expand as waste handling facilities are placed into operation.

In addition to the restricted area boundary, other physical barriers and procedural safety controls will be used, as appropriate, to prevent construction activities from adversely affecting surface facility operations and to ensure that construction worker doses are maintained ALARA during this process.

Radiation protection, security, and isolation barriers are erected between the operational portion of the subsurface facility and those portions under construction to protect the operating facility from construction-initiated hazards and protect construction workers from the hazards of the emplacement drifts (Section 1.3.1). Isolation barriers are used to separate the ventilation flow that is directed to the operating emplacement drifts from the ventilation that is provided to the development areas.

These separated areas have individual and unique ventilation systems. A pressure differential is maintained between the two systems. The emplacement area has a negative air pressure relative to the atmosphere, while the development area has a positive pressure. The pressure differential ensures that airflow leakage between the systems is into the emplacement areas.

1.10.2.11.3 Public

Members of the public (individuals who are not occupationally exposed) given access to the DOE-controlled area, outside of the restricted area, may receive direct radiation exposures from the operation of the repository. The principal fixed sources of radioactivity outside the surface waste handling facilities are the various transportation casks in the buffer areas that are shielded to meet transportation limits (10 CFR Part 71) and the aging overpacks that are shielded to meet repository aging limits. SNF and HLW canisters and waste packages inside the surface waste handling facilities are surrounded by concrete walls that allow routine occupancy of the repository open area.

DOE ensures that members of the public inside the preclosure controlled area boundary receive no more than 100-mrem/yr dose (10 CFR Part 20). Figure 1.1-3 illustrates the phased development and associated changes of the protected area, the restricted area, and the surface GROA. Figure 1.1-2 shows the boundaries of the surface GROA at maximum extent of the restricted area. The surface GROA includes the restricted area. Potential public doses from normal operations and Category 1 event sequences inside the preclosure controlled area are provided in Table 1.8-28. Potential offsite public doses from normal operations and Category 1 event sequences are provided in Table 1.8-29.

1.10.3 Surface and Subsurface Shielding Design

[NUREG-1804, Section 2.1.1.5.1.3: AC 2(4), (5); Section 2.1.1.5.2.3: AC 2(5); Section 2.1.1.6.3: AC 1(2)(c); Section 2.1.1.7.3.1: AC 1(6); Section 2.1.1.7.3.3(I): AC 1(1); Section 2.1.1.7.3.3(III): AC 1(7)]

Facility shielding at the repository is designed to reduce dose rates from radiation sources, including waste packages, casks, canisters, overpacks, and uncanistered SNF, to levels consistent with the radiation zoning characteristics presented in Table 1.10-1. Specific area or item dose rate criteria used in the evaluation of shielding are presented in Table 1.10-2.

Facility shielding includes concrete walls, floors, and ceilings; shielded viewing windows; shield doors; slide gates in concrete floors; canister transfer machines; waste package trolleys; and penetration designs to allow items, such as piping, HVAC ducts, and electrical raceway, to pass through walls or floors. Design attributes for these shielding features are addressed in Sections 1.2.3 to 1.2.8. This section provides the results of shielding evaluations performed to ensure the capability of existing designs to meet shielding criteria. The specific evaluations performed are not intended to provide a design solution, but only to ensure that an adequate space envelope is identified along with structural loads for shielding features. In some cases, structural concrete is supplemented by the addition of plate steel, or other shielding material, to meet specific shielding dose rate criteria. Design of concrete for radiation shielding is in accordance with ACI-349-01/349R-01 and ANSI/ANS-6.4-2006.

1.10.3.1 Shielding Design Objectives

The objective of radiation shielding is to reduce worker dose, in conjunction with a program of controlled personnel access to, and occupancy of, restricted areas, to levels that are ALARA within the dose standards of 10 CFR Part 20. Shielding and equipment layout and design are evaluated for normal operations, maintenance and inspection activities utilizing the design recommendations in accordance with Regulatory Guide 8.8, Position C.2. Specific shielding design objectives are:

- Ensure that radiation doses to workers, contractors, administrators, visitors, and members of the public are ALARA and within the limits of 10 CFR Part 20
- Ensure worker access and occupancy times allow normal anticipated waste handling, maintenance, and inspection operations
- Minimize the possibility of radiation damage to components not intended for (higher) radiation fields.

1.10.3.1.1 Shielding Design Considerations

The bases for shielding configurations are discussed in this section. Neutron and gamma shielding design considerations are as follows:

- Shielding is provided to reduce dose rates to levels consistent with expected occupancy during normal operation.
- Shielding is provided to reduce dose rates from external sources to levels less than or comparable to dose rates resulting from sources within a compartment.
- Shielding is provided to minimize radiation effects on components consistent with their materials of construction.
- Shielding is based on bounding source terms applicable to each location and operation.

Shielding is provided to attenuate direct and scattered radiation to levels consistent with the radiation zones established for each area. Locations of equipment and facility areas discussed in this section are shown in the general arrangement drawings presented in Sections 1.2.3 to 1.2.8 for each handling facility and the Low-Level Waste Facility (LLWF). Shielding features are in accordance with the recommendations of Regulatory Guide 8.8.

The material used for shielding evaluation is ordinary Type 04 concrete with a bulk density of 2.35 gm/cm^3 (approximately 146.6 lb/ft^3) based on *Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants,* ANSI/ANS-6.4-2006, Table 5.2. Water is used as the primary shield material for radioactive sources in the WHF pool. Materials used in the design of the waste handling facilities, the LLWF, and in other shielding features, such as aging overpacks, are addressed in Sections 1.2.3 to 1.2.8.

Shielding analyses use bounding source terms to evaluate the adequacy of facility designs to meet radiation zoning requirements and specific shielding design criteria. For a waste handling area involving a variety of casks and canisters, the radial, top, and bottom dose rates are considered for applicable casks and/or canisters. Bounding dose rates for shielding purposes may result from one or more casks or canisters depending on their design, waste form, and shielding characteristics. Shielding evaluations also consider areas where normal operations could upend or downend a cask or canister, and whether this operation affects the bounding source terms. Note that concrete walls may be thicker than required for shielding purposes due to structural design requirements for seismic forces or other loads.

The following activities are considered in shielding design:

- Waste receipt and staging
- Normal waste handling
- Aging
- Waste emplacement (including activities in the subsurface environment)
- Maintenance and inspection
- Repair and rework.

1.10.3.2 Calculation Methodology and Computer Codes

Shielding ensures compliance with radiation zoning and specific shielding design criteria to provide ALARA personnel exposure based on bounding source activities. The design of shield walls, floors, and other shielding features surrounding radioactive equipment is determined by approximating the geometry and physical condition of sources. Isotopic concentrations are converted to energy group sources which are then totaled to establish a source intensity. Flux-to-dose-rate conversion factors used in the shielding analyses are in accordance with pages 4 to 5 of *American National Standard for Neutron and Gamma-Ray Flux-to-Dose-Rate Factors,* ANSI/ANS-6.1.1-1977. The use of ANSI/ANS-6.1.1-1977 flux-to-dose-rate conversion factors provides conservative dose rates compared to those calculated using ICRP-74 (ICRP 1997) flux-to-dose-rate conversion factors, based on ICRP-60 (ICRP 1991) organ weighting factors.

Shielding requirements for an area are established based on the point of maximum or peak radiation dose. Therefore, the overall radiation level in an area is less than this maximum dose point and consequently less than the upper limit specified for the radiation zone classification of an area.

Shielding analyses use industry accepted methods and codes appropriate for radiation types, sources, facility geometry, and materials at the repository. Analytical tools include codes that use Monte Carlo, deterministic transport, and point-kernel integration techniques. Simple and scoping-type gamma shielding problems are handled with point-kernel integration codes. Complex or deep-penetration shielding problems use MCNP or deterministic transport codes, especially for problems involving neutron and secondary gamma dose contributions.

The following commonly accepted shielding codes are used to support the repository design:

- **MCNP (Monte Carlo N-Particle) Transport Code**—This is a general purpose Monte Carlo code for neutron, photon, or coupled neutron-photon transport problems that is suitable for complex three-dimensional geometry and a variety of radiation source types. MCNP is widely used in the nuclear industry for various radiation transport, scattering, and absorption applications. Shielding evaluations utilized MCNP V. 4B2LV and MCNP5 V. 1.40.

- **SCALE (Standardized Computer Analysis for Licensing Evaluation)**—The SCALE system was developed to provide a standardized method of analysis for the evaluation of a nuclear fuel facility. SCALE is a modular system with three shielding modules (SAS1, SAS4, and QADS) and a depletion module (ORIGEN-S). SCALE contains modules for performing both source term calculations and shielding calculations. Shielding evaluations utilized SCALE V. 4.3and SCALE V. 4.4A.

Shielding design codes and standards have been applied to the design of surface and subsurface facilities to ensure appropriate protection of workers and the public. Shielding source terms for the surface and subsurface facilities design are based on bounding source terms as discussed in Section 1.10.3.4.

1.10.3.3 Radiation Sources

The following sections describe the radiation sources encountered at the repository. These radiation sources establish permanent shielding requirements for facilities, as well as the dose rate at a distance from contained sources, such as transportation casks, aging overpacks, or waste packages in open areas. Figure 1.10-18 summarizes the radiation sources received and handled at the repository. A detailed discussion of SNF and HLW is in Section 1.5.1. A detailed discussion of the various waste packages used for emplacement is in Section 1.5.2.

Pressurized water reactor (PWR) SNF contains the highest inventory of radioactive material compared to other waste forms (Section 1.5.1) and produces the highest unshielded dose rates. PWR SNF, therefore, typically requires the most shielding to reduce dose rates consistent with personnel exposures allowed in radiation zones. However, shielding source terms for surface and subsurface facility design are based on bounding source terms handled in each area, including evaluation of bounding waste forms, their physical configuration and containment, and radionuclide inventory. The repository receives, handles, processes, packages, and emplaces a variety of waste forms, including commercial PWR and boiling water reactor (BWR) SNF, naval SNF, DOE SNF, and HLW, any of which could be the bounding source term for an area.

Radiation sources take different forms at the repository. In the simplest form, a radiation source would be the individual nuclides that, via a natural process of decay, result in the production of radiation (i.e., beta, alpha, neutron, and photon particles). Uncharged particles (i.e., photons and neutrons) travel substantially farther through shielding material than charged particles and tend to establish shielding requirements.

Radiation sources include the containers (or packaging) of SNF and HLW received and handled at the repository. SNF and HLW containers include:

- Transportation casks (truck and railcar)
- Waste canisters (commercial SNF (e.g., TAD canisters and DPCs), naval SNF, DOE SNF, and HLW)

- Aging overpacks and shielded transfer casks
- Waste packages.

Waste forms received at the repository include:

- Commercial SNF
- Naval SNF, DOE SNF, and HLW.

Low-level wastes generated during operation include:

- HVAC filters
- Filter cartridges resulting from the cleanup of WHF pool water
- Ion exchange resins resulting from the cleanup of WHF pool water
- Dry active waste, including contaminated personnel protective equipment and other materials.

1.10.3.3.1 Spent Nuclear Fuel and High-Level Radioactive Waste Containers

SNF and HLW are received in transportation casks regulated and certified by the NRC. After receipt, these wastes are placed inside a waste package for permanent disposal. The movement of packaged waste in both surface and subsurface environments is evaluated to determine shielding requirements.

1.10.3.3.1.1 Transportation Casks

1.10.3.3.1.1.1 Rail

Evaluation of NRC-certified transportation cask systems is performed as part of the shielding evaluation. Shielding evaluations modeled a representative transportation rail cask containing 21 PWR SNF assemblies. Figures 1.10-19 and 1.10-20, respectively, display the radial and axial configurations used for evaluation.

1.10.3.3.1.1.2 Truck

Transportation casks for commercial SNF must meet the same NRC external dose limits whether they are truck or rail transported casks. However, since truck transport casks are smaller (typically below 10 commercial BWR and even fewer PWR SNF assemblies), the shielding evaluation used a representative rail cask, containing 21 PWR SNF assemblies, as the bounding source for evaluation.

1.10.3.3.1.2 Transportation, Aging, and Disposal Canister

A TAD canister (Figure 1.5.1-5) may contain either 21 PWR or 44 BWR spent fuel assemblies. For shielding evaluations, PWR fuel bounds BWR fuel. Each PWR fuel assembly (Figure 1.5.1-1) is modeled as containing four axial regions: top nozzle, gas plenum, active fuel and bottom nozzle. The height of the top nozzle is 4.0 in., the height of the gas plenum is 6.5 in., the height of the active fuel region is 144.0 in., and the height of the bottom nozzle is 5.0 in. The fuel region is homogenized in the MCNP models. The effective radius of the homogenized fuel region is approximately 28 in. Region materials and radiation sources of each assembly region are homogenized inside a volume defined by the SNF basket radius and region height inside a TAD canister.

The effect of source homogenization is twofold: (1) it decreases the mass density of the source region, reducing radiation attenuation; and (2) it places source points slightly closer to detector locations which results in a conservatively higher radiation field. The active fuel region is homogenized with fresh, unirradiated fuel (assumed for the material composition but not for the source term). The modeling of unirradiated fuel increases the fission to absorption coefficient in the fuel region which thus increases the conservatism of the shielding calculations. The composition of the homogenized fuel does not include basket materials and, therefore, does not take credit for any attenuation that these materials may provide.

The TAD canister also contains an 11-in. thick shield plug to reduce the axial top surface dose rate of the TAD canister to 1 rem/hr. The TAD canister is modeled as a cylinder 66.5 in. in diameter and 212 in. long. The assembly regions are homogenized within the 66.5 in. limit.

1.10.3.3.1.3 Dual-Purpose Canister

Evaluation of NRC-certified DPC systems determined that a representative transportation cask containing commercial SNF is assumed to bound DPCs capable of shipment to the repository. Therefore, shielding evaluations modeled a DPC as using the same materials and dimensions of a representative transportation cask.

1.10.3.3.1.4 Naval SNF Canister

The naval SNF canister (Figures 1.5.1-29 and 1.5.1-30) is modeled as a right circular cylinder with a central void. The naval SNF canister inside a transportation cask (Table 1.10-3) is shown in Figures 1.10-21 and 1.10-22. The Naval Long waste package dimensions and material selections are shown on the waste package configuration drawings (BSC 2007a; BSC 2007b; BSC 2007c).

1.10.3.3.1.5 DOE Spent Nuclear Fuel Canister

The standardized SNF canister (Figure 1.5.1-9) used to evaluate shielding in the surface facilities contains a homogenized TRIGA-FLIP fuel composition. The DOE standardized SNF short canister is a right circular cylinder. The physical dimensions and material specifications of the DOE SNF canister are summarized in Table 1.10-4. TRIGA-FLIP fuel and baskets are modeled in MCNP as homogenized inside the cavity of the DOE SNF canister. This approach is conservative because it decreases the fuel self shielding and places the radiation source closer to the outer surfaces of the canister increasing the dose rate outside the canister.

1.10.3.3.1.6 High-Level Radioactive Waste Canister

Standardized HLW canisters (Figure 1.5.1-8) consist of a steel canister containing HLW glass. Savannah River Site canisters are placed inside a 5-DHLW/DOE Short Codisposal waste package, while Hanford canisters are placed inside a 5-DHLW/DOE Long Codisposal waste package.

Savannah River Site HLW is modeled as a right circular cylinder that fills a canister interior from the bottom to a height determined by the amount of glass in the canister. The HLW glass density used in this calculation is 2.57 g/cm^3. The total mass of Savannah River Site HLW glass in a canister is modeled as 1814 kg (Ray 2007, Table 2) with a resulting glass height of 101.3 in. (257.30 cm). Table 1.5.1-16 provides physical parameters for the different HLW canisters while Table 1.10-5 summarizes the dimensions of a Savannah River Site HLW canister used in the shielding evaluation. Internal structural components (i.e. internal dividers or support tubes) are omitted from the model which results in conservative dose rates due to the lack of attenuation from internal structural components.

Although the Savannah River Site canister is utilized in most shielding evaluations, the Hanford canister is utilized in the canister slide gate evaluation for the Canister Receipt and Closure Facility (CRCF), as it is the top axial bounding source. The HLW canister containing Hanford glass waste is filled to the top of the canister cavity. The waste packages share the same radial and axial dimensions with the exception of the internal cavity length of the inner vessel and outer corrosion barrier. Figures 1.10-23 to 1.10-26 depict the Savannah River Site canister and Hanford canister. Figure 1.10-27 depicts a representative 5-DHLW/DOE Codisposal waste package used to evaluate shielding requirements in the surface facilities. The same 5-DHLW/DOE Codisposal waste package with a DOE SNF canister in the center is also evaluated.

1.10.3.3.1.7 Aging Overpack and Shielded Transfer Cask

During transport between handling facilities, and on the aging pads, TAD canisters are placed inside aging overpacks. Table 1.10-6 presents the dimensions of a representative aging overpack (Figure 1.2.7-6). Figures 1.10-28 and 1.10-29 show a TAD canister inside a representative aging overpack.

Shielded transfer casks are used to move TAD canisters (Figure 1.2.5-77) and DPCs (Figure 1.2.5-76) within the WHF. The shielded transfer cask is required to shield a TAD canister in axial and radial directions. The TAD canister is utilized in these calculations, modeled with a 21-PWR assembly maximum source term. Shielded transfer cask design is bounded in the radial direction by dimensions listed in Table 1.10-7. Shielded transfer cask shielding is designed to meet a 100-mrem/hr dose rate on all surfaces including the lid in accordance with Table 1.10-2. A summary of shielding options that meet the radial dose rate requirement is shown in Table 1.10-8.

1.10.3.3.1.8 HLW and SNF Waste Packages

Table 1.10-9 summarizes the dimensions and materials of the 5-DHLW/DOE Short Codisposal waste package. The 5-DHLW/DOE Short Codisposal waste package is modeled with five DOE HLW canisters around the periphery with nothing in the center (Figure 1.10-27) for the Initial Handling Facility (IHF), and is modeled with five DOE HLW canisters surrounding a single, centralized DOE SNF canister for the CRCF. The 5-DHLW/DOE Codisposal waste packages with SNF centers are modeled without internal divider plates for conservatism. The waste package contents also differ. Instead of Savannah River Site HLW in a short waste package, the long waste package contains Hanford HLW, and instead of TRIGA DOE SNF in a short waste package, the long waste package contains Fast Flux Test Facility DOE SNF.

1.10.3.3.1.9 TAD Waste Packages

A 21-PWR/44-BWR TAD waste package contains a TAD canister conservatively modeled with 21 PWR spent fuel assemblies inside. The TAD waste package is modeled as shown in Table 1.10-10.

1.10.3.3.1.10 Naval Waste Packages

The Naval Long waste package is designed to contain the Naval Long SNF canister, while the Naval Short waste package is designed to contain the Naval Short SNF canister. This section describes the Naval Long waste package as the only major difference is the axial dimension between waste packages.

The fuel inside the naval SNF canister is modeled as a central void with reflective boundary conditions on the canister surfaces. Modeling the canister in this way transforms the default isotropic source for a cylindrical volume distribution into an isotropic distribution only in the outward direction. Modeling a void also conservatively neglects any attenuation from the canister and fuel materials. Table 1.10-11 shows the Naval Long waste package dimensions. The dimensions used for the canister were the maximum allowed. Figures 1.10-30 and 1.10-31 show the model for a waste package with a naval SNF canister inside.

1.10.3.3.2 Waste Forms Received for Disposal

The radioactive waste received for disposal takes two principal forms: (1) SNF (including commercial, DOE, or naval SNF), or (2) HLW.

1.10.3.3.2.1 Commercial Spent Nuclear Fuel

Table 1.10-12 provides the design parameters for the B&W 15 × 15 Mark B fuel assemblies. Commercial SNF is modeled as either a maximum PWR with fuel enrichment of 5.0 wt % initial ^{235}U enrichment or a design basis PWR with fuel enrichment of 4.0 wt % initial ^{235}U enrichment (Table 1.10-13). In modeling commercial SNF, fuel enrichment conservatively assumes unirradiated SNF for material composition. This assumption does not apply to the source term, which is based on irradiated spent fuel.

1.10.3.3.2.2 Naval Spent Nuclear Fuel

Surface fluxes for the naval SNF canister are provided by the Naval Nuclear Propulsion Program for shielding evaluations, based on the dimensions for the naval SNF canister and the radionuclide inventory provided in Table 1.5.1-32.

1.10.3.3.2.3 DOE Spent Nuclear Fuel

The DOE SNF canister containing a homogenized TRIGA-FLIP fuel composition is used as the representative DOE SNF in the 5-DHLW/DOE Codisposal waste package. Table 1.10-14 presents the isotopic inventory used to model the TRIGA fuel.

1.10.3.3.2.4 DOE High-Level Radioactive Waste

HLW composition and radionuclide inventory is presented in Tables 1.5.1-14 and 1.5.1-21. The elemental weight percents, for Savannah River Site and Hanford respectively, used in shielding models is presented in Tables 1.10-15 and 1.10-16.

1.10.3.3.3 Low-Level Radioactive Waste

Low-level radioactive waste is generated at the repository from receiving, repackaging, and emplacing of SNF and HLW. Low-level radioactive waste forms are both solid and liquid. The solid waste includes HEPA filters, WHF pool filters, WHF pool spent resins, and dry active waste, such as contaminated trash, disposable personal protective equipment, and empty DPCs. The liquid waste includes water from WHF pool water treatment, floor drainage, sampling activities, and liquids resulting from decontamination of repository areas, equipment, and personnel. The solid and liquid low-level radioactive waste forms generated in the GROA are evaluated for shielding purposes to ensure personnel protection.

GROA surface facilities that generate low-level radioactive waste from receipt, handling, or emplacement of SNF and HLW are:

- Canister Receipt and Closure Facility (CRCF)
- Receipt Facility (RF)
- Initial Handling Facility (IHF)
- Wet Handling Facility (WHF)
- Aging Facility
- Low-Level Waste Facility (LLWF).

In addition, the subsurface area consisting of access mains and emplacement drifts, may also generate low-level radioactive waste.

Low-level radioactive waste is generated primarily in the WHF because uncanistered commercial SNF assemblies, and DPCs containing commercial SNF assemblies that are transferred to a TAD canister, are processed in WHF. The WHF is expected to generate the greatest quantity of low-level radioactive waste that also has the highest level of radioactivity in the low-level radioactive waste stream. The highest activity low-level radioactive waste occurs on WHF HEPA filters, and on process filters and ion exchangers used in the pool water treatment system (Section 1.2.5). Other surface facilities also generate low-level radioactive waste from HEPA filters, decontamination activities, disposable personal protective equipment, and contaminated trash.

The LLWF receives low-level radioactive waste generated by surface handling facilities and the subsurface. Solid low-level radioactive waste is first collected in the handling facilities then transported to the LLWF. Liquid low-level radioactive waste is also collected at the handling facilities and transported to the LLWF, or in the case of the WHF, piped directly to tanks located at the LLWF. Since the LLWF performs handling, sorting, and packaging of low-level radioactive waste, additional low-level radioactive waste is also generated in this facility. The LLWF is designed with remote handling equipment and shielded holding areas for temporary storage of the waste forms received including; liquid wastes, WHF pool water treatment system dewatered spent resins and filter cartridges placed into sealed containers or drums (as appropriate), HEPA filters, empty DPCs, and other forms of dry active waste stored in drums and boxes. WHF pool water treatment system spent resins and filter cartridges are disposed of directly from the WHF or may be temporarily held in the LLWF prior to disposal.

1.10.3.3.3.1 Pool Water Treatment System Filter Cartridges

The WHF pool water treatment system includes both roughing and polishing cartridge filters (Table 1.10-17 and Figures 1.2.5-59, 1.2.5-60, and 1.2.5-61) that are periodically changed out based on increased external dose rates or differential pressure. The dose rates encountered on the filters may be substantial and remote filter cartridge change out capabilities are provided to protect workers from excessive doses. The filter process rooms are segregated from routine operational areas, shielded, and provided with remote filter handling features to minimize worker exposure.

1.10.3.3.3.2 Pool Water Treatment System Spent Resins

The WHF pool water treatment system also includes ion exchangers (Table 1.10-17 and Figure 1.4.5-2). The spent radioactive resins from the ion exchange beds are periodically changed out based on increased external dose rates or degraded performance. The dose rates encountered on the ion exchangers may be substantial and remote resin change out capabilities are provided to protect workers from excessive doses. The ion exchanger areas are segregated from routine operational areas, shielded, and provided with remote resin handling features to minimize worker exposure. Spent resins are processed for offsite disposal (Section 1.4.5.1.1.1). The WHF design does not include spent resin storage tanks or other forms of intermediate holding that require shielding.

1.10.3.3.3.3 Liquid Low-Level Radioactive Waste

The majority of low-level radioactive liquid waste results from cask wash down and decontamination activities in GROA facilities. This liquid waste is collected through facility floor drain systems. Liquid waste is transported in tanker trucks from the waste handling facilities, or in the case of the WHF piped directly, to collection tanks at the LLWF. Holding tanks are provided at each surface handling facility to collect liquid waste prior to transfer to the LLWF. Tank design includes provisions for removing accumulated bottoms to maintain dose rates ALARA.

1.10.3.3.3.4 Dry Active Wastes

Dry wastes generated at the repository include contaminated trash, disposable personal protective equipment, and decontamination materials, such as dewatered mop heads, used floor coverings, laboratory wastes, and swipes. Each facility is provided with appropriately shielded areas designed to collect and temporarily hold these wastes, prior to transport to the LLWF.

1.10.3.3.3.5 Empty DPCs and Miscellaneous Items

Empty DPCs, and associated SNF assembly baskets, are considered contaminated objects for offsite disposal purposes. To minimize the potential spread of contamination between facilities, residual SNF is removed and the exterior surfaces cleaned to reduce contamination levels prior to transportation to the LLWF. Empty DPCs are transported to the LLWF for storage and further decontamination, if necessary, to comply with surface contamination limits for transportation and disposal. A shielded staging area is provided in the LLWF for empty DPCs.

As a result of equipment repair or replacement activities, other contaminated components and materials may also be generated which exhibit dose rates and/or contamination levels that require temporary storage at the LLWF prior to offsite disposal. Radiation protection personnel determine appropriate handling and shielding requirements for these items.

1.10.3.4 Source Terms

1.10.3.4.1 Commercial SNF

Commercial SNF source terms are based on two source specifications (Table 1.10-13) as follows:

- Maximum PWR source (5.0 wt % initial ^{235}U enrichment, 80-GWd/MTU burnup, and 5-year decay time). Table 1.10-18 lists the gamma source terms for the fuel and nonfuel regions and the neutron source terms for the fuel region. The neutron source is in the fuel region only, whereas both fuel and nonfuel regions contain the gamma sources.

- Design basis PWR source (4.0 wt % initial ^{235}U enrichment, 60-GWd/MTU burnup, and 10-year decay time). Table 1.10-19 lists the gamma source terms for the fuel and nonfuel region and the neutron source terms for the fuel region. The neutron source is in the fuel region only, whereas both fuel and nonfuel regions contain the gamma sources.

For dose rate calculations, fuel assemblies in a waste package are uniformly modeled as having the same characteristics, except for the TEV which evaluated a single high burnup SNF source. The source terms presented in Tables 1.10-18 and 1.10-19 are multiplied by the number of assemblies in a waste package to generate a source intensity. Gamma and neutron source profiles that account for the axial distribution of gamma and neutron sources in the active fuel region are provided in Table 1.10-20.

1.10.3.4.2 Naval Spent Nuclear Fuel

Tables 1.10-21 and 1.10-22 present the Naval Long canister source spectra used to evaluate shielding (McKenzie 2007). Figure 1.10-32 depicts the top surface of a naval SNF canister and corresponds to the regions of flux presented in Tables 1.10-21 and 1.10-22.

1.10.3.4.3 DOE Spent Nuclear Fuel

The bounding source DOE SNF waste used to evaluate shielding is a homogenized TRIGA-FLIP SNF composition. Table 1.10-23 presents the source intensity used for a homogenized TRIGA-FLIP fuel.

1.10.3.4.4 DOE High-Level Radioactive Waste

The two bounding sources of DOE waste used to evaluate shielding are Savannah River Site HLW and Hanford HLW. The source intensities for HLW are presented in Tables 1.10-24 and 1.10-25.

1.10.3.4.5 Low-Level Radioactive Waste

The estimated annual volume of low-level radioactive waste is presented in Table 1.4.5-1. The estimated radionuclide concentration of low-level radioactive waste is presented in Table 1.4.5-2. Because of the large time out of reactor (minimum of 5 years) for the commercial SNF received at the repository, short lived radionuclides decay to low levels. The WHF pool treatment system filter cartridges and spent resins are typically the most highly radioactive waste forms found in the low-level radioactive waste stream.

To protect workers from radiation exposure and ensure that doses are ALARA, shielding estimates utilize conservative source term assumptions that account for uncertainty in estimated low-level radioactive waste quantities and radionuclide constituents. The radiological characteristics of the forms of low-level radioactive waste, including dry active waste and both filter cartridges and spent resins from WHF pool operations, are assumed to be similar to those generated by a typical power reactor. As a result, low-level radioactive waste quantities used in the evaluation of facility shielding are conservatively higher than the annual quantities presented in Section 1.4.5.

Conservatively estimated radionuclide concentrations for the WHF pool treatment system filter cartridges are presented in Table 1.10-26. Conservatively estimated radionuclide concentrations for the WHF pool treatment system spent resins are presented in Table 1.10-27. The associated gamma intensity (energy spectrum) for filter cartridges and spent resins used in the shielding evaluation are presented in Tables 1.10-28 and 1.10-29, respectively.

HEPA filters are used in the confinement ventilation systems of each of the surface handling facilities. However, because of the handling of uncanistered SNF in the WHF, HEPA filters in the WHF are bounding. Conservatively estimated radionuclide concentrations for the WHF HEPA filters are presented in Table 1.10-30. The associated gamma intensity for WHF HEPA filters used in the shielding evaluation is presented in Table 1.10-31.

The LLWF design has four low-level radioactive waste staging areas that are assumed to be 50-ft long by 30-ft wide by 20-ft high. Based on a comparison of source intensities the highest radiation source staged in the LLWF is WHF pool treatment system filter cartridges. For the shielding evaluation, the source is modeled as a 50-ft long, by 30-ft wide, by 6-ft high homogenized source, based on the annual estimated production of filter cartridges. Conservatively estimated radionuclide concentrations for a LLWF staging area are presented in Table 1.10-32. The associated gamma intensity (energy spectrum) for LLWF staging areas used in the shielding evaluation is presented in Table 1.10-33.

Liquid waste stored in the LLWF collection tanks is modeled as containing ^{60}Co at a concentration of 1.0×10^{-3} μCi/mL and ^{137}Cs at a concentration of 1.5×10^{-3} μCi/mL. The associated gamma intensity (energy spectrum) for a LLWF collection tank used in the shielding evaluation is presented in Table 1.10-34.

1.10.3.5 Shielding Evaluation of Surface Repository Areas

1.10.3.5.1 Open Areas

This section addresses waste handling activities in open areas of the repository, from the time waste is received to the time that waste is transported to the subsurface for emplacement, excluding activities inside waste handling facilities.

The waste received at the repository complies with U.S. Department of Transportation (49 CFR Part 173) and NRC (10 CFR Part 71) requirements for shipping radioactive materials. NRC licensing of transportation casks determines allowable materials, source strengths, and thermal loads. The NRC licensing process also verifies that casks design meet regulatory criteria. Dose rate limits are specified for radial contact and distance exposures, as well as for top and bottom ends. Shipments received by the DOE in transportation casks could arrive by truck or rail with varying quantities of waste in the casks. Separate buffer areas are used for truck and rail shipments.

1.10.3.5.1.1 Cask Receipt Security Station

The Cask Receipt Security Station processes a single cask (truck or rail) at a time. The bounding model of the Cask Receipt Security Station consists of a single representative rail transportation cask (with impact limiters in place) positioned in contact with the wall of the station. The wall is modeled as 24 in. concrete. Dose rates are evaluated and extrapolated through the wall to determine required thickness to meet a dose rate of 0.05 mrem/hr on the inner surface. The results are presented in Table 1.10-35.

1.10.3.5.1.2 Aging Facility

The Aging Facility temporarily holds commercial SNF in DPCs and TAD canisters that are placed into shielded aging overpacks. Horizontal canisters are placed into separate horizontal aging modules. No permanent shielding is anticipated around the Aging Facility because of the shielding provided by the aging overpacks and aging modules, combined with the location which provides significant distance attenuation from normally occupied areas. Dose fields around a single aging overpack are determined with a TAD canister inside containing a maximum SNF source term that meets the 40-mrem/hr dose requirement for accessible surfaces. To determine the dose field around the Aging Facility, a row containing 13 groups of aging overpacks arranged in a four by four matrix is modeled. The aging overpacks contained TAD canisters with a maximum SNF source term. Results are presented in Tables 1.10-35 and 1.10-36 as shielding and offset distance requirements, respectively.

1.10.3.5.1.3 Buffer Areas

No permanent shielding is anticipated for the rail or truck buffer areas. These above-surface locations temporarily hold waste forms in transportation casks, which are shielded for personnel protection. Transportation casks received at the repository pass through the buffer areas.

One model is used to evaluate the dose field around both buffer areas. The model consisted of 25 representative rail transportation casks. Dose rates are then evaluated from the radial direction of the casks, as this produces the highest dose rates. Offset distance results are presented in Table 1.10-36.

1.10.3.5.1.4 Movement Corridors

Paths of transient sources in the open areas of the repository are limited to identified rail and roadway movement corridors. This includes standard rail lines, TEV rail lines, truck transport roads, and site transporter roads. Higher radiation levels are present during waste movement and may temporarily extend outside of surface buildings. No permanent shielding of these waste movement paths is planned. Dose to personnel is controlled by a combination of distance, shielding provided by buildings, and administrative controls.

Representative rail transportation casks are used to evaluate shielding from transient sources. A dose field is calculated around one, two, or three casks to determine the dose rates with no shielding present. Then, 14-in. thick concrete walls are placed at distances of 50, 75, 100, and 150 ft radially from a single cask to determine the amount of shielding required to meet dose rates of 0.25 mrem/hr and 0.05 mrem/hr. Table 1.10-35 presents a summary of the sources used and shielding thicknesses determined or verified in the GROA. Table 1.10-36 presents offset distances required to meet the radiation zoning specified for inside the GROA. Using this information, building walls are determined to be sufficient to reduce interior dose rates below 0.05 mrem/hr.

Specific details of the administrative access controls employed to protect workers vary depending on the waste container to be moved and its exterior radiation level. Transportation casks licensed by the NRC for the safe movement of radioactive materials in public areas meet radiation limits in accordance with 10 CFR Part 71. Shielded waste containers that do not leave the GROA meet radiation levels licensed by the NRC for the repository.

1.10.3.5.2 Initial Handling Facility

The IHF (Section 1.2.3) only receives naval SNF canisters and HLW canisters. They are placed directly into waste packages and closed (sealed). The IHF is designed to provide radiation protection to workers, the public, and the environment and to minimize occupational doses in accordance with ALARA principles.

Shielding of specific areas in the IHF, and the other waste handling facilities, meets the radiation zoning criteria (Table 1.10-1), which are established based on required worker access and exposure potential including: (1) types of worker activities to be performed, (2) use of remotely operated equipment, (3) source strengths and necessary worker stay times, (4) need for active (versus passive) radiation protection design features, and (5) potential maximum exposures to workers due to an equipment malfunction or personnel error. Table 1.10-37 presents a summary of shielding evaluation for the IHF.

1.10.3.5.3 Canister Receipt and Closure Facility

Activities conducted in the CRCF (Section 1.2.4) include receiving and handling HLW canisters, DOE SNF canisters, and commercial SNF in TAD canisters; placing these canisters into waste packages; closing (sealing) the waste packages; and transferring them to a TEV. The CRCF also has the capability to receive DPCs and transfer them to the Aging Facility or to the WHF.

The CRCF is designed to provide radiation protection to workers, the public, and the environment and to minimize occupational doses in accordance with ALARA principles. Features for minimization and control of radioactive contamination within the CRCF are incorporated into the design (Section 1.2.4). Shielded work areas, as required, are also incorporated into the design. A summary of the shielding evaluation for the CRCF is presented in Table 1.10-38.

1.10.3.5.4 Receipt Facility

The RF (Section 1.2.6) receives and handles commercial SNF in TAD canisters or DPCs. Commercial SNF (TAD canisters or DPCs) is transferred to a CRCF for processing or may be transferred to the WHF or the Aging Facility.

The RF is designed to provide radiation protection to workers, the public, and the environment and to minimize occupational doses in accordance with ALARA principles. Features for minimization and control of radioactive contamination within the RF are incorporated into the design (Section 1.2.6). Shielded work areas, as required, are also incorporated into the design. A summary of the shielding evaluation for the RF is presented in Table 1.10-39.

1.10.3.5.5 Wet Handling Facility

The WHF (Section 1.2.5) provides the space, layout, structures, and systems to support uncanistered commercial SNF assemblies and canistered commercial SNF in DPCs, opening DPCs, SNF transfer operations, and closure of TAD canisters containing SNF assemblies. The WHF is designed to provide radiation protection to workers, the public, and the environment and to minimize occupational doses in accordance with ALARA principles.

Features for minimization and control of radioactive contamination within the WHF are incorporated into the design. Shielded work areas are also incorporated into the design. Waste handling operations performed in the WHF pool are performed under borated water to provide radiation protection and to maintain subcriticality. The minimum water depth above an SNF assembly during transfer operations is 10 ft-6 in. for a dose rate of 0.25 mrem/hr at the pool surface and 9 ft-2 in. for a dose rate of 2.5 mrem/hr. The minimum depth for waste movement underwater does not apply to the movement of SNF inside shielded casks.

Commercial SNF may also arrive in sealed DPCs. These DPCs are received at the RF or the WHF. DPCs received at the RF are transferred to the WHF in aging overpacks and placed into a shielded transfer cask for further processing in the WHF. A DPC cutting machine in the WHF operates remotely in a dry environment on a DPC. However, the DPC is filled with water prior to cutting the inner lid of the DPC, thus providing additional shielding for the cutting operation. The cut DPC, still inside a shielded transfer cask, is moved into a special area of the WHF pool. Pool water preferentially flows through this DPC holding area to allow releases from a DPC to be removed by the pool water treatment system. This cleanup of DPC releases also minimizes the potential for contamination of staged waste, canisters, and tools used in the normal area of the pool. The SNF contents of a DPC are repackaged into TAD canisters underwater. The handling of uncanistered commercial SNF assemblies may result in the release of fission product and/or crud isotopes to the pool water. These radionuclides are removed by the pool water treatment system by recirculation of pool water through both filters and ion exchangers (Figures 1.2.5-60 to 1.2.5-63). Waste fragments or discrete radioactive particles are also collected on filters or demineralizers, or settle out in other devices. The buildup of radionuclides on these filters and ion exchangers can represent a substantial source term for shielding in the WHF.

Airborne releases, resulting from noble gases and the small fraction of particulates not retained in the pool water, are collected and exhausted by the WHF HVAC system to prevent an airborne buildup and minimize worker exposures. A summary of the shielding evaluation for the WHF is presented in Table 1.10-40.

1.10.3.5.6 Low-Level Waste Facility

The LLWF is designed for the collection, packaging, and shipment of low-level radioactive waste streams generated during the handling of HLW. The LLWF is capable of storing packaged low-level radioactive waste. Dry active waste is typically received at the LLWF in bags or drums, sorted, and repackaged for disposal. DPC carcasses are received, decontaminated if necessary, and staged prior to disposal as low-level radioactive waste. Spent HEPA filters brought to the LLWF are packaged for disposal. Spent pool water treatment system resins and cartridge filters inside suitable containers may also be transferred to the LLWF for storage prior to disposal.

The bounding source anticipated to be stored in the LLWF is pool water treatment system cartridge filters packaged in shielded containers that are awaiting disposal. Spent resins from the pool water treatment system packaged into suitable shielded containers may also be stored in the LLWF while awaiting disposal. However, the source term from spent resins are bounded by the source term from the cartridge filters. Up to 50,000 gal of low-level liquid waste may also be stored at the LLWF (Section 1.4.5).

A summary of the shielding evaluation for the LLWF is presented in Table 1.10-41.

1.10.3.5.7 Common Shielding Features

The surface facilities use similar shielding features. These features include shield doors, slide gates, shielded viewing windows, canister transfer machines (shield bell and its slide gate), and waste package transfer trolleys. Each feature is analyzed with the expected maximum source term to meet specified external dose rate criteria (Table 1.10-2). The shielding evaluation demonstrates that specified mechanical envelopes are sufficient for common shielding materials and establishes limits on expected weights for facility design. Expected thicknesses of shielding material are provided in each facilities' shielding summary tables.

1.10.3.5.7.1 Shield Doors

Shield doors (Figures 1.2.4-82 and 1.2.4-85) are used in areas where access control is combined with design to reduce personnel dose. Various thicknesses of steel and borated polyethylene are evaluated. Typically, neutron absorbing material is sandwiched between steel for structural strength with a thicker layer of steel placed furthest away from the source to reduce neutron dose and associated secondary gamma production in the outer steel layer.

1.10.3.5.7.2 Shielded Viewing Windows

Shielded viewing windows (Figure 1.2.4-147) are designed to provide the shielding and containment necessary to protect operators from elevated radiation levels while providing wide-angle viewing of process and maintenance activities in accordance with the requirements of ASTM C 1572-04. The window design includes an independent shield glass housing frame that is installed into a window liner that is cast into the building concrete walls.

A shielded viewing window provides biological-shielding that meets dose rate criteria equivalent to that of the wall into which it is installed. To ensure that this criteria is met, the design prevents streaming paths and provides equivalent attenuation to the shield wall into which it is installed.

1.10.3.5.7.3 Slide Gates

Slide gates (Figures 1.2.4-57 and 1.2.4-62) are used in the design of surface facilities to allow movement of canisters between operating areas of different elevations. Minimal access is required around the slide gates. Shielding is specified to protect operators from unshielded radiation sources when working in areas close to slide gates. Dose rate limits prevent the area protected by a slide gate from being a high radiation area, thus reducing worker doses and the need for more restrictive controls. Various thicknesses of steel and borated polyethylene are modeled for slide gates.

1.10.3.5.7.4 Canister Transfer Machine

The canister transfer machine (Figure 1.2.4-50) moves canisters between cask unloading rooms, waste package positioning rooms, and canister staging areas. The canister transfer machine shield bell is designed to reduce doses from unshielded canisters such that the external contact dose rate is 100 mrem/hr, or less. This design prevents an operating area from being classified as a high radiation area. The lower portion of the canister transfer machine includes a rectangular shield skirt to also lower the dose rate at its outer edges to below 100 mrem/hr. This design is provided to reduce streaming at the bottom from a 2-in. clearance. The canister transfer machine also features an additional slide gate, called the canister transfer machine slide gate, which serves to protect other areas below the canister transfer room (such as galleries) as the canister transfer machine moves an unshielded canister from one port to another. Borated polyethylene and/or steel are modeled in canister transfer machine shielding.

1.10.3.5.7.5 Transfer Trolley

The waste package transfer trolley (Figures 1.2.4-88 and 1.2.4-88) moves waste packages between positioning rooms and loadout rooms. Waste package transfer trolley shielding is designed to reduce doses from unshielded waste packages such that the external contact dose rate is 100 mrem/hr, or less. This design prevents an operating area from being classified as a high radiation area. Both steel and borated polyethylene are modeled in trolley shielding.

1.10.3.6 Shielding Evaluation of Subsurface Repository Areas

Waste packages are moved into the subsurface for emplacement by a remotely controlled TEV. The waste packages are loaded into a TEV in a surface facility while inside the shielded portion of the facility. The TEV provides shielding of the waste package during its movement from the surface to a subsurface emplacement drift. No permanent shielding other than that provided by the TEV is anticipated for waste package movement from a surface facility to emplacement.

Dose rate evaluations are performed for the turnouts and access mains to ensure worker protection and to comply with dose rate criteria. The evaluation determines the effects on dose rate at the curvatures and intersections of the emplacement drifts and the access main to verify that the geometric layout reduces dose rates to meet goals. Dose rate limits for the subsurface facility are:

1. Below 100 mrem/hr for intermittent access in restricted areas, such as the turnout

2. Below 20 mrem/hr at the turnout bulkhead location on the access side of the door, where access is only expected for door maintenance

3. Below 5 mrem/hr at the access main in the area facing each emplacement drift, when a loaded TEV is not present.

Dose rates calculated in the turnouts and access main consider the first three (closest to the access main) emplaced waste packages. The waste packages are oriented with the waste package bottom toward the access main and turnout. The TAD waste package is selected as the bounding waste package. Determination of the selected emplaced waste package is based upon comparison of dose rates on the bottom surface of various waste package configurations.

The features of the underground layout include emplacement drifts (Table 1.10-42), turnouts and the access main drift. The typical angle of departure of the turnout from the access main is 25°, which is representative of the majority of the subsurface repository turnouts. Remaining turnouts are distinguished by a lengthened straight segment of turnout, dictated by the turnout location in the repository. The typical turnout, is composed of the following design elements and is shown in Figure 1.10-33:

1. The access main represents the combined allowance for the TEV rail turnout and the excavation departure segment. The access main tunnel has an inner radius of 3.81 m (12.5 ft). The initial departure angle of the turnout drift is approximately 12°.

2. The turnout begins with two smaller drifts, the access–turnout junction and the launch chamber drift. The access–turnout junction has an inner radius of 3.81 m (12.5 ft) and the launch chamber has an inner radius of 3.35 m (11 ft).

3. The turnout drift has an inner radius of 2.74 m. The turnout drift follows an arc with a radius of 60.96 m (200 ft).

The emplacement drift is the final emplacement location for the waste packages, which are situated 18.28 m (60 ft) from the point where the turnout straightens from a curved tunnel to a straight tunnel. Figures 1.10-34 and 1.10-35 illustrate transversal sections inside emplacement drifts used in MCNP models. Waste package emplacement pallets are conservatively excluded from the MCNP models and the volume is replaced by air. Table 1.10-43 provides the chemical compositions and densities of materials modeled in the subsurface drifts.

The turnout and access main have significant design features that affect dose rates. The angle of turnout eliminates a direct radiation pathway into the access main. This geometry ensures that radioactive particles emanating from the waste packages have at least one collision in the turnout. These particle collisions reduce the energy (if not cause the elimination) of the particles in the turnout system, thus leading to a reduction in dose rate. The shielding evaluation demonstrates that the length and curvature of the turnouts are sufficient and that no additional shielding is required.

1.10.3.6.1 Transport and Emplacement Vehicle

TEV (Figures 1.3.4-19 and 1.3.4-20) shielding ensures dose rates below 100 mrem/hr at 30 cm from the external surfaces of the TEV (Table 1.10-2). The TEV design is a mobile enclosure with four shielded sides (two side walls and doors on each end), a shielded floor, a shielded roof and two tapered sides which connect the side walls to the roof. Dimensions relating to the TEV and TEV shielding are provided in Tables 1.10-44 and 1.10-45. A view of the TEV and its shielding are provided in Figures 1.10-36 and 1.10-37. The TEV floor is modeled 4.53 in. (11.5 cm) below the waste package. The waste package is centered in both the axial and radial directions within the TEV.

The TEV is centered inside the drift. The drift material employed in the model is tuff with a density of 2.21 gm/cm^3, and the drift wall is modeled as 30 cm (11.8 in.) thick. TEV dose rates are calculated using a fully loaded waste package containing 21 PWR fuel assemblies centered in both the radial and axial directions inside the TEV with the TEV centered inside a repository drift. Materials compositions that are modeled for the TEV are provided in Table 1.10-46.

1.10.3.7 Event Sequence Considerations

No permanent shielding is provided for postulated Category 2 event sequences.

1.10.4 As Low As Is Reasonably Achievable Principles in Operations
[NUREG-1804, Section 2.1.1.8.3: AC 3; HLWRS-ISG-03, Section 2.1.1.8.3: AC 4]

The Operational Radiation Protection Program discussed in Section 5.11 includes repository processes and procedures that govern work in the restricted area in accordance with ALARA principles (Ghanooni and Carl 2002). Applicable regulatory guidance and guidelines contained in Regulatory Guides 8.8 and 8.10 on ALARA, including experience and lessons learned at other nuclear facilities, will be incorporated into the applicable processes and procedures.

Worker and public doses that result from repository operation will be within regulatory performance objectives and will be minimized in accordance with ALARA principles. The Operational Radiation Protection Program will embody a continuous improvement and lessons learned process. Implementation of ALARA principles at the repository will ensure individual and collective doses are minimized, consistent with ALARA goals.

1.10.4.1 Operational ALARA Considerations

Implementation of ALARA principles into repository operations ensures personnel are trained, procedures are followed, doses are controlled, and management support is provided. To ensure an effective ALARA program, personnel doses are constantly monitored and evaluated, and applicable policies and procedures are part of a continuous improvement program. Operational considerations include:

- An ALARA committee oversees ALARA implementation. This committee includes members from affected organizations and will include senior management personnel.
- Worker and public doses are monitored and evaluated to identify adverse trends and opportunities for reduction in dose. Area dose rates are also monitored and evaluated to verify the effectiveness of engineered and operational controls and to identify adverse trends in radiation levels or in the build-up of radioactive material in the workplace.
- ALARA goals are established and monitored. Progress in achieving the goals is incorporated into the decision-making process and incorporated into worker and work group input to further reduce dose accumulation.
- Administrative limits are established to control occupational radiation doses. These administrative limits are structured to ensure periodic review of worker and job dose methodologies and, as appropriate, require senior management approval for dose controls to be exceeded.
- Worker access controls within the restricted area are established to prevent inadvertent exposure. Worker and equipment egress controls from restricted areas ensure that radioactive material is controlled and not inadvertently removed from the area.
- Radioactive contamination is prevented or minimized. The size and number of contaminated areas is minimized to limit impacts to workers and to minimize generation of low-level radioactive waste.
- Methods to monitor and reduce radioactive waste production at the source are implemented.

1.10.4.1.1 ALARA Program Administration

It is the responsibility of management to implement a program to maintain occupational radiation doses ALARA, in accordance with Regulatory Guide 8.8, Position C.1 and Regulatory Guide 8.10.

To verify that the overall radiation protection program is functioning properly, management performs audits of the ALARA program.

The authority to prevent unsafe practices and to direct steps to prevent unnecessary radiation exposures rests with the radiation protection organization. Radiation protection personnel report to the Radiation Protection Manager and handle the day-to-day implementation of the Operational Radiation Protection Program (Section 5.11). To ensure compliance with this expectation, the radiation protection manager and supervisors are charged with the responsibility to promptly advise higher management of unsafe practices. It is also the obligation of radiation protection personnel and radiation workers to halt operations that, in their judgment, are unsafe.

In addition to reviews by management, employees are encouraged to express their concerns through a formal feedback program. The ALARA program provides the basis to evaluate proposed ALARA improvements. Prior to allowing personnel inside restricted areas, they are trained and tested in radiation protection procedures and techniques to verify that they understand how these procedures relate to the safe performance of their jobs. Personnel, who are required by their assignments, undergo periodic retraining in radiation protection procedures and techniques. This training program is in accordance with 10 CFR 19.12.

Maintenance and operating procedures are reviewed to verify adherence to ALARA policy prior to their use.

1.10.4.1.2 Design Change Review

Formal ALARA reviews, similar to those conducted during original design (Section 1.10.2.1.3.3) of the facility and equipment are performed to ensure that proposed design changes meet ALARA guidelines. ALARA reviews are performed of design changes that impact maintenance, inspections, routine operations, processing radioactive wastes, decontamination, and decommissioning. This formal ALARA design change review process ensures that ALARA objectives and considerations are applied and documented prior to the approval of a design change.

1.10.4.1.3 ALARA Training

A comprehensive worker ALARA training program addresses various aspects of radiation safety, access control, and procedural compliance concerning risks to workers from occupational exposure to ionizing radiation. Individuals who work in restricted areas are classified as radiation workers and receive radiation protection training commensurate with the safe performance of their jobs. Individuals who work outside the restricted area are classified as on site members of the public and are instructed in site emergency procedures and other relevant site information. Additional information on training is addressed in Sections 5.3 and 5.11.

1.10.4.1.4 General ALARA Guidelines

General ALARA guidelines followed during operations are:

- Temporary shielding is considered if the total dose, including installation and removal of the temporary shielding, is reduced.

- Piping systems and other pieces of equipment which are subject to particulate buildup (such as the WHF pool cleanup system) are flushed to reduce personnel exposure.

- Work involving significant worker doses is preplanned. The purpose of preplanning is to ensure that the work can be performed in a safe manner with personnel doses minimized in accordance with ALARA principles. Job preplanning includes dose estimates and debriefing sessions to capture ALARA good practices and lessons learned.

- On complex jobs or jobs with significant collective doses, dry-run training may be utilized, including mockup training, as appropriate. The intent of dry-run or mockup training is to improve worker efficiency, minimize worker stay times, avoid unnecessary and potentially harmful actions, and minimize overall doses.

- As much as practicable, tasks are performed in lower radiation areas. Workers move to lower radiation areas for such activities as reading instruction manuals and maintenance procedures, adjusting tools or jigs, and performing maintenance or repair activities.

- Wherever possible, equipment is removed from higher radiation areas and moved to maintenance or laydown spaces in lower radiation areas. These maintenance or laydown spaces will have appropriate tools and support services (e.g., electrical power, compressed air, and nitrogen) to ensure an efficient task completion. However, moving of equipment not previously evaluated includes an evaluation of the estimated dose for removal and reinstallation, including necessary inspections.

- Special tools or jigs are used to prevent personnel errors, reduce time spent in high radiation–zoned areas, or increase the distance from the source to the worker.

- Entry and exit points are established to allow personnel access and exit in as low a radiation level as practicable.

- Contamination containments (e.g., glove bags, polybottles, tents, and floor coverings) are used to allow personnel to work on contaminated equipment while minimizing the spread of contamination.

- Areas within work areas with higher radiation levels are identified.

1.10.4.1.5 Administrative Controls to Maintain Doses ALARA

Operating, maintenance, and radiation protection procedures are reviewed to identify situations in which potential exposures could be reduced. Administrative controls are implemented through the use of radiological work permits during operations and maintenance:

- **Restricted Area**—Restricted areas, as defined in 10 CFR Part 63 and 10 CFR Part 20, are established for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Specific areas within a restricted area may warrant additional controls due to the potential for significant exposures to radiation or radioactive materials. Radiation areas and high radiation areas are identified within restricted areas in accordance with 10 CFR Part 20, utilizing access control and posting of radiation area and high radiation area signs. Access to high radiation areas and very high radiation areas is controlled in accordance with 10 CFR 20.1601 and 10 CFR 20.1602.

- **Radiological Work Permits**—Procedures will require radiological work permits for entry into restricted areas. These permits are a principal administrative means of managing personnel radiation dose and describe the controls required to perform an activity while maintaining personnel radiation doses ALARA. The permit contains the most current information pertinent to the activity such as radiation and/or contamination levels in the area, allowable stay times, protective clothing requirements, respiratory protection equipment requirements, special personnel monitoring requirements, temporary shielding requirements, and personnel authorized to receive radiation exposure while performing the activity for which the radiological work permit was issued.

- **Operating and Maintenance Activities**—Operating and maintenance activities are controlled by written procedures. Procedures controlling routine and off-normal waste handling, repair and rework, radiochemical sampling, inspections, maintenance, and calibrations that are expected to require issuance of a radiological work permit are reviewed for ALARA considerations.

Administrative controls are implemented to maintain personnel doses ALARA. These controls ensure protection of radiation workers and onsite members of the public. Controls are documented and changes are reviewed and approved to ensure that protective features are not inadvertently compromised.

1.10.4.1.6 Task Planning and Preparation

Task planning will reflect the following considerations:

- To provide the bases for planning activities, surveys are performed to ascertain information with respect to radiation, contamination, and airborne radioactive material that might be encountered while performing services.

- Personnel planning preparation includes study, as appropriate, of blueprints, drawings, photographs, videotapes, previous inspection reports, previous radiation and contamination surveys, or previous radiological work permits.

- Prejob briefings for personnel who work in higher radiation areas will ensure that these individuals understand the tasks to be performed, associated hazards, and special instructions.

- Procedures address access controls and dose limitations for workers in restricted areas and the need for special permits.

- Consideration of potential off-normal occurrences and contingency planning to facilitate personnel egress.

- Portable or temporary ventilation systems or contamination enclosures and expendable floor coverings control the spread of contamination and limit the intake by workers through inhalation.

- Data and experience attained in previous operations.

- Improving the ease of access to a work area to reduce the dose accumulated during movement of personnel between a work site and a control point, including installation of scaffolding, removal of interferences, and/or establishment of different access control points. However, the estimated dose for installing and removing access improvements is weighed against expected dose reductions.

- Evaluate the size of the projected work crews, and consider reducing the number in the work crew.

- Consider the use of remotely operated equipment and systems. Use electric hoists, where possible, instead of equipment such as manual chain falls.

- Consider decontaminating the work area or equipment prior to the commencement of work. In addition to direct worker dose reduction due to the removal of the contamination, decontamination may allow work crews to forego more restrictive protective clothing or respiratory protection or both. The estimated dose for decontamination is weighed against the expected dose reduction.

1.10.4.1.7 Radiation Surveys

Radiation protection personnel perform radiation and contamination surveys as delineated in procedures. Surveys consist of radiation or contamination measurements, or both, as appropriate for the specific area. Survey information is factored into exposure stay time determinations and radiological work permit specifications. A radiological work permit may specify the need for additional surveys for specific operations or maintenance activities or both. Radiation surveys are performed to determine beta, gamma, and/or neutron radiation levels. Contamination surveys are performed to determine alpha and beta-gamma contamination levels. Air samples are taken to determine airborne concentrations of radioactive materials.

1.10.4.1.8 Housekeeping

Operating procedures address housekeeping requirements. Personnel will be trained on the importance of a clean and well-organized work area to keeping worker doses ALARA.

1.10.4.2 Operational Radiation Protection Program

The organization, equipment, instrumentation, facilities, and policies and procedures of the Operational Radiation Protection Program are described in Section 5.11.

1.10.4.3 Recovery from Event Sequences

As shown in Section 1.7, there are no Category 1 event sequences associated with the waste handling at the repository. The probability of a preclosure Category 2 event sequence has been evaluated for waste handling areas (Section 1.7). The ALARA process does not apply to the termination of these postulated low probability events that could result in a partial loss of containment of the waste being handled with a consequent increase in airborne radioactivity and a possible increase in direct radiation levels in affected areas. As part of the emergency plan (Section 5.7), noncritical personnel will be evacuated from the vicinity of the event. The ALARA process is applied for recovery actions that are reviewed to ensure that personnel are appropriate for the tasks identified and that measures are taken to reduce doses.

1.10.4.4 Decommissioning

As described in Section 1.12, the overall ALARA objective for decommissioning is to:

- Prevent the uncontrolled release of radioactive materials offsite
- Ensure that radiation doses to workers and to the public from decommissioning activities are below the performance objectives and maintained ALARA.

Reviews of planned decommissioning activities for contaminated structures allow personnel to fully understand the methods and procedures undertaken to remediate or dispose of a contaminated structure. These reviews assess decommissioning activities for safety and appropriate ALARA measures.

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Table 1.10-1. Classification of Radiation Zones

Classification	Dose Rate Range (mrem/hr)	Description
R1	Background to <0.05	Unlimited Occupancy
R2	0.05 to 2.5	Routine Occupancy
R3	>2.5 to 15	Occasional Occupancy
R4	>15 to 100	Infrequent Occupancy
R5	>100	Limited or No Occupancy

Table 1.10-2. Shielding Evaluation Criteria

Description	Criterion	Basis
Dose rates exterior to SNF/HLW process facilities at personnel level	≤0.25 mrem/hr	To allow continuous occupational access in support of ALARA goal of 500 mrem/yr.
Dose rates exterior to SNF/HLW process facilities above the personnel level	≤2.5 mrem/hr	Higher dose rate is allowed above personnel level, provided the contribution from the high level will not cause the dose rate on the personnel level to exceed the criterion. Does not include areas that impact external stairways.
Operating galleries, support rooms, offices at personnel level	≤0.25 mrem/hr	To allow continuous occupational access and meet the ALARA goal of 500 mrem/yr.
Operating galleries, support rooms, offices above personnel level	≤2.5 mrem/hr	Higher dose rate is allowed above personnel level, provided the contribution from the high level will not cause the dose rate on the personnel level to exceed the criterion.
Intermittent access in restricted areas	≤100 mrem/hr	Dose rate criterion will vary with the access requirement for each area provided the general dose criteria are met.
Canister transfer machine contact dose rate	≤100 mrem/hr	Minimal access is required around the canister transfer machine. Shielding is to protect operators when working around the canister transfer machine. This limit will prevent the area around the canister transfer machine from being a high radiation area thus eliminating the need for additional controls around the canister transfer machine.
TAD canister port slide gates contact dose rate	≤100 mrem/hr	Minimal access is required around the slide gates. Shielding is to protect operators when working around the slide gates. This limit will prevent the area around the slide gates from being a high radiation area thus eliminating the need for additional controls around the slide gates.
Outside or beyond the restricted area boundary	≤0.05 mrem/hr	Applicable to controlled and unrestricted areas where members of the public have access to comply with 10 CFR 20.1301. Includes normal operations and Category 1 event sequences.
Shielded transfer cask	≤100 mrem/hr	Shielding on all sides of the shielded transfer cask is to protect operators when working around the shielded transfer casks. This limit will prevent the area around the shielded transfer cask from being a high radiation area thus eliminating the need for additional controls around the shielded transfer cask.
Aging overpack and horizontal aging module for vertical and horizontal DPCs	≤40 mrem/hr[a]	The combined neutron and gamma contact dose rate on any accessible exterior surface shall not exceed 40 mrem/hr at any location on a loaded aging overpack. This includes air circulation ducts, penetrations and any other potential streaming paths on the overpack surface. This limit will prevent the area around the aging overpack from being a high radiation area during transport to the aging pad and once the aging overpack is placed on the pad.

NOTE: [a]This criteria is limited to the top and sides for an aging overpack containing a TAD canister and is not applicable from the bottom of a TAD aging overpack during transport.

Table 1.10-3. Naval SNF Canister and Transportation Overpack Model

Dimension	Maximum Value in. (cm)	Modeled Value in. (cm)
Canister Outer Length	212.0 (538.48)	212.0 (538.48)
Canister Outer Diameter	66.5 (168.91)	66.5 (168.91)
Bolt Hole Diameter	3.0 (7.62)	3.0 (7.62)
Diameter for Bolt Hole Centers	13.0 (33.02)	13.0 (33.02)
Seal Weld Inside Diameter	52.75 (133.99)	52.75 (133.99)
Seal Weld Outside Diameter	63.68 (161.75)	63.68 (161.75)
Trans-Overpack Outer Diameter	90.0 (228.6)	88.0 (223.52)
Trans-Overpack Outer Height	240.0 (609.6)	226.0 (574.04)

Table 1.10-4. DOE SNF Short Canister Dimensions

Component	Material	Parameter	Dimension (cm)	Dimension (in.)
Canister Shell	Stainless Steel	Outer Diameter	45.7	18.0
		Thickness	0.953	0.375
		Length	299.9	118.07
		Internal Cavity Length	257.5	101.38
Top/Bottom Impact Plate	Carbon Steel	Thickness	5.00	2.00
Upper/Lower Head	Stainless Steel	Thickness	0.953	0.375

Table 1.10-5. Savannah River Site HLW Canister Dimensions

Component	Material	Parameter	Dimension (cm)	Dimension (in.)
Top Lid	Stainless Steel	Thickness	1.59	0.626
Bottom Lid	Stainless Steel	Thickness	1.27	0.500
Shell	Stainless Steel	Outer Diameter	61.00	24.02
		Thickness	0.95	0.37
		Length	300.00	118.11
Glass Log	Savannah River Site Glass	Height	257.30	101.3

Table 1.10-6. Dimensions of Aging Overpack

Component[a]	Dimension	
	(in.)	(cm)
Outer length of aging overpack	258	655.32
Thickness of steel liner in aging overpack	1.25	3.175
Thickness of radial concrete shielding	37.5	95.25
Thickness of aging overpack top lid	10.5	26.67

NOTE: [a]Designed to meet dose rate limit of 40 mrem/hr on accessible surfaces when placed on the aging pad (DOE 2008, Section 3.3.4).

Table 1.10-7. Shielded Transfer Cask Bounding Radial Dimensions

Specification	Dimension/Constraint
Cavity Inner Diameter	5 ft to 7.5 in. (minimum) 171.45 cm
Shield Outer Diameter	9 ft (maximum) 274.32 cm

Table 1.10-8. Radial Dose Rates of Shielded Transfer Cask Shielding Design Combinations for a TAD Canister with Maximum Source

Shielding Combination	Total Dose Rate (mrem/hr)
12 in. Borated Polyethylene + 7 in. Lead	99
12 in. Borated Polyethylene + 7.5 in. Lead	62
2 in. Stainless Steel + 10 in. Borated Polyethylene + 6 in. Lead	69
8 in. Borated Polyethylene + 12 in. Stainless Steel	80

Table 1.10-9. 5-DHLW/DOE Short Codisposal Waste Package Description

Component	Material	Parameter	Dimension (cm)	Dimension (in.)
Outer Corrosion Barrier	Alloy 22 (UNS N06022)	Shell Outer Diameter	204.47	80.5
		Shell Inner Diameter	199.39	78.5
		Bottom Plate Thickness	2.54	1.0
Inner Vessel	Stainless Steel Type 316	Shell Outer Diameter	198.44	78.13
		Shell Inner Diameter	188.28	74.13
		Bottom Plate Thickness	5.08	2.0
Inner Vessel Shield Plug		Thickness	22.86	9.0
Divider Tube	Carbon Steel SA 516	Outer Diameter	56.50	22.24
		Inner Diameter	50.15	19.74
		Length	300.04	118.13

Table 1.10-10. 21-PWR/44-BWR TAD Waste Package Description

Component	Material	Characteristic	Dimension/Unit
Outer corrosion barrier	Alloy 22	Outer diameter	74.08 in. (188.16 cm)
		Inner diameter	72.08 in. (183.08 cm)
Inner vessel	Stainless Steel Type 316	Outer diameter	71.70 in. (182.12 cm)
		Inner diameter	67.70 in. (171.96 cm)
		Length	218.50 in. (554.99 cm)
Outer barrier top lid	Alloy 22	Thickness	1.00 in. (2.54 cm)
Air gap over inner vessel	Air	Thickness	3.38 in. (8.59 cm)
Inner vessel top lid	Stainless Steel Type 316	Thickness	2.00 in. (5.08 cm)
Air gap over TAD canister	Air	Thickness	2.50 in. (6.35 cm)
Inner vessel cavity	Air	Length	213.00 in. (541.02 cm)
Inner vessel bottom lid	Stainless Steel Type 316	Thickness	2.00 in. (5.08 cm)
Air gap below inner vessel	Air	Thickness	2.69 in. (6.83 cm)
Outer barrier bottom lid	Alloy 22	Thickness	1.00 in. (2.54 cm)

Table 1.10-11. Naval Long Waste Package Dimension

Component	Material	Characteristic	Dimension in. (cm)
Outer corrosion barrier	Alloy 22	Thickness	1.00 (2.54)
		Outer diameter	74.08 (188.16)
		Inner diameter	72.08 (183.08)
Inner vessel	Stainless Steel Type 316	Thickness	2.00 (5.08)
		Outer diameter	71.70 (182.12)
		Inner diameter	67.70 (171.96)
Inner vessel/outer corrosion barrier radial gap	Air	Thickness	0.38 (0.96)
Outer barrier top lid	Alloy 22	Thickness	1.00 (2.54)
Inner vessel/outer corrosion barrier top gap	Air	Thickness	3.38 (8.59
Inner vessel top lid	Stainless Steel Type 316	Thickness	2.00 (5.08)
Inner vessel cavity	Air	Height	213.0 (541.02)
Inner vessel bottom lid	Stainless Steel Type 316	Thickness	2.00 (5.08)
Inner vessel/outer corrosion barrier bottom gap	Air	Thickness	2.69 (6.83)
Outer barrier bottom lid	Alloy 22	Thickness	1.00 (2.54)

Table 1.10-12. B&W 15x15 Mark B Fuel Assembly Description

Component	Material	Characteristic	Value/Unit
Assembly	—	Array Size	15 × 15
		Fuel Pins/Assembly	208
		Guide Tubes/Assembly	16
		Instrument Tubes/Assembly	1
		Pin Pellet Diameter	0.936244 cm (0.3686 in.)
		Pin Pitch	1.44272 cm (0.568 in.)
		Active Fuel Height	360.172 cm (141.8 in.)
		Assembly Pitch	21.81098 cm (8.587 in.)
		Assembly Height	165.625 in. (420.6875 cm)
		Mass U/Assembly	463.63 kg
Guide tube	Zircaloy-4	Outer Diameter	1.3462 cm (0.53 in.)
		Inner Diameter	1.26492 cm (0.498 in.)
Instrument tube	Zircaloy-4	Outer Diameter	1.38193 cm (0.5441 in.)
		Inner Diameter	1.12014 cm (0.441 in.)
Cladding	Zircaloy-4	Inner Diameter	0.95758 cm (0.377 in.)
		Outer Diameter	1.0922 cm (0.43 in.)
Plenum region	—	Length	11.720 in. (29.7688 cm)
Top nozzle	Stainless Steel CF3M	Mass/Assembly (Top)	7.48 kg
Bottom nozzle	Stainless Steel CF3M	Mass/Assembly (Bottom)	8.16 kg
Guide tubes	Zircaloy-4	Mass/Assembly (in Core)	8.0 kg
Instrument tube	Zircaloy-4	Mass/Assembly (in Core)	0.64 kg
Spacer-plenum	Inconel-718	Mass/Assembly (Plenum)	1.04 kg
Spacer-bottom	Inconel-718	Mass/Assembly (Bottom)	1.3 kg
Spacer-in core	Inconel-718	Mass/Assembly (in Core)	4.9 kg
Spring retainer	Stainless Steel CF3M	Mass/Assembly (Top)	0.91 kg
Holddown spring	Inconel-718	Mass/Assembly (Top)	1.8 kg
Upper end plug	Stainless Steel Type 304	Mass/Assembly (Top)	0.06 kg
Upper nut	Stainless Steel Type 304	Mass/Assembly (Top)	0.51 kg
Lower nut	Stainless Steel Type 304	Mass/Assembly (Bottom)	0.15 kg

Table 1.10-12. B&W 15x15 Mark B Fuel Assembly Description (Continued)

Component	Material	Characteristic	Value/Unit
Grid supports	Zircaloy-4	Mass/Assembly (in Core)	0.64 kg
Plenum spring	Stainless Steel Type 302	Mass/Assembly (Plenum)	0.042 lb (0.01905 kg)

Table 1.10-13. Comparison of Design Basis and Maximum Commercial SNF Assemblies

SNF Assembly	Initial Enrichment (%)	Burnup (GWd/MTU)	Decay Time (Years)
Design Basis PWR	4.0	60	10
Maximum PWR	5.0	80	5

Table 1.10-14. Homogenized TRIGA-FLIP Fuel Compositions

Element/Isotope	Mass (g)	wt %
C	50,251	4.312
Mn	17,076	1.465
P	384.21	0.033
S	256.14	0.022
Si	6,403.5	0.550
Cr	146,922	12.608
Ni	100,014	8.583
Mo	19,125	1.641
N	853.8	0.073
Fe	562,465	48.269
Zr	235,731	20.230
^{235}U	15,207	1.305
^{238}U	6,549	0.562
H	4,042	0.347
TOTAL	**1,165,279**	**100.00**
Density	3.001 g/cm^3	

Table 1.10-15. Savannah River Site HLW Composition

Material	Density (g/cm^3)	Element	wt %
Savannah River Site HLW Glass	2.57	Al	3.745
		B	2.154
		Ba	0.108
		Ca	0.752
		Cd	0.002
		Cr	0.058
		Cs	0.069
		Cu	0.200
		Fe	5.160
		K	1.772
		La	0.075
		Li	2.147
		Mg	0.876
		Mn	1.607
		Na	6.112
		Ni	0.315
		O	46.557
		P	0.021
		Pb	0.056
		Pd	0.031
		Pu	0.051
		Rh	0.015
		Ru	0.082
		S	0.064
		Si	25.423
		Sn	0.002
		Sr	0.007
		Tc	0.002
		Th	0.484
		Ti	0.330
		U	0.858
		Y	0.030
		Zn	0.013
		Zr	0.275
		Pr	0.108
		Ce	0.108
		Nd	0.108
		Sm	0.109
		Eu	0.109

Table 1.10-16. Hanford HLW Composition

Material	Density (g/cm^3)	Element	wt %
Hanford HLW Glass	2.81	Ag	0.048
		As	0.017
		Al	4.379
		B	1.918
		Ba	0.108
		Be	0.002
		Bi	0.011
		Ca	0.380
		Cl	0.005
		Cd	1.049
		Ce	0.128
		Co	0.007
		Cu	0.042
		Cr	1.310
		F	0.075
		Fe	13.622
		K	0.373
		La	0.547
		Li	1.084
		Mg	0.167
		Mo	0.295
		Mn	0.158
		Na	11.666
		Nd	0.432
		Ni	0.761
		O	40.219
		Pb	0.048
		P	0.046
		Pr	0.071
		Rb	0.010
		Rh	0.030
		Ru	0.211
		Si	14.824
		S	0.197
		Sr	0.058
		Sb	0.001
		Se	0.046
		Ta	0.001
		Te	0.013
		Th	0.043
		Ti	0.012
		Tl	0.001
		U	1.186
		V	0.008
		Zn	0.021
		Zr	4.369

Table 1.10-17. Pool Water Treatment System Major Components (1 Train)

Item	Quantity Per Train[a]	Type	Dimensions
Roughing Filter	2	Cartridge 2 μm	Cylindrical; 6-in. diameter; 40-in. high
Polishing Filter	2	Cartridge 0.1 μm	Cylindrical; 6-in. diameter; 40-in. high
Ion Exchanger[b]	1	Mixed bed	Cylindrical; 60-in. diameter; 96-in. high

NOTE: [a]Three trains total: 1 train for the DPC handling area, 1 train for the main pool, and 1 train on standby that can function in either area (Section 1.2.5).
[b]Active resin volume is 50 ft^3.

Table 1.10-18. Maximum PWR SNF Assembly Gamma and Neutron Sources

Upper Energy Boundary (MeV)	Gamma Intensity (photons per second)				Neutron Intensity (neutrons per second)	
	Bottom-End Fitting Region	Active Fuel Region	Plenum Fuel Region	Top-End Fitting Region	Upper Energy Boundary (MeV)	Active Fuel Region
5.00×10^{-2}	5.94×10^{11}	2.33×10^{15}	5.28×10^{11}	3.79×10^{11}	1.00×10^{-8}	0.00
1.00×10^{-1}	1.16×10^{11}	6.44×10^{14}	6.09×10^{10}	7.43×10^{10}	3.00×10^{-8}	0.00
2.00×10^{-1}	2.83×10^{10}	5.22×10^{14}	3.52×10^{10}	1.79×10^{10}	5.00×10^{-8}	0.00
3.00×10^{-1}	1.41×10^{9}	1.48×10^{14}	1.96×10^{9}	8.91×10^{8}	1.00×10^{-7}	0.00
4.00×10^{-1}	1.90×10^{9}	9.85×10^{13}	5.86×10^{9}	1.17×10^{9}	2.25×10^{-7}	0.00
6.00×10^{-1}	1.91×10^{9}	1.53×10^{15}	1.10×10^{11}	7.41×10^{7}	3.25×10^{-7}	0.00
8.00×10^{-1}	4.35×10^{9}	4.70×10^{15}	5.95×10^{10}	2.37×10^{9}	4.00×10^{-7}	0.00
1.00	1.37×10^{11}	7.08×10^{14}	8.03×10^{9}	7.66×10^{10}	8.00×10^{-7}	0.00
1.33	3.38×10^{13}	4.55×10^{14}	1.74×10^{13}	2.17×10^{13}	1.00×10^{-6}	0.00
1.66	9.53×10^{12}	1.30×10^{14}	4.91×10^{12}	6.12×10^{12}	1.13×10^{-6}	0.00
2.00	1.87×10^{3}	1.44×10^{12}	9.19×10^{2}	1.13×10^{3}	1.30×10^{-6}	0.00
2.50	2.26×10^{8}	2.49×10^{12}	1.16×10^{8}	1.45×10^{8}	1.77×10^{-6}	0.00
3.00	3.51×10^{5}	1.10×10^{11}	1.81×10^{5}	2.25×10^{5}	3.05×10^{-6}	0.00
4.00	7.66×10^{-8}	1.39×10^{10}	1.00×10^{-8}	4.16×10^{-8}	1.00×10^{-5}	0.00
5.00	0.00	7.09×10^{7}	0.00	0.00	3.00×10^{-5}	0.00
6.50	0.00	2.86×10^{7}	0.00	0.00	1.00×10^{-4}	0.00
8.00	0.00	5.58×10^{6}	0.00	0.00	5.50×10^{-4}	0.00
10.00	0.00	1.19×10^{6}	0.00	0.00	3.00×10^{-3}	0.00
Total	$\mathbf{4.42 \times 10^{13}}$	$\mathbf{1.13 \times 10^{16}}$	$\mathbf{2.31 \times 10^{13}}$	$\mathbf{2.84 \times 10^{13}}$	1.70×10^{-2}	0.00
					1.00×10^{-1}	0.00
					4.00×10^{-1}	8.05×10^{7}
					9.00×10^{-1}	4.11×10^{8}
					1.40×10^{0}	3.76×10^{8}
					1.85	2.76×10^{8}
					3.00	4.85×10^{8}
					6.43	4.43×10^{8}
					2.00×10^{1}	3.93×10^{7}
					Total	$\mathbf{2.11 \times 10^{9}}$

NOTE: 5.0 wt % initial enrichment, 80 GWd/MTU burnup and 5 years cooling time.

Table 1.10-19. Design Basis PWR Assembly Gamma and Neutron Sources

Gamma Intensity (photons per second)					Neutron Intensity (neutrons per second)	
Upper Energy Boundary (MeV)	**Bottom End-Fitting Region**	**Active Fuel Region**	**Plenum Fuel Region**	**Top End-Fitting Region**	**Upper Energy Boundary (MeV)**	**Active Fuel Region**
5.00×10^{-2}	2.73×10^{11}	1.21×10^{15}	1.88×10^{11}	1.75×10^{11}	1.00×10^{-8}	0.00
1.00×10^{-1}	5.28×10^{10}	3.29×10^{14}	2.77×10^{10}	3.39×10^{10}	3.00×10^{-8}	0.00
2.00×10^{-1}	1.28×10^{10}	2.45×10^{14}	1.17×10^{10}	8.19×10^{9}	5.00×10^{-8}	0.00
3.00×10^{-1}	6.39×10^{8}	7.13×10^{13}	6.33×10^{8}	4.07×10^{8}	1.00×10^{-7}	0.00
4.00×10^{-1}	8.50×10^{8}	4.55×10^{13}	1.64×10^{9}	5.33×10^{8}	2.25×10^{-7}	0.00
6.00×10^{-1}	4.92×10^{8}	2.26×10^{14}	2.69×10^{10}	3.37×10^{7}	3.25×10^{-7}	0.00
8.00×10^{-1}	2.91×10^{9}	2.37×10^{15}	1.60×10^{10}	1.86×10^{9}	4.00×10^{-7}	0.00
1.00	5.40×10^{9}	1.22×10^{14}	2.48×10^{9}	3.41×10^{9}	8.00×10^{-7}	0.00
1.33	1.54×10^{13}	1.95×10^{14}	7.97×10^{12}	9.90×10^{12}	1.00×10^{-6}	0.00
1.66	4.35×10^{12}	4.50×10^{13}	2.25×10^{12}	2.80×10^{12}	1.13×10^{-6}	0.00
2.00	2.35	1.52×10^{11}	1.49×10^{2}	2.15×10^{-2}	1.30×10^{-6}	0.00
2.50	1.03×10^{8}	5.17×10^{10}	5.34×10^{7}	6.64×10^{7}	1.77×10^{-6}	0.00
3.00	1.60×10^{5}	3.79×10^{9}	8.29×10^{4}	1.03×10^{5}	3.05×10^{-6}	0.00
4.00	9.43×10^{-10}	4.97×10^{8}	1.55×10^{-10}	5.19×10^{-10}	1.00×10^{-5}	0.00
5.00	0.00	2.82×10^{7}	0.00	0.00	3.00×10^{-5}	0.00
6.50	0.00	1.13×10^{7}	0.00	0.00	1.00×10^{-4}	0.00
8.00	0.00	2.22×10^{6}	0.00	0.00	5.50×10^{-4}	0.00
10.00	0.00	4.71×10^{5}	0.00	0.00	3.00×10^{-3}	0.00
Total	**2.0099×10^{13}**	**4.8590×10^{15}**	**1.0495×10^{13}**	**1.2923×10^{13}**	1.70×10^{-2}	0.00
					1.00×10^{-1}	0.00
					4.00×10^{-1}	3.16×10^{7}
					9.00×10^{-1}	1.61×10^{8}
					1.40	1.48×10^{8}
					1.85	1.09×10^{8}
					3.00	1.91×10^{8}
					6.43	1.74×10^{8}
					2.00×10^{1}	1.54×10^{7}
					TOTAL	**8.30×10^{8}**

NOTE: 4 wt % initial enrichment, 60 GWd/MTU burnup and 10 years cooling.

Table 1.10-20. Axial Source Terms Profile for a Typical PWR Fuel Assembly

Axial boundaries (from mid-plane) (cm)	Top		Bottom	
	Neutron	Gamma	Neutron	Gamma
0	1.554	1.117	1.554	1.117
11.43	1.537	1.114	1.571	1.12
22.86	1.521	1.111	1.588	1.123
34.29	1.504	1.108	1.605	1.126
45.72	1.486	1.104	1.622	1.129
57.15	1.464	1.1	1.636	1.131
68.58	1.438	1.095	1.648	1.133
80.01	1.401	1.088	1.657	1.135
91.44	1.35	1.078	1.654	1.134
102.87	1.277	1.063	1.625	1.129
114.3	1.165	1.039	1.554	1.117
125.73	0.998	0.9995	1.414	1.091
137.16	0.769	0.9365	1.172	1.041
148.59	0.492	0.8375	0.816	0.951
160.02	0.22	0.685	0.402	0.797
171.45	0.046	0.4625	0.092	0.551
182.88	0	0	0	0

NOTE: Source terms are relative power ratios with respect to one another and are unitless.

Table 1.10-21. Naval SNF Canister Gamma Source Spectrum

Upper Energy Boundary (MeV)	Side Surface Over Assembly Mid-Section (photons per $cm^2 \cdot s$)	Bottom Surface (photons per $cm^2 \cdot s$)	Top, Above Bolt Holes (photons per $cm^2 \cdot s$)	Top, 18 in. from Centerline (photons per $cm^2 \cdot s$)	Top, Above Outer Seal Plate (photons per $cm^2 \cdot s$)
3.85	2.11×10^1	0.00	0.00	0.00	0.00
3.35	1.29×10^3	0.00	0.00	0.00	0.00
2.95	8.67×10^3	2.24×10^2	1.23	2.91×10^{-2}	1.23
2.65	4.09×10^4	1.01×10^3	5.02	1.08×10^{-1}	5.23
2.35	2.33×10^6	5.20×10^4	1.93×10^2	3.21	2.20×10^2
2.03	5.04×10^5	2.10×10^4	1.31×10^2	2.77	1.49×10^2
1.77	4.31×10^6	8.19×10^4	2.31×10^2	3.65	3.05×10^2
1.57	9.45×10^5	5.53×10^4	1.76×10^2	3.00	2.45×10^2
1.43	4.23×10^7	4.70×10^5	5.63×10^2	1.77	9.43×10^2
1.31	4.50×10^7	6.16×10^5	7.74×10^2	6.86×10^{-1}	1.39×10^3
1.19	4.21×10^7	6.54×10^5	9.22×10^2	8.81	1.81×10^3
1.07	5.66×10^7	8.50×10^5	1.11×10^3	2.01×10^1	2.44×10^3
0.95	4.30×10^7	7.86×10^5	1.04×10^3	2.51×10^1	2.51×10^3
0.85	5.95×10^8	3.27×10^6	1.48×10^3	2.28×10^1	4.39×10^3
0.75	1.76×10^8	1.81×10^6	1.04×10^3	2.83×10^1	3.43×10^3
0.69	1.54×10^9	8.00×10^6	1.42×10^3	1.03×10^2	6.11×10^3
0.63	1.07×10^9	7.52×10^6	1.79×10^3	1.20×10^2	8.46×10^3
0.57	1.98×10^9	1.88×10^7	4.01×10^3	1.60×10^2	2.42×10^4
0.45	1.93×10^9	2.09×10^7	3.53×10^3	2.96×10^1	2.52×10^4
0.35	2.15×10^9	2.41×10^7	4.27×10^3	1.00×10^3	2.92×10^4
0.25	8.78×10^8	9.95×10^6	9.23×10^3	1.02×10^4	2.26×10^4
0.21	6.03×10^8	7.01×10^6	1.51×10^4	1.75×10^4	2.92×10^4
0.18	5.26×10^8	6.10×10^6	1.57×10^4	1.91×10^4	3.23×10^4
0.15	4.03×10^8	4.53×10^6	1.53×10^4	1.92×10^4	3.19×10^4
0.12	1.89×10^8	2.14×10^6	1.10×10^4	1.43×10^4	2.34×10^4
0.09	1.93×10^7	2.28×10^5	3.21×10^3	4.24×10^3	6.80×10^3
0.05	3.57×10^4	4.19×10^2	4.48×10^1	5.72×10^1	8.65×10^1
Total	**1.23×10^{10}**	**1.18×10^8**	**9.23×10^4**	**8.62×10^4**	**2.57×10^5**

Source: McKenzie 2007, Enclosure 1, Table 1.

Table 1.10-22. Naval SNF Canister Neutron Source Spectrum

Upper Energy Boundary (MeV)	Side Surface Over Assembly Mid-Section (neutrons per $cm^2 \cdot s$)	Bottom Surface (neutrons per $cm^2 \cdot s$)	Top, Above Bolt Holes (neutrons per $cm^2 \cdot s$)	Top, 18 in. from Centerline (neutrons per $cm^2 \cdot s$)	Top, Above Outer Seal Plate (neutrons per $cm^2 \cdot s$)
21.17	9.96×10^{-2}	4.08×10^{-3}	3.73×10^{-4}	6.01×10^{-5}	2.63×10^{-4}
12.84	6.15×10^{-1}	2.28×10^{-2}	1.87×10^{-3}	2.89×10^{-4}	1.38×10^{-3}
10	2.58	8.79×10^{-2}	6.40×10^{-3}	9.62×10^{-4}	4.95×10^{-3}
7.79	7.37	2.16×10^{-1}	1.32×10^{-2}	1.85×10^{-3}	1.10×10^{-2}
6.07	1.63×10^{1}	4.53×10^{-1}	2.47×10^{-2}	3.39×10^{-3}	2.16×10^{-2}
4.72	7.13×10^{1}	2.17	1.19×10^{-1}	1.75×10^{-2}	1.01×10^{-1}
2.86	1.69×10^{2}	8.03	5.56×10^{-1}	9.95×10^{-2}	4.12×10^{-1}
1.74	5.53×10^{2}	7.06×10^{1}	9.96	2.78	5.27
8.2085×10^{-1}	7.76×10^{2}	2.51×10^{2}	7.77×10^{1}	2.91×10^{1}	2.95×10^{1}
3.8774×10^{-1}	6.36×10^{2}	3.42×10^{2}	1.56×10^{2}	6.71×10^{1}	5.26×10^{1}
1.8316×10^{-1}	4.15×10^{2}	2.43×10^{2}	1.35×10^{2}	5.59×10^{1}	4.12×10^{1}
6.738×10^{-2}	2.81×10^{2}	1.47×10^{2}	1.12×10^{2}	3.82×10^{1}	2.96×10^{1}
5.530×10^{-3}	5.76×10^{1}	3.31×10^{1}	4.40×10^{1}	1.14×10^{1}	7.52
2.260×10^{-5}	2.59	2.62	4.27	1.10	7.47×10^{-1}
6.250×10^{-7}	5.06×10^{-2}	1.49×10^{-2}	2.72×10^{-2}	6.33×10^{-3}	4.40×10^{-3}
Total	$\mathbf{2.99 \times 10^{3}}$	$\mathbf{1.10 \times 10^{3}}$	$\mathbf{5.40 \times 10^{2}}$	$\mathbf{2.06 \times 10^{2}}$	$\mathbf{1.67 \times 10^{2}}$

Source: McKenzie 2007, Enclosure 1, Table 2.

Table 1.10-23. Homogenized TRIGA-FLIP Fuel Gamma Source Terms

Upper Energy Boundary (MeV)	**Intensity (photons per second)**
0.02	3.85×10^{13}
0.03	8.61×10^{12}
0.05	9.14×10^{12}
0.07	7.92×10^{12}
0.10	5.52×10^{12}
0.15	6.77×10^{12}
0.30	4.57×10^{12}
0.45	2.36×10^{12}
0.70	1.84×10^{13}
1.00	1.15×10^{13}
1.50	1.15×10^{13}
2.00	5.82×10^{10}
2.50	2.28×10^{11}
3.00	1.05×10^{9}
4.00	1.12×10^{8}
8.00	2.61×10^{3}
11.00	3.00×10^{2}
14.00	3.45×10^{1}
TOTAL	$\mathbf{1.25 \times 10^{14}}$

Source: DOE 1999.

Table 1.10-24. Savannah River Site HLW Source Term

Gamma		Neutron	
Upper Energy Boundary (MeV)	**Intensity (photons per second)**	**Upper Energy Boundary (MeV)**	**Intensity (neutrons per second)**
0.01 – 0.05	6.51×10^{14}	0.017 – 0.10	1.35×10^{5}
0.10	1.78×10^{14}	0.40	2.51×10^{6}
0.20	1.18×10^{14}	0.90	1.05×10^{7}
0.30	3.67×10^{13}	1.40	1.04×10^{7}
0.40	2.57×10^{13}	1.85	8.30×10^{6}
0.60	1.80×10^{13}	3.00	2.48×10^{7}
0.80	1.31×10^{15}	6.43	3.08×10^{7}
1.00	5.37×10^{12}	20.0	8.07×10^{5}
1.33	8.39×10^{12}	**Total**	**8.83×10^{7}**
1.66	1.35×10^{12}		
2.00	6.55×10^{10}		
2.50	3.36×10^{9}		
3.00	1.84×10^{7}		
4.00	4.35×10^{6}		
5.00	1.47×10^{6}		
6.50	5.88×10^{5}		
8.00	1.15×10^{5}		
10.00	2.45×10^{4}		
Total	**2.36×10^{15}**		

Table 1.10-25. Hanford HLW Source Term

Gamma		Neutron	
Upper Energy Boundary (MeV)	**Intensity (photons per second)**	**Upper Energy Boundary (MeV)**	**Intensity (neutrons per second)**
0.01 – 0.05	1.2921×10^{15}	0.017 – 0.10	4.244×10^{4}
0.10	3.8720×10^{14}	0.40	2.854×10^{5}
0.20	2.5791×10^{14}	0.90	7.427×10^{5}
0.30	8.2278×10^{13}	1.40	9.194×10^{5}
0.40	5.9095×10^{13}	1.85	9.064×10^{5}
0.60	4.0147×10^{13}	3.00	4.959×10^{6}
0.80	1.7860×10^{15}	6.43	6.558×10^{6}
1.00	7.9800×10^{12}	20.0	5.744×10^{3}
1.33	4.6310×10^{12}	**Total**	**1.442×10^{7}**
1.66	9.0308×10^{11}		
2.00	1.5150×10^{11}		
2.50	7.7499×10^{9}		
3.00	6.1385×10^{6}		
4.00	1.2262×10^{4}		
5.00	3.7649×10^{3}		
6.50	1.4177×10^{3}		
8.00	2.6174×10^{2}		
10.00	5.3193×10^{1}		
Total	**3.9184×10^{15}**		

Table 1.10-26. Maximum Activity for Pool Water Treatment System Filters

Nuclide	Maximum Filter Activity (Ci/filter)	Maximum Filter Activity (Ci/m^3)
^{39}Ar	2.59×10^{-10}	1.42×10^{-8}
^{134}Cs	1.36	7.47×10^{1}
^{135}Cs	2.67×10^{-5}	1.47×10^{-3}
^{137}Cs	4.00	2.20×10^{2}
^{3}H	7.71×10^{-3}	4.24×10^{-1}
^{129}I	3.46×10^{-5}	1.90×10^{-3}
^{85}Kr	5.38×10^{-2}	2.95
^{87}Rb	3.91×10^{-10}	2.15×10^{-8}
^{220}Rn	1.45×10^{-7}	7.99×10^{-6}
^{125}Sb	3.03×10^{-2}	1.66
^{126}Sb	1.52×10^{-6}	8.35×10^{-5}
^{126m}Sb	1.09×10^{-5}	5.97×10^{-4}
^{54}Mn	1.33×10^{-1}	7.30
^{55}Fe	1.35×10^{2}	7.42×10^{3}
^{60}Co	1.73×10^{1}	9.52×10^{2}
^{63}Ni	1.08×10^{-1}	5.93
^{65}Zn	8.52×10^{-3}	4.68×10^{-1}

NOTE: Based on the annually removed radionuclide inventory divided by the estimated number of filters to be used on a yearly basis.

Table 1.10-27. Maximum Radionuclide Concentration for Pool Water Treatment System Ion Exchanger Resin

Nuclide	Maximum Resin Concentration (Ci/m^3)
^{39}Ar	1.27×10^{-9}
^{134}Cs	6.65×10^{0}
^{135}Cs	1.31×10^{-4}
^{137}Cs	1.96×10^{1}
^{3}H	3.78×10^{-2}
^{129}I	1.69×10^{-4}
^{85}Kr	2.63×10^{-1}
^{87}Rb	1.91×10^{-9}
^{220}Rn	7.12×10^{-7}
^{125}Sb	1.48×10^{-1}
^{126}Sb	7.44×10^{-6}
^{126m}Sb	5.32×10^{-5}
^{54}Mn	6.51×10^{-1}
^{55}Fe	6.62×10^{2}
^{60}Co	8.49×10^{1}
^{63}Ni	5.29×10^{-1}
^{65}Zn	4.17×10^{-2}

Table 1.10-28. Gamma Intensity for Pool Water Treatment System Filters

Upper Energy Boundary (MeV)	Gamma Intensity (photons per second)
5.00×10^{-2}	4.0928×10^{10}
1.00×10^{-1}	6.4445×10^{9}
2.00×10^{-1}	2.2678×10^{9}
3.00×10^{-1}	3.0504×10^{8}
4.00×10^{-1}	1.6932×10^{8}
6.00×10^{-1}	4.4724×10^{10}
8.00×10^{-1}	1.6790×10^{11}
1.00	2.5668×10^{10}
1.33	1.0158×10^{12}
1.66	2.8784×10^{11}
2.00	1.6934×10^{-2}
2.50	6.7975×10^{6}
3.00	1.0542×10^{4}
4.00	8.5598×10^{-10}
5.00	2.1650×10^{-10}
6.50	6.2383×10^{-11}
8.00	7.9347×10^{-12}
10.00	1.0589×10^{-12}
Total	$\mathbf{1.5921 \times 10^{12}}$

Table 1.10-29. Gamma Intensity for Pool Water Treatment System Ion Exchanger Resin

Upper Energy Boundary (MeV)	**Gamma Intensity (photons per second)**
5.00×10^{-2}	2.0063×10^{11}
1.00×10^{-1}	3.1596×10^{10}
2.00×10^{-1}	1.1113×10^{10}
3.00×10^{-1}	1.4941×10^{9}
4.00×10^{-1}	8.2939×10^{8}
6.00×10^{-1}	2.1869×10^{11}
8.00×10^{-1}	8.2222×10^{11}
1.00	1.2553×10^{11}
1.33	4.9850×10^{12}
1.66	1.4125×10^{12}
2.00	8.3131×10^{-2}
2.50	3.3359×10^{7}
3.00	5.1726×10^{4}
4.00	4.2031×10^{-9}
5.00	1.0631×10^{-9}
6.50	3.0632×10^{-10}
8.00	3.8962×10^{-11}
10.00	5.1997×10^{-12}
Total	$\mathbf{7.8097 \times 10^{12}}$

Table 1.10-30. Maximum Expected Activity for WHF HEPA Filters

Nuclide	Maximum Activity (Ci/Filter)	Maximum Concentration (Ci/m^3)	Nuclide	Maximum Activity (Ci/Filter)	Maximum Concentration (Ci/m^3)	Nuclide	Maximum Activity (Ci/Filter)	Maximum HEPA Concentration (Ci/m^3)
^{241}Am	3.72×10^{-4}	3.28×10^{-3}	^{154}Eu	7.43×10^{-4}	6.56×10^{-3}	^{242}Pu	5.29×10^{-7}	4.67×10^{-6}
^{242}Am	2.29×10^{-6}	2.02×10^{-5}	^{155}Eu	1.56×10^{-4}	1.37×10^{-3}	^{106}Ru	0.00	0.00
^{242m}Am	2.30×10^{-6}	2.03×10^{-5}	^{55}Fe	3.29×10^{-2}	2.91×10^{-1}	^{125}Sb	1.23×10^{-4}	1.08×10^{-3}
^{243}Am	7.25×10^{-6}	6.40×10^{-5}	^{3}H	0.00	0.00	^{79}Se	1.50×10^{-8}	1.32×10^{-7}
^{137m}Ba	1.80×10^{-2}	1.59×10^{-1}	^{129}I	0.00	0.00	^{151}Sm	7.72×10^{-5}	6.81×10^{-4}
^{14}C	1.33×10^{-7}	1.17×10^{-6}	^{85}Kr	0.00	0.00	^{126}Sn	1.25×10^{-7}	1.10×10^{-6}
^{113m}Cd	4.38×10^{-6}	3.87×10^{-5}	^{93m}Nb	1.08×10^{-7}	9.57×10^{-7}	^{90}Sr	1.29×10^{-2}	1.14×10^{-1}
^{144}Ce	2.29×10^{-5}	2.02×10^{-4}	^{94}Nb	1.99×10^{-11}	1.75×10^{-10}	^{99}Tc	2.94×10^{-6}	2.59×10^{-5}
^{36}Cl	0.00	0.00	^{237}Np	7.97×10^{-8}	7.04×10^{-7}	^{230}Th	2.03×10^{-11}	1.79×10^{-10}
^{242}Cm	1.90×10^{-6}	1.68×10^{-5}	^{239}Np	7.25×10^{-6}	6.40×10^{-5}	^{232}U	7.69×10^{-9}	6.78×10^{-8}
^{243}Cm	4.95×10^{-6}	4.37×10^{-5}	^{231}Pa	9.45×10^{-12}	8.34×10^{-11}	^{233}U	7.75×10^{-12}	6.84×10^{-11}
^{244}Cm	8.16×10^{-4}	7.20×10^{-3}	^{107}Pd	2.72×10^{-8}	2.41×10^{-7}	^{234}U	1.89×10^{-7}	1.67×10^{-6}
^{245}Cm	1.06×10^{-7}	9.37×10^{-7}	^{147}Pm	2.00×10^{-3}	1.77×10^{-2}	^{235}U	2.41×10^{-9}	2.13×10^{-8}
^{246}Cm	3.65×10^{-8}	3.23×10^{-7}	^{144}Pr	2.29×10^{-5}	2.02×10^{-4}	^{236}U	5.70×10^{-8}	5.03×10^{-7}
^{60}Co	2.66×10^{-3}	2.35×10^{-2}	^{238}Pu	8.73×10^{-4}	7.70×10^{-3}	^{238}U	4.63×10^{-8}	4.09×10^{-7}
^{134}Cs	0.00	0.00	^{239}Pu	5.67×10^{-5}	5.01×10^{-4}	^{90}Y	1.29×10^{-2}	1.14×10^{-1}
^{135}Cs	0.00	0.00	^{240}Pu	1.01×10^{-4}	8.90×10^{-4}	^{93}Zr	2.63×10^{-7}	2.32×10^{-6}
^{137}Cs	0.00	0.00	^{241}Pu	1.64×10^{-2}	1.45×10^{-1}	—	—	—

NOTE: The maximum expected activity for WHF HEPA filters is based on a 10-month changeout frequency.

Table 1.10-31. Gamma Intensity for WHF HEPA Filters

Upper Energy Boundary (MeV)	Gamma Intensity (photons per second)	Upper Energy Boundary (MeV)	Neutron Intensity (neutrons per second)
5.00×10^{-2}	2.3451×10^{8}	1.00×10^{-1}	0
1.00×10^{-1}	8.0649×10^{7}	4.00×10^{-1}	4.412
2.00×10^{-1}	6.2210×10^{7}	9.00×10^{-1}	2.254×10^{1}
3.00×10^{-1}	1.8788×10^{7}	1.40	2.066×10^{1}
4.00×10^{-1}	1.2448×10^{7}	1.85	1.525×10^{1}
6.00×10^{-1}	1.2558×10^{7}	3.00	2.703×10^{1}
8.00×10^{-1}	1.2493×10^{7}	6.43	2.445×10^{1}
1.00	9.6788×10^{6}	2.00×10^{1}	2.154
1.33	1.7166×10^{8}	**Total**	$\mathbf{1.165 \times 10^{2}}$
1.66	4.5001×10^{7}		
2.00	3.3404×10^{4}		
2.50	9.2333×10^{3}		
3.00	2.0712×10^{1}		
4.00	1.1933×10^{1}		
5.00	4.0280		
6.50	1.6165		
8.00	3.1708×10^{-1}		
10.00	6.7321×10^{-2}		
Total	$\mathbf{6.6005 \times 10^{8}}$		

NOTE: The gamma intensity for WHF HEPA filters documented in this table is based on Table 1.10-30.

Table 1.10-32. Maximum Radionuclide Activity for Each Low-Level Waste Staging Area

Nuclide	Maximum LLWF Activity (Ci)
^{39}Ar	5.39×10^{-8}
^{134}Cs	2.83×10^{2}
^{135}Cs	5.56×10^{-3}
^{137}Cs	8.33×10^{2}
^{3}H	1.60
^{129}I	7.20×10^{-3}
^{85}Kr	1.12×10^{1}
^{87}Rb	8.13×10^{-8}
^{220}Rn	3.02×10^{-5}
^{125}Sb	6.29
^{126}Sb	3.16×10^{-4}
^{126m}Sb	2.26×10^{-3}
^{54}Mn	2.76×10^{1}
^{55}Fe	2.81×10^{4}
^{60}Co	3.61×10^{3}
^{63}Ni	2.25×10^{1}
^{65}Zn	1.77

NOTE: The maximum radionuclide activity for each low-level waste staging area documented in this table utilizes a homogenized source based on 208 pool water treatment system filter cartridges.

Table 1.10-33. Gamma Intensity for Each Low-Level Waste Staging Area

Upper Energy Boundary (MeV)	Gamma Intensity (photons per second)
5.00×10^{-2}	8.5295×10^{12}
1.00×10^{-1}	1.3434×10^{12}
2.00×10^{-1}	4.7249×10^{11}
3.00×10^{-1}	6.3525×10^{10}
4.00×10^{-1}	3.5263×10^{10}
6.00×10^{-1}	9.3062×10^{12}
8.00×10^{-1}	3.4957×10^{13}
1.00	5.3385×10^{12}
1.33	2.1196×10^{14}
1.66	6.0061×10^{13}
2.00	3.5264
2.50	1.4184×10^{9}
3.00	2.1997×10^{6}
4.00	1.7828×10^{-7}
5.00	4.5092×10^{-8}
6.50	1.2993×10^{-8}
8.00	1.6526×10^{-9}
10.00	2.2055×10^{-10}
Total	$\mathbf{3.3207 \times 10^{14}}$

NOTE: The gamma intensity for each low-level waste staging area documented in this table is based on Table 1.10-32.

Table 1.10-34. Gamma Intensity for Liquid Low-Level Waste Collection Tank

Upper Energy Boundary (MeV)	Gamma Intensity (photons per second)
5.00×10^{-2}	7.9210×10^{8}
1.00×10^{-1}	9.6575×10^{7}
2.00×10^{-1}	3.8690×10^{7}
3.00×10^{-1}	6.3953×10^{6}
4.00×10^{-1}	2.4239×10^{6}
6.00×10^{-1}	6.4073×10^{5}
8.00×10^{-1}	4.2147×10^{9}
1.00	2.6311×10^{5}
1.33	5.5204×10^{9}
1.66	1.5590×10^{9}
2.00	0
2.50	3.6996×10^{4}
3.00	5.7366×10^{1}
4.00	0
5.00	0
6.50	0
8.00	0
10.00	0
Total	$\mathbf{1.2231 \times 10^{10}}$

Table 1.10-35. Summary of Geologic Repository Operations Area Shielding Results

Area	Component	Source	Distance from Source to Shielding (ft)	Evaluated Shielding Thickness	Shielding Material	Radiation Zone
Cask Receipt Security Station	Walls	Transportation Cask (radial)	0	22 in.	Concrete	R2
			0	29 in.	Concrete	R1
Transient	Walls	Transportation Cask (radial)	50	12.5 in.	Concrete	R2
			75	9 in.	Concrete	R2
			100	7 in.	Concrete	R2
			150	3.5 in.	Concrete	R2
Aging Overpack	Radial Overpack Design	TAD Canister (maximum source, radial)	0	1.25 in.	Steel	40 mrem/hr contact
			0	37.5 in.	Concrete	40 mrem/hr contact

Table 1.10-36. Summary of Geologic Repository Operations Area Offset Results

Area	Source	Offset Distance	Dose Rate mrem/hr
33A/33B (Railcar and Truck Buffer Areas)	25 Transportation Casks (radial)	165 ft	0.25
		411 ft	0.05
Transient	1 Transportation Cask (radial)	70 ft	0.25
		160 ft	0.05
	2 Transportation Casks (radial)	90 ft	0.25
		260 ft	0.05
	3 Transportation Casks (radial)	100 ft	0.25
		320 ft	0.05
Aging Overpack	Aging Overpack (maximum source, radial)	94 ft	0.25
	Aging Overpack (design basis source, radial)	58 ft	0.25
		132 ft	0.05
	Aging Overpack (average source, radial)	18 ft	0.25
		50 ft	0.05
Aging Facility	208 Aging Overpacks (maximum source, radial)	493 ft	0.25
		821 ft	0.05

NOTE: All cases evaluated through air.

Table 1.10-37. Summary of IHF Shielding Results

Room	Component	Source	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1008 (Cask Unloading Room)	Shield Door	Naval Long Canister (radial)	11.5 in. Steel	R4
	Walls	Naval Long Canister (radial)	48 in. Concrete	R2 (0.25 mrem/hr)
	Port Slide Gate	Savannah River Site HLW Canister (top axial)	8 in. Steel	R4
2005 (Canister Transfer Area)	Canister Transfer Machine Slide Gate	Savannah River Site HLW Canister (bottom axial)	2 in. Steel	R4
	Canister Transfer Machine Radial Shielding	Naval Long Canister (radial)	11.5 in. Steel[a]	R4
1005 (Waste Package Loadout Room)	Walls	Naval Long Waste Package (radial) (from source centerline)	48 in. Concrete	R2 (0.25 mrem/hr)
	Roof	Naval Long Waste Package (radial) (from source centerline)	48 in. Concrete	R3
	Entrance	Naval Long Waste Package (bottom axial)	5 in. Steel	R2
	Exit	Naval Long Waste Package (top axial)	8 in. Steel	R2 (0.25 mrem/hr)
	Waste Package Transfer Trolley Radial Shielding	Naval Long Waste Package (radial)	11.5 in. Steel[a]	R4
1200 Series (Support Area)	South Wall	Naval Long Canister Transportation Cask (radial) (from source centerline)	12 in. Concrete	R1
Not applicable	Personnel Fence	Naval Long Canister Transportation Cask (radial) (from source centerline)	80 ft (offset) Air	R2 (0.25 mrem/hr)

NOTE: [a]Shielding thickness is based on the results for the IHF cask unloading room shield door.

Table 1.10-38. Summary of CRCF Shielding Results

Room	Component	Source	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1036	East Wall (to rooms 1040/1052)	Rail Cask	48 in. Concrete	R2
	West Wall (to rooms 1034/1035)		24 in. Concrete	
	Ceiling (to room 2012)		18 in. Concrete	
1026	North and South Personnel Entrances	Rail Cask	12 in. Concrete	R2
	Corridor 2006D Floor	Rail Cask	18 in. Concrete and 1.5 in. Steel	R2
1023/1024	Door to 1026	21-PWR/44-BWR TAD Canister	11 in. Steel and 5 in. Borated Polyethylene	R4
	West Wall (to corridors 1005E/1006)	21-PWR/44-BWR TAD Canister	48 in. Concrete	R2
2004	Canister Transfer Machine Radial Shielding	21-PWR/44-BWR TAD Canister	12 in. Steel and 8 in. Borated Polyethylene	R4
	Roof	Hanford Canister	18 in. Concrete	R3
	East Wall (to corridor 2006D)	21-PWR/44-BWR TAD Canister	48 in. Concrete	R2
	Port Slide Gate	5-DHLW/DOE Long and Short and 21-PWR/44-BWR TAD Waste Packages	8 in. Steel and 2 in. Borated Polyethylene	R4
	Canister Transfer Machine Slide Gate	21-PWR/44-BWR TAD Canister	8 in. Steel	R4
	Canister Slide Gate	Hanford Canister	7 in. Steel	R4
1017/1021 1022/1025	Walls to rooms 1023/1024/1018/1019	21-PWR/44-BWR TAD Canister	48 in. Concrete	R4
	Walls to corridors 1005D/1005E/1005F	21-PWR/44-BWR TAD Canister	48 in. Concrete and 5 in. Steel	R2 (0.25 mrem/hr)
1018/1019	WP Transfer Trolley Radial Shielding	21-PWR/44-BWR TAD Waste Package	9 in. Steel and 7.5 in. Borated Polyethylene	R4
	Door to 1015	21-PWR/44-BWR TAD Canister	10 in. Steel and 2 in. Borated Polyethylene	R4

Table 1.10-38. Summary of CRCF Shielding Results (Continued)

Room	Component	Source	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1015	Door to 1014	21-PWR/44-BWR TAD Waste Package	5 in. Steel and 5 in. Borated Polyethylene	R2
	Corridor 2006J Floor	21-PWR/44-BWR TAD Waste Package	24 in. Concrete and 1 in. Steel	R2
	North Wall (to Rooms 1007/1007A)	21-PWR/44-BWR TAD Waste Package	48 in. Concrete	R2
	Roof	21-PWR/44-BWR TAD Waste Package	18 in. Concrete	R3
	1st Floor Personnel Door	21-PWR/44-BWR TAD Waste Package	7 in. Steel and 2 in. Borated Polyethylene	R2
	2nd Floor Personnel Door	21-PWR/44-BWR TAD Waste Package	5 in. Steel and 5 in. Borated Polyethylene	R2

NOTE: For the 21-PWR/44-BWR TAD canister or waste package, only the PWR source term was evaluated because the PWR source term bounds the BWR source term.

Table 1.10-39. Summary of RF Shielding Results

Room	Component	Source[a]	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1021	West Wall (to Corridor 1003G)	Rail Cask	20 in. Concrete	R2 (0.25 mrem/hr)
	Ceiling (to Room 2012)			
1017	Personnel Entrances to Corridors 1003B and 1003E	Rail Cask	3.25 in. Steel 4.75 in. Borated Polyethylene	R2 (0.25 mrem/hr)
1015	Door to 1017	21-PWR/44-BWR TAD Canister	11 in. Steel and 5 in. Borated Polyethylene	R4
	North and West Wall (to Corridors 1003B/1003D/1003H)	21-PWR/44-BWR TAD Canister	48 in. Concrete	R2 (0.25 mrem/hr)
2007	Canister Transfer Machine Radial Shielding	21-PWR/44-BWR TAD Canister	12 in. Steel and 8 in. Borated Polyethylene	R4
	Port Slide Gate	Evaluated for CRCF[b]	8 in. Steel and 2 in. Borated Polyethylene	R4
1013	Door to 1002	21-PWR/44-BWR TAD Canister	11 in. Steel and 5 in. Borated Polyethylene	R4

NOTE: [a]For the 21-PWR/44-BWR TAD canister or waste package, only the PWR source term was evaluated because the PWR source term bounds the BWR source term.
[b]Shielding thicknesses are based on results for the CRCF using a bounding source term.

Table 1.10-40. Summary of the WHF Shielding Design

Room	Component	Source	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1001	Personnel shield door	Transportation cask	2 in. Stainless Steel and 2.5 in. Borated Polyethylene	R2 (0.25 mrem/hr)
	Transportation cask vestibule wall (dose rates into the support area as a transportation cask travels past the support area entrance (Rooms 1202 to 1204, 1035 to 1037, 1201, 1218A, and 1026)	Transportation cask	18 in. Concrete	R1
	Transportation cask vestibule equipment door (equipment door to the cask preparation area, Room 1016)	Transportation cask	No shielding is required	R2
1016	North, south, east and west walls around pool area	Transportation cask	25 in. Concrete	R2
	The personnel doors to south vestibule (1022/1218C)	Transportation cask	4 in. Stainless Steel and 11 in. Borated Polyethylene	R2 (0.25 mrem/hr)
	The personnel door to the west corridor (1045)	Transportation cask	2 in. Stainless Steel and 5 in. Borated Polyethylene	R2
	The personnel door to Cask Unloading Room (1009), Canister Transfer Machine Maintenance Room (1010) and Maintenance Room vestibules (1018)	Transportation Cask	1 in. Stainless Steel and 4 in. borated Polyethylene	R2
	TAD Canister Closure Station—axial (top of platform)	TAD canister	5 in. Stainless Steel and 4.5 in. Borated Polyethylene	R2
	TAD Canister Closure Station—radial (side shield walls)	TAD canister	1 in. Stainless Steel and 4 in. Borated Polyethylene	R2
	DPC Cutting Station—axial (top of platform)	Transportation cask	4 in. Stainless Steel and 5 in. Borated Polyethylene	R2
	DPC Cutting Station—radial (side shield walls)	Transportation cask	2 in. Stainless Steel and 5 in. Borated Polyethylene	R2

Table 1.10-40. Summary of the WHF Shielding Design (Continued)

Room	Component	Source	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1044A / 1044B / 1044C	Doors and walls	Ion exchanger	12 in. Concrete and 8.5 in. Stainless Steel, or 18 in. Concrete and 6.5 in. Stainless Steel, or 24 in. Concrete and 4.75 in. Stainless Steel	R2
	Corridor and ceiling	Ion exchanger	18 in. Concrete and 8.75 in. Stainless Steel, or 24 in. Concrete and 7 in. Stainless Steel	R2 (0.25 mrem/hr)
1007 / 1008	Shield door	TAD canister	10 in. Stainless Steel and 5 in. Borated Polyethylene	R4
	Shield walls	TAD canister	48 in. Concrete	R2
	Shield walls	Transportation cask	18 in. Concrete	R2 (0.25 mrem/hr)
1043A / 1043B / 1043C	Doors and walls	Pool water treatment system filter	12 in. Concrete and 6.5 in. Stainless Steel, or 18 in. Concrete and 4.5 in. Stainless Steel, or 24 in. Concrete and 2.75 in. Stainless Steel	R2
	Corridor and ceiling	Pool water treatment system filter	18 in. Concrete and 6.75 in. Stainless Steel, or 24 in. Concrete and 5 in. Stainless Steel, or 36 in. Concrete and 1.25 in. Stainless Steel	R2 (0.25 mrem/hr)
1045A / 1045B / 1045C / 1045D	Access doors (from pool area)	Transportation cask	2 in. Stainless Steel and 5 in. Borated Polyethylene	R2

Table 1.10-41. Summary of the LLWF Shielding Design

Room	Component	Source	Evaluated Shielding Thickness and Shielding Material	Radiation Zone
1008, 1009, 1010, 1011	LLWF staging room exterior walls	Homogenized source 50-ft × 30-ft × 6-ft, equivalent to 208 spent pool water treatment system filter cartridges	24 in. Concrete and 9.25 in.Stainless Steel, or 36 in. Concrete and 5.5 in. Stainless Steel	R2 (0.25 mrem/hr)
	Corridor (1007) between staging rooms	Homogenized source 50-ft × 30-ft × 6-ft, equivalent to 208 spent pool water treatment system filter cartridges	24 in. Concrete and 7 in. Stainless Steel, or 36 in. Concrete and 3.25 in. Stainless Steel	R2

Table 1.10-42. Repository Underground Layout Description

Component	Characteristic	Value / Unit
Emplacement drift	Excavation diameter	18 ft (5.49 m)
	Azimuth angle (Angle from access main)	252° (69°)
	Drift invert	4 ft 4 in. (1.32 m)
	Distance between adjacent emplacement drifts (center-to-center)	81 m (265 ft 9 in.)
	Springline elevation	0 cm
	Minimum distance between waste packages	0.1 m (3.94 in.)
	21-PWR/44-BWR TAD waste package centerline to concrete invert	46.93 in. (1.19 m)

Table 1.10-43. Repository Underground Structural and Shielding Materials

Material	Region	Density (g/cm^2)	wt %
Air Space	Dry Air	0.000889	N: 75.52 O: 23.18 C: 0.01 Ar: 1.29
Subsurface Surroundings Invert (Crushed Tuff Ballast)	Tuff	2.21 1.99[a]	H: 0.0716 C: 0.0027 F: 0.0400 Na: 2.6113 Mg: 0.0772 Al: 6.6421 Si: 35.6989 P: 0.0218 S: 0.0500 Cl: 0.0200 K: 4.0096 Ca: 0.3573 Ti: 0.0659 Mn: 0.0542 Fe: 0.7873 O: 49.4899

NOTE: [a]A density of 1.97 is used in the model.

Table 1.10-44. Transport and Emplacement Vehicle Dimensions

Characteristic	Dimension in. (cm)
Internal Width	91 (231)
Internal Length	270 (685)
Internal Height	102 (259)
Internal Side Wall Height	75.05 (191)
External Width	111 (282)
External Length	290 (736)
External Height	122 (310)

Table 1.10-45. Transport and Emplacement Vehicle Shielding Materials and Material Thicknesses

Shield Layer	Dimension – in. (cm)
Inner Stainless Steel	1.5 (3.8)
Depleted Uranium	1.5 (3.8)
Middle Stainless Steel	0.5 (1.27)
NS-4-FR	6 (15.2)
Outer Stainless Steel	0.5 (1.27)

NOTE: The transport and emplacement vehicle shielding material thicknesses documented in this table are based on the cited dimension, or equivalent.

Table 1.10-46. Transport and Emplacement Vehicle Shielding and Subsurface Materials

Material	Region(s) Used	Density (g/cm^3)	Element and wt %
NS-4-FR	TEV	1.68	H: 6.0 C: 27.7 N: 2.0 O: 42.2 ^{10}B: 0.11 Al: 21.5 ^{11}B: 0.49
Depleted Uranium	TEV	18.95	^{235}U: 0.2 ^{238}U: 99.8
Tuff	Host Rock	2.21	Si: 46.75 O:53.25 Note: Based on silicon oxide (SiO_2) compound
Dry Air	Atmosphere Inside TEV and Outside TEV	0.001204	N: 75.522 O: 23.177 C: 0.013 Ar: 1.288 Note: These wt % values are converted from vol %
Holtite-A	Potential TEV Shield Material	1.63	C: 27.66 H: 5.92 Al: 21.28 N: 1.98 ^{10}B: 0.14 O: 42.37 ^{11}B: 0.64
Borated Polyethylene	Potential TEV Shield Material	0.95	H: 11.6 O: 22.2 ^{10}B: 0.92 C: 61.2 ^{11}B: 4.08

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-111.

NOTE: Cask unloading rooms, waste package positioning rooms, and the waste package loadout room are classified as R5 areas and are very high radiation areas (>500 rads/hr at 1 m) during transfer and loadout operations. Canister staging areas are classified as R5 areas and are very high radiation areas (>500 rads/hr at 1 m) when canisters are being staged. Since the unshielded vestibules have an exterior metal structure, elevated dose rates may be present during transportation cask receipt or aging overpack receipt, or when the TEV is exiting, which may require access controls in accordance with 10 CFR Part 20.
LLW = low-level radioactive waste; PCM = personnel contamination monitor; RA = radiation area; WP = waste package.

Figure 1.10-1. Canister Receipt and Closure Facility Radiation Zones—1st Floor

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This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-112.

NOTE: WP = waste package.

Figure 1.10-2. Canister Receipt and Closure Facility Radiation Zones—2nd Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-113.

Figure 1.10-3. Canister Receipt and Closure Facility Radiation Zones—3rd Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-114.

NOTE: The cask unloading room, waste package loading room, and waste package loadout room are classified as R5 areas and are very high radiation areas (>500 rads/hr at 1 m) during transfer and loadout operations. The waste package positioning room is classified as an R5 area and is a high radiation area (≥100 mrem/hr at 30 cm) during closure operations.
CTM = canister transfer machine; IHF = Initial Handling Facility; LLW = low-level radioactive waste; MCC = motor control center: RP = radiation protection; WP = waste package.

Figure 1.10-4. Initial Handling Facility Radiation Zones—1st Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-115.

NOTE: WP = waste package.

Figure 1.10-5. Initial Handling Facility Radiation Zones—2nd Floor (Elevation 37′–0″)

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-116.

Figure 1.10-6. Initial Handling Facility Radiation Zones—3rd Floor (Elevation 73′–6″)

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-117.

NOTE: The cask unloading room and loading room are classified as R5 areas and are very high radiation areas (>500 rads/hr at 1 m) during transfer and loadout operations. Since the unshielded transportation cask vestibule annex has an exterior metal structure, elevated dose rates may be present during transportation cask receipt and handling operations, which may require access controls, in accordance with 10 CFR Part 20.
AO = aging overpack; CTM = canister transfer machine; LLW = low-level radioactive waste; PCM = personnel contamination monitor; RA = radiation area; STC = shielded transfer cask.

Figure 1.10-7. Receipt Facility Radiation Zones—1st Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-118.

Figure 1.10-8. Receipt Facility Radiation Zones—2nd Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-119.

Figure 1.10-9. Receipt Facility Radiation Zones—3rd Floor

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This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-120.

NOTE: Rooms are zoned R5 during cask, DPC, and fuel transfer operations in the pool due to the potential for elevated dose rates. Otherwise, the areas will be zoned accordingly. LLW = low-level radioactive waste.

Figure 1.10-10. Wet Handling Facility Radiation Zones—Basement

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-121.

NOTE: Cask unloading room and loading room are classified as R5 zones and are very high radiation areas (>500 rads/hr at 1 m) during transfer and loadout operations. Since the unshielded vestibules have an exterior metal structure, elevated dose rates may be present during transportation cask receipt or aging overpack receipt and handling operations, which may require access controls, in accordance with 10 CFR Part 20.
LLW = low level radioactive waste; STC = shielded transfer cask.

Figure 1.10-11. Wet Handling Facility Radiation Zones—1st Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-122.

Figure 1.10-12. Wet Handling Facility Radiation Zones—2nd Floor (Elevation 40′–0″)

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This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-123.

NOTE: Staging rooms(1008–1010) for LLW are classified as R5 zones and may be high radiation areas (≥100 mrem/hr at 30 cm), if higher sources are present. During operations, access to these areas is controlled based on radiation measurements. Corridor 1007 is zoned R2 since it is shielded from waste held in the staging rooms. The zone classification increases to R3 during waste movement into or out of a staging room. LLW = low-level radioactive waste.

Figure 1.10-13. Low-Level Waste Facility Radiation Zones—1st Floor

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-124.

Figure 1.10-14. Low-Level Waste Facility Radiation Zones—2nd Floor

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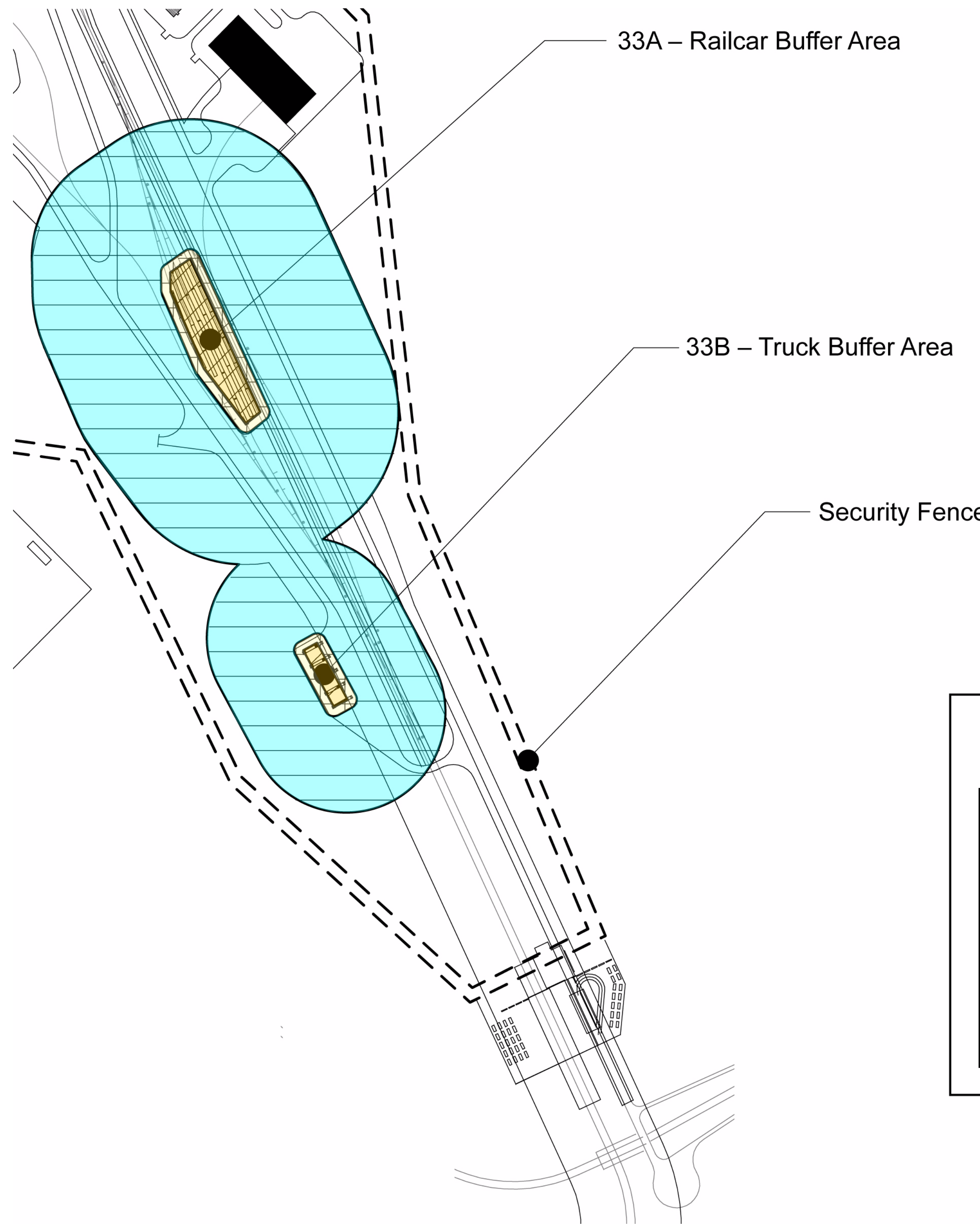

RADIATION ZONE LEGEND

SYMBOL	DESCRIPTION	(mrem/hr)
R1	Unlimited occupancy possible.	Background to <0.05
R2	Routine occupancy possible	0.05 to 2.5
R3	Occasional occupancy possible	>2.5 to 15
R4	Infrequent occupancy	>15 to 100
R5	Limited or no occupancy, access restricted, occupancy not normally allowed	>100

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NOTE: Areas outside of the R2 boundary are considered R1.

Figure 1.10-15. North Portal Operations Area Radiation Zones—Buffer Area

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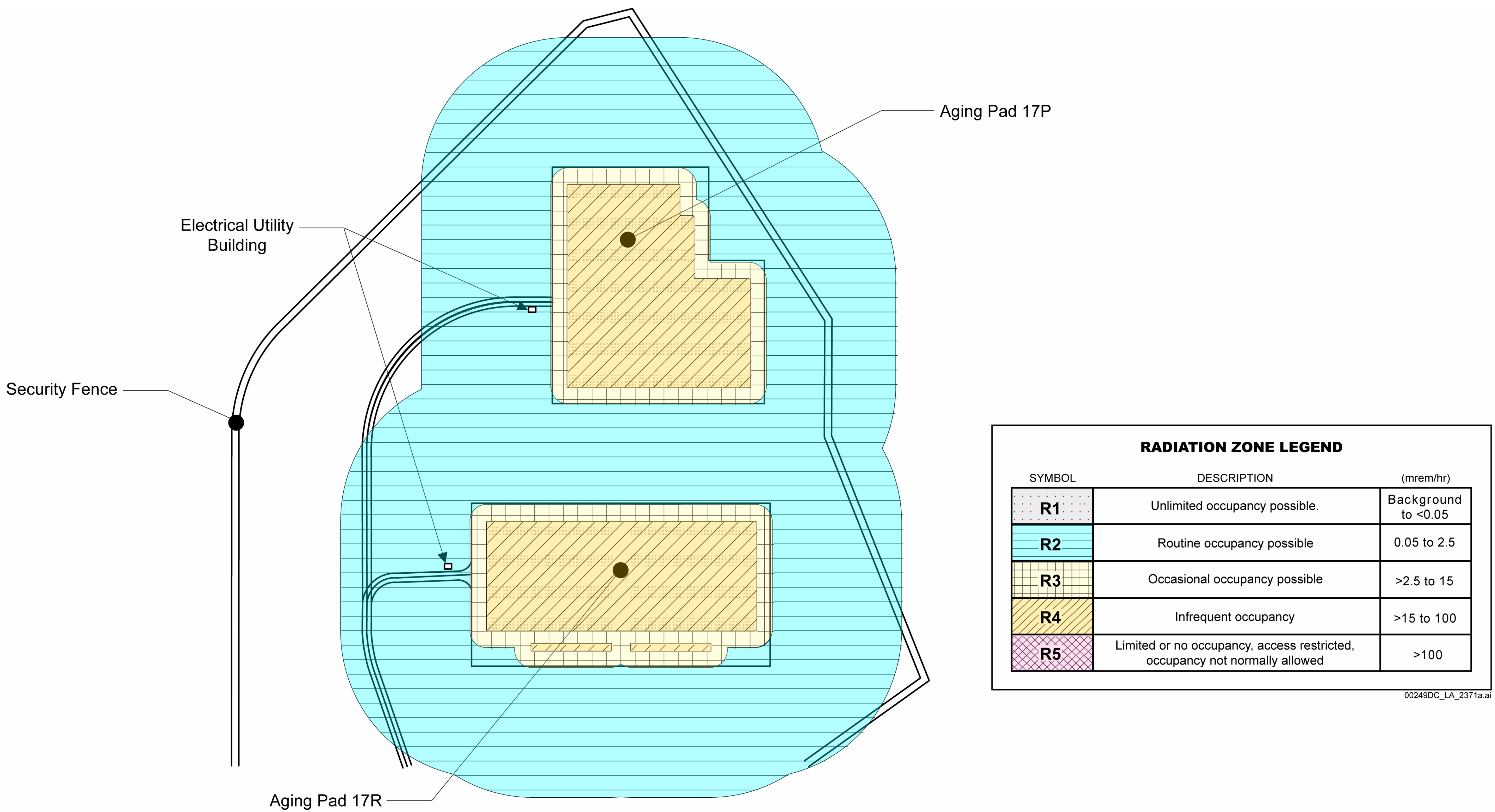

NOTE: Areas outside the R2 boundary are considered R1. Possible localized R5 areas around aging overpacks may occur during movements.

Figure 1.10-16. North Portal Operations Area Radiation Zones—Aging Facility

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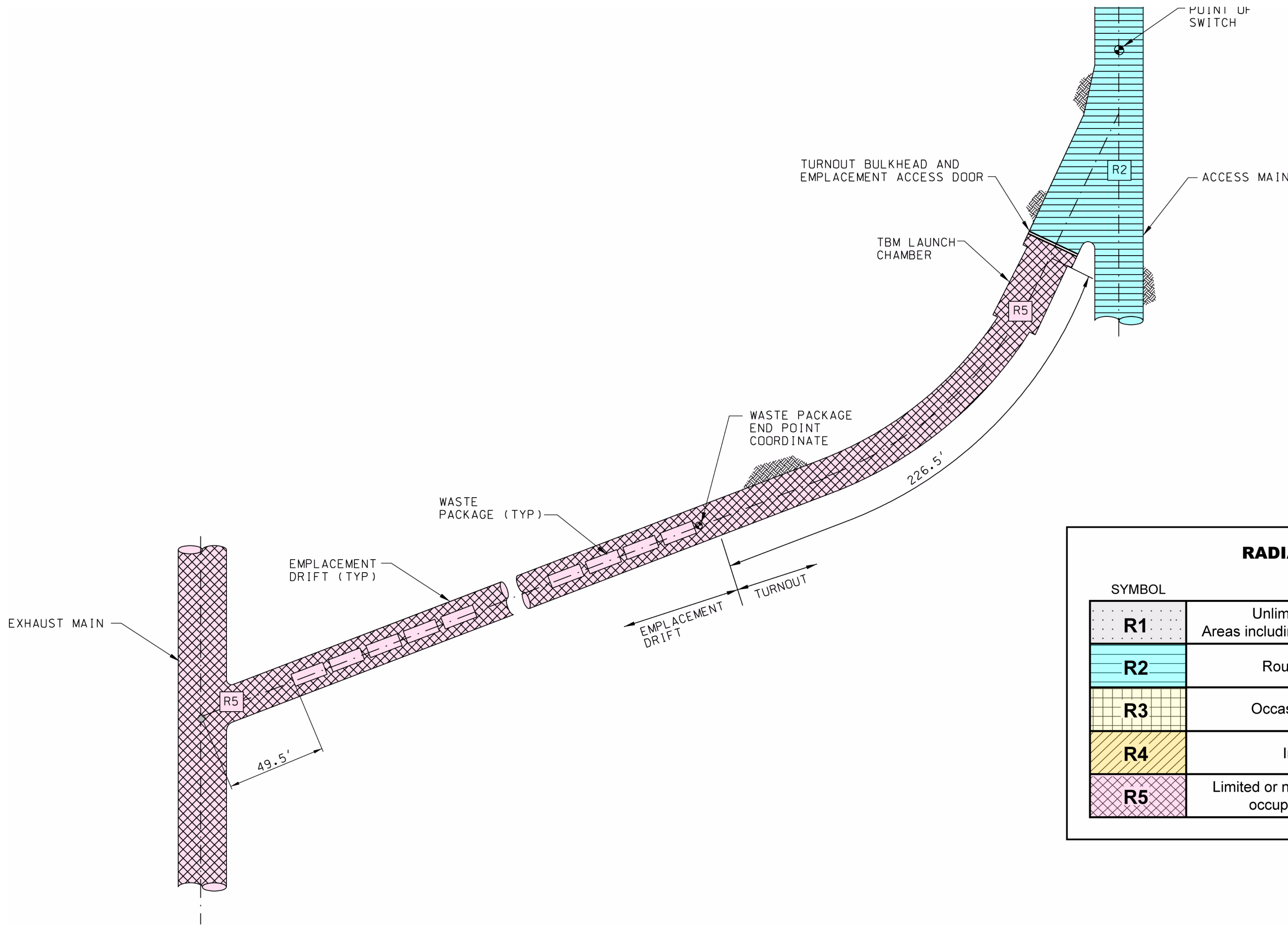

RADIATION ZONE LEGEND

SYMBOL	DESCRIPTION	(mrem/hr)
R1	Unlimited occupancy possible. Areas including those outside restricted areas.	Background to <0.05
R2	Routine occupancy possible	0.05 to 2.5
R3	Occasional occupancy possible	>2.5 to 15
R4	Infrequent occupancy	>15 to 100
R5	Limited or no occupancy, access restricted, occupancy not normally allowed	>100

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NOTE: This figure is representative of any configuration of access mains, emplacement drift turnouts, and exhaust mains. Access mains and ramps (up to and including emplacement drift turnout bulkhead/doors) and other nonemplacement openings not specifically identified are designated radiation zone R2 (occupancy 2,000 hr/yr possible). Dose rates will vary. Maximum dose rate occurs at the emplacement drift turnout bulkhead. The TEV creates a transient R4 radiation zone in the access mains and ramps (up to and including emplacement drift turnout bulkhead/doors) when emplacing or retrieving waste packages (TEV loaded with a waste package). Emplacement drifts with emplaced waste packages are designated very high radiation areas per 10 CFR 20.1602 within 1 m of the emplaced waste packages.
TBM = tunnel boring machine.

Figure 1.10-17. Subsurface Facilities Access Main Radiation Zones

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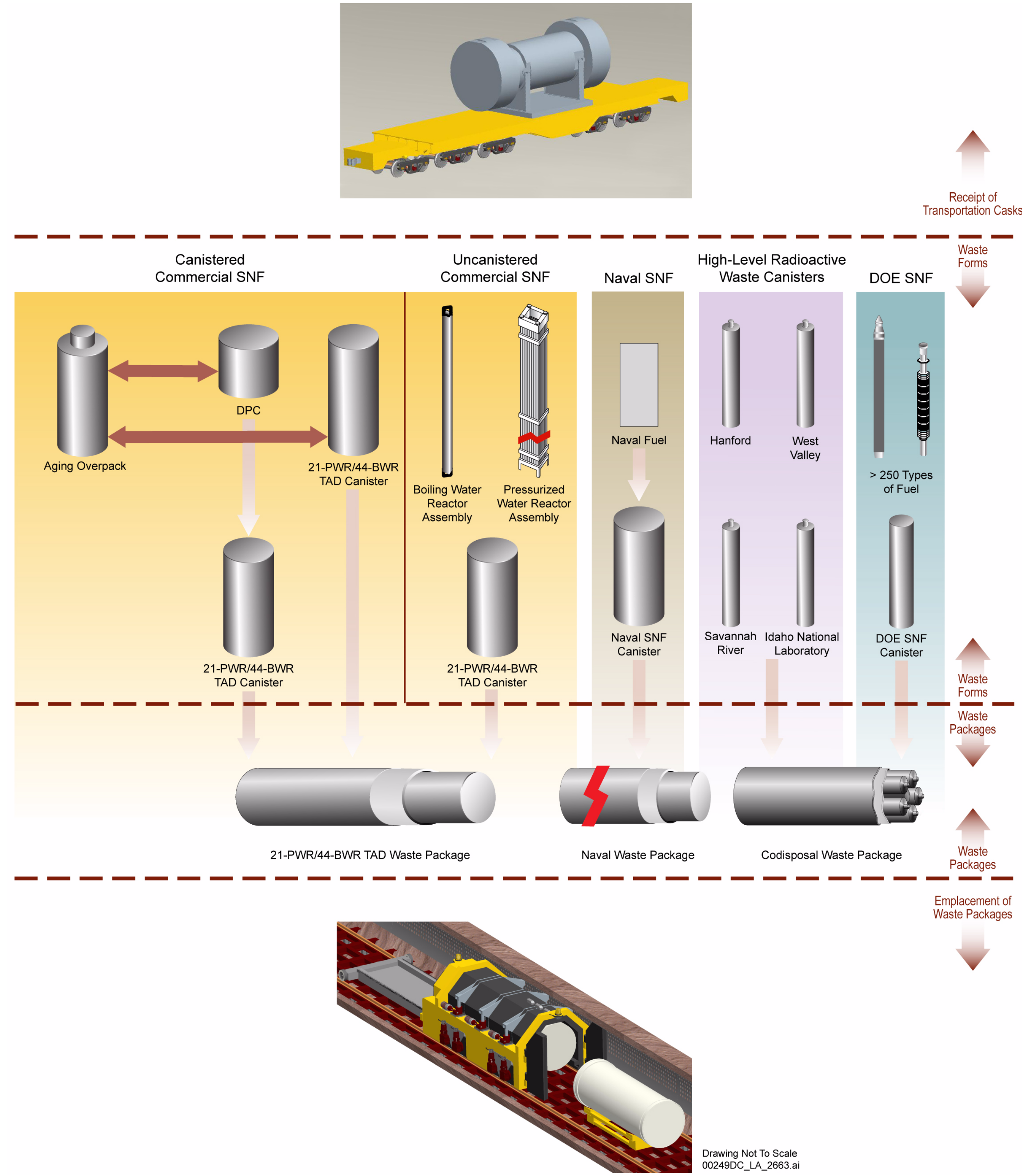

Figure 1.10-18. Summary of Radiation Sources

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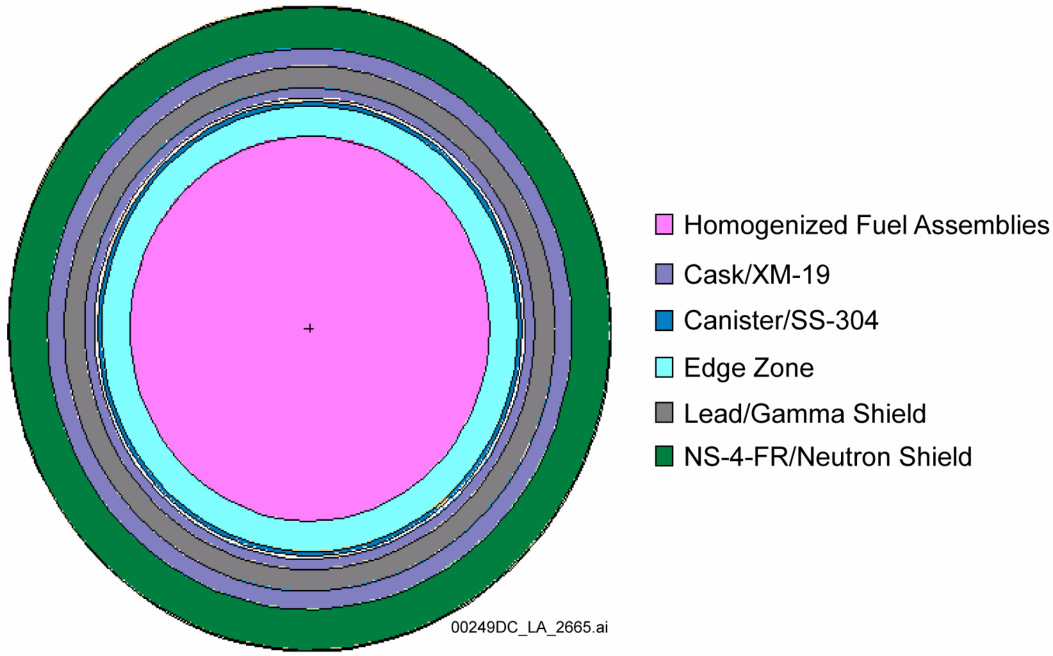

Figure 1.10-19. Transportation Cask and Canister Radial Configuration at Midplane

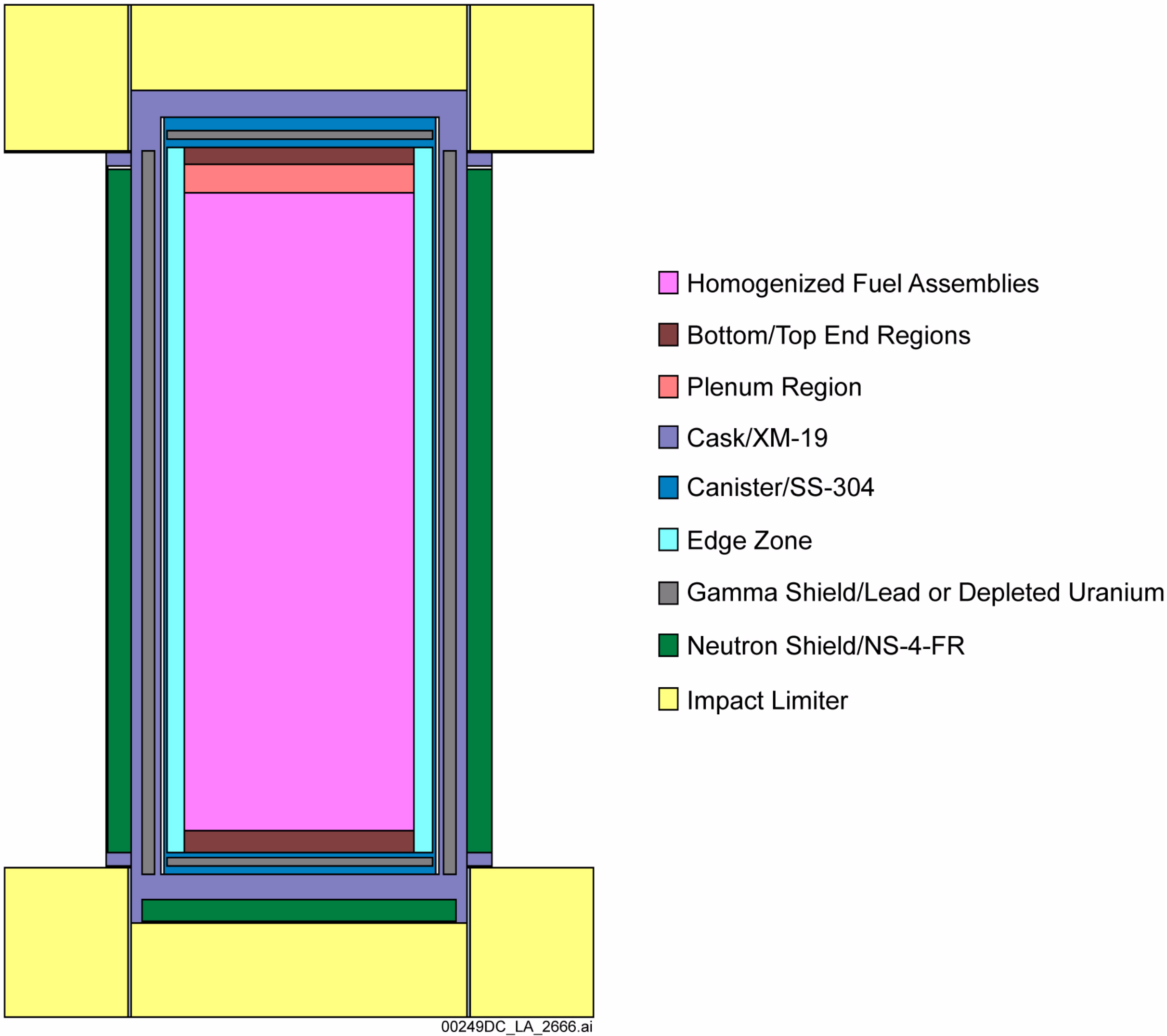

Figure 1.10-20. Transportation Cask Axial Configuration

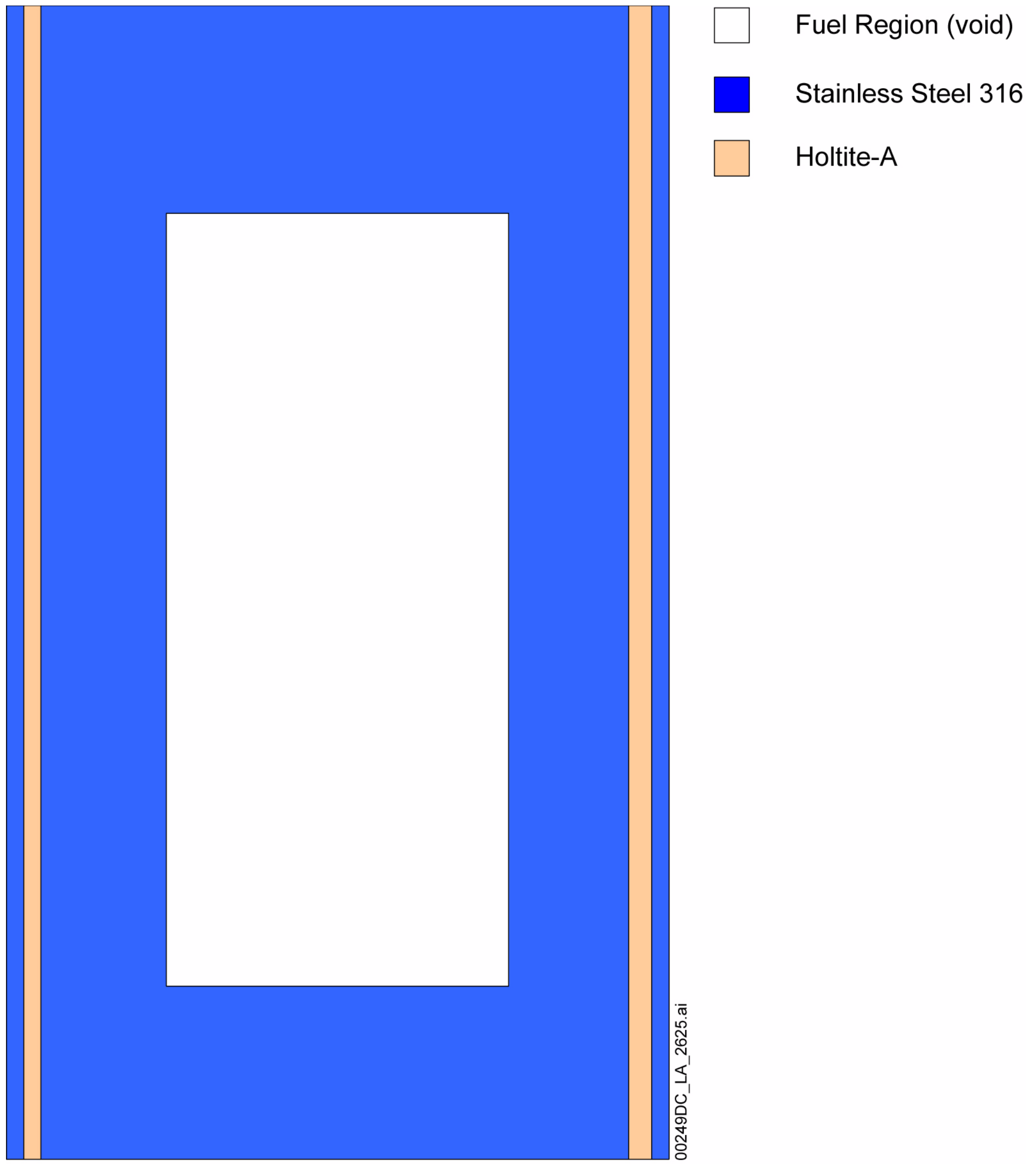

Figure 1.10-21. Axial Cross Section of Naval Canister with Transportation Overpack

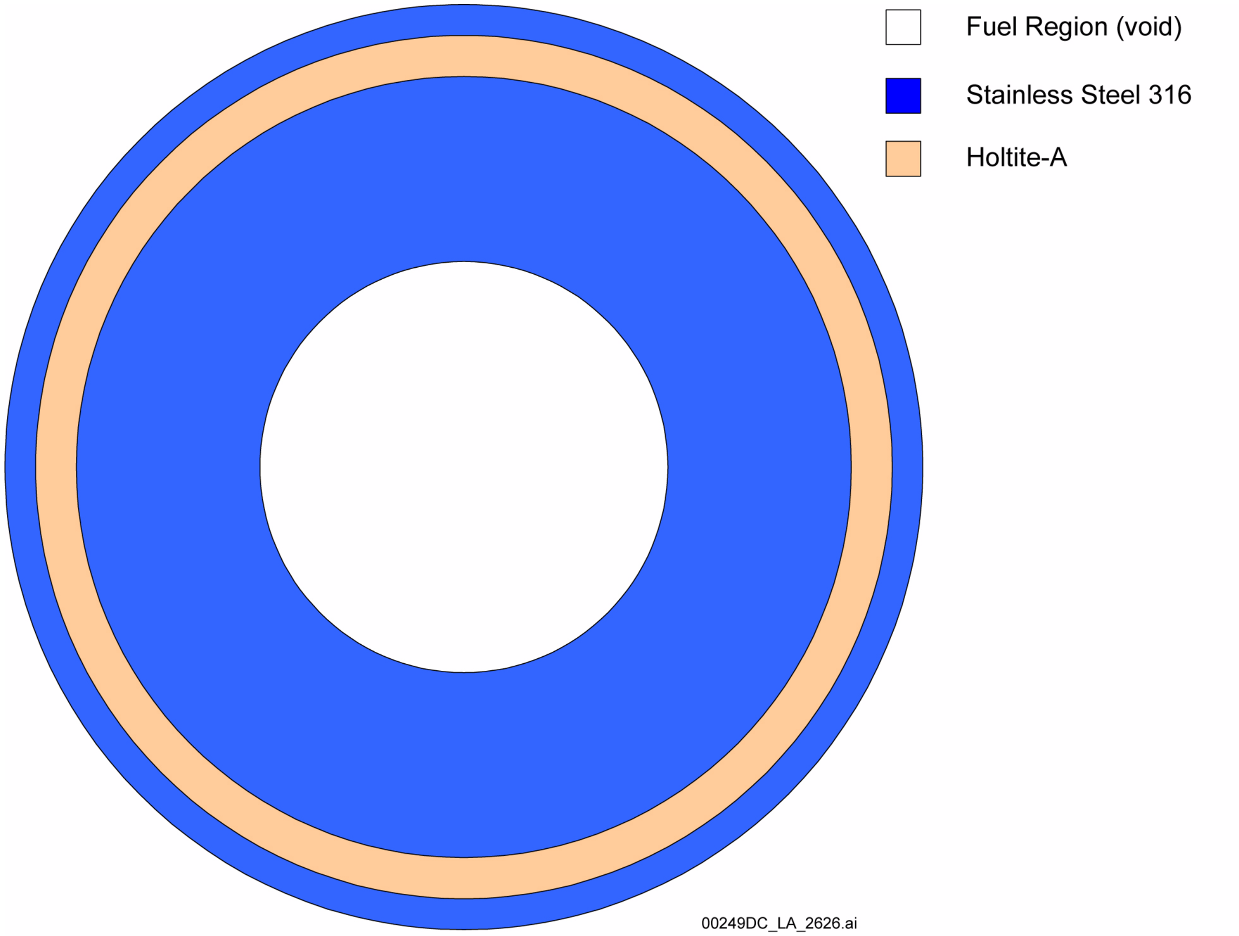

Figure 1.10-22. Radial Cross Section of Naval Canister with Transportation Overpack

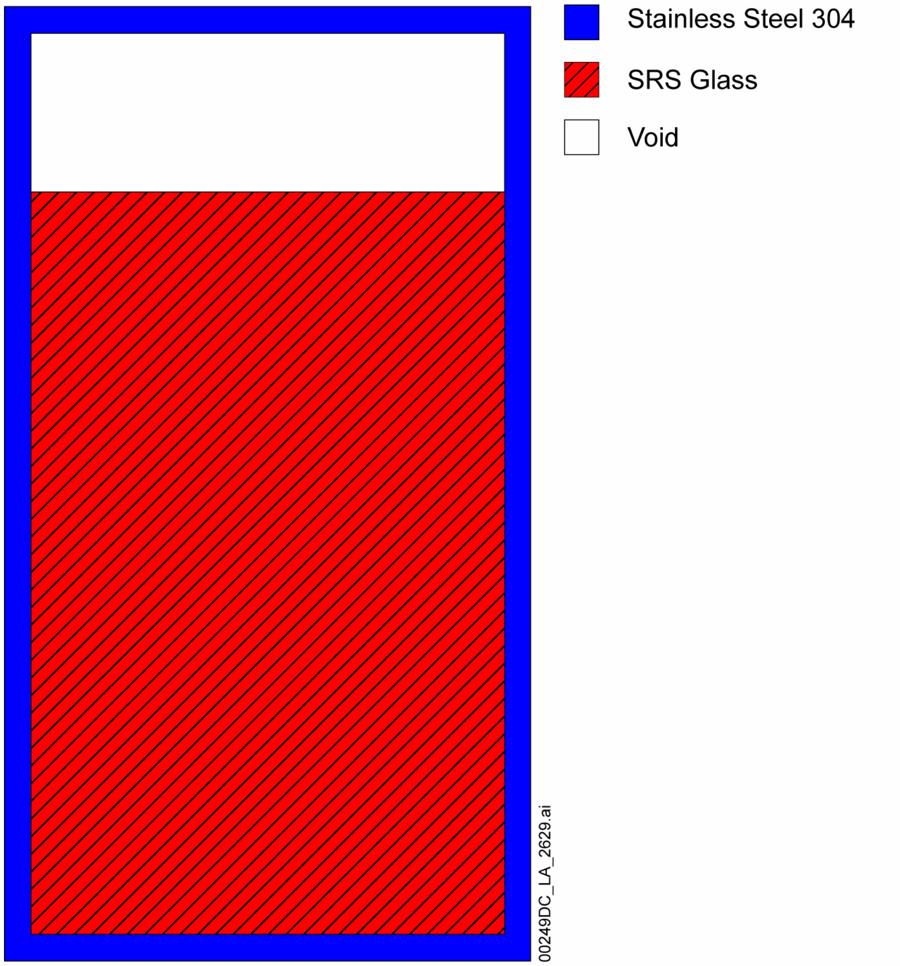

Figure 1.10-23. Axial Cross Section of Savannah River Site HLW Canister

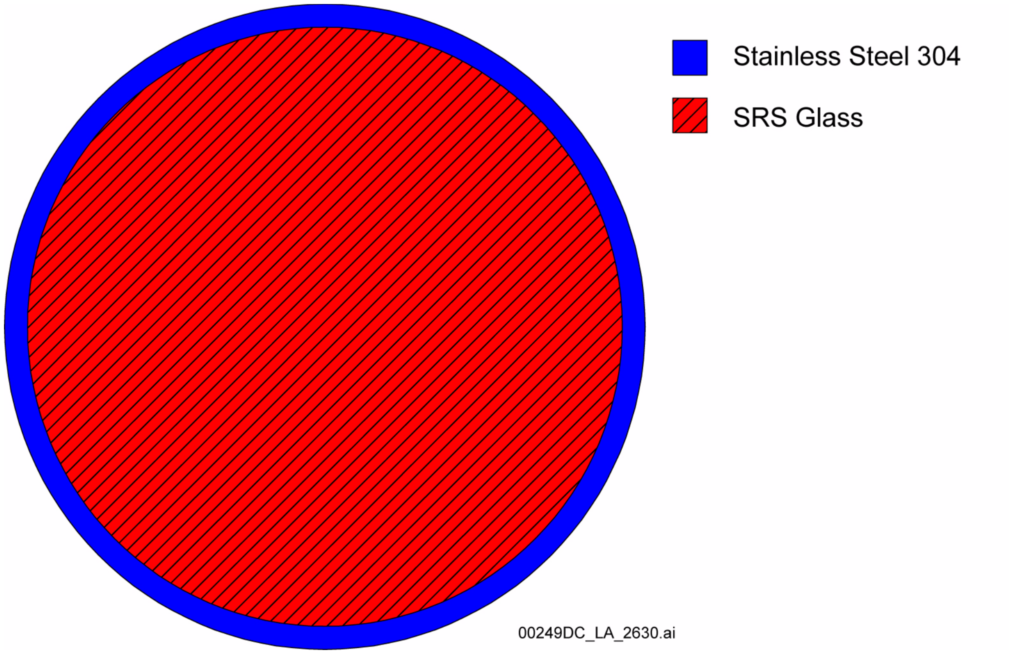

Figure 1.10-24. Radial Cross Section of DOE Savannah River Site Canister at Midplane

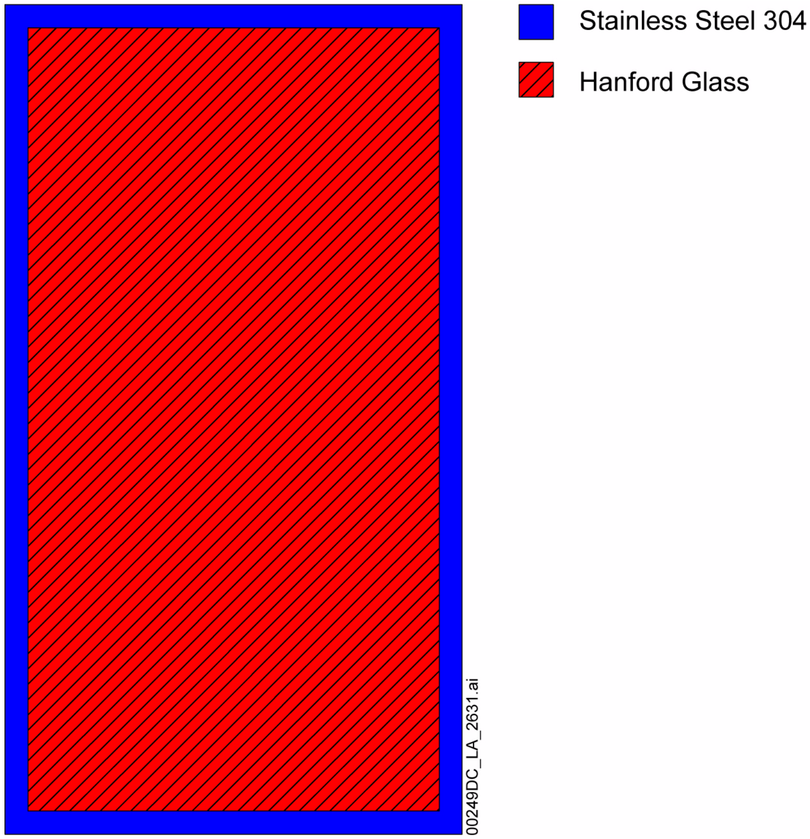

Figure 1.10-25. Axial Cross Section of Hanford HLW Canister

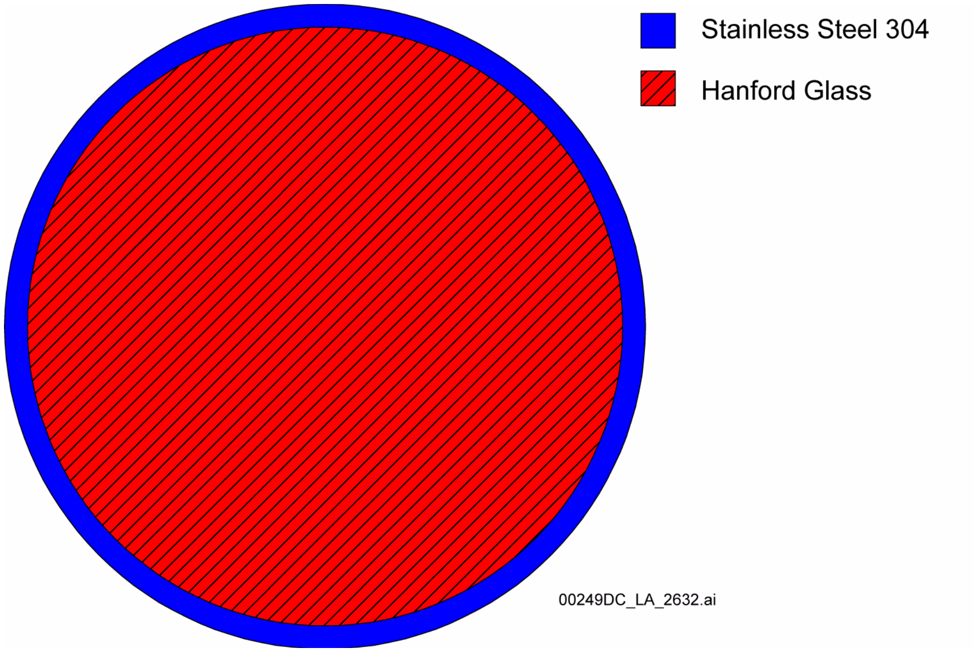

Figure 1.10-26. Radial Cross Section of Hanford HLW Canister

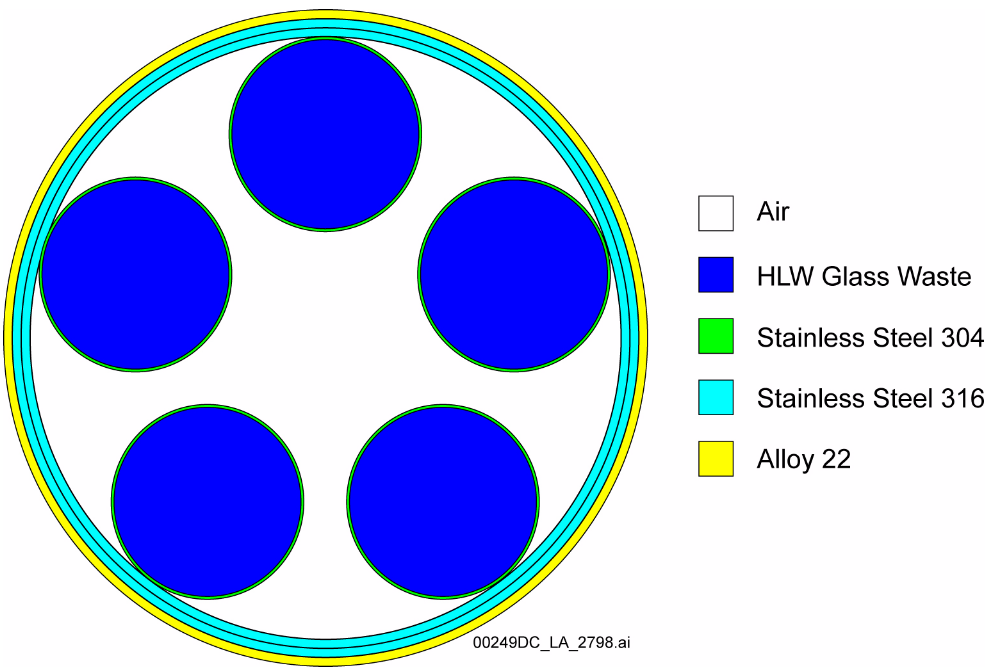

Figure 1.10-27. Radial Cross Section of 5-DHLW/DOE Codisposal Waste Package with Savannah River Site HLW Glass (IHF)

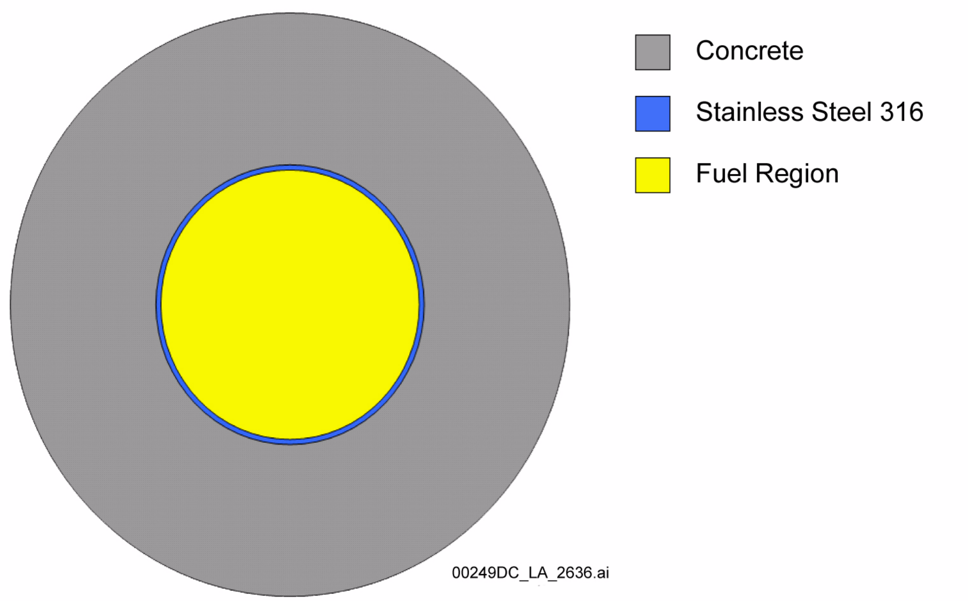

Figure 1.10-28. Radial Cross Section at Midplane of an Aging Overpack Containing a TAD Canister

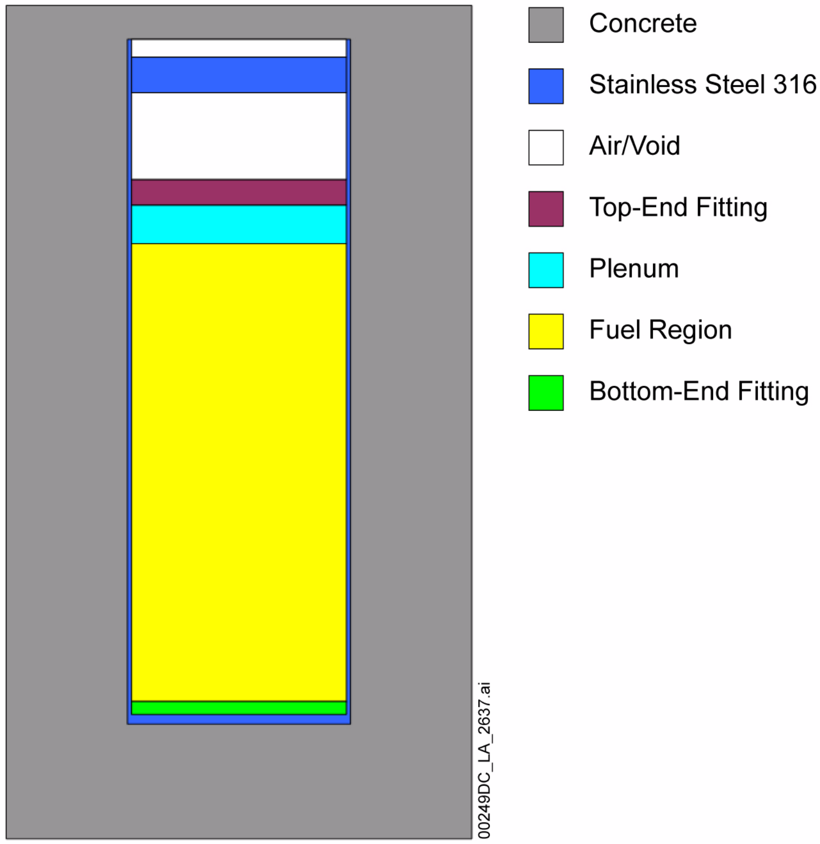

Figure 1.10-29. Axial Cross Section of an Aging Overpack Containing a TAD Canister

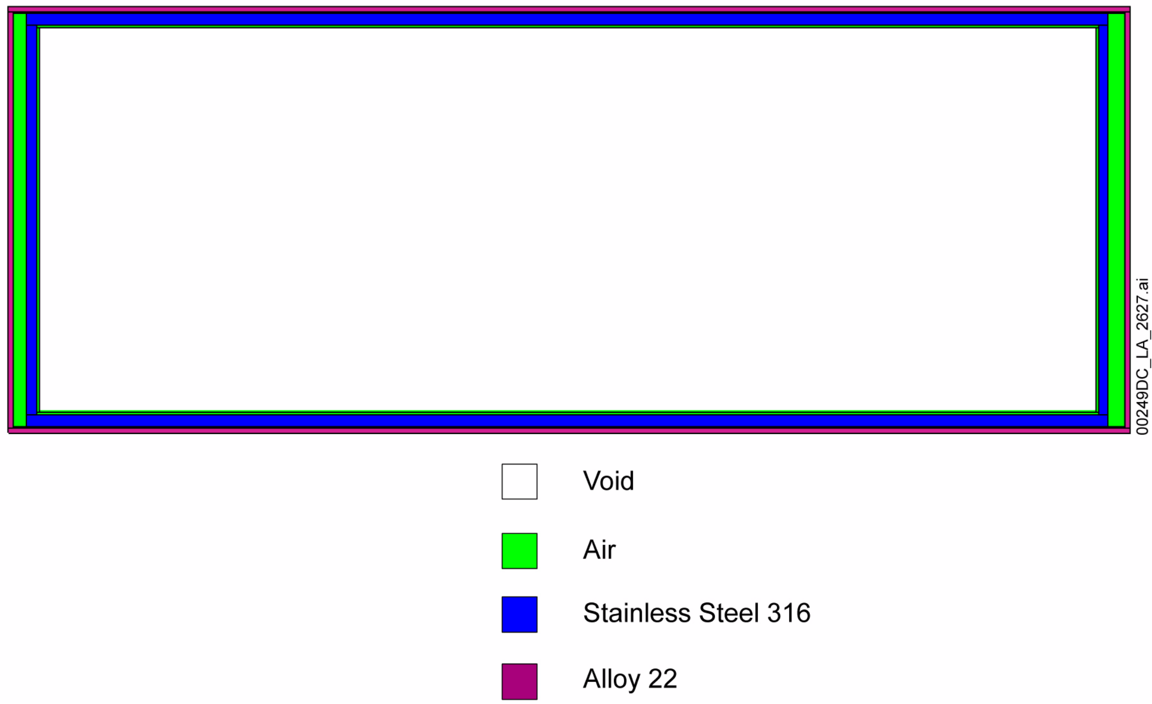

Figure 1.10-30. Axial Cross Section of Naval Canister and Waste Package

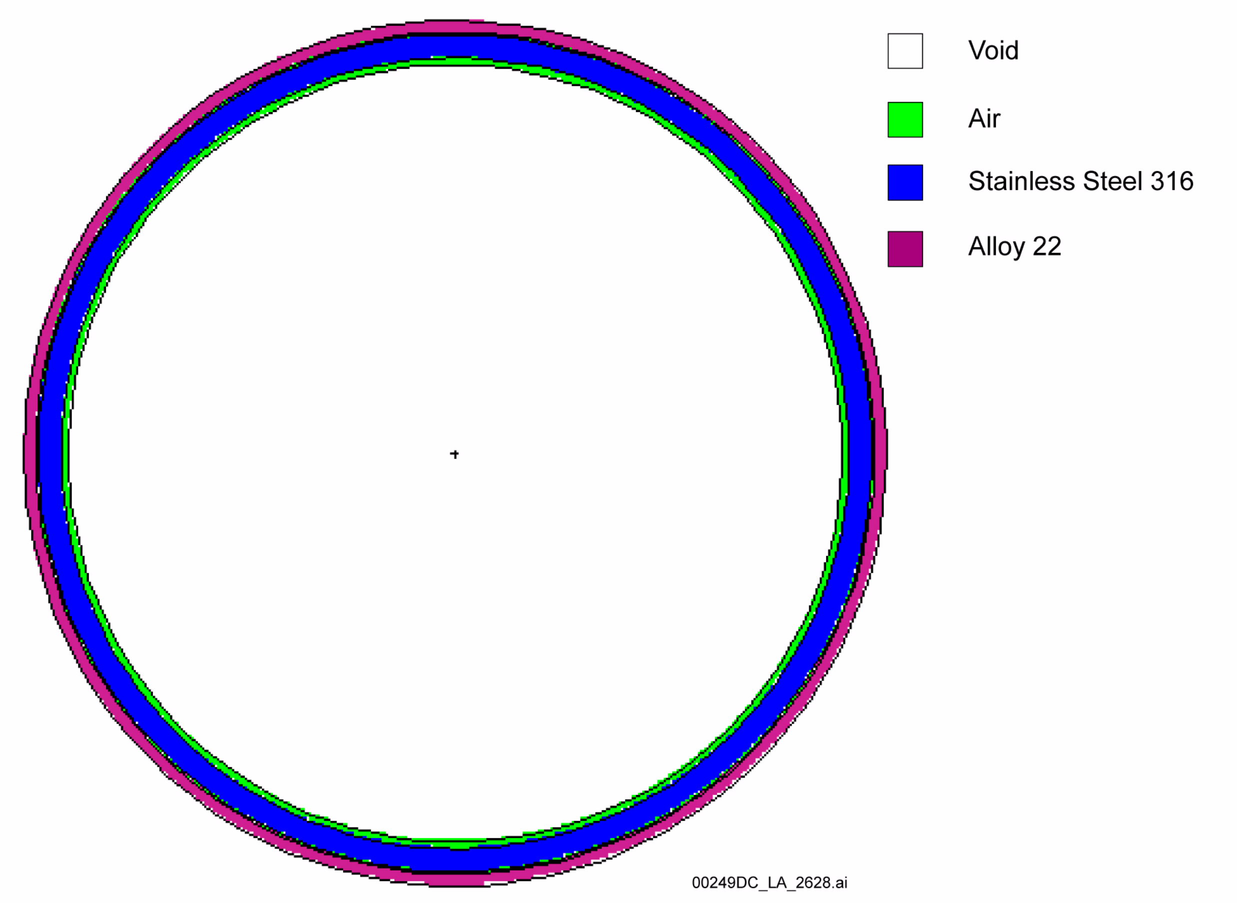

Figure 1.10-31. Radial Cross Section of Naval Canister and Waste Package

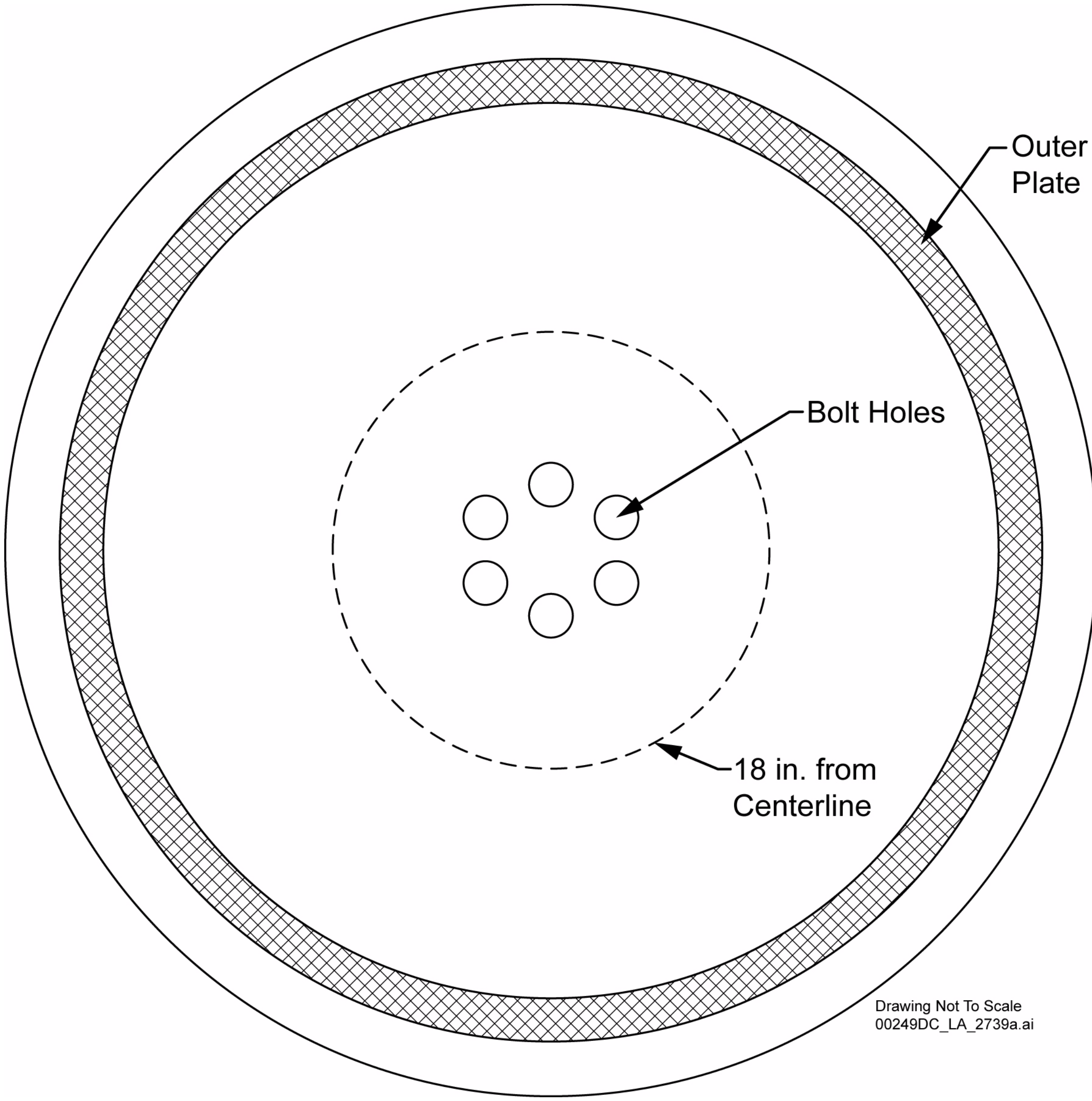

Figure 1.10-32. Naval Canister Top Source Distribution Geometry

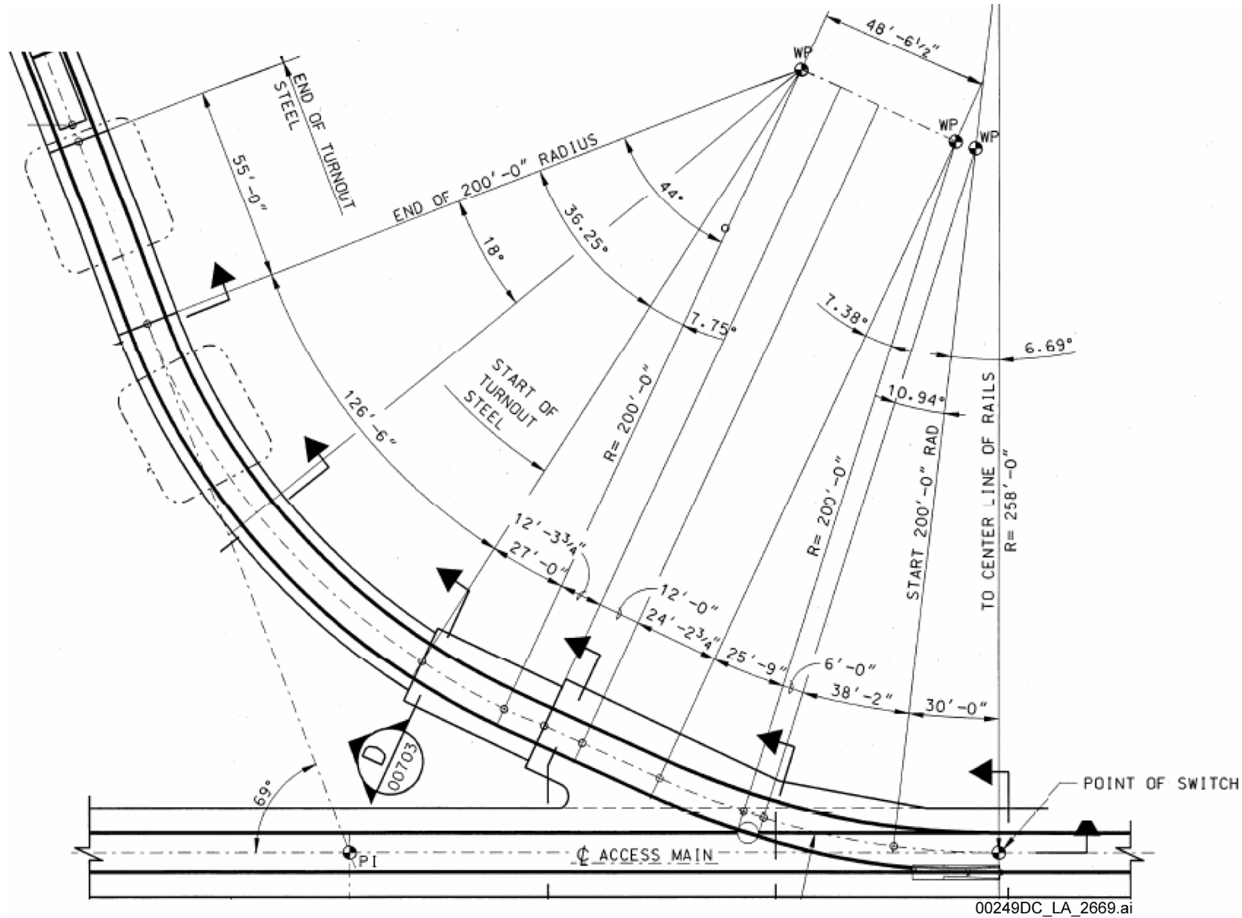

Figure 1.10-33. Emplacement Drift, Turnout Drift, and Access Main Plan

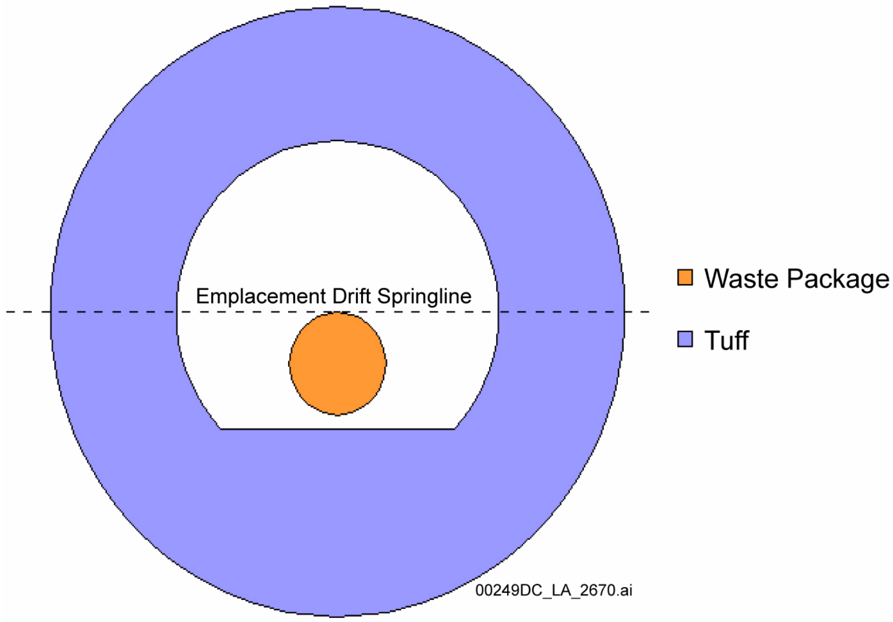

Figure 1.10-34. Waste Package in the Emplacement Drift/Transversal Section

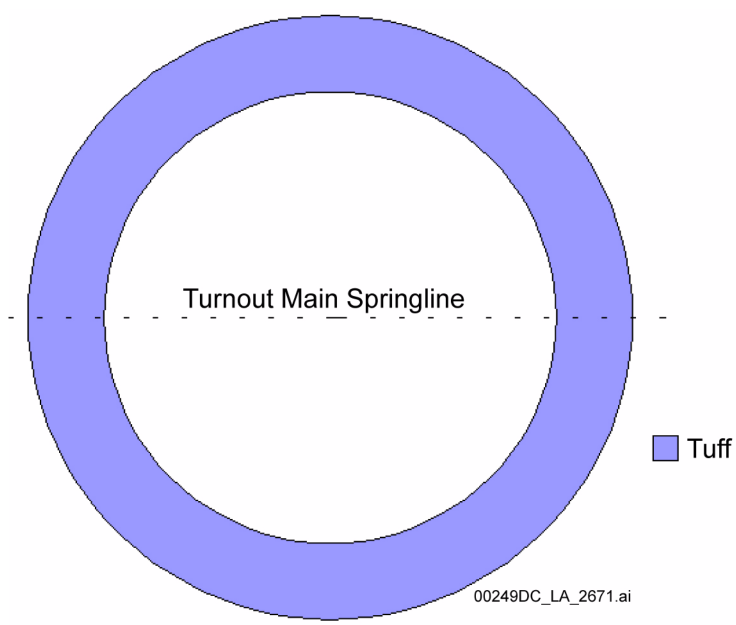

Figure 1.10-35. Turnout Main Drift/Transversal Section

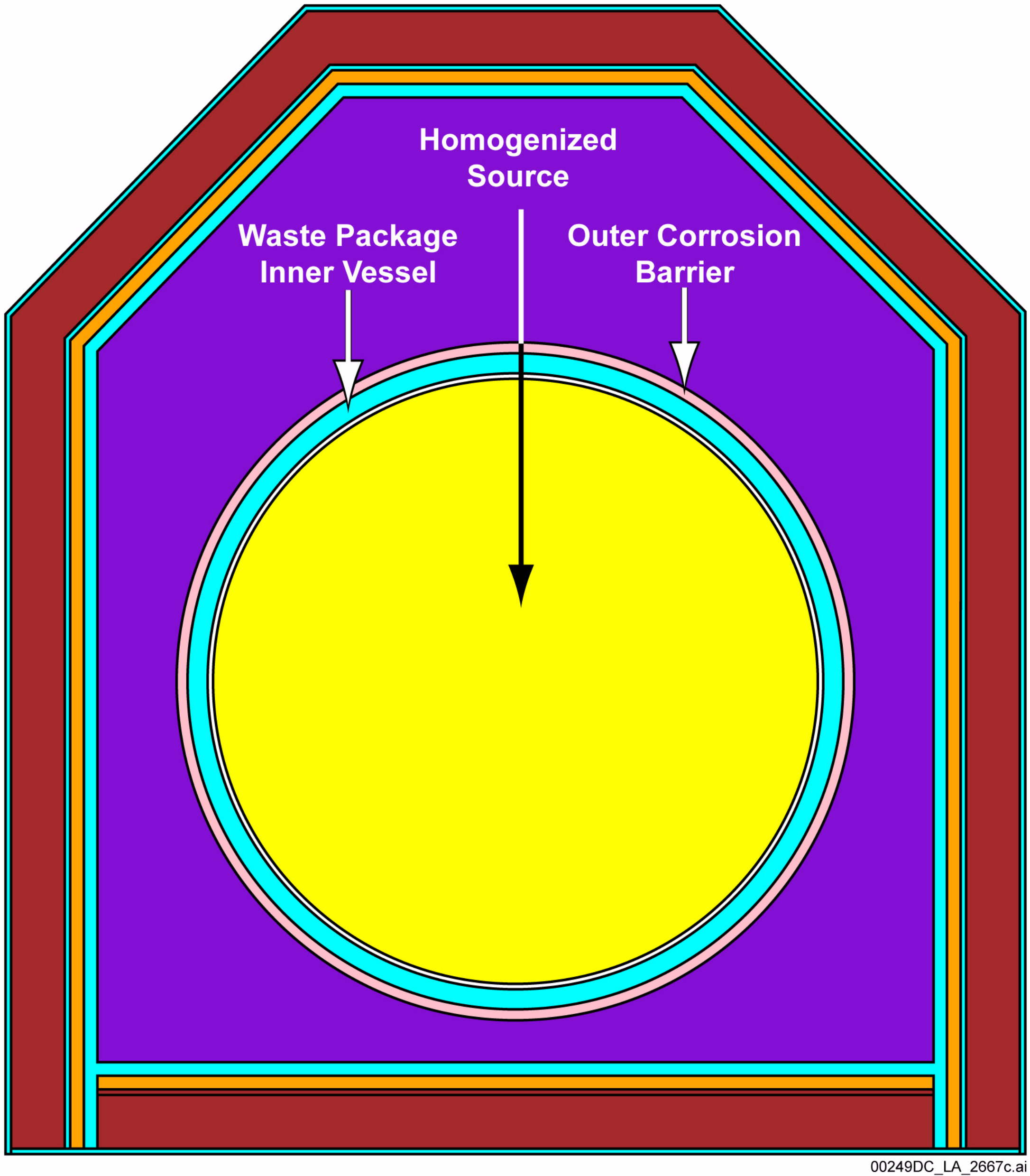

Figure 1.10-36. Axial View of Waste Package Inside the Transport and Emplacement Vehicle

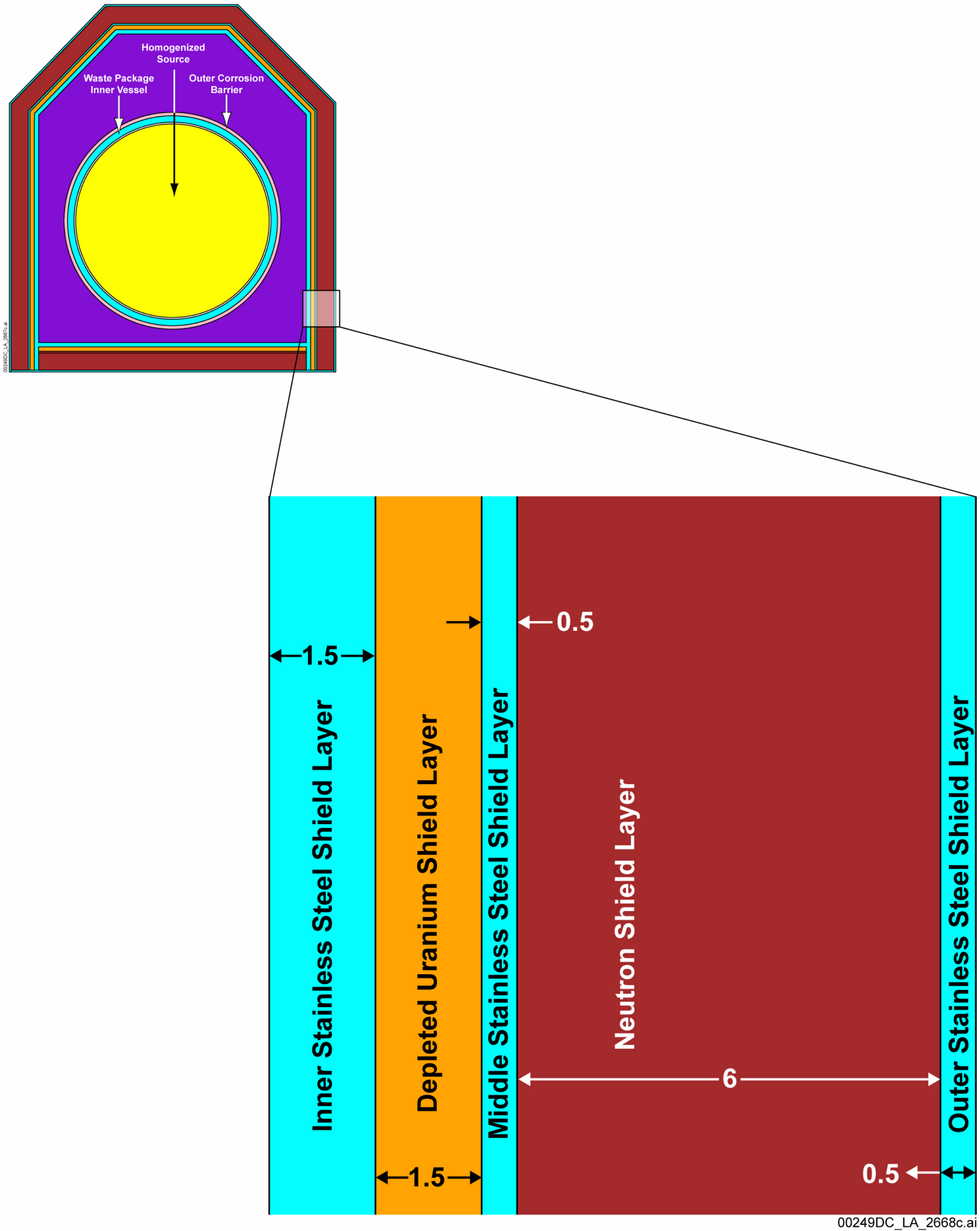

Figure 1.10-37. Proposed Transport and Emplacement Vehicle Shielding Material Arrangement

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1.11 PLANS FOR RETRIEVAL AND ALTERNATE STORAGE OF RADIOACTIVE WASTES

This section presents information on how the structures, systems, and components (SSCs) built and operated in the facilities implement a safe operating approach and maintain the capability to retrieve waste. Safety issues that are addressed in this section include the safe removal of waste packages from the subsurface and transport to an alternate storage facility. The actions to be taken in the event of an off-normal event during the retrieval process are also addressed.

This section discusses approaches for retrieval and alternate storage of radioactive wastes, as required by 10 CFR 63.21(c)(7), that will be developed should a decision to retrieve be made. This section provides information that addresses specific acceptance criteria in Section 2.1.2 of NUREG-1804. The information presented in this section also addresses the requirements of elements of 10 CFR 63.111(e). The following table lists the information provided in this section, the corresponding regulatory requirements, and the applicable acceptance criteria from NUREG-1804.

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
1.11.1	Retrieval Plans	63.21(c)(7) 63.111(e)(1)	Section 2.1.2.3: Acceptance Criterion 1 Acceptance Criterion 2
1.11.2	Alternate Storage Plans	63.21(c)(7) 63.111(e)(1)	Section 2.1.2.3: Acceptance Criterion 3
1.11.3	Retrieval Operations Schedule	63.21(c)(7) 63.111(e)(3)	Section 2.1.2.3: Acceptance Criterion 4

The information in this section is generally based on *Concepts for Waste Retrieval and Alternate Storage of Radioactive Waste* (BSC 2008).

The approach for performing retrieval at the repository was examined during the early design phases. It was determined that the approach most likely to support successful retrieval operations was to develop the subsurface facility in such a manner that access to the waste was maintained throughout the preclosure period. The design requirements established for the subsurface facility therefore incorporated this philosophy. For example, a design life of 100 years (including maintenance) has been established for the ground support system in the access mains, ventilation mains, and emplacement drifts, to ensure access to the emplaced waste packages. A maintenance plan to test, inspect, and repair ground support as necessary in the future has also been planned to support this design strategy. Similarly, the subsurface communication and transportation infrastructure is designed for the preclosure operating life and supports access for maintenance or equipment replacement as needed. The design described in Sections 1.3.1 through 1.3.6 is based on compliance with design requirements that ensure that accessibility to the waste is maintained throughout the preclosure operating period.

The retrieval discussions presented in Section 1.11 are based on information that is currently available. Specific plans for retrieval will be developed and defined in detail should the need for retrieval be identified. In the event of a decision to retrieve, safety analyses will be performed for those retrieval actions and operations necessary to safely remove the waste from the underground emplacement area to an alternate storage facility on the surface. These analyses will include the specific details about how retrieval operations would be performed and will include material control and accounting, and physical protection considerations.

Since the specific cause for retrieval may vary, this section demonstrates the feasibility of retrieval under the designed operational conditions, using the proposed facility equipment. This section discusses the approach used to ensure that the repository design and operations do not preclude retrieval of waste and discusses a potential approach for future retrieval actions consistent with the current design.

Studies of retrieval and alternate storage of radioactive wastes have been performed, and the findings have been considered in development of the approaches presented (BSC 2008, Section 1.2). During the design process, off-normal situations that could require remediation or interfere with retrieval were conceptualized. Such events were outside the design bases and would be considered off-normal events. Further, they were considered to be localized events (e.g., a local collapse of the ground support). Such conditions were the subject of studies that determined that, with proper planning and development of specialized equipment, remedies that restored access to the waste packages could be achieved.

1.11.1 Retrieval Plans
[NUREG-1804, Section 2.1.2.3: AC 1, AC 2]

Since the design philosophy is to ensure access to the waste packages for retrieval throughout the preclosure operating period, development of detailed plans for the retrieval of waste packages would be driven by the reason for retrieval. In other words, if a determination was made that a retrieval action needed to provide long-term storage, then facilities for handling and storing waste would be designed to accommodate those needs. If the reason for retrieval was a desire to recover the resource, then the retrieval action would likely involve transporting the waste to a remote location. This requirement would involve a different set of facilities and operations. In either case, the procedural and operating capabilities to retrieve the waste from the subsurface facility would already be in place. Section 1.11.3 describes the time lines that would be planned to assess the retrieval needs and provide appropriate design and licensing documentation to implement and accomplish retrieval.

The repository design described in the safety analysis report demonstrates the ability to handle and emplace 70,000 MTHM of waste within the preclosure safety limits imposed by 10 CFR Part 63. It is expected that a retrieval action will be able to similarly meet the preclosure safety limits. This is a reasonable expectation since the operations performed during retrieval are not significantly different than those of operations performed during the preclosure period. The aspects of retrieval as related to subsurface operations closely parallel those of emplacement, so few if any additional hazards are expected to be identified. In the surface facilities, the strategy for minimizing the number of lifts, minimizing the lift heights, and providing confinement for controlling potential releases during retrieval operations would be the safety approach established while processing waste packages for emplacement.

As required by 10 CFR 63.111(e), and as described in Sections 1.3.1 through 1.3.6, the repository design preserves the ability to retrieve any or all of the emplaced waste prior to closure based on a reasonable schedule. The design approach to address this requirement is to ensure that the repository design and emplacement processes do not preclude the retrieval of any or all waste packages during that period. Retrieval is defined in 10 CFR 63.2 as the act of permanently removing radioactive waste from the underground location at which the waste had been previously emplaced for disposal. Activity that involves removing or relocating selective waste packages from the subsurface as a result of concerns related to an off-normal condition will be addressed as a recovery action (Section 1.3.4.8).

Two hypothetical situations for retrieving waste are considered. In the first, a policy decision could be made to recover the resource value of the waste or approach disposal in a different manner. In the second situation, the Performance Confirmation Program could determine that postclosure performance of the natural barriers or Engineered Barrier System may not achieve regulatory compliance following disposal (Section 4.1). In both situations, retrieval of the waste to surface storage is the outcome of the decision process.

If a decision for retrieval is made for any or all of the waste, the waste that is to be retrieved would be placed in a storage facility designed in accordance with the regulations applicable at the time. The concept presented here is consistent with current practices and regulations for the protection of public health and safety and the environment and demonstrates the feasibility of such a facility, if required, and has considered keeping radiation exposure as low as is reasonably achievable (ALARA).

1.11.1.1 Operational Equipment and Processes

Conditions within the repository during the preclosure period are expected to be within the design bases, with no leakage of radioactivity expected from waste packages. However, conditions beyond the licensing bases are considered in Section 1.11.1.2.1. Radiation protection controls will be implemented during retrieval to limit the potential for radiation exposure to repository workers and the public in accordance with ALARA principles. Information regarding the ALARA program during retrieval is addressed in Section 1.11.1.3.2.

For a retrieval action, the SSCs that make up the subsurface facility will be operating as designed and capable of performing in accordance with their intended functions. This assumption is supported by the design basis requirement to maintain the capability to install drip shields within the emplacement drifts at the time of closure. The design and maintenance bases include maintaining this capability for the entire preclosure period so that drip shields can be installed should the decision to close the repository be made. Analyses, performed to evaluate the potential extent of drift degradation that may occur during the preclosure period, indicate that minimal, if any, degradation may be expected (Section 1.3.4.4.1.3). The operational process for retrieving the waste packages is to perform those steps executed in the waste emplacement process, using the same equipment (Sections 1.3.3 and 1.3.4), but in reverse order. Off-normal conditions, if any, will require an assessment to identify specific conditions and determine an appropriate operational strategy.

1.11.1.1.1 Retrieval Equipment

Retrieval operations would be performed using mobile transportation and emplacement equipment used for emplacement or equipment developed for retrieval and to be described in an amendment to the license application. Such equipment and facilities, along with measures that may need to be implemented to support operational readiness, for the purpose of retrieval, will be evaluated at the time a decision to retrieve is made.

As part of routine transport and emplacement operations, and in order to facilitate the capability to retrieve, procedures will guide development and maintenance of records that document equipment operations, usage, and operating histories. Compilation of these records will ensure that design and operations information will be available and adequate to support a retrieval action if such a decision is made.

Surface facilities will be needed for preparing waste packages for surface storage or for remediation of waste packages should conditions at the time of retrieval require such actions. These facilities and mobile equipment currently expected to be used are discussed in the following sections.

1.11.1.1.1.1 Mobile Equipment

The transport and emplacement vehicle (TEV) or a similar conveyance is the mobile equipment that would be used for waste package retrieval from the emplacement drifts. The TEV is the mobile equipment used for emplacement operations. The advisability of using the TEV will be reevaluated at the time a decision to retrieve is made (BSC 2008, Section 6.3.1).

Detailed descriptions of the TEV are presented in Sections 1.3.3 and 1.3.4.

1.11.1.1.1.2 Surface Facilities

It is possible that a retrieval action will be necessary after the surface facilities used to process waste packages have been decommissioned. Regardless of when a retrieval action occurs, surface facilities will be needed to prepare a retrieved waste package for alternate storage or shipment to a remote location. The function to be performed by these facilities will depend on what is needed to prepare waste packages for storage. Conditions will be assessed at the time of retrieval, and appropriate facility designs will be developed and analyzed as part of the retrieval safety analysis. The redesigned or new facilities design will place the existing waste packages into suitable surface storage containers and place them in alternate storage. The design of a facility that could perform handling and storage of retrieved waste packages is discussed in Section 1.11.2.

1.11.1.1.2 Retrieval Operations Overview

Waste retrieval operations would begin by cooling the emplacement drifts with the installed ventilation to meet the thermal operating limits of the retrieval equipment. The current design does not require the installation of drip shields until a decision to close the repository is made and approved by the U.S. Nuclear Regulatory Commission (GI Section 1.1.3.2). Therefore, no drip shield removal is anticipated for retrieval except for those drip shields that may be installed to support the Performance Confirmation Program. More detailed discussions regarding drip shields and subsurface ventilation are presented in Sections 1.3.4 and 1.3.5, respectively. Once the emplacement drift is sufficiently cool, the TEV would be prepared for waste retrieval operations.

Waste package retrieval would be initiated with the removal of the waste package emplaced nearest the emplacement drift entrance. The waste package emplacement process does not allow the lifting of one waste package over another, so the waste packages would be removed in reverse sequence to emplacement. The TEV would perform retrieval by loading waste packages in the emplacement drifts and transporting them to the surface. There, the retrieved waste packages would be handled and moved to an alternate storage facility. Upon retrieval of waste packages from an emplacement drift, the TEV would move to the next available drift to repeat the retrieval cycle. The retrieval operational processes would be the reverse of the emplacement operations performed to emplace the waste package into the waste package emplacement drifts. Procedures would be prepared, approved, and tested prior to initiation of retrieval operations (Section 1.3.4.8) (BSC 2008, Section 10).

Records of emplacement operations will be maintained (Section 5.2.1), including evaluation of operating conditions and lessons learned from operating problems that are encountered, to assist in retrieval planning and operations.

1.11.1.2 Identification of Design and Operational Conditions for Retrieval

The specific design and operational plan needed for disposition of the retrieved waste depends upon the reasons for retrieval, consideration of any associated hazards, and consideration of licensing regulations applicable at the time.

During the preclosure period, drifts will be monitored and corrective maintenance performed, if necessary, to ensure that the capability to retrieve is maintained (Sections 1.3.3.3 and 1.3.4.4). Since retrieval is the reversal of the emplacement process, operational interferences that may be expected are the same as those encountered during emplacement operations.

Preclosure operations have been analyzed to consider events that could credibly occur during the preclosure period. The subsurface facility is designed to withstand the effects of disruptive events that may occur during the preclosure period without significant impact on the subsurface facilities; that is, no significant drift failure would occur. Analyses indicate that rockfall events that may occur in the emplacement drifts will not cause failure of the waste packages (Section 1.3.4.4). Information regarding seismic events and potential rockfall occurrences is presented in Sections 1.3.2, 1.3.3, and 1.3.4.

Furthermore, during the preclosure period, the waste package is housed in a well-ventilated, low-humidity environment and is unlikely to experience significant, if any, corrosion. In addition, the Performance Confirmation Program (Chapter 4) includes activities to ensure that the impacts of loading waste into the emplacement drifts do not affect retrieval capability. Were some deleterious effects to the drift or waste package to be identified, they would be addressed in the detailed retrieval plans.

1.11.1.2.1 Considerations of Conditions Beyond the Licensing Bases

The subsurface facility is designed and will be maintained to facilitate retrieval (Sections 1.3.3.3 and 1.3.4.4). The design bases, operations, and inspection and maintenance programs work together to maintain access and functionality of the equipment necessary to support retrieval (Sections 1.3.3, 1.3.4, and 5.6).

Even though the expected behavior of the subsurface facility will preclude most deleterious conditions, three potential conditions have been postulated that could interfere with retrieval. Although none of the conditions is expected to occur during preclosure operations, the facility has the capability to respond to them.

The first condition that was considered was possible drift collapse. It has been calculated that drift collapse due to rockfall that may result because of seismic loads induced by a design basis ground motion 2 (DBGM-2) event does not occur because of the nature of the rock at the repository horizon and the capability of the installed ground support (BSC 2007a, Section 7). However, in the very unlikely event of significant drift collapse, an appropriate recovery and remedial action strategy will be developed (Section 1.11.1.2.3).

The second condition that was considered was the possibility of removing backfill. The current design and analysis for the postclosure period do not include backfill of the emplacement drifts. This is a deliberate action that the repository would not undertake unless the decision to close had been made and a need for backfill was determined. Accordingly, no special provisions have been made for backfill removal.

The third condition that was considered was the possible existence of loose surface contamination. It is unlikely that a waste package would fail from outer barrier or inner shell degradation, considering its environment (Section 2.1.1); nor is any preclosure drift collapse expected to breach a waste package. Even if a waste package were breached, there would be no expected impact on retrieval operations, because all retrieval operations will be performed remotely. Ventilation is designed for normal and retrieval operations to move air away from occupied areas in the underground; that is, from the development areas and access mains to the emplacement drifts and then to the exhaust mains. This flow will direct potential surface contamination that may become airborne away from the normally accessible operating areas through the emplacement drifts and potentially into the ventilation exhaust shafts, helping to prevent the spread of contamination inside the normally accessible areas of the repository (Section 1.3.5.1). During the retrieval process, it is expected that the waste package would be placed in a closed and shielded conveyance, such as the TEV, to protect the workers from direct radiation. This situation would be evaluated to determine appropriate additional radiation protection controls and design activities that may be required prior to commencing recovery actions.

1.11.1.2.2 Potential Retrieval Interference

In general, no major impacts to operational processes are anticipated. However, a number of operational events have been identified that, while not likely to happen, may potentially impact retrieval operations. These events include (Section 1.6):

- Derailment of the TEV
- Waste package drop
- Damage to the TEV by impacts
- Impact between the TEV and facility structures, equipment, or objects.

These same or derivative events might be encountered during normal emplacement operations, so lessons learned during emplacement operations will be documented in retrieval plans and applied in implementing recovery actions during retrieval. Each event occurrence, beginning at the time of initial startup activities and testing, will be assessed to identify specific event conditions and to determine an appropriate approach for recovery and remediation. Analyses of events that are potentially adverse to retrieval will continue through operations until permanent closure. Records of emplacement operations will be maintained, including evaluation of operating conditions and lessons learned from operating problems that are encountered, to assist in retrieval planning and operations (Section 5.2.1).

1.11.1.2.3 Methodologies for Identifying and Analyzing Potential Retrieval Problems

Should they occur, retrieval problems will be evaluated to identify specific conditions and to determine an appropriate recovery and remedial action strategy. The complexity and duration of the recovery and remedial action for a potential retrieval problem will relate directly to the severity and impact of the specific event or issue occurrence. Potential solutions have been considered that might be implemented for the potential retrieval operational events listed in Section 1.11.1.2.2 (BSC 2008, Section 6.4).

Although each event or issue will be evaluated in detail, the strategy for recovery from off-normal events includes the following activities (BSC 2008, Section 6.3):

- Assessing the immediate status of an involved waste package for personnel safety

- Developing a detailed recovery plan, which will include:

  – Assessing radiological and industrial safety conditions to ensure the recovery action is implemented in accordance with ALARA and industrial safety requirements

  – Assessing security impacts during retrieval

  – Assessing impacts of retrieval on the environment at or near the site

  – Establishing access control and isolating the event area from continued operations, if required, to ensure worker safety

  - Confining contamination

  - Collecting technical data

- Completing actions in accordance with the recovery strategy and returning the repository to normal operations

  - Formulating a mitigation plan
  - Designing and providing any additional specialized equipment needed for mitigation
  - Implementing the mitigation plan.

As part of the preclosure safety analysis, event sequence analyses were performed for a wide range of potentially credible event sequences, including impact occurrences such as rockfall (Section 1.6). These event sequence analyses address the potential retrieval problems listed in Section 1.11.1.2.2.

While no significant rockfall events are expected during preclosure operations, studies indicate that fallen material from an emplacement drift rockfall can be safely and effectively removed to allow resumption of operations (BSC 2007b, Section 3.4).

1.11.1.2.4 Repository Conditions

At the time the decision is made to retrieve waste, one of the first actions will be to assess the condition of the existing facilities and determine the need for refurbishment or upgrade to support retrieval operations. This evaluation will include the subsurface facility, mobile equipment, surface facilities, and alternate storage facilities necessary for waste retrieval. These facilities and equipment will be assessed, including safety analyses, radiation monitoring and protection, and access control and security. Measures will be taken to ensure support for planned retrieval operations.

1.11.1.2.4.1 Subsurface Facility

The subsurface facility will be maintained until repository closure. During the period between the completion of emplacement and closure of the repository, several activities that require access and use of the facilities will continue. These activities include the Performance Confirmation Program (Chapter 4), installation of drip shields, and backfilling and sealing openings where applicable (Section 1.3.6). The same maintenance programs that are in place for emplacement will be ongoing during retrieval (Section 1.3.4). No adverse subsurface conditions are expected that would impact retrieval and storage.

During repository operations, selected emplacement drifts will be monitored for environmental conditions as part of the Performance Confirmation Program (Section 4.2.1.8). Monitoring will be accomplished using instrumentation or remotely operated vehicles for visual inspections to detect rockfall. These monitoring and inspection activities provide the necessary information for evaluating drift degradation effects, as well as any required maintenance and repair of the ground support components that may be needed to provide continued accessibility to the emplacement drifts for possible waste retrieval operations.

1.11.1.2.4.2 Mobile Equipment

The assessment of mobile equipment will be similar to the assessment performed for the surface facilities. Reconditioning and upgrade of mobile equipment will be determined by the results of the assessment.

1.11.1.2.4.3 Surface Facilities

The condition of the surface facilities at the time any decision is made to retrieve waste will be determined by a number of factors, including:

- The status of emplacement
- Facilities available for retrieval
- The status of equipment within the facilities with regard to being serviceable and usable for retrieval.

If retrieval begins while the emplacement process is still ongoing, reconditioning of surface facilities may be limited to specific modifications. If retrieval begins after emplacement operations are complete, reconditioning requirements may be more extensive (Section 1.11.1.2.2).

1.11.1.2.5 Maintenance Plans

Maintenance plans will be developed during the retrieval planning stage to support the completion of retrieval in a manner that will protect health and safety, as well as keep radiation exposures ALARA. These plans will include maintenance of the operating environment, such as the ground support system, as well as the mobile equipment.

1.11.1.2.5.1 Ground Support Maintenance

Ground support SSCs will be required to be operational throughout the retrieval period.

1.11.1.2.5.2 Equipment Maintenance

The surface facilities include a maintenance area where maintenance activities are performed on the TEV to ensure operational readiness (Section 1.3.3.5.2). Routine maintenance is performed on a scheduled basis.

1.11.1.2.6 Backfill Option

The repository design does not include backfilling of the emplacement drifts. Accordingly, the current approach for retrieval does not consider the need for removal of backfill at the time of retrieval.

1.11.1.2.7 Performance Confirmation Program Effects

The Performance Confirmation Program and its support systems will continue to operate throughout the retrievability period. The Performance Confirmation Program does not affect the time frame specified by 10 CFR 63.111(e) for the period of retrievability. However, the Performance Confirmation Program will continue until a license amendment to close the repository is approved. Maintaining retrieval capability will not impose adverse impacts on the Performance Confirmation Program. The Performance Confirmation Program monitors subsurface conditions and performs tests to confirm geotechnical and design assumptions to ensure the preservation of the retrievability option (Section 4.2.1.8).

1.11.1.3 Compliance with Preclosure Performance Objectives

Performance objectives for waste package retrieval functions are based on compliance with requirements specified in 10 CFR 63.111(a), (b), and (e).

The design bases requirements for repository SSCs are developed from Category 1 and Category 2 event sequences that are identified through preclosure safety analyses. The consequences of the Category 1 and Category 2 event sequences are evaluated against the respective 10 CFR Part 63 preclosure dose performance objectives. The SSCs involved in event sequences and required to prevent or mitigate a dose from exceeding the 10 CFR Part 63 preclosure dose performance objectives are classified as important to safety. The design bases for SSCs important to safety are provided in Section 1.9. The design criteria and design descriptions of the repository SSCs associated with the design bases are discussed in Sections 1.2, 1.3, 1.4, and 1.5.

1.11.1.3.1 Preclosure Performance

Demonstrating compliance with 10 CFR 63.111(a) and (b) during waste retrieval is accomplished by categorizing event sequences, performing consequence analysis, and imposing design requirements.

Categorization of event sequences is an essential step in demonstrating compliance with 10 CFR 63.111, because the risk-informed performance objectives are correlated to the categories of event sequences. Section 1.7 identifies repository preclosure event sequences and describes the categorization, in accordance with the categories identified in 10 CFR 63.2. The current preclosure safety analyses have not identified any new events for retrieval activities based on the use of the same equipment and operating approaches involved in emplacement. However, should retrieval be required, part of the planning process will include development of additional calculation of doses for event sequences associated with retrieval (Section 1.8.2.2.2). These calculations will be developed prior to initiating retrieval to ensure that any potential event sequences are considered. In addition to categorization of event sequences and consequence analysis, imposed design and operating requirements will be used to limit the potential for an event sequence during retrieval.

1.11.1.3.2 Implementation of As Low As Is Reasonably Achievable Concepts

A combination of engineering features, administrative controls, and radiation safety considerations will be employed in the design and operation of waste retrieval processes to implement ALARA principles.

Occupational dose estimates have not been developed for retrieval. However, the radiation exposure considerations applicable for emplacement operations would also apply for retrieval, along with any additional considerations based on preretrieval surveys.

The ventilation system is designed to minimize the spread of contamination into occupied areas. Retrieval of waste packages from the emplacement drifts would be done through use of remote technology to eliminate the presence of occupational workers in high-radiation areas. Further administrative controls, such as surveying and monitoring, would be used in conjunction with engineering controls to reduce individual and collective dose to occupational workers during retrieval operation. The regulatory requirements for conducting surveys and monitoring contamination levels and radiation doses would be followed.

1.11.2 Alternate Storage Plans
[NUREG-1804, Section 2.1.2.3: AC 3]

The specific design and the operational plan for surface facilities depend upon the specific needs for the storage and hazards encountered in the retrieval process. Modifications to facilities existing at the repository may be appropriate, or new facilities may be needed. This planning will commence should retrieval be necessary. The storage concepts considered are based on documented assumptions, engineering studies, and regulatory requirements (BSC 2008, Section 7.2).

A comprehensive repository assessment, operational analysis, and detailed design for the Alternate Storage Facility will be performed should the decision to retrieve waste be made. This effort will include ALARA and criticality considerations and may lead to other concepts of operations. For example, a horizontal waste package handling and storage system may be preferred, but other orientations will be considered. The nature of the waste package handling system and the size and configuration of the storage area will be evaluated in the comprehensive assessment. The purpose of the Alternate Storage Facility will be to store the retrieved waste in a manner that protects the health and safety of workers and the public and that maintains the quality of the environment.

1.11.2.1 Alternate Storage Facility Location

Figure 1.2.1-1 shows the locations for aging up to 21,000 MTHM of commercial spent nuclear fuel (Section 1.2.1). This area could be used for initial alternate storage, if necessary. As shown in Figure 1.11-1, the conceptual alternate waste retrieval and storage area has been located in the vicinity of the surface facilities and aging area in Midway Valley. This area is sufficient to store the planned 70,000 MTHM of waste. Although much of the identified area may have already been characterized, location selection will include appropriate characterization, siting studies, and surface preparation activities to meet retrieval goals (BSC 2008, Section 7.3).

Location selection criteria considered for the Alternate Storage Facility include:

- Proximity to the repository North Portal
- Retrievability of repository waste in the allocated time frame
- Space for dry storage of retrieved waste
- Storing the waste at locations in the proximity of a central processing facility
- Identifying optimal environmental conditions.

1.11.2.2 Alternate Storage Facility Size and Operations

The following sections discuss the size and operations of an Alternate Storage Facility.

1.11.2.2.1 Alternate Storage Facility Size

The size of an Alternate Storage Facility will depend on the amount of waste already received at the geologic repository operations area at the time of the decision to retrieve waste. A detailed evaluation of the storage area size requirements and the storage system configuration would be performed during the comprehensive assessment of an Alternate Storage Facility.

The conceptual layout of the Alternate Storage Facility contains space for a Waste Retrieval Transfer Building, rail spur, staging area, long-term storage area, support facilities, and administrative and security facilities.

An Alternate Storage Facility location would be furnished with both rail and road access. A new rail line would be constructed from the existing repository main gate to the Alternate Storage Facility.

1.11.2.2.2 Alternate Storage Facility Operations

As is the case for facility size, handling operations of the Alternate Storage Facility will also depend on the amount of waste already received at the geologic repository operations area at the time of the decision to retrieve waste.

Retrieval of waste packages from the subsurface emplacement drifts will be performed by the TEV or similar mobile equipment. During retrieval, waste packages are loaded into the TEV in the emplacement drifts, transported to the surface, and unloaded in the Alternate Storage Facility. This facility is equipped for waste package unloading from the TEV, transfer to a long-term storage unit that provides shielding for the waste package, and transport of each storage unit to a dry storage pad at the storage area. Potential locations for the storage areas are shown in Figure 1.11-1. During retrieval planning, a determination will be made of materials used for the long-term storage units and the storage configuration implemented at the storage area (BSC 2008, Sections 7.1 and 7.2).

Emplacement pallets will be processed independently of waste packages. The emplacement pallets will be surveyed for potential contamination, and managed and processed for disposal in accordance with applicable program and procedural requirements.

1.11.2.3 Public and Repository Worker Safety

Although a specific operational hazard analysis and a preclosure safety analysis have not been developed for the Alternate Storage Facility, the similar nature of emplacement and retrieval operational processes indicate that work activities for retrieval can be performed to protect the public and repository workers within the regulatory limits. However, a preclosure safety analysis of the Alternate Storage Facility will be performed during the retrieval planning stage should a decision to retrieve waste be made (Section 1.8.2.2.2).

1.11.3 Retrieval Operations Schedule
[NUREG-1804, Section 2.1.2.3: AC 4]

The retrieval operations schedule, including its duration, is based on the requirements of 10 CFR 63.111(e) (BSC 2008, Section 4.2).

The anticipated period of retrievability will extend until the decision is made to perform permanent closure. This period is consistent with the minimum 50-year period specified in 10 CFR 63.111(e)(1) and up to the 100-year preclosure period envisioned for the repository. The period of retrieval is approximately equivalent to the planned schedule for construction of the repository and emplacement of waste.

The development of retrieval plans would include:

- An evaluation of hazards associated with retrieval, an evaluation of facilities and equipment available for use in retrieval operations, and an evaluation of licensing requirements for retrieval applicable at the time of retrieval
- A license amendment request to include a design with supporting safety analyses to implement an operational retrieval plan
- U.S. Nuclear Regulatory Commission review and approval, attendant with any other regulatory action, of a license amendment prior to initiating retrieval operations
- Construction of facilities and development of procedures governing the retrieval process.

A retrieval planning time line for planning, developing the necessary facilities, and commencing retrieval operations is shown in Figure 1.11-2.

1.11.4 General References

BSC (Bechtel SAIC Company) 2007a. *Ground Control for Emplacement Drifts for LA*. 800-K0C-SSE0-00100-000-00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070925.0082.

BSC 2007b. *Strategies for Recovery After an Off-Normal Event to the Waste Package Transport and Emplacement Vehicle*. 800-30R-HE00-01800-000-000. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070531.0043.

BSC 2008. *Concepts for Waste Retrieval and Alternate Storage of Radioactive Waste.* 800-30R-HER0-00100-000-007. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080109.0010.

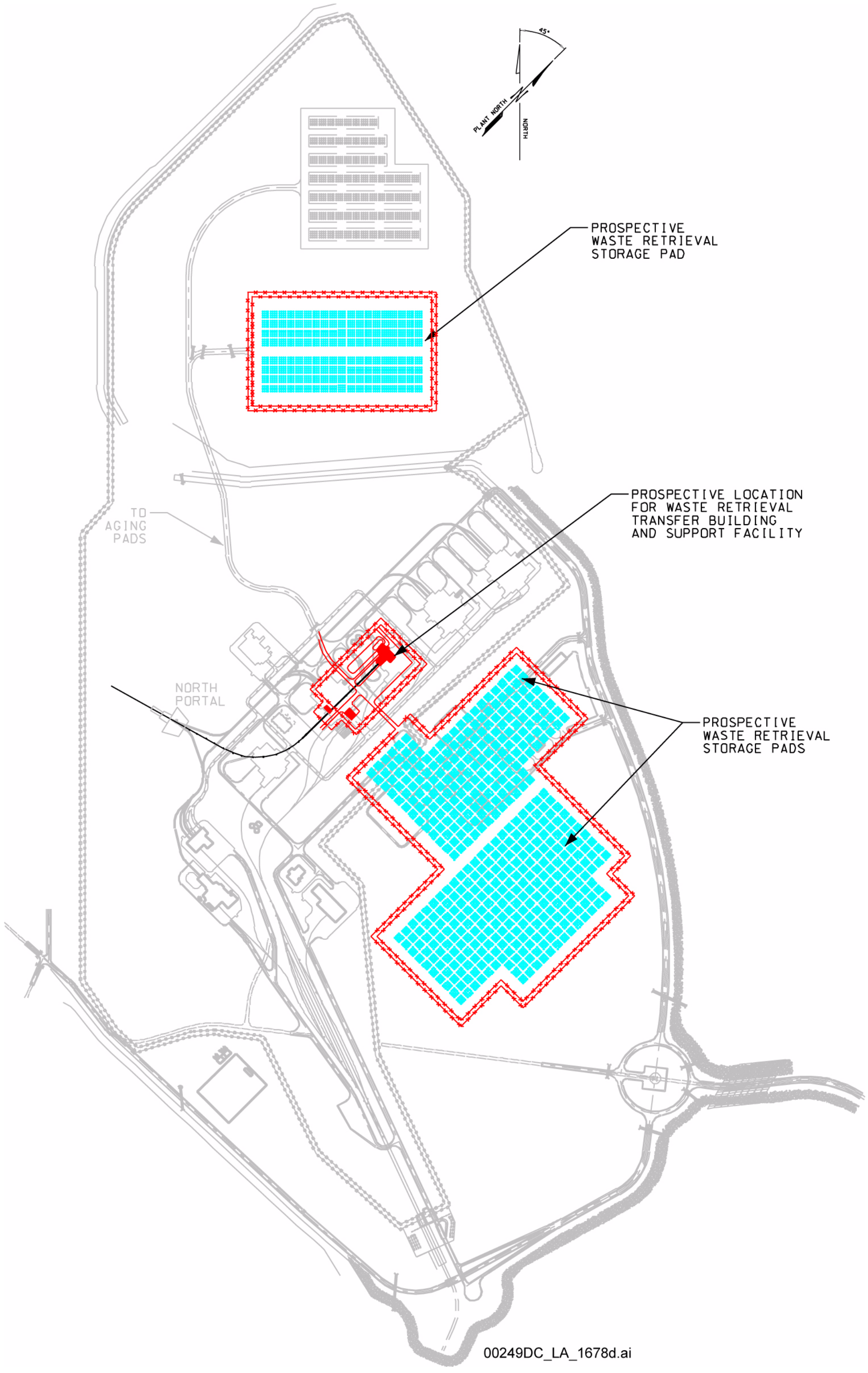

Figure 1.11-1. Alternate Storage Facility—Conceptual Layout

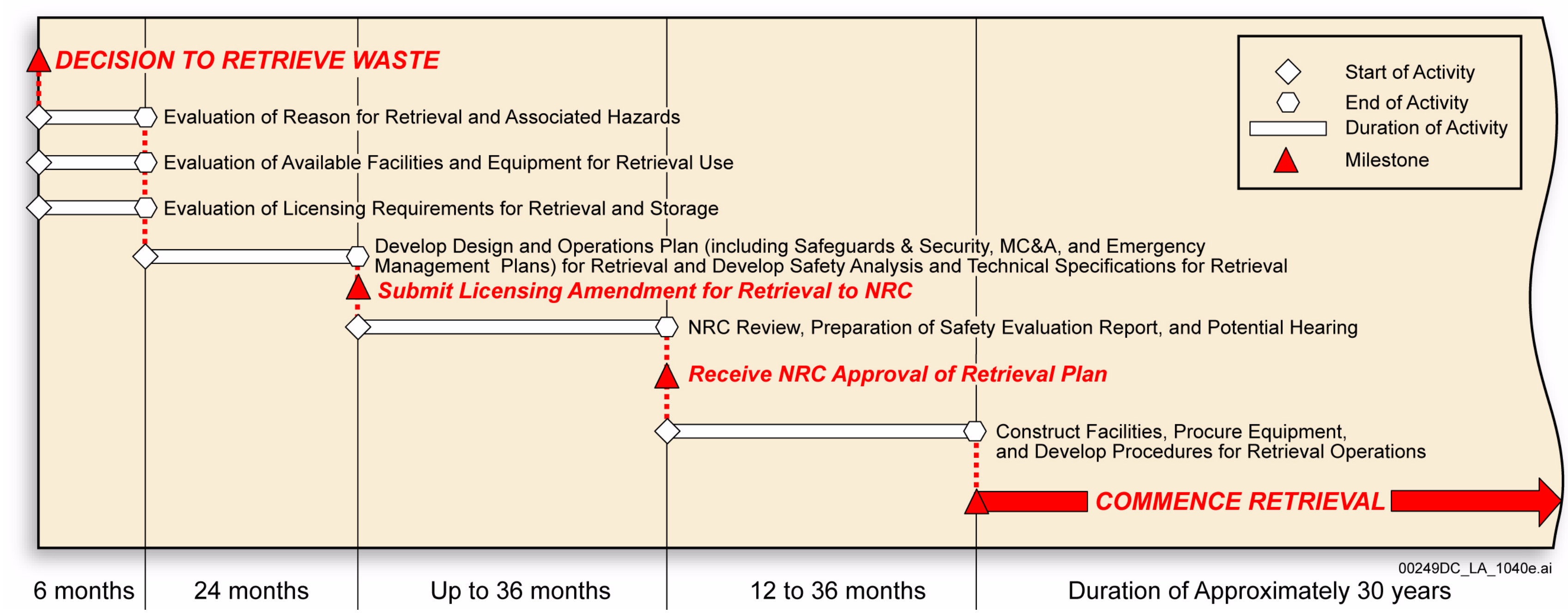

Figure 1.11-2. Retrieval Planning Time Line

NOTE: MC&A = Material Control and Accounting.

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1.12 PLANS FOR PERMANENT CLOSURE, DECONTAMINATION, AND DISMANTLEMENT OF SURFACE FACILITIES

This section provides information that addresses specific acceptance criteria in Section 2.1.3.3 of NUREG-1804. The information presented in this section also addresses requirements of 10 CFR 63.21(c)(8), 10 CFR 63.21(c)(22)(vi), and 10 CFR 63.51. The following table lists the information provided in this section, the corresponding regulatory requirements, and the applicable acceptance criteria from NUREG-1804.

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
1.12.1	Design Considerations to Facilitate Permanent Closure and Dismantlement	63.21(c)(8) 63.21(c)(22)(vi)	Section 2.1.3.3: Acceptance Criterion 1
1.12.2	Plans for Permanent Closure	63.21(c)(8) 63.21(c)(22)(vi) 63.51	Section 2.1.3.3: Acceptance Criterion 2
1.12.3	Plans for Decontamination and Dismantlement of Surface Facilities	63.21(c)(8) 63.21(c)(22)(vi)	Section 2.1.3.3: Acceptance Criterion 2

As part of site decommissioning, the repository will develop and implement a plan for the final radiological survey of the repository. The plan will conform to NUREG-1575 (NRC 2000), in recognition of the provisions in Section 2.6 that allow users the flexibility to customize guidance based on specific site characteristics as needed. Key considerations for development of the decontamination and dismantlement plan described in this section comply with that manual.

For the ancillary surface facilities that support radioactive waste disposal, the repository will follow decommissioning program policies and guidance in NUREG-1757 (Banovac et al. 2006; Schmidt et al. 2006; Fredrichs et al. 2003), with the exception of those sections that are not applicable because the U.S. Department of Energy (DOE) is a government agency and those sections that address radioactive sources or radioactive quantity limits not applicable to the repository. Key considerations for development of the decommissioning plans described in this section are consistent with that guidance.

In accordance with 10 CFR 63.21(c)(8), Section 1.12.1 describes the design considerations and features incorporated into the design of the handling facilities intended to facilitate decontamination and dismantlement of the facilities at permanent closure of the repository. Section 1.12.2 discusses plans for permanent closure. In accordance with 10 CFR 63.21(c)(22)(vi), Section 1.12.3 discusses plans for decontamination and dismantlement of the handling facilities and other facilities in the geologic repository operations area (GROA) and identifies and discusses key elements of such plans. As noted in Section 2.1.3.1 of NUREG-1804, preliminary plans discussed in the initial license application are prospective in nature and do not have the same level of detail as the final plans, which will reflect the knowledge gained over the operating life of the surface facilities. The final plans will also incorporate experience, technology, and techniques developed at other DOE sites and nuclear industry facilities that are decontaminated and dismantled during the operating life of the repository. Development of the final plans will include a regulatory assessment to define relevant and updated guidance in more detail.

Final plans for the decontamination and dismantlement of repository surface facilities in the GROA will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The final plans will identify the types, extent, and locations of radiological contamination, along with descriptions of the methods and procedures to be used to achieve decontamination and dismantlement. If necessary, to take into account the environmental impact of substantial changes in the activities planned to be carried out or significant new information regarding the environmental impacts, the DOE will also supplement its environmental impact statement and submit that supplement with the application for a license amendment for permanent closure in accordance with 10 CFR 63.46 and 63.51. Planning, implementation and performance of decontamination and dismantlement activities will be conducted in accordance with applicable regulations and regulatory guidance. The planning process will include updating the Operational Radiation Protection Program (Section 5.11) to encompass the scope of decontamination and dismantlement activities.

1.12.1 Design Considerations to Facilitate Permanent Closure and Dismantlement *[NUREG-1804, Section 2.1.3.3: AC 1]*

The design features of the repository are compatible with the objectives of safe and economical permanent closure, decontamination, and dismantlement, while maintaining radiation doses to workers and the public as low as is reasonably achievable (ALARA). Design features that support closure, decontamination, and dismantlement are discussed in this section. Evaluation and selection of alternatives for appropriate design features are documented during the design process. Those design features that support closure, decontamination, and dismantlement will be selected, where feasible and economical, over competing alternatives. During the design process, structures, systems, and components are reviewed for decontamination and dismantlement considerations to ensure that features that support waste minimization and worker safety are incorporated and ALARA principles are considered for decontamination and dismantlement activities.

The following requirements and criteria will be applied as the design progresses and more information becomes available to ensure that the design features facilitate and support permanent closure and decontamination and dismantlement, while maintaining radiation doses to workers and the public ALARA:

- Selection of materials and processes to minimize waste production
- Minimizing materials that are susceptible to neutron activation to minimize production of radioactive waste
- Selection of materials and incorporation of features intended to ease decontamination and dismantlement or waste processing procedures, consistent with required stringent seismic design requirements; for example, reinforced concrete structures that facilitate demolition techniques

- Use of construction materials and surface finishes to minimize porosity, crevices, and rough machine marks on structures, systems, and components to limit the potential for contamination and facilitate ease of decontamination

- Use of smooth or special protective coatings or polished stainless steel metal surfaces, where applicable, that preclude penetration into porous materials by radioactive gas, condensate, deposited aerosols, or spills to facilitate decontamination by surface treatment

- Stainless-steel-lined wet handling pool with a leak-detection drainage system to minimize the contamination of concrete around the pool

- Use of confinement systems to contain and minimize the spread of potential radioactive contamination generated during process operations and to isolate noncontaminated areas of the surface facilities from potentially contaminated areas

- Incorporation of features to contain leaks and spills, such as curbs and berms, to minimize the number and extent of contaminated areas

- Incorporation of waste minimization techniques

- Use of exhaust ducting and high-efficiency particulate air (HEPA) filters for the exhaust ventilation system of areas or rooms that may become contaminated (Sections 1.2.3.4.1, 1.2.4.4.1, 1.2.5.5.1, and 1.2.6.4.1)

- Incorporation of features that would maintain occupational and public radiation doses ALARA during decommissioning.

1.12.2 Plans for Permanent Closure
[NUREG-1804, Section 2.1.3.3: AC 2]

An application for a license amendment will be submitted before permanent closure of the repository in accordance with 10 CFR 63.51. The application will include:

- An updated assessment of the performance of the repository for the period after permanent closure. The updated assessment will include performance confirmation data collected under the program required by 10 CFR 63, Subpart F, and pertinent to compliance with 10 CFR 63.113.

- A description of the program for postpermanent closure monitoring of the repository.

- A detailed description of the measures to be employed, such as land use controls, construction of monuments, and preservation of records, to regulate or prevent activities that could impair the long-term isolation of emplaced waste within the repository and to ensure relevant information will be preserved for the use of future generations. At a minimum, these measures will include:

  – Identification of the site and GROA by monuments that have been designed, fabricated, and placed to be as permanent as is practicable.

  – Placement of records in the archives and land record systems of local, state, and federal government agencies and archives elsewhere in the world that would be likely to be consulted by potential human intruders, such records to identify the location of the GROA, including the underground facility, boreholes, shafts and ramps, the site boundaries, and the nature and hazard of the waste.

  – Continued oversight to prevent activities at the site that may pose an unreasonable risk of breaching the engineered barriers of the repository or increasing the exposure of individual members of the public to radiation beyond allowable limits.

- Geologic, geophysical, geochemical, hydrologic, and other site data that are obtained during the operational period, pertinent to compliance with 10 CFR 63.113.

- The results of tests, experiments, and other analyses relating to backfill of excavated areas; drip shields; waste packages; interactions between natural features and engineered systems; and other tests, experiments, or analyses pertinent to compliance with 10 CFR 63.113. A description of the planned activities to support closure of the subsurface facility is provided in Section 1.3.6.

- Revisions that are substantial in the plans for permanent closure.

- Other information bearing on permanent closure that was not available at the time a license to receive and possess was issued.

1.12.3 Plans for Decontamination and Dismantlement of Surface Facilities
[NUREG-1804, Section 2.1.3.3: AC 2]

The preparation for decontamination and dismantlement begins with the design and carries through the operation of the facility. Proper design and effective radiological operation of the facility aid the eventual permanent closure, decontamination, and dismantlement actions, while maintaining radiation doses to workers and the public consistent with ALARA principles.

Operation of the facilities will be conducted in a manner that ensures that work is performed safely and in a manner that provides adequate protection for the employees, public, and environment. This operational practice ensures that aspects of environmental management, including pollution prevention and waste minimization, will be implemented, to the extent practicable, to operate the surface nuclear facilities as “clean” facilities (BSC 2008, Section 5). This operational approach will support a need for minimal decontamination during the decontamination and dismantlement process.

An Operational Radiation Protection Program will be established (Section 5.11) that is focused on the concept of maintaining radiation doses ALARA. The Operational Radiation Protection Program is implemented through management and engineering controls that ensure activities related to the radiological aspects of design, construction, operations, maintenance, and decontamination and dismantlement of the repository are conducted to keep individual and collective doses to workers and the public consistent with ALARA principles.

One of the ALARA design goals is to minimize the number and the extent of areas that become radioactively contaminated during routine operations. Containing radioactive contamination in a few designated areas or rooms minimizes the impact to ongoing operations, reduces worker doses, minimizes the amount of low-level radioactive waste for decontamination and dismantlement, and also reduces the resources that will be required to implement decontamination and dismantlement activities. The Operational Radiation Protection Program will include the necessary controls to minimize the potential spread of contamination. More information regarding ALARA program objectives is presented in Section 1.10.

The decision to implement a predominantly canister-based approach for handling SNF and HLW using a transportation, aging, and disposal (TAD) canister system was in large part made to (1) minimize, to the extent practicable, contamination of the surface waste handling facilities in the GROA and potential releases of radioactive materials into the surrounding environment; (2) facilitate eventual decontamination and dismantlement of the surface waste handling facilities prior to decommissioning and permanent closure of the repository; and (3) minimize, to the extent practicable, the generation of radioactive waste during operations and prior to permanent closure (all as required in 10 CFR 20.1406).

Specific examples of the use of facility design and operations to facilitate eventual decontamination and dismantlement and minimizing the amount of low-level radioactive waste generated during operations and decommissioning include (BSC 2008, Appendices A, C, D, E, F, and K):

1. The Initial Handling Facility (IHF) and the Canister Receipt and Closure Facilities (CRCFs) will receive, handle, and package only canistered wastes. Approximately 90% of the commercial SNF will be received at the repository in TAD canister systems. Commercial HLW from West Valley, Naval Nuclear Propulsion Program SNF, and defense HLW will be received in canisters. With the exception of some DOE-managed commercial SNF, DOE-managed SNF will also be received in canisters. As a result, the potential for radiological contamination within the IHF and CRCFs is minimized, and, there will be no significant low-level radioactive waste resulting from operations or decontamination and dismantlement of the IHF and CRCFs.

2. The Receipt Facility (RF) is designed to transfer, without opening, dual-purpose canisters (DPCs) and TAD canisters from rail transportation casks to aging overpacks. The RF will only receive and handle canistered wastes. As a result, the potential for radiological contamination within the RF is minimized, and there will be no significant low-level radioactive waste resulting from operations or decontamination and dismantlement of the RF.

3. The approximately 10% of the commercial SNF that will not be in TAD canisters, including some of the DOE-managed commercial SNF, will be shipped to the repository as uncanistered SNF assemblies in NRC-certified transportation systems. The Wet Handling Facility (WHF), which includes a deep water-filled pool of borated water, is designed to receive, handle, and place the commercial SNF assemblies into TAD canisters. The pool provides a well-shielded, non-oxidizing, cooling environment for handling and transferring commercial SNF assemblies into TAD canisters and minimizes the potential for spread of radiological contamination outside of the pool.

4. The estimated throughput of the WHF will be approximately 400 MTHM per year, which equates to a production rate of less than one TAD canister per week. This provides the opportunity, if necessary, to use the WHF to open any damaged transportation casks, canisters, TAD canisters, or waste packages and to remove the contained wastes and place them into TAD canisters underwater in the pool. As a result, the potential for radiological contamination in other surface waste handling facilities is minimized.

5. After welding, the TAD canisters will be decontaminated, as necessary, as they are removed from the pool in the WHF.

6. Any radiological contaminants released during normal and remediation operations, including any radiological contaminants in the transportation casks and DPCs opened in the pool, will be captured in the pool and continuously removed by filters in its pool water treatment and cooling system. As a result, the potential for deposition of radiological contaminants on system surfaces is minimized.

7. A decision was made to treat, package, transport, and dispose offsite all of the low-level radioactive waste resulting from waste receipt and handling operations in the GROA. There will be no onsite, near-surface low-level radioactive waste disposal facility that requires decommissioning prior to permanent closure.

8. Most of the low-level radioactive waste generated at the repository will result from waste handling operations in the WHF. The low-level radioactive waste will primarily consist of dry, mildly contaminated cloth and personnel protective equipment. The low-level radioactive waste will also include dry, surface-contaminated, empty DPC shells and wet filter canisters from the recirculating water filtering system for the deep-water pool. In addition, the low-level radioactive waste will include any dry, contaminated HEPA filters from the other surface waste handling facilities.

The DOE recognizes that certain information needs to be gathered and retained over the life of the project to properly plan and execute decontamination or decontamination and dismantlement of the facility. Specific types of information that will be gathered and preserved are identified in the following subsections. The DOE will ensure the necessary information will be available and defensible at the time of decontamination and dismantlement. Records will be maintained as described in Section 5.2.1.

A determination must be made that the surface facilities are no longer required to support SNF and HLW handling, processing, emplacement, retrieval operations, or subsurface closure. Following this determination, a decontamination and dismantlement management organization will be established, and a decontamination and dismantlement plan will be developed.

Key considerations for development of the decontamination and dismantlement plan include:

- Evaluation completed and decision made that surface facilities are no longer required to support SNF and HLW handling or a retrieval action
- Establishment of the decontamination and dismantlement management organization
- Evaluation of the radiological status of the surface facilities to determine the extent of radioactive contamination
- Evaluation of decontamination and dismantlement strategies
- Development of the decontamination and dismantlement plan and submittal to the NRC for review and approval.

Information that will be available to plan and execute decontamination and dismantlement, while maintaining radiation doses to workers and the public consistent with ALARA principles, is described in the sections that follow.

1.12.3.1 Facility Operating History

The information that will be available to facilitate decontamination and dismantlement includes the records documenting radioactive material and contamination at the facility and records of the facility operating history. Also included will be the type of radiation monitoring equipment used and activities that could have led to residual contamination or radioactivity being present at the site.

The records that document when the radioactive material is received, processed, and emplaced, as well as the locations of the processing activities, will include:

- The types of radioactive material received and processed at the GROA
- The nature of the authorized use of radioactive materials at the GROA
- The activities at the GROA, including routine and nonroutine activities, such as spills or releases, that could have contributed to residual radioactive material being present at the GROA, and the measures immediately taken to remove such contamination
- The activities authorized under the license
- Past authorized activities using licensed radioactive material at the site

- Activities involving radioactive material that could contribute to residual radioactivity being present at the site prior to the start of licensed operation

- Previous decontamination, dismantlement, or remedial activities at the site.

To facilitate future decontamination and dismantlement activities, contamination of areas at the facility will be documented, including the radiological surveys performed routinely during the operational period of the facility and surveys performed in response to events such as leaks and spills. The surveys document and quantify contaminated areas and are used to help plan actions to keep the contamination levels consistent with ALARA principles. More information regarding facility survey protocols established by the Operational Radiation Protection Program are described in Section 5.11.

The records of the facility operating history will be created and maintained in accordance with the records management and document control processes discussed in Section 5.2.1 and will be available at the time of permanent closure and decommissioning.

1.12.3.2 Facility Description

The information related to the facility and its environs that is required to estimate doses to onsite and offsite populations during and at the time of decontamination and dismantlement will include:

- A description of the GROA

- A description of the population distribution

- A summary of uses and potential future uses of land in and around the site

- Descriptions of the site meteorology, geology, seismology, climatology, surface and groundwater hydrology, and geotechnical characteristics

- Descriptions of the natural and water resources at the site

- The radiological impacts of the planned decontamination activities or the planned decontamination and dismantlement activities for the GROA and its surrounding areas

- Impacts of the environment on the site (e.g., those due to floods, tornadoes, and earthquakes).

This information, along with the original design records, documentation of modifications, and maintenance history for the facility, will be created and maintained in accordance with the records management and document control processes described in Section 5.2.1.

1.12.3.3 Radiological Status of the Facility

The information concerning the radiological status of the facility at the time of decontamination and dismantlement that will be available to facilitate the decontamination and dismantlement process is described in this section and includes the types and the extent of radioactive contamination at the facility. Decontamination and dismantlement actions will be performed for the surface nuclear facilities. The information to perform these actions will be based on the facilities operational records and data, radiological surveys and assessments, and safety and hazards analyses. Evaluations of this information will provide a basis for the anticipated magnitude of decontamination activities or the anticipated magnitude of decontamination and dismantlement activities. Records associated with the radiological status of the facility, including those for soil contamination and surface and groundwater contamination, will be created and maintained in accordance with the records management and document control processes described in Section 5.2.1.

1.12.3.3.1 Structures and Buildings

Information concerning structures and buildings that will be available to facilitate decontamination and dismantlement will include:

- A list or description of structures that contain residual radioactive material in excess of site background levels, including the types of materials that are expected to be generated during decontamination and dismantlement operations, such as structural and component metal, concrete, activated components, contaminated piping, wood, and plastic
- A summary of the structures and locations that have not been impacted by licensed operations and the rationale for the conclusion that they have not been impacted
- A list or description of each room or work area within each of these structures
- A summary of the background radiation levels used during scoping or characterization surveys, including baseline information for preexisting radiation levels (BSC 2006, Section 4)
- A summary of the locations of contamination, such as walls, floors, wall and floor joints, structural steel surfaces, ceilings, ventilation ducting, and glove boxes, in each room or work area
- A summary of the radionuclides present at each location, the maximum and average radionuclide activities in disintegrations per minute per 100 cm^2 for removable and for fixed contamination, the chemical form of the radionuclide, and, if multiple radionuclides are present, the radionuclide ratios
- The mode of contamination for each surface, including whether the radioactive material is present only on the surface of the material or if it has penetrated the material
- Approximate quantities of contaminated materials by type at each location
- A summary of the access control measures that may be implemented during remedial action, a description of the radiation protection program developed to protect workers and the public, and the identification of the regulatory requirements that guide the program

- The maximum and average radiation levels in each room or work area
- A scale drawing or map, including compass direction indicators, of the rooms or work areas showing the locations of radioactive contamination and radiation levels.

1.12.3.3.2 Systems and Components

Information concerning systems and components that will be available to facilitate decontamination and dismantlement will include:

- A list or description and the location of systems or components at the facility that contain residual radioactive material in excess of site background levels
- A summary of the radionuclides present in each system or on the component at each location; the maximum and average radionuclide activities in disintegrations per minute per 100 cm^2 for removable and for fixed contamination; the chemical form of the radionuclide; and, if multiple radionuclides are present, the radionuclide ratios
- The maximum and average radiation levels at the surface of each component
- A summary of the access control measures that may be implemented during remedial action, a description of the radiation protection program to be developed to protect workers and the public, and identification of the regulatory requirements that guide the program
- A summary of the background radiation levels used during scoping or characterization surveys (BSC 2006, Section 4)
- A scale drawing or map, including compass direction indicators, of the rooms or work areas, showing the locations of the contaminated systems or components
- Types and approximate quantities of contaminated materials at each location.

1.12.3.3.3 Soil Contamination

Information concerning surface soil contamination that will facilitate decontamination and dismantlement will include:

- A list or description of locations at the facility at which soil contains residual radioactive material in excess of site background levels
- A summary of the background radiation levels used during scoping or characterization surveys, including baseline information for preexisting radiation levels (BSC 2006, Section 4)

- A summary of the radionuclides present at each location; the maximum and average radionuclide activity concentrations; the chemical form of the radionuclide; and, if multiple radionuclides are present, the radionuclide ratios
- The maximum and average radiation dose rate levels at each location
- A summary of the access control measures that may be implemented during remedial action, a description of the radiation protection program developed to protect workers and the public, and identification of the regulatory requirements that guide the program
- A scale drawing or map, including compass direction indicators, of the site, showing the locations of radionuclide material contamination in soil
- Soil characteristics at each contaminated soil location
- Approximate quantities of contaminated soil at each location
- Identification of potential borrow materials (uncontaminated materials from a nearby location that can be used to backfill excavations and reestablish area surfaces), sources, and quantities
- Grading and contouring considerations at each contaminated soil location
- The depth of the soil contamination at each location.

1.12.3.3.4 Potential Water Contamination from Process Operations

There are no natural surface water bodies at the site (Section 1.1.1.2). Stormwater drainage diversion channels will protect the GROA from runoff from the slopes above the facilities. Since the runoff will be diverted around the process operations area, the potential for this water to become contaminated is precluded by design (Section 1.1.4.1.2.2).

A stormwater detention impoundment will collect runoff from the North Portal pad operations area for the purpose of evaporation (Figure 1.2.1-2). Sample analysis of the retained runoff in the impoundment and vicinity surface water and groundwater will be performed and the records of the results will be created and maintained in accordance with the records management and document control processes. Characterization of the impoundment to support determination of the radiological status of the repository will be performed in a manner similar to that for soil contamination.

A second detention impoundment will collect and evaporate cooling tower blowdown and nonradioactive wastewater (Figure 1.2.1-2). The potential for the water in this impoundment to become contaminated is precluded by design. Sample analysis of the contents in this impoundment will be performed and the records of the results will be created and maintained in accordance with the records management and document control processes. The characterization of this impoundment will be performed in a manner similar to that for structures and buildings.

1.12.3.4 Dose Modeling Evaluations

Dose models will be developed 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 at permanent closure decontamination and dismantlement. The model will use the following information:

- Source-term information that includes radionuclides of interest, configuration of sources, and variability of the sources
- A description of the exposure scenario that includes a description of the critical exposure group, including dose rates and time estimates to complete various decontamination and dismantling activities
- A description of the conceptual model of the site that includes the source terms, the physical features reflected in modeling the exposure pathways, and the critical exposure group
- Identification, description, and justification of the mathematical model used
- A description of the parameters used in the analysis
- A discussion of the accuracy and quality control of the dose modeling results
- Input and output files or printouts, if a computer program is used.

The records of the development, review, validation, verification, approval, and use of the dose modeling computer programs will be created and maintained in accordance with the records management and document control processes described in Section 5.2.1.

1.12.3.5 Alternatives for Decontamination and Dismantlement

The decontamination and dismantlement of the facilities will be performed in a manner that will keep radiation doses to the workers and the public consistent with ALARA principles. The information required to facilitate decontamination and dismantlement by evaluating alternative decontamination and dismantlement strategies will include:

- Determining the effort required to decontaminate the facilities to levels that are consistent with ALARA principles, while minimizing the amount of low-level radioactive waste requiring disposal
- Determining the anticipated physical condition of the facilities, components, and structures over time

- Determining environmental impacts
- Determining low-level radioactive waste disposal methods that meet regulatory requirements.

Records associated with alternative decontamination strategies or with alternative decontamination and dismantlement strategies will be created and maintained in accordance with the records management and document control processes.

1.12.3.6 As Low As Is Reasonably Achievable Analyses

An ALARA assessment will be performed to demonstrate that the decontamination and dismantlement plan dose goals for repository workers and members of the public are consistent with ALARA principles. The ALARA assessment will address the target residual radioactivity, the planned remediation activities, the decontamination and dismantlement guidelines to be employed at the facility, and assumptions and justifications required to support the evaluation. The following information will be provided to support the decontamination and dismantlement ALARA goal:

- A description of the ALARA goals
- A description of how the ALARA program will be implemented
- A quantitative cost–benefit analysis, considering ALARA goals
- The assumptions, methods, and information used to estimate costs for lowering doses
- An evaluation that confirms that doses to the public are consistent with ALARA principles.

Records associated with ALARA analyses will be created and maintained in accordance with the records management and document control processes, as described in Section 5.2.1.

1.12.3.7 Planned Decontamination and Dismantlement Activities

The information required to facilitate planned closure, decontamination, and dismantlement activities includes the methods, procedures, schedules, and contractor resources that the repository intends to use to remove residual radioactive material at the GROA to levels that allow for site remediation. Records associated with the planned decontamination and dismantlement activities will be created and maintained in accordance with the records management and document control processes, as described in Section 5.2.1.

In addition to the plans developed for decontamination or decontamination and dismantlement of GROA surface facilities, plans will be developed for decontamination or decontamination and dismantlement of facilities that support the subsurface. The subsurface facility closure process includes the removal of noncommitted materials (Section 1.3.6). These materials will be removed prior to placement of backfill (Section 1.3.6). In addition, the noncommitted material will be characterized prior to removal in order to determine potential levels of radioactive contamination and identify necessary subsequent actions, such as performing decontamination or disposing of the material as low-level radioactive waste.

1.12.3.7.1 Contaminated Structures

Noncommitted subsurface material will be removed prior to permanent closure. Information regarding closure of subsurface structures and facilities is presented in Section 1.3.6. Information concerning contaminated structures and facilities that will be available to facilitate decontamination and dismantlement will include:

- A summary of the remediation tasks planned for each area in the contaminated structure, in the order in which they will occur
- A summary of unique safety or remediation issues associated with remediating a room or an area
- A description of the remediation techniques, such as scabbling, hydrolazing, or grit blasting, that will be employed in each area of the contaminated structure
- A summary of the worker radiation protection methods that will be employed in each room or area, such as personal protection equipment, stepoff pads, and exit monitoring
- A summary of the control procedures that will be employed during decontamination in each room or area, such as scabbler shrouds, HEPA-vented enclosures, or superfine water misting
- A summary of the procedures already authorized for use and those for which approval may be required
- Decontamination activities or decontamination and dismantlement activities will be conducted in accordance with written, approved procedures.

1.12.3.7.2 Contaminated Systems and Components

The methods, procedures, and techniques information concerning contaminated systems or components that will be available to facilitate decontamination and dismantlement will include:

- A description of the techniques, such as scabbling, hydrolazing, or grit blasting, that will be employed to remediate each system
- A summary of unique safety or remediation issues associated with remediating systems or components
- A description of the radiation protection methods, such as personal protective equipment, stepoff pads, and exit monitoring, that will be employed while remediating each system

- A description of the control procedures, such as scabbler shrouds, HEPA-vented enclosures, or superfine water misting, that will be employed while remediating each system

- A summary of the components that will be removed or decontaminated and how the decontamination process will be accomplished

- A summary of the procedures already authorized under the existing license and those for which approval may be required

- A commitment to conduct decontamination and dismantlement activities in accordance with written, approved procedures.

In addition to methods, procedures, and techniques information, development of decontamination and dismantlement plans will incorporate experience and lessons learned from decontamination and dismantlement of other nuclear facilities. The decontamination and development plans will also address application of innovative technologies or methodologies that may be appropriate to improve safety or enhance performance of the decontamination and dismantlement process to support project goals.

1.12.3.7.3 Contaminated Soil

The information concerning plans for remediating contaminated soil will include:

- A summary of the removal or remediation tasks planned for soil at the site, in the order in which they will occur, including which activities will be conducted by licensee staff and which will be performed by a contractor

- A summary of unique safety, removal, or remediation issues associated with remediating the soil

- A description of the techniques that will be employed to remove or remediate soil at the site

- A description of the radiation protection methods, such as personal protective equipment or area exit monitoring, that will be employed during soil removal or remediation

- A description of the control procedures, such as the use of HEPA-vented enclosures during excavation or covering soil piles to prevent wind dispersion, that will be employed during soil removal or remediation

- A summary of the procedures already authorized under the existing license and those for which approval may be required

- A commitment to conduct decontamination and dismantlement activities in accordance with written, approved procedures.

1.12.3.7.4 Surface Water

There are no natural surface water bodies at the site; therefore, no information for surface water contamination is expected to be included in the plan.

A water retention impoundment will be constructed in the GROA for the purpose of water collection and evaporation (Figure 1.2.1-2). This impoundment will be decommissioned after the need to support facility operations is no longer required. Water remaining in the impoundment will be processed according to the appropriate methods based on the sampling results. Information to support planned decontamination and dismantlement activities for the impoundment will be developed following the guidance for contaminated structures or contaminated soil, as appropriate.

1.12.3.7.5 Schedules

Information for the schedule for decontamination and dismantlement will include:

- The remediation tasks in the order in which they will occur, the time required to perform the tasks, and the initiation and completion dates for the tasks
- A statement acknowledging that the dates in the schedule are contingent upon NRC approval of the decontamination and dismantlement plan
- A statement acknowledging that circumstances can change during decontamination and dismantlement, and, if it is determined that the decontamination and dismantlement cannot be completed as outlined in the schedule, that an updated schedule will be provided.

1.12.3.8 Project Management and Organization

The plan for conducting and managing the activities associated with the closure, decontamination, and dismantlement of the repository is to develop a management organization that is responsible for task management. The management organization will be responsible for the design and implementation of the programs necessary to ensure applicable legal and regulatory requirements are met, including programs that manage radioactive waste generated through closure, decontamination, and dismantlement activities. The following sections describe the information that will be provided concerning the detailed plans for the repository project management and organization.

1.12.3.8.1 Management Organization

Information concerning the management organization that will facilitate decontamination and dismantlement will include:

- A description of the management organization, including descriptions of the individual project units within the decontamination and dismantlement project organization, such as project management, health and safety, and remedial activities
- A description of the responsibilities of each of the project units
- A description of the reporting hierarchy within the decontamination and dismantlement project management organization, including a chart or diagram showing the relationship of each project unit to other project units and project management
- A description of the responsibility and authority of each project unit to ensure activities are conducted in a safe manner and in accordance with approved written procedures, including both stop-work authority of each unit and the manner in which concerns about safety issues are managed within the overall decontamination and dismantlement project.

1.12.3.8.2 Decontamination and Dismantlement Task Management

Information concerning decontamination and dismantlement task management will include:

- A description of the manner in which the tasks will be managed, such as through the use of radiological work permits
- A description of how individual decontamination and dismantlement tasks will be evaluated and how the radiological work permits will be developed for each task
- A description of how the radiological work permits will be reviewed and approved by the project management organization
- A description of how radiological work permits will be managed throughout the decontamination and dismantlement project, including how they will be issued, maintained, revised, and terminated
- A description of how individuals performing the decontamination and dismantlement tasks will be informed of the requirements in the radiological work permit, including how they will be initially informed and how they will be informed when a radiological work permit is revised or terminated (Section 5.11).

1.12.3.8.3 Decontamination and Dismantlement Management Positions and Qualifications

Information concerning decontamination and dismantlement organization management positions and qualifications will include:

- A description of the duties and responsibilities of management positions and the reporting responsibility of the positions
- A description of the duties and responsibilities of chemical, radiological, physical, and occupational safety-related positions and the reporting responsibility of the positions
- A description of the duties and responsibilities of engineering, quality assurance, and waste management positions and the reporting responsibility of the positions
- The minimum qualifications for each of the positions described above, along with the qualifications of replacements for these management positions
- A description of decontamination and dismantlement safety committees, including the membership of the committees, the duties and responsibilities of each committee, and the authority of each committee
- A description of the qualifications, authority, and responsibilities of the designated project manager responsible for radiation safety.

1.12.3.8.4 Training

Information concerning training will include:

- A description of the radiation safety training that the licensee will provide to each employee, including preemployment, annual or periodic training, and specialized training to comply with 10 CFR Part 19
- A description of worker briefings that will be provided at the beginning of each workday or job task, as appropriate, to familiarize workers with job-specific procedures or safety requirements
- A description of the documentation that will be maintained to demonstrate that training commitments are met.

1.12.3.9 Radiological Health and Safety Program during Decontamination and Dismantlement

Information provided will include a description of the radiological health and safety program to comply with 10 CFR Part 20 that will be implemented during decontamination and dismantlement. The preclosure Operational Radiation Protection Program will be modified to address decontamination and dismantlement activities and will include:

- Workplace air sampling
- Respiratory protection
- Internal exposure determination
- External dose determination
- ALARA principles
- A contamination control program
- Radiation protection instrumentation use
- Nuclear criticality safety
- Radiation protection audits, inspections, and a record-keeping program.

1.12.3.10 Environmental Monitoring and Control Program

Information required to facilitate decontamination and dismantlement with respect to environmental monitoring and control will include:

- A description of ALARA goals and implementation plans for effluent control

- A description of the procedures, engineering controls, and process controls to maintain doses consistent with ALARA principles

- A description of the ALARA reviews and reports to management.

The Environmental Radiological Monitoring Program (Section 5.11.3.11) established before the opening of the repository and maintained throughout its operating period will be evaluated and revised, as necessary, to measure and record potential impacts to the site environment during closure and during decontamination and dismantlement. At the time of decontamination and dismantlement, in accordance with document control procedures (Section 5.2.1), records of the Environmental Radiological Monitoring Program in and around the facility will be available to evaluate and use for closure planning.

1.12.3.11 Low-Level Radioactive Waste Management Program

Information required to facilitate decontamination and dismantlement with respect to the management of low-level radioactive waste generated through planned closure, decontamination, and dismantlement activities will be maintained in accordance with the records management and document control processes described in Section 5.2.1.

1.12.3.11.1 Preliminary Estimates of the Types and Quantities of Low-Level Radioactive Waste

The preliminary estimated volume of low-level radioactive waste generated during the repository closure phase is approximately 3,500 m^3 after treatment (DOE 2002, Volume 1, Chapter 4, Table 4-42). The low-level radioactive waste expected to be generated during repository operations is addressed in Section 1.4.5.1. Updated estimates of the types and quantities of low-level radioactive waste that may be generated during closure, decontamination, and dismantlement activities will be made during development of the plan for those activities. That information will include:

- The types of low-level radioactive waste that are expected to be generated, including solidified liquids, soil, structural and component metal, concrete, activated components, contaminated piping, wood, and plastic
- The estimated volume of each solid low-level radioactive waste type
- The radionuclides, including the estimated activity of each radionuclide, in each estimated solid low-level radioactive waste type
- The volumes of Class A, Class B, and Class C solid low-level radioactive waste that will be generated
- A description of how and where each of the solid low-level radioactive wastes will be stored at the GROA prior to shipment for disposal
- A description of how each of the solid low-level radioactive wastes will potentially be treated and packaged to meet disposal site acceptance criteria prior to shipment for disposal.

1.12.3.11.2 Preliminary Plans for Minimizing and Disposing of the Quantities of Low-Level Radioactive Waste

Plans for minimizing the quantities of low-level radioactive waste and for disposing of the low-level radioactive waste will include:

- A description of how volumetrically contaminated material will be managed
- A description of how contaminated soil or other loose solid low-level radioactive waste will be prevented from being redisbursed after exhumation and collection
- A description of the waste volume reduction techniques to be used to minimize the amount of waste requiring burial
- The name and location of the disposal facility intended to be used for each solid low-level radioactive waste type
- A description of the methods intended to be used to package and transport each waste type to its designated disposal facility.

1.12.3.12 Quality Assurance Program

Information required to facilitate decontamination or decontamination and dismantlement with respect to quality assurance will be integrated with the preclosure Quality Assurance Program (Section 5.1) and will include:

- A description of the organization that will be responsible for implementing the Quality Assurance Program
- A description of the Quality Assurance Program, including descriptions of the manner in which quality assurance activities are controlled
- A description of the manner in which Quality Assurance Program documents will be controlled
- A description of how measuring and test equipment will be controlled
- A description of how conditions adverse to quality will be corrected
- A description of the quality assurance records that will be maintained
- A description of the audits and surveillance that will be performed as part of the Quality Assurance Program.

1.12.3.13 Facility Radiation Surveys

Radiological information concerning the repository facilities will be obtained from:

- Historical records gathered during the preoperational and operational period of the facility
- Characterization surveys performed during planning for decontamination and dismantlement
- Routine and special radiological surveys performed during decontamination and dismantlement
- Final radiological surveys in support of license termination.

Records associated with facility radiation surveys will be created and maintained in accordance with the records management and document control processes.

1.12.3.14 Development of a Decontamination and Dismantlement Plan

The anticipated period of decontamination and dismantlement will extend from NRC approval to permanently close the repository until termination of the license as set forth in the approval document. Nuclear facilities will be decontaminated and dismantled on a basis that is determined by operational needs for their process functions. The operating period for the nuclear facilities is approximately 50 years after which their need and subsequent removal would be determined on a case by case basis. Potential new construction for decontamination and dismantlement would need to be completed in time to support completion of decontamination and dismantlement.

The development of decontamination and dismantlement plans will include the following:

- During preparation of final decontamination and dismantlement plans, a review and evaluation of regulatory guidance will be performed to identify applicable references and information that are appropriate to guide and support the decontamination and dismantlement process. The requirements, considerations, procedures, and protocols identified during the guidance review and evaluation will form the basis for final decontamination and dismantlement plans and will guide implementation, execution, and completion of the decontamination and dismantlement work.

- Lessons learned from nuclear facilities decontaminated and dismantled, including hazards associated with such decontamination and dismantlement activities; facilities and equipment available for use in decontamination and dismantlement operations; and licensing requirements for decontamination and dismantlement.

- An application for a license amendment which includes a design with supporting safety analysis and specifications to implement an operational decontamination plan or an operational decontamination and dismantlement plan.

- After NRC issuance of the license amendment, decontamination or decontamination and dismantlement would commence.

A planning timeline for decontamination or for decontamination and dismantlement is shown in Figure 1.12-1.

1.12.4 General References

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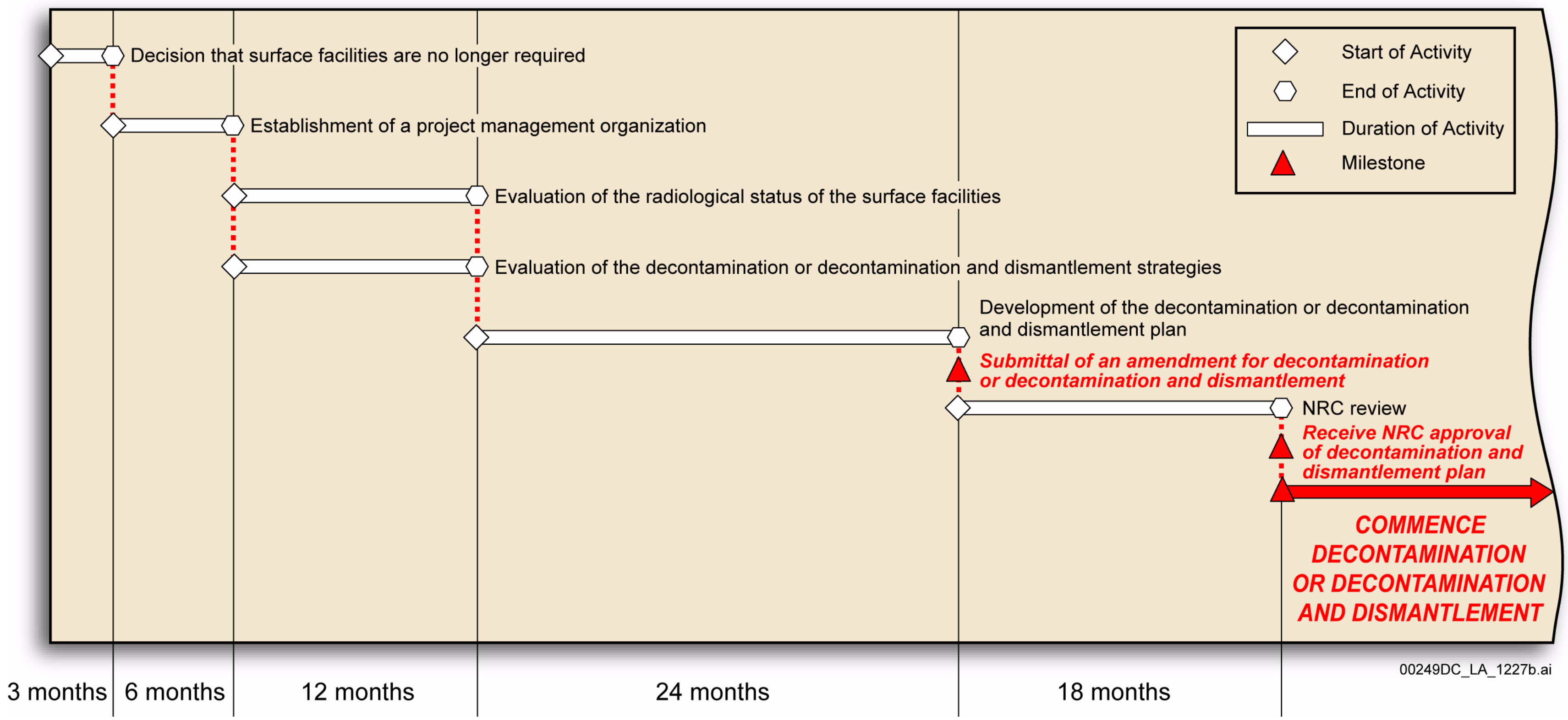

Figure 1.12-1. Decontamination or Decontamination and Dismantlement Timeline

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CONTENTS

Page

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TABLES

Page

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FIGURES

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1.13 EQUIPMENT QUALIFICATION PROGRAM

This section provides information that addresses specific regulatory acceptance criteria in Section 2.1.1.6.3 of NUREG-1804 and 10 CFR 63.112(e) to ensure structures, systems, and components (SSCs) important to safety (ITS) have the ability to perform their intended safety functions of preventing the initiation or mitigating the consequences of event sequences. This section also provides information that addresses the quality assurance requirements of 10 CFR 63.142(d) on the design, selection, and suitability of materials, parts, and equipment for active ITS functions. These requirements include those in 10 CFR 63.142(d)(2)(i) to verify the adequacy of design through one or more measures such as verifying a specific design feature through suitable qualifications testing of a prototype unit under the most adverse design conditions. The following table lists the information provided in this section, the corresponding regulatory requirements, and the applicable acceptance criteria from NUREG-1804.

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
1.13.1	Functions of the Equipment Qualification Program	63.112(e)(8) 63.112(e)(13) 63.142(d)(2)(i)	Section 2.1.1.6.3: Acceptance Criterion 1(2)(h) Acceptance Criterion 1(2)(m)
1.13.2	Equipment Qualification Program Requirements	63.112(e)(8) 63.112(e)(13) 63.142(d)(2)(i)	Section 2.1.1.6.3: Acceptance Criterion 1(2)(h) Acceptance Criterion 1(2)(m) Section 2.1.1.7.3.1 Acceptance Criterion 1(5)
1.13.3	Environmental Qualification Process	63.112(e)(8) 63.112(e)(13) 63.142(d)(2)(i)	Section 2.1.1.6.3: Acceptance Criterion 1(2)(h) Acceptance Criterion 1(2)(m)
1.13.4	Seismic Qualification Process	63.112(e)(8) 63.112(e)(13) 63.142(d)(2)(i)	Section 2.1.1.6.3: Acceptance Criterion 1(2)(h) Acceptance Criterion 1(2)(m)

1.13.1 Functions of the Equipment Qualification Program
[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(h), (m)]

An equipment qualification program, which applies to active electrical and active mechanical SSCs that are ITS (passive SSCs do not require this type of specialized qualification), will be prepared and implemented to:

- Ensure the ability of ITS SSCs to perform their intended safety functions under applicable environmental, seismic, and event sequence conditions
- Ensure the availability, reliability, and component-aging management of ITS SSCs

- Ensure the materials, parts, and equipment used as ITS SSCs are suitable for the application
- Verify the adequacy of the design through qualification testing or analysis.

1.13.2 Equipment Qualification Program Requirements
[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(h), (m); Section 2.1.1.7.3.1: AC 1(5)]

The equipment qualification program for ITS active equipment located in a harsh environment will be applied to ITS SSCs and will include evaluating age-related sensitivity; demonstrating performance under applicable environmental, seismic, and event sequence conditions; and maintaining the qualification for the duration of the service life of the SSC. Qualification plans will be developed for ITS SSCs to account for the unique materials, environments, functions, and performance requirements.

The equipment qualification program will be developed consistent with the guidelines contained in Regulatory Guide 1.89, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*, and the updated IEEE Std 323-2003, *IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*, as appropriate, for application to the repository's environmental conditions. ITS active electrical and active mechanical equipment located in a mild environment will be qualified to the applicable provisions of IEEE Std 323-2003.

The equipment qualification program for ITS active electrical and active mechanical located in harsh environments will also be used to satisfy the requirements of codes and standards governing the design, fabrication, installation, and testing of mechanical SSCs. For example, ASME NOG-1-2004, *Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)*, will be the adopted standard for the design of ITS overhead and gantry cranes (Section 1.2.2). ASME NOG-1-2004 identifies a range of environmental conditions that must be considered in the design and construction of cranes (e.g., radiation (total integrated dose), temperature and duration, humidity, and loads due to seismic or off-normal events). In addition, IEEE Std 323-2003 will be used in the design of ITS cranes instrumentation and controls that are relied upon to perform safety design bases functions. The equipment qualification program will be used to implement these requirements through a combination of design specifications, analyses, tests, and inspections.

The equipment qualification program will use the guidelines of Regulatory Guide 1.100, *Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants*, and IEEE Std 344-2004, *IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*, for seismic qualification for those active components required to function during and after a design basis ground motion seismic event.

The equipment qualification program or other appropriate administrative controls will be implemented prior to initiating procurements involving ITS SSCs to ensure that the design of ITS SSCs will adequately incorporate qualification requirements before fabrication, construction, or installation into a repository facility.

The equipment qualification program will prevent common-failures attributable to environmental and seismic conditions by:

- Qualifying ITS equipment located in a harsh environment for conditions that envelope the harsh environments (which include radiation). Examples of harsh radiation environments based upon preliminary assessments are provided in Table 1.13-1.

- Addressing age-related degradation of ITS equipment elastomeric parts in harsh environments.

- Qualifying ITS equipment located in a harsh environment that performs under conditions of normal operations and off-normal operations and during event sequences.

- Including the interfaces and interactions among ITS and non-ITS equipment in harsh environments to ensure performance of nuclear safety design basis functions.

- Including the effects of installation, surveillance, and maintenance for ITS equipment in harsh environments to ensure performance of nuclear safety design basis functions.

- Validating the qualified-life basis for reliability of active ITS components.

1.13.2.1 Harsh and Mild Environments

The equipment qualification program will divide the repository into two environments: harsh and mild. Items in the equipment qualification program will be evaluated to determine if their location in the repository is a mild or harsh environment (IEEE Std 323-2003). Section 1.13.4 discusses the seismic qualification process for ITS SSCs requiring seismic qualification.

A harsh environment is an environment that is postulated to (1) experience significant increase in radiation or an increase in temperature exceeding 130°F, or both, because of event sequences; (2) experience significant increase in radiation or an increase in temperature, or both, because of off-normal environments; or (3) experience an event sequence radiation dose greater than 10^4 rads (gamma) or a total event sequence radiation dose plus the 50-year total integrated operating dose greater than 5×10^4 rads (gamma).

A mild environment is an environment that would be at no time significantly more severe than the environment that would occur during normal operations, including off-normal operations (IEEE Std 323-2003).

The harsh environment qualification process is discussed in Section 1.13.3.1 and will consist of an analysis to identify significant aging mechanisms. The qualification process will include demonstration by type test or analysis that encompasses aging for significant aging mechanisms, and testing and analysis of safety functions for normal, off-normal, and applicable environmental and seismic event sequences.

The mild environment qualification documentation is discussed in Section 1.13.3.4 and will ensure that appropriate performance specifications are prepared for ITS SSCs and that vendors provide certification of conformance to the specifications. Startup testing will be performed to verify design capability.

Figures 1.13-1 through 1.13-17 identify areas of harsh and mild environments, following an event sequence, based upon preliminary heating, ventilation, and air-conditioning design information and expected radiation studies. Thermal and radiological analyses will be developed to confirm the preliminary assessment. During normal operations and during anticipated operational occurrences all areas are in a mild state. Table 1.13-1 shows examples of preliminary bounding harsh radiation environments for active SSCs required to prevent the initiation or mitigate the consequences of event sequences.

1.13.2.2 Equipment Qualification Records

Equipment qualification records will be prepared for equipment subject to the equipment qualification program. These records will provide the objective evidence of qualification of equipment that is ITS.

1.13.2.3 Environmental Qualification Lists

Lists will be developed and maintained to identify the repository locations that contains ITS SSCs in harsh environments and ITS SSCs required to prevent or mitigate event sequences. These lists will contain information related to normal, off-normal, and event sequence environmental parameters per IEEE Std 323-2003.

1.13.2.4 Testing and Analyses

A testing and analysis process for equipment subject to equipment qualification will be prepared to ensure that the ITS SSCs located in a harsh environment are able to perform their intended safety functions in the range of operating environments. Equipment qualification testing and/or analysis will be performed to verify that ITS equipment located in a harsh environment will perform as expected. If qualification testing is performed, qualification plans will be prepared prior to qualification testing and will include applicable environmental conditions, performance requirements, and acceptance criteria (IEEE Std 323-2003).

1.13.2.5 Procurement, Installation, Maintenance, Surveillance, and Monitoring

Equipment qualification requirements will be defined prior to procurement and will be integrated with processes for installation, startup testing, maintenance, surveillance, aging management, and condition monitoring to ensure ITS SSCs in the equipment qualification program perform their intended safety functions. The equipment qualification program will evaluate the results of environmental monitoring in spent nuclear fuel and high-level radioactive waste handling areas at Yucca Mountain to determine whether equipment remains in a qualified condition (IEEE Std 323-2003).

1.13.2.6 Corrective Action Program Evaluation

Equipment failures will be documented and evaluated by the corrective action program.

1.13.2.7 Equipment Qualification Margin

An equipment qualification margin will be included in the equipment qualification program for ITS equipment located in a harsh environment. The equipment qualification margin will account for reasonable uncertainties in demonstrating satisfactory performance, normal variations in commercial production, and uncertainties in measurement and test equipment. Equipment qualification margins will consider temperature, radiation levels, electrical power supply surges, or other physical parameters important to the performance of the ITS SSCs (IEEE Std 323-2003).

1.13.3 Environmental Qualification Process
[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(h), (m)]

Active electrical and active mechanical equipment in the equipment qualification program will be designed to have the capability of performing their safety functions under normal operations, off-normal operations, and event sequence environments and for the duration for which the event sequence safety functions are required. The active electrical and active mechanical equipment environmental capability for equipment located in a harsh environment will be demonstrated by appropriate testing and analyses. The environmental qualification process for ITS active electrical and active mechanical equipment located in a harsh environment will follow the guidelines of Regulatory Guide 1.89 and the updated IEEE Std 323-2003, as appropriate, for application to the repository environmental conditions. ITS active electrical and active mechanical equipment located in a mild environment will be qualified to the applicable provisions of IEEE Std 323-2003.

1.13.3.1 Harsh Environment Qualification

The preliminary assessment shows that the harsh environments at the repository are due to radiation, temperature, or both.

The harsh environment qualification process will consist of an analysis to identify significant aging mechanisms for the ITS active electrical and active mechanical SSCs. The qualification process will include a type test program and/or analysis, including aging for significant aging mechanisms and demonstration of operability for normal, off-normal, and applicable event sequence conditions. A significant aging mechanism is an aging mechanism that, under normal and off-normal service conditions, causes degradation of equipment that progressively and appreciably renders the equipment vulnerable to failure to perform its safety functions during normal conditions and during off-normal and event sequence conditions (IEEE Std 323-2003).

1.13.3.2 Qualification Methods

ITS active electrical and active mechanical equipment located in a harsh environment will be qualified by a combination of the methods discussed below.

1.13.3.2.1 Analysis

Qualification by analysis requires an assessment or mathematical model of the equipment to be qualified. The analysis typically includes application of physical laws, results of test data, operating experience, and environmental condition indicators. Analysis of data and tests for material properties, equipment rating, and environmental tolerance can be used to demonstrate qualification (IEEE Std 323-2003).

1.13.3.2.2 Type Testing

A type test subjects a representative sample of equipment, including interfaces, to be qualified to a series of tests, including age conditioning, to simulate the effects of significant aging mechanisms during normal operation. The sample is subsequently subjected to event sequence testing that simulates and thereby establishes the tested configuration for installed equipment service, including mounting, orientation, interfaces, and expected environments. A successful type test demonstrates that equipment can perform its nuclear safety design basis functions for the required service life (IEEE Std 323-2003).

1.13.3.2.3 Operating Experience

Performance data from equipment of similar design that has successfully operated under known service conditions may be used in qualifying other equipment to equal or less severe conditions. Applicability of this data depends on the adequacy of documentation establishing past service conditions, equipment performance, and similarity against the equipment to be qualified and upon which operating experience exists. When qualification for an event is required, the equipment qualification program will require demonstration of functionality during an event sequence based on prior equipment operating experience under equivalent or more severe environmental conditions (IEEE Std 323-2003).

1.13.3.2.4 Combined Methods

Equipment may be qualified by using combinations of a type test, operating experience, and analysis. For example, where type test of a complete assembly is not possible, component testing supplemented by analysis may be used (IEEE Std 323-2003).

1.13.3.3 Qualified Life

The qualified life will be established based on the significant aging mechanisms and will include consideration for temperature time-dependent, radiation-dependent, and operational cycle–dependent aging mechanisms. A qualified life determination considers degradation of equipment capability prior to and during service (IEEE Std 323-2003). Inherent in establishing a qualified life is that a qualified condition is also established. This qualified condition is the state of degradation for which successful performance during an event sequence is demonstrated.

Instead of a qualified life, condition monitoring may be used to determine if qualified equipment is suitable for further service. Condition monitoring for environmental qualification purposes will monitor one or more condition indicators to determine whether equipment remains in a qualified condition.

Initial environmental equipment qualification may yield a qualified life that is less than the anticipated service life of the equipment. Prior to the end of qualified life, the equipment will be maintained, replaced, or life extension analysis will be performed.

1.13.3.4 Environmental Qualification Documentation

The purchase specifications will contain a description of the active ITS safety functions and the identification of the normal environmental conditions and conditions that may exist during off normal and event sequences for those areas where qualified ITS SSCs are located.

The documentation for harsh environmental qualification will provide evidence that the ITS active mechanical and electrical equipment are qualified for their design bases applications; meet specification requirements; and have a qualified life and periodic surveillance, maintenance, or condition monitoring interval established. Test data used to demonstrate the qualification of the equipment will be organized in a traceable manner that permits independent auditing of the conclusions presented (IEEE Std 323-2003).

The mild environment documentation requirements to demonstrate the qualification of ITS SSCs will be the design and purchase specifications, seismic test reports (if applicable), and an evaluation or certificate of conformance. The design or purchase specifications will contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and conditions that may exist during off-normal and event sequence conditions (Section 7.1 of IEEE Std 323-2003).

1.13.4 Seismic Qualification Process

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(h), (m)]

Active electrical and active mechanical equipment that are ITS and credited with preventing the initiation of or mitigating the consequences of a seismically initiated event sequence will be designed to perform their safety functions during and after the appropriate design basis ground motion seismic event. The active electrical and active mechanical equipment seismic capability will be demonstrated by appropriate testing and analyses. The seismic qualification process will follow the guidelines of Regulatory Guide 1.100 and IEEE Std 344-2004, *IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*, as appropriate for the repository seismic design bases.

1.13.4.1 Methods

The methods for seismic qualification that will be utilized will be to:

- Predict the equipment performance by analysis
- Test the equipment under simulated seismic conditions
- Qualify the equipment by a combination of test and analysis
- Qualify the equipment through the use of experience data.

Method selection will be based on the practicality of the method for the type, size, shape, and complexity of the equipment configuration, whether the ITS function can be assessed in terms of operability or structural integrity alone, and the reliability of the conclusions.

1.13.4.2 Acceptance Criteria

The acceptance criteria for qualification of ITS active electrical and active mechanical equipment will be satisfied by test and analysis.

Testing will be the preferred method to qualify equipment. Both dynamic and static approaches will be used to ensure structural integrity and operability of ITS mechanical and electrical equipment. Test fixtures will be designed to simulate actual service mounting. Test samples will be selected according to type, load, level, and size.

Equipment that has been previously seismically qualified by means of test and analysis equivalent to conditions appropriate for the ITS equipment being qualified will be acceptable, provided that proper documentation is submitted.

1.13.4.3 Process

The seismic qualification process at the repository will utilize the seismic qualification guidance in NRC Regulatory Guide 1.100, and IEEE Std 344-2004, *IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*, for seismic qualification for those active components required to function during and after a design basis ground motion seismic event.

1.13.4.4 Qualification Methods

Using the methods described below, the ITS equipment for facilities at Yucca Mountain will be qualified for seismic ground motion obtained from the in-structure response spectra.

The response spectra for equipment that has experienced a seismic event and performed its intended safety function will be compared to the repository in-structure response spectra to determine if the seismic spectra that equipment experienced envelops the repository in-structure response spectra.

Testing and/or analysis will be conducted to determine equipment ability to perform its intended safety function when subjected to seismic motion consistent with the repository in-structure response spectra.

The response spectra for equipment that has been previously seismically qualified will be compared to the repository in-structure response spectra to determine if the qualified equipment spectra envelops the repository in-structure response spectra.

1.13.4.5 Qualified Life

Establishing the qualified life for the seismic process is the same as described in Section 1.13.3.3.

1.13.4.6 Seismic Qualification Documentation

The seismic qualification documentation requirements will follow IEEE Std 344-2004 and are similar to the documentation provisions described in Section 1.13.3.4.

1.13.5 General References

ASME NOG-1-2004. 2005. *Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).* New York, New York: American Society of Mechanical Engineers. TIC: 257672.

BSC (Bechtel SAIC Company) 2007. *Preliminary Equipment Qualification Environment Bounding Design Basis Values for YMP ITS Surface and Subsurface Facility SSCs*. 000-30R-MGR0-02900-000-000. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070516.0027.

IEEE Std 323-2003. 2004. *IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.* New York, New York: Institute of Electrical and Electronics Engineers. TIC: 255697.

IEEE Std 344-2004. 2005. *IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*. New York, New York: Institute of Electrical and Electronics Engineers. TIC: 258050.

Regulatory Guide 1.89, Rev. 1. 1984. *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.* Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 238593.

Regulatory Guide 1.100, Rev. 2. 1988. *Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants*. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 4636.

Weed Instrument Company, Inc. 2003. *Nuclear Environmental and Seismic Qualification Test Report of Weed Model N97 Pressure (PA/PG) and Differential Pressure (DP) Transmitters.* Test Report Number 3077-RD-0281-003 REV. 1. Round Rock, Texas: Weed Instrument Company, Inc. TIC: 260029.

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Table 1.13-1. Examples of Preliminary Bounding Harsh Radiation Environments for Active ITS SSCs Required to Prevent the Initiation or Mitigate the Consequences of Event Sequences

Components/Area Exposed to Bounding Environment	Harsh Due To	Bounding Design Basis Qualification Levels Yucca Mountain (Gamma)	Maximum Industry Qualification Levels (Gamma)	Margin
Waste Transfer–Canister Transfer Machine/Initial Handling Facility	Radiation	3.01×10^7 Rad Total Integrated Dose	20.0×10^7 Rad Total Integrated Dose	564%
Spent Fuel Transfer Crane/Wet Handling Facility	Radiation	3.01×10^7 Rad Total Integrated Dose	20.0×10^7 Rad Total Integrated Dose	564%

Source: BSC 2007; Weed Instrument Company 2003.

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This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-125.

Figure 1.13-1. Initial Handling Facility (Harsh and Mild Environments)—1st Floor

NOTE: CTM = canister transfer machine; HEPA = high-efficiency particulate air; LLW = low-level waste.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-126.

Figure 1.13-2. Initial Handling Facility (Harsh and Mild Environments)—2nd Floor

NOTE: HVAC = heating, ventilation, and air-conditioning.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-127.

Figure 1.13-3. Initial Handling Facility (Harsh and Mild Environments)—3rd Floor

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-128.

Figure 1.13-4. Wet Handling Facility (Harsh and Mild Environments)—1st Floor

NOTE: CTM = canister transfer machine; DECON = decontamination; HVAC = heating, ventilation, and air-conditioning; LLW = low-level waste.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-129.

Figure 1.13-5. Wet Handling Facility (Harsh and Mild Environments)—2nd Floor Below Elevation 40 ft

NOTE: HVAC = heating, ventilation, and air-conditioning.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-130.

Figure 1.13-6. Wet Handling Facility (Harsh and Mild Environments)—2nd Floor at Elevation 40 ft

NOTE: COMM = communications; HVAC = heating, ventilation, and air-conditioning.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-131.

Figure 1.13-7. Wet Handling Facility (Harsh and Mild Environments)—Floor Below 93 ft

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-132.

Figure 1.13-8. Wet Handling Facility (Harsh and Mild Environments)—Basement

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-133.

Figure 1.13-9. Receipt Facility (Harsh and Mild Environments)—1st Floor

NOTE: CTM = canister transfer machine; HVAC = heating, ventilation, and air-conditioning; LLW = low-level waste.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-134.

Figure 1.13-10. Receipt Facility (Harsh and Mild Environments)—2nd Floor

NOTE: HVAC = heating, ventilation, and air-conditioning.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-135.

Figure 1.13-11. Receipt Facility (Harsh and Mild Environments)—3rd Floor

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-136.

Figure 1.13-12. Canister Receipt and Closure Facility (Harsh and Mild Environments)—1st Floor

NOTE: HVAC = heating, ventilation, and air-conditioning; LLW = low-level waste.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-137.

Figure 1.13-13. Canister Receipt and Closure Facility (Harsh and Mild Environments)—2nd Floor

NOTE: HVAC = heating, ventilation, and air-conditioning.

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-138.

Figure 1.13-14. Canister Receipt and Closure Facility (Harsh and Mild Environments)—3rd Floor

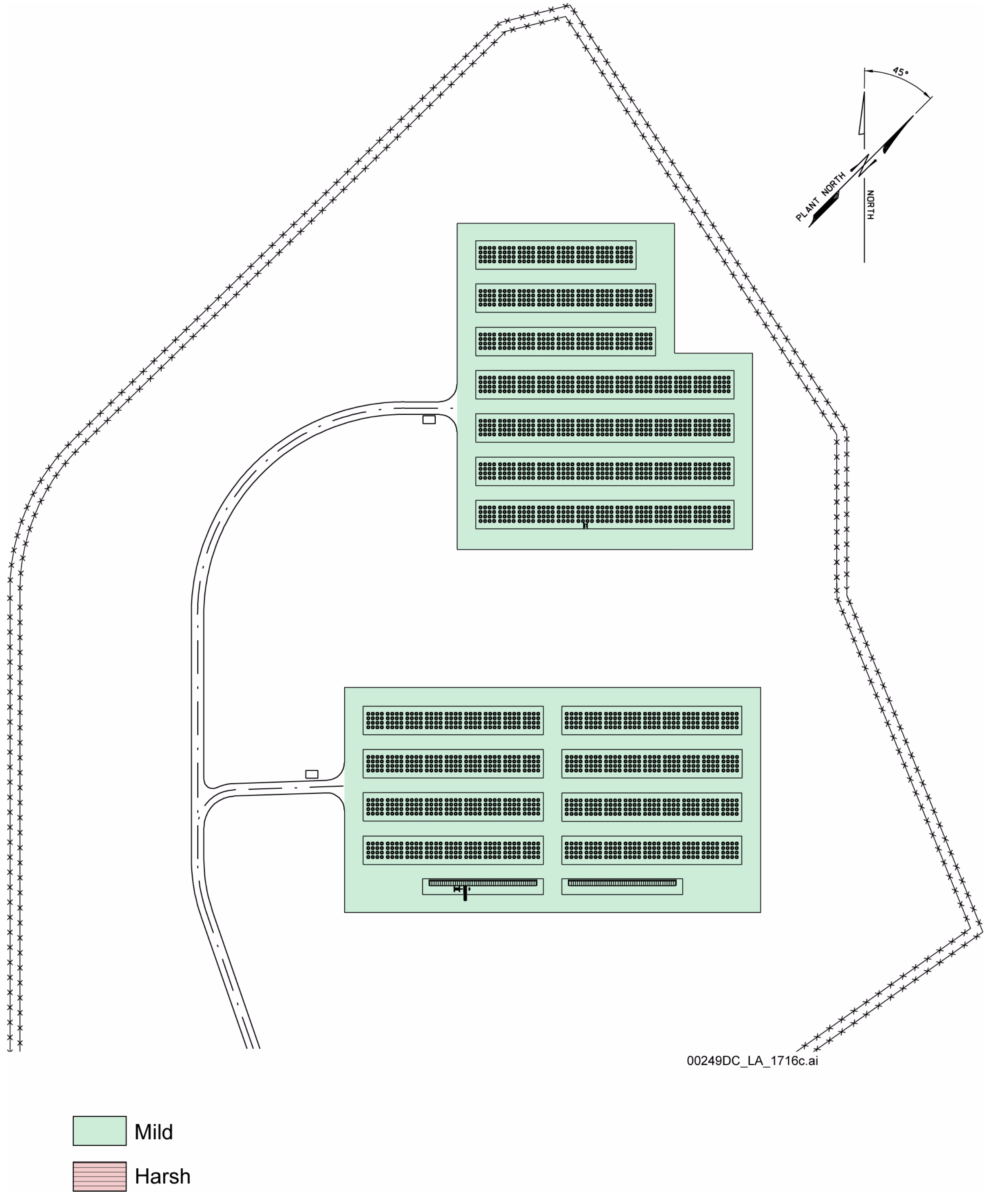

Figure 1.13-15. Aging Pads (Harsh and Mild Environments)

This figure has been designated Official Use Only under the Freedom of Information Act (5 U.S.C. 552), Exemption 2, Circumvention of Statute.

This figure is included in Appendix A: Information Designated as Official Use Only, as Figure A-139.

Figure 1.13-16. Emergency Diesel Generator Facility (Harsh and Mild Environments)

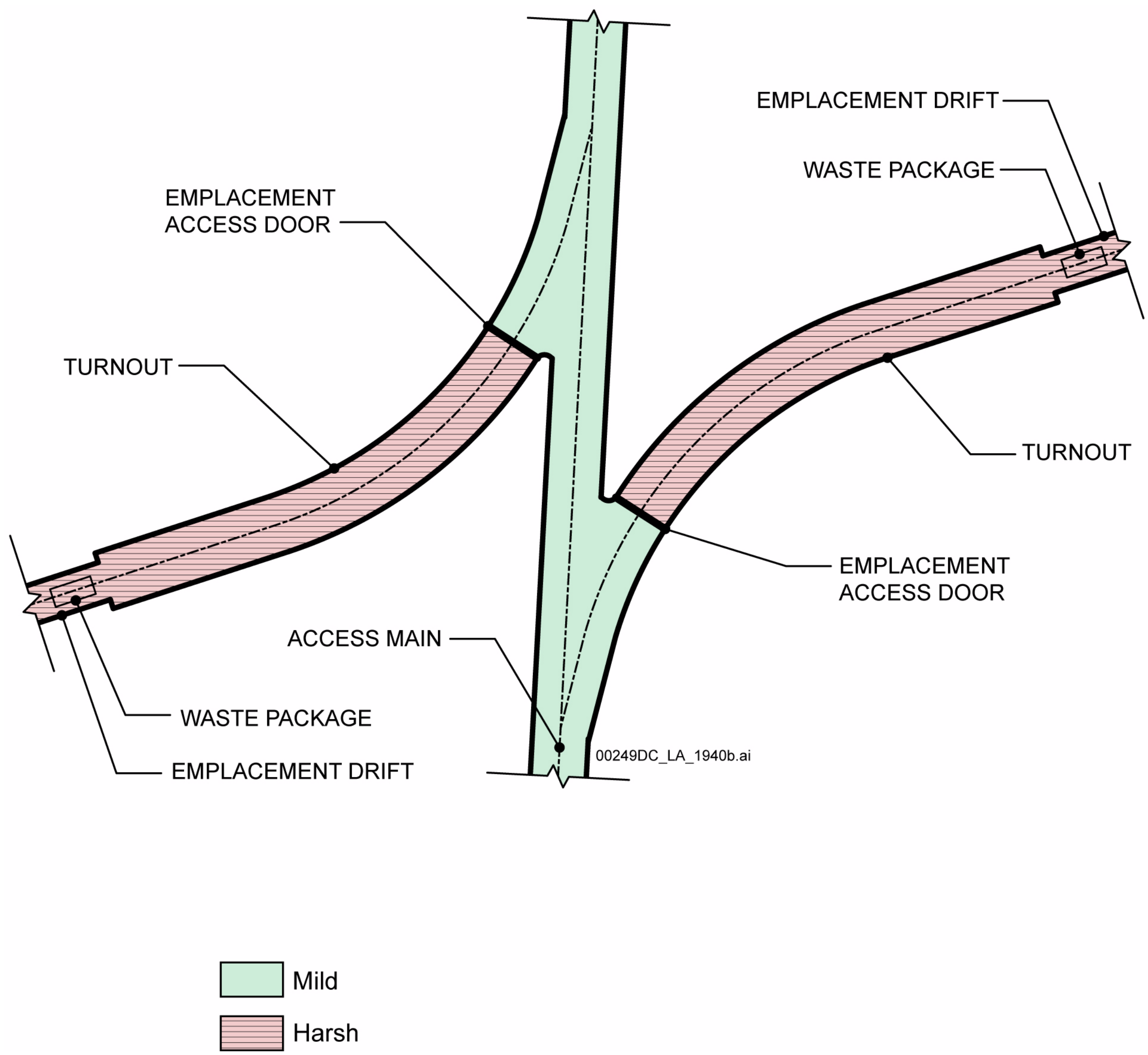

Figure 1.13-17. Subsurface Facilities (Harsh and Mild Environments)

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CONTENTS

Page

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TABLES

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FIGURES

Page

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1.14 NUCLEAR CRITICALITY SAFETY

This section describes the nuclear criticality safety program for the repository during the preclosure period. Postclosure criticality is discussed in Section 2.2.1.4.1. The preclosure criticality safety program is in accord with standard industry practice. The major program areas described are (1) organization and administration; (2) the technical program; and (3) nuclear criticality safety regulations, codes, standards, and guidance. The following table lists the information provided in this section, the corresponding regulatory requirements, and the applicable acceptance criteria from NUREG-1804.

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
1.14.1	Nuclear Criticality Safety Organization and Administration	63.112(e)(6)	Section 2.1.1.6.3: Acceptance Criterion 1(2)(f)
1.14.2	Nuclear Criticality Safety Technical Program	63.112(e)(6)	Section 2.1.1.2.3: Acceptance Criterion 5(2) Section 2.1.1.6.3: Acceptance Criterion 1(2)(f) Section 2.1.1.7.3.1: Acceptance Criterion 1(8) Acceptance Criterion 1(9) Section 2.1.1.7.3.3(I): Acceptance Criterion 1 Acceptance Criterion 4(1) Acceptance Criterion 4(2) Acceptance Criterion 4(4) Acceptance Criterion 4(5) Section 2.1.1.7.3.3(III): Acceptance Criterion 1(1) Acceptance Criterion 1(2) Acceptance Criterion 1(6)
1.14.3	Nuclear Criticality Safety Regulations, Codes, and Standards	63.112(e)(6)	Section 2.1.1.6.3: Acceptance Criterion 1(2)(f) Section 2.1.1.7.3.3(I): Acceptance Criterion 1

The goal of the nuclear criticality safety program is the prevention of a nuclear criticality during the preclosure time period. The goal of preventing preclosure nuclear criticality is achieved through ensuring that, under normal conditions and Category 1 and Category 2 event sequences important to criticality, the calculated effective neutron multiplication factor, k_{eff}, does not exceed the upper subcritical limit as described in Section 1.14.2.3.4.

This section identifies and screens criticality control parameters based on review and analysis of waste forms, canister designs, facility designs and characteristics, and the operational sequences in the various handling facilities. Section 1.6 identifies the hazards and initiating events that impact the criticality parameters that must be controlled. The development, quantification, and categorization of event sequences that impact the parameters that must be controlled are described in Section 1.7, which also identifies those design features and procedural safety controls relied upon to prevent preclosure criticality. These procedural safety controls and design features along with their important to safety (ITS) designation are listed in Section 1.9.

1.14.1 Nuclear Criticality Safety Organization and Administration

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f)]

During the detailed design and construction of the repository, the nuclear criticality safety design functions are performed by the preclosure safety analysis (PCSA) organization with close coordination with the engineering organization. This ensures nuclear criticality safety is integrated into the design process.

Prior to U.S. Nuclear Regulatory Commission (NRC) issuance of a license to receive and possess source, special nuclear, or byproduct material, i.e., spent nuclear fuel (SNF) and high-level radioactive waste (HLW), the existing criticality safety design organization will be expanded to include operational components. This nuclear criticality safety organization will be responsible for development and implementation of administrative practices, procedures, and training for nuclear criticality safety.

Section 5.3 describes the organizational structure for the repository during its operational phase. The criticality safety organization reports to the Criticality Safety manager, who is stationed at the site and reports to the Operations manager. The Criticality Safety manager is responsible for developing and implementing the program for nuclear criticality safety. The minimum qualifications of the Criticality Safety manager are listed in Section 5.3.2.1.8.

As discussed in Section 5.3, nuclear criticality safety training will be developed for repository personnel in accordance with ANSI/ANS-8.20-1991, *American National Standard, Nuclear Criticality Safety Training*.

Nuclear criticality safety administrative practices and procedures for the repository will be developed in accordance with the Quality Assurance Program discussed in Section 5.1 and with ANSI/ANS-8.19-2005, *American National Standard, Administrative Practices for Nuclear Criticality Safety*.

Nuclear criticality safety audits and assessments will be performed in accordance with the Quality Assurance Program described in Section 5.1 and with ANSI/ANS-8.19-2005.

1.14.2 Nuclear Criticality Safety Technical Program

[NUREG-1804, Section 2.1.1.2.3: AC 5(2); Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.1: AC 1(8), (9); Section 2.1.1.7.3.3(I): AC 1, AC 4(1), (2), (4), (5); Section 2.1.1.7.3.3(III): AC 1(1), (2), (6)]

This section addresses the technical aspects of the preclosure nuclear criticality safety program. The requirements, analysis process, and criticality safety evaluation results are presented in Sections 1.14.2.1, 1.14.2.2, and 1.14.2.3, respectively. A detailed example for a specific operation is presented in Section 1.14.2.4 to clarify the application of the preclosure criticality analysis process. The criticality safety analysis presented in this section reflects the current facility designs, expected fuel operations, a conceptual transportation, aging, and disposal (TAD) canister design, a representative dual-purpose canister (DPC) design, and a representative group of SNF types owned by the U.S. Department of Energy (DOE). A future set of evaluations will be performed, as indicated in Table 5.10-3 to demonstrate that actual designs and fuel characteristics, as accepted for receipt, comply with the criticality safety requirements in Section 1.14.2.1. If the transportation cask and canister designs, fuel characteristics, or fuel operations are not bounded by the analysis presented in this section, an update to the safety analysis will be conducted.

The information in this section is based principally on:

- *Preclosure Criticality Analysis Process Report* (BSC 2008a)
- *Preclosure Criticality Safety Analysis* (BSC 2008b).

The preclosure criticality analysis process and criticality evaluation for naval SNF are described in Section 1.14 of the Naval Nuclear Propulsion Program Technical Support Document.

1.14.2.1 Nuclear Criticality Safety Requirements

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.1: AC 1(8), (9)]

The means to prevent and control criticality must be addressed as part of the PCSA required for compliance with 10 CFR Part 63, where the preclosure period covers the time prior to permanent closure activities. One of the requirements of the PCSA as stated in 10 CFR 63, Subpart E, Section 112(e) is to perform an analysis of the performance of the structures, systems, and components (SSCs) to identify those that are ITS. This analysis must include consideration of means to prevent and control criticality.

In order to comply with the preclosure criticality safety requirement of 10 CFR 63, Subpart E, Section 112(e)(6), nuclear criticality is prevented through a combination of the ITS SSCs and procedural safety controls. The project preclosure criticality safety requirement for all canistered and uncanistered SNF is:

> ... the SNF and canister designs, in conjunction with the facility SSCs, shall provide the basis for ensuring subcriticality at the time of delivery to the geologic repository and during all subsequent handling operations, including all event sequences that are important to criticality and have at least one chance in 10,000 of occurring before permanent closure (BSC 2008a, Section 1.4).

1.14.2.2 Nuclear Criticality Analysis Process

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.1: AC 1(8), (9); Section 2.1.1.7.3.3(I): AC 4(1), (2), (4); Section 2.1.1.7.3.3(III): AC 1(1), (2), (6)]

The preclosure nuclear criticality analysis process is summarized in this section. The detailed preclosure criticality safety analysis process is described in *Preclosure Criticality Analysis Process Report* (BSC 2008a). Figure 1.14-1 provides an overview of the preclosure criticality safety analysis process used for all facilities and waste forms. The starting point for the preclosure criticality analysis process is to define criticality design and operational criteria based on review and analysis of waste forms, canister designs, facility designs and characteristics, and the operational sequences in the various handling facilities. In order to determine the criticality potential for each specific waste form and associated facility and handling operations, criticality sensitivity calculations are performed. These calculations evaluate the impact on system reactivity of variations in each of the parameters important to criticality during the preclosure period, which are:

1. **Waste Form Characteristics**—Waste form characteristics including physical form (e.g., size and shape), chemical form (e.g., oxide), mass, density, fissile material type (e.g., ^{235}U) and enrichment. The k_{eff} sensitivity calculations use either bounding or representative fuel characteristics. If waste form characteristics are not bounded in the calculations and the system being evaluated is subject to potential misloads based on the mistaken availability of more reactive SNF than can be handled, then this parameter is identified as needing to be controlled.

2. **Moderation**—Potential moderators that could be present in the geologic repository operations area (GROA) including type (e.g., water), composition (e.g., borated water), density, volume, and location with respect to the SNF. The k_{eff} sensitivity calculations determine maximum or optimum moderation conditions (e.g., type, mass, volume, density) that maintain subcriticality as a function of other relevant parameters.

3. **Neutron Absorber**—Potential fixed neutron absorbers (e.g., borated stainless steel) and soluble neutron absorbers (e.g., borated water) varying from as-designed to complete omission. The k_{eff} sensitivity calculations determine minimum neutron absorber characteristics (e.g., type, loading, concentration) that maintain subcriticality as a function of other relevant parameters.

4. **Geometry**—Potential geometric rearrangement of the SNF (e.g., varying pin or plate pitch) and fuel baskets (e.g., varying flux trap gap). The k_{eff} sensitivity calculations determine the most limiting geometric conditions that maintain subcriticality as a function of other relevant parameters.

5. **Interaction**—Potential neutronic coupling conditions in the GROA between containers (casks, canisters, or waste packages) of similar or different waste forms. The k_{eff} sensitivity calculations determine the most limiting interaction conditions that maintain subcriticality as a function of other relevant parameters.

6. **Reflection**—Potential reflection conditions in the GROA including material type (e.g., concrete), density, and thickness. The k_{eff} sensitivity calculations examine potential reflection conditions as a function of other relevant parameters.

These criticality calculations determine the sensitivity of k_{eff} to variations in any parameter(s) as a function of other relevant parameters in order to provide guidance to hazards identification (Section 1.6), and to event sequence development, quantification, and categorization analyses (Section 1.7) on whether each parameter:

- Does not need to be controlled because it is bounded (i.e., its analyzed value is greater than or equal to the design limit) or its effect is bounded,

- Needs to be controlled if another parameter is not controlled (conditional control), or

- Needs to be controlled because it is the primary criticality control parameter.

Based on internal and external hazards identification and screening analyses (Section 1.6) and on event sequence development and quantification analyses, the event sequences that impact the criticality control parameters that need to be controlled are identified, developed, quantified, and categorized (Section 1.7). If an event sequence important to criticality cannot be screened out as beyond Category 2 (less than one chance in 10,000 of occurring during the preclosure period), criticality evaluations are performed for those end-state configurations over the range of parameters that characterize the event sequence. A configuration is considered acceptably subcritical if the maximum calculated effective neutron multiplication factor (k_{eff}) plus calculational uncertainties is less than or equal to the configuration-specific upper subcritical limit (see Section 1.14.2.3.4 for additional detail).

For end-state configurations where the maximum k_{eff} value exceeds the upper subcritical limit and the probability of occurrence of the end-state configuration exceeds the Category 2 criterion, the event sequence is further extended or refined to credit additional design features or procedural safety controls such that the event sequence probability is reduced to less than the Category 2 criterion. The probability of the extended or refined event sequence may include the additional probability of occurrence of parameters important to criticality, such as degree of moderation, extent of fuel rearrangement, and fuel basket geometric reconfiguration. The end-state configuration is acceptable provided that the probability of occurrence of the extended or refined event sequence does not exceed the Category 2 screening criterion. If the probability of an extended or refined event sequence exceeds the Category 2 screening criterion, design or operational requirements are imposed to reduce the probability of the event sequence to below the Category 2 screening criterion.

The analysis process is continued until all facilities and waste forms have been evaluated, criticality control parameters have been established, and event sequences important to criticality have been identified and evaluated as acceptable. The surface and subsurface facility designs are acceptable with respect to criticality when: (a) each event sequence important to criticality has been shown to be beyond Category 2 or (b) the maximum effective neutron multiplication factor, including calculational uncertainty, for normal operations and end-state configurations of all Category 1 and Category 2 event sequences important to criticality is less than or equal to the configuration-specific upper subcritical limit.

The Monte Carlo N-Particle transport code, MCNP, was used to calculate k_{eff} for the various waste form configurations because it is designed to perform Monte Carlo simulations of particle transport, including k_{eff} calculations for fissile materials (Briesmeister 1997).

1.14.2.3 Nuclear Criticality Safety Evaluation

[NUREG-1804, Section 2.1.1.2.3: AC 5(2); Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.1: AC 1(8), (9); Section 2.1.1.7.3.3(I): AC 1, AC 4(1), (2), (4), (5); Section 2.1.1.7.3.3(III): AC 1(1), (2), (6)]

This section discusses the nuclear criticality safety evaluation for the surface and subsurface facilities, including the Initial Handling Facility (IHF), the Receipt Facility (RF), the Canister Receipt and Closure Facilities (CRCFs), and the Wet Handling Facility (WHF). In addition to operations performed within waste-handling buildings, intrasite operations, including aging, are part of surface operations. Following completion of necessary surface operations to place waste form canisters within waste packages, those waste packages are moved to and emplaced within the subsurface facility. Because transportation casks, canisters, and waste packages are common to more than one facility, a discussion of those components is given in Section 1.14.2.3.1. Further, because many of the facilities and operations are similar from a criticality safety viewpoint, the criticality safety analysis focuses on common waste form-based criticality control parameters as described in Section 1.14.2.3.2.

Other facilities such as the Central Control Center Facility and Emergency Operations Center, the Heavy Equipment Maintenance Facility, and the Warehouse and Non-Nuclear Receipt Facility have no operations with fissile material and are not included in this criticality safety evaluation.

Criticality safety evaluations are performed for normal operations and for the event sequences and configurations that are identified as important to criticality in Section 1.7, using the process described in Section 1.14.2.2. The detailed criticality safety evaluation is documented in *Preclosure Criticality Safety Analysis* (BSC 2008b).

1.14.2.3.1 Transportation Casks, Canisters, and Waste Packages

Transportation casks, DPCs, DOE SNF canisters, HLW canisters, naval SNF canisters, and most TAD canisters are loaded prior to shipment to the repository. Some TAD canisters are loaded at the repository in the WHF. Canister loading is performed in a manner to meet the 10 CFR Part 63 requirements for preclosure and postclosure. In addition, transportation casks and canisters loaded elsewhere for shipment to the repository will also meet 10 CFR Part 71 transportation requirements. Postclosure criticality loading requirements for TAD canisters are discussed in Section 2.2.1.4.1.1.3. Loading criteria for fuel in DOE SNF canisters and transportation casks will be developed by the DOE prior to shipment.

1.14.2.3.1.1 Transportation Casks

Commercial SNF assemblies that are not loaded into TAD canisters at utility sites can be handled and shipped to the repository in transportation casks certified by the NRC (Section 1.2.1). Criticality safety design for transportation casks containing commercial SNF is similar to the representative DPCs discussed in Section 1.14.2.3.1.2 and the TAD canisters discussed in Section 1.14.2.3.1.3. The results for analyses of TAD canisters and DPCs with close-fitting full-thickness reflectors in the safety analysis (BSC 2008b) are expected to be representative or bounding for transportation casks containing commercial SNF.

DOE SNF canisters will also be shipped to the repository in transportation casks certified by the NRC (Section 1.2.1). Criticality safety evaluations for specific DOE SNF transportation cask designs will demonstrate compliance with the preclosure criticality safety requirements in Section 1.14.2.1.

1.14.2.3.1.2 Dual-Purpose Canisters

The general characteristics of DPCs are described in Section 1.5.1.1.1.2.1.2. They are stainless steel cylinders of welded construction and are available in both vertical and horizontal configurations. Criticality control for commercial SNF in the DPCs relies primarily upon moderator control in dry areas of the surface facilities and upon the presence of soluble boron in the WHF pool. Criticality safety control features provided by the DPCs include moderator control (included in the containment provided by the shell) as well as geometry control and fixed neutron absorbers (provided by the basket). Analysis of operations involving DPCs is performed using representative DPC designs for pressurized water reactor (PWR) and boiling water reactor (BWR) SNF.

1.14.2.3.1.3 Transportation, Aging, and Disposal Canisters

The characteristics of the TAD canisters are described in Section 1.5.1.1.1.2.1.3. There are two configurations of TAD canisters, both of which fit inside the same TAD waste package configuration (Section 1.5.2.1.1). The 21-PWR TAD canister contains 21 PWR commercial SNF assemblies, and the 44-BWR TAD canister contains 44 BWR commercial SNF assemblies. Criticality control for commercial SNF in the TAD canisters relies primarily upon moderator control in dry areas of the surface facilities and upon the presence of soluble boron in the WHF pool. Criticality design control features provided by the TAD canister include moderator control (provided by the shell of a sealed canister), geometry controls, and fixed borated stainless steel neutron absorber (provided by the basket). Analysis of operations involving TAD canisters was performed using conceptual representations of the PWR and BWR TAD canisters compliant with the criticality performance requirements of the TAD canister performance specification (DOE 2008). Details of the TAD canister basket designs and dimensions of many internal components were assumed (BSC 2008b, Section 1.4.1) because they are not given in the performance specification. The assumed basket design is similar to existing transportation cask designs that have fuel compartments with absorber panels on four sides of the assembly and a gap (flux trap) between adjacent compartments. Nonetheless, the criticality safety results presented in Section 1.14.2.3.3 demonstrate that the internal structure of the TAD canister and the fixed neutron absorber are not required to maintain subcriticality based on moderator control for dry operations and soluble boron for operation in the WHF facility pool.

1.14.2.3.1.4 High-Level Radioactive Waste Canisters

HLW canisters are described in Section 1.5.1.2.1.2.1. Criticality safety control features are not necessary for HLW canisters because criticality is not possible due to the low concentration of fissile isotopes in an HLW canister (Section 1.14.2.3.2.4).

1.14.2.3.1.5 DOE Standardized Spent Nuclear Fuel Canisters

The DOE standardized SNF canister accommodates the representative fuel types for eight of the nine DOE fuel groups used for criticality analysis, as described in Section 1.14.2.3.2.3.1. The DOE multicanister overpack (MCO), described in Section 1.14.2.3.1.6, accommodates the ninth group.

The characteristics of the DOE standardized SNF canister shell are described in Section 1.5.1.3.1.2.1.1. The containment provided by the shell includes the criticality safety control feature of moderator control. The characteristics of the DOE standardized canister internals are described in Section 1.5.1.3.1.2.1.2. The criticality safety control features provided by the internals include geometry control, fissile material limits, and type and quantity of fixed neutron absorber.

1.14.2.3.1.6 DOE Multicanister Overpacks

The characteristics of the DOE MCO shell are described in Section 1.5.1.3.1.2.1.3. The shell provides the criticality safety control feature of moderator control. The characteristics of the DOE MCO canister internals are described in Section 1.5.1.3.1.2.1.4. The criticality safety control features provided by the internals include geometry control and fissile mass limits. The MCO is designed to hold fuel from the U metal fuel group, for which N Reactor fuel is the representative type, as described in Section 1.14.2.3.2.3.1. Prior to receipt and acceptance of MCOs, criticality safety analyses of MCOs containing U metal fuels will be performed to demonstrate compliance with the criticality safety requirements in Section 1.14.2.1.

1.14.2.3.1.7 Waste Package Configurations

There is a single waste package design with six configurations as described in Section 1.5.2.1.1:

- 21-PWR/44-BWR TAD waste package holding either a 21-PWR TAD canister or a 44-BWR TAD canister
- 5-DHLW/DOE Short Codisposal waste package nominally holding a single 18-in.-diameter short (10-ft) DOE standardized SNF canister surrounded by five 24-in.-diameter short HLW canisters
- 5-DHLW/DOE Long Codisposal waste package nominally holding a single 18-in.-diameter long (15-ft) DOE standardized SNF canister surrounded by five 24-in.-diameter long HLW canisters
- 2-MCO/2-DHLW waste package holding two 25.51-in.-diameter MCOs and two 24-in.-diameter long HLW canisters
- Naval Short waste package holding a single short naval SNF canister
- Naval Long waste package holding a single long naval SNF canister.

The criticality safety control feature provided by sealed waste packages is moderator control. The 5-DHLW/DOE Short Codisposal and 5-DHLW/DOE Long Codisposal waste packages also provide interaction control with a basket structure that maintains separation between canisters.

1.14.2.3.2 Criticality Control Parameters

As summarized in Section 1.14.2.2, in order to identify the parameters that are important to criticality control during the preclosure period, a series of k_{eff} calculations was performed for each specific waste form that covered the possible conditions to which the waste form may be exposed during handling operations in the surface and subsurface facilities. These calculations evaluated the impact on system reactivity of variations in each of the parameters that could be important to criticality during the preclosure period (i.e., they determined the sensitivity of k_{eff} to variations in any of these parameters as a function of the other relevant parameters). These sensitivity calculations demonstrated that each parameter:

- Does not need to be controlled because it is bounded (i.e., its analyzed value is greater than or equal to the design limit) or its effect on k_{eff} is bounded,
- Needs to be controlled if another parameter is not controlled (conditional control), or
- Needs to be controlled because it is the primary criticality control parameter.

Hazards identification and screening, as described in Section 1.6, followed by event sequence development and quantification, as described in Section 1.7, are performed only if a parameter must be controlled. Event sequences impacting criticality parameters that must be controlled are identified, developed, quantified, and categorized. These event sequences are referred to as event sequences important to criticality and are summarized in Section 1.7.5.

As presented in Section 1.14.2.3.4.1, the upper subcritical limit for all commercial SNF operations is 0.93, whereas the upper subcritical limit for all DOE SNF operations is 0.89.

1.14.2.3.2.1 Criticality Control Parameters for Commercial SNF Dry Operations

1.14.2.3.2.1.1 Waste Form Characteristics for Commercial SNF Dry Operations

For commercial SNF, the k_{eff} calculations and analysis considered the following bounding waste form parameters:

- 5 wt % enriched ^{235}U fresh fuel (i.e., maximum commercial SNF enrichment and no credit for burnup)
- UO_2 density of 10.751 g/cm^3 (i.e., 98% of full theoretical density)
- Use of full assembly length as active fuel length
- No burnable poison

- No credit for the presence of ^{234}U or ^{236}U absorbers
- Fuel pellet stack modeled as a simple cylinder with no density correction for dished ends
- Gap between fuel and clad filled with unborated water
- Simplified fuel assembly model neglecting spacer grids and end fittings.

These bounding parameters were used in two models of PWR commercial SNF assemblies: a simplified Westinghouse 17 × 17 optimized fuel assembly and a simplified Babcock & Wilcox 15 × 15 assembly; and in two models of BWR commercial SNF assemblies: a simplified General Electric 7 × 7 BWR assembly and a simplified Advanced Nuclear Fuel 9 × 9 BWR assembly. These assembly types were shown to be the most reactive assembly designs based on a survey of commercial SNF assemblies in various potential preclosure configurations (BSC 2008b, Section 2.3.1.1.1).

Because these combinations of assembly types and waste form parameters are considered bounding, waste form characteristics do not need to be controlled, and no misload of commercial SNF could lead to a criticality. Therefore, no initiating events in Section 1.6 or event sequences in Section 1.7 involving misloads of commercial SNF need to be identified.

1.14.2.3.2.1.2 Moderation for Commercial SNF Dry Operations

The analysis showed that moderation is the primary criticality control parameter during dry operations (BSC 2008b, Section 2.3.2.5). Because moderator control is required to maintain subcriticality, event sequences involving introduction of moderator into breached or open DPCs, TAD canisters, or transportation casks containing uncanistered commercial SNF must be identified (Section 1.6), and, if necessary, developed, quantified, and categorized (Section 1.7).

No Category 1 or Category 2 event sequences were identified that introduce moderator into breached or open DPCs, TAD canisters, or transportation casks containing uncanistered commercial SNF during dry operations in any surface or subsurface facility (Tables 1.7-9 to 1.7-18). For operations with a single canister of commercial SNF in the absence of moderation, subcriticality is maintained with a maximum k_{eff}, including calculational uncertainty, less than 0.60 (BSC 2008b, Section 2.3.2.1.2).

1.14.2.3.2.1.3 Fixed Neutron Absorber and Geometry for Commercial SNF Dry Operations

Geometry control is provided in transportation casks, TAD canisters, and DPCs by the structure of the cask or canister baskets and of the fuel assemblies themselves. Fixed neutron absorber is present in these containers in the form of plates. The k_{eff} sensitivity calculations examined the impact of fixed neutron absorber and geometry on system reactivity as a function of other relevant parameters to determine the minimum neutron absorber characteristics (e.g., loading) and geometry configurations necessary to maintain subcriticality and the extent to which neutron absorber and geometry need to be controlled. In those calculations, aspects of the geometry such as pin pitch and flux trap gap size were varied between their nominal and most reactive or physically limiting state, and the quantity of fixed neutron absorber was varied between its nominal value and complete omission. In accordance with NUREG-1567 (NRC 2000, Section 8.4.1.1) only 75% credit was conservatively taken for the nominal quantity of fixed neutron absorber present.

For dry operations, the analysis showed that the need for fixed neutron absorber and geometry control depends upon the control of moderation (BSC 2008b, Section 2.3.2.1.3). Because no Category 1 or Category 2 event sequences were identified that introduce moderator into any breached or open commercial SNF containers (Tables 1.7-9 through 1.7-18), no initiating events in Section 1.6 or event sequences in Section 1.7 impacting fixed neutron absorber effectiveness and geometry need to be identified.

1.14.2.3.2.1.4 Interaction for Commercial SNF Dry Operations

The models for TAD canisters and DPCs include infinite hexagonal planar arrays of close-fitting units that effectively bound the interaction between these units. In the absence of moderation, subcriticality is maintained for interaction of containers of commercial SNF during surface operations with a maximum k_{eff}, including calculational uncertainty, less than 0.60. For interaction of containers of commercial SNF with a single container of DOE SNF during surface operations, subcriticality is maintained with a maximum k_{eff}, including calculational uncertainty, less than 0.84 (BSC 2008b, Section 2.3.2.1.4).

To determine the effect of moderator presence between sealed commercial SNF canisters, TAD canister planar arrays were modeled with varying amounts of interstitial water. The safety analysis showed that adding interstitial moderator decreases reactivity (BSC 2008b, Section 2.3.2.1.4).

Interaction between waste packages containing the same waste form in an emplacement drift is bounded by the use of mirror boundary conditions applied to the axial ends of close-fitting, radially reflected waste packages. There is no statistically significant difference between the k_{eff} for a single waste package and the k_{eff} for a waste package with mirror boundary conditions at the axial ends of the waste package. Thus, a waste package containing commercial SNF is always effectively infinite in length, so that its interaction with other waste packages (of commercial or DOE SNF) in an emplacement configuration is bounded. In the absence of moderation, subcriticality is maintained for interaction of waste packages containing commercial SNF in an emplacement configuration with a maximum k_{eff}, including calculational uncertainty, less than 0.50 (BSC 2008b, Section 2.3.2.1.4).

Therefore, interaction is considered bounded during dry operations with commercial SNF and no initiating events in Section 1.6 or event sequences in Section 1.7 involving interaction need to be identified.

1.14.2.3.2.1.5 Reflection for Commercial SNF Dry Operations

The reflectors considered in the analysis of transportation casks, DPCs, TAD canisters, and waste packages (Sections 1.14.2.3.2.1.2 through 1.14.2.3.2.1.4) include materials that could be present during dry operations (i.e., stainless steel, concrete, lead, uranium, water, Alloy 22 (UNS N06022), HLW glass, titanium, and tuff). Reflector materials were modeled as close-fitting reflectors that are effectively full-thickness; that is, the thickness is greater than or equal to any dimension that may be encountered during dry operations or the thickness is sufficient to be considered infinite. Commercial SNF canisters were modeled both as fully reflected single canisters and as axially reflected infinite hexagonal planar arrays (BSC 2008b, Section 2.3.2.1.5).

Therefore, reflection is considered bounded for commercial SNF during dry operations and no initiating events in Section 1.6 or event sequences in Section 1.7 involving reflection need to be identified.

1.14.2.3.2.2 Criticality Control Parameters for Commercial SNF Wet Operations

1.14.2.3.2.2.1 Waste Form Characteristics for Commercial SNF Wet Operations

For wet operation with commercial SNF, the k_{eff} calculations considered the same bounding waste form parameters and the same assembly types discussed in Section 1.14.2.3.2.1.1 for dry operations. Because these combinations of assembly types and waste form parameters are considered bounding, no initiating events in Section 1.6 or event sequences in Section 1.7 involving misloads of commercial SNF need to be identified.

1.14.2.3.2.2.2 Moderation for Commercial SNF Wet Operations

Control of moderation is not applicable to wet operations with commercial SNF.

1.14.2.3.2.2.3 Soluble Neutron Absorber for Commercial SNF Wet Operations

The k_{eff} sensitivity calculations examined the impact of soluble neutron absorber on system reactivity as a function of other relevant parameters (e.g., geometry and interaction) to determine the minimum soluble neutron absorber characteristics (i.e., concentration and ^{10}B enrichment) necessary to maintain subcriticality and the extent to which soluble neutron absorber must be controlled.

Soluble neutron absorber in the form of orthoboric acid, H_3BO_3, is present in the WHF pool and transportation cask/DPC fill water. The soluble boron concentration required to maintain subcriticality is dependent upon the limiting conditions and/or control of other parameters important to criticality safety. Therefore, the soluble boron sensitivity calculations are discussed in conjunction with fixed neutron absorber, geometry, and interaction in Section 1.14.2.3.2.2.4, which indicate that a soluble boron concentration of 2,500 mg/L of boron enriched to 90 atom % ^{10}B is sufficient to maintain subcriticality with bounding geometry, neutron absorber, and interaction conditions for normal operations and potential event sequences.

Because soluble neutron absorber is required to maintain subcriticality, event sequences that dilute the concentration of soluble boron in the WHF pool and transportation cask/DPC fill water need to be identified (Section 1.6), and, if necessary, developed, quantified, and categorized (Section 1.7). No Category 1 or Category 2 event sequences were identified that result in boron dilution to a concentration insufficient to maintain subcriticality (Tables 1.7-13 and 1.7-14) because boron dilution was screened out as an initiating event (Table 1.7-1). A procedural safety control is relied upon to ensure that a minimum concentration of 2,500 mg/L of soluble boron (enriched to

90 atom % ^{10}B) is maintained in the WHF pool and the transportation cask/DPC fill water (Table 1.9-10).

The safety analysis showed that increasing the void fraction in the borated water at various levels of soluble boron concentration results in decreased reactivity (BSC 2008b, Section 2.3.2.2.2). Therefore, no initiating events in Section 1.6 or event sequences in Section 1.7 that result in increased void fraction (e.g., boiling) in the WHF pool or transportation cask/DPC fill water need to be identified.

1.14.2.3.2.2.4 Fixed Neutron Absorber, Geometry, and Interaction for Commercial SNF Wet Operations

Because event sequences in the WHF pool will likely impact fixed neutron absorber, geometry, and interaction at the same time, the k_{eff} sensitivity calculations considered these three criticality control parameters collectively.

Fixed neutron absorber is present in transportation casks, TAD canisters, DPCs, and in the storage racks in the WHF pool in the form of plates. In the k_{eff} sensitivity calculations, the quantity of fixed neutron absorber was varied between its nominal value and complete omission. In accordance with NUREG-1567 (NRC 2000, Section 8.4.1.1) only 75% credit was conservatively taken for the nominal quantity of fixed neutron absorber present.

Geometry control is provided by the structure of the canister baskets and the pool staging racks, as well as by the structure of the assemblies themselves. In the k_{eff} sensitivity calculations, aspects of the geometry such as pin pitch and flux trap gap size were varied between their nominal and most reactive or physically limiting state.

The models for TAD canisters, DPCs, and WHF pool staging racks included infinite planar arrays of close-fitting units that effectively bound the interaction of these units with similar units.

In order to bound geometry for normal operations and potential event sequences associated with transferring single assemblies from DPCs or transportation casks to the staging racks or into a TAD canister or from the staging racks to a TAD canister, criticality calculations were performed with close-fitting full-thickness reflection with borated water, unborated water, concrete, stainless steel, lead, and natural uranium modeled on all six sides of an assembly with optimized pin pitch. For the limiting PWR assembly in this configuration (i.e., Babcock & Wilcox 15 × 15) and the most limiting reflection conditions, no more than 15% of the minimum required soluble boron concentration is necessary to maintain subcriticality (BSC 2008b, Section 2.3.2.2.4).

For the same configuration, with the most limiting reflection conditions, BWR assemblies remain subcritical without any credit for soluble boron, with a maximum k_{eff}, including calculational uncertainty, less than 0.80 (BSC 2008b, Section 2.3.2.2.4).

End states of potential event sequences resulting in interaction of single assemblies with the staging racks or shielded transfer casks containing TAD canisters or DPCs remain subcritical, crediting no more than 15% of the minimum required soluble boron concentration (BSC 2008b, Section 2.3.2.2.4).

Potential event sequences including drops or seismic events during operations with shielded transfer casks containing either TAD canisters or DPCs associated with their transfer in and out of the pool could impact the geometry inside these canisters as well as the effectiveness of the fixed neutron absorber. Subcriticality is maintained for these operations crediting no more than 10% of the minimum required soluble boron with the geometry control and the fixed neutron absorber provided by the canister baskets. In order to bound geometry, fixed neutron absorber, and interaction effects inside these canisters, the canister baskets and the fixed neutron absorber are omitted. This configuration is neutronically similar to a hypothetical conservative representation in which the entire contents of the shielded transfer cask are optimally rearranged outside of the confinement of the shielded transfer cask on the bottom of the pool reflected by stainless steel or concrete and borated water. With the optimally spaced maximum number of PWR and BWR fuel pins in a DPC, which has a larger capacity than a TAD canister, subcriticality is maintained with a soluble boron (enriched to 90 atom % ^{10}B) concentration of 2,500 mg/L for the most limiting fuel type, pin design, and reflection condition (BSC 2008b, Section 2.3.2.2.4).

Event sequences that result in damage to the staging racks are conservatively represented with optimally spaced fuel pins within the fuel compartments, fixed neutron absorber omission, and complete flux trap gap collapse, for which subcriticality is maintained crediting no more than 30% of the minimum required soluble boron concentration (BSC 2008b, Section 2.3.2.2.4).

Therefore, control of geometry, fixed neutron absorber, and interaction in the WHF pool is not required for operations with single assemblies and shielded transfer casks containing either TAD canisters or DPCs unless Category 1 or Category 2 event sequences are identified that result in boron dilution that will lower the concentration to a value that is insufficient to maintain subcriticality. In addition, because the staging racks are designed to maintain confinement of the fuel assemblies within the fuel compartments for all Category 1 and Category 2 event sequences (Table 1.9-4), a concentration of 2,500 mg/L of soluble boron enriched to 90 atom % ^{10}B is also sufficient to maintain subcriticality without any additional control of geometry, interaction or fixed neutron absorber (BSC 2008b, Section 2.3.2.2.4).

No Category 1 or Category 2 event sequences were identified that result in boron dilution that will lower the boron concentration to a value insufficient to maintain subcriticality (Tables 1.7-13 and 1.7-14) because boron dilution was screened out as an initiating event (Table 1.7-1).

1.14.2.3.2.2.5 Reflection for Commercial SNF Wet Operations

The reflectors considered in the analysis of transportation casks, DPCs, commercial SNF assemblies, and TAD canisters (Sections 1.14.2.3.2.2.2 through 1.14.2.3.2.2.4) include materials that could be present during wet operations (i.e., borated and unborated water, stainless steel, concrete, lead, and uranium). Reflector materials were modeled as close-fitting reflectors that are effectively full-thickness, that is, the thickness is greater than or equal to any dimension that may be encountered during wet operations or the thickness is sufficient to be considered infinite. In the analysis, TAD canisters and DPCs were modeled as axially reflected infinite hexagonal planar arrays with no interstitial material, thus bounding any possible radial reflection (BSC 2008b, Section 2.3.2.2.5).

Therefore, reflection is considered bounded for commercial SNF and no initiating events in Section 1.6 or event sequences in Section 1.7 involving reflection need to be identified.

1.14.2.3.2.3 Criticality Control Parameters for DOE SNF

1.14.2.3.2.3.1 Waste Form Characteristics for DOE SNF

Due to the variety of DOE-owned SNF, the National Spent Nuclear Fuel Program has designated nine representative fuel groups for criticality analyses (Section 1.5.1.3.1.1.3). For each fuel group, a fuel type that represents the characteristics of the fuels in that group has been selected for detailed analysis. The nine fuel groups, the representative fuel types for criticality analysis, and the associated basket designs are described below (BSC 2008b, Section 2.3.1.1.2):

- **Criticality Group 1: U Metal**—N Reactor fuel is the representative fuel type for the U metal fuel group. There are two design variants of fuel elements (Mark IV and Mark IA), each of which is made of two concentric tubes of uranium metal co-extruded with Zircaloy 2 cladding. Initial enrichments range from 0.947 to 1.25 wt % ^{235}U. An MCO holds five stainless steel baskets. There are two basket designs, one for intact and one for scrap fuel elements.

- **Criticality Group 2: Mixed Oxide Fuels**—Fast Flux Test Facility fuel is the representative type for the mixed oxide fuel group. There are four basic types of Fast Flux Test Facility fuel pins and one experimental variant. Each pin contains mixed oxide fuel ($UO_{1.96}$ and $PuO_{1.96}$) surrounded by stainless steel clad. The Fast Flux Test Facility standard driver fuel assembly contains 217 Type 4.1 fuel pins (which have the highest fissile content) within a Stainless Steel Type 316 hexagonal duct. Some assemblies have been disassembled, and up to 217 fuel pins have been placed in a 5-in. stainless steel pipe known as an Ident-69 container. A long DOE standardized SNF canister contains a spoked-wheel basket constructed of nickel-gadolinium alloy holding five driver fuel assemblies surrounding a single Ident-69 container. As noted in Section 1.5.1.3.1.2.1.2, only five of the six basket compartments will be used for any fully-loaded canister. The space not occupied by the fuel assemblies, the Ident-69 container, and the basket are filled with aluminum shot containing $GdPO_4$ that is used as a neutron absorber.

- **Criticality Group 3: U-Mo/U-Zr Alloy Fuels**—Enrico Fermi fast reactor fuel is the representative type for the U-Mo and U-Zr alloy fuel group. The Enrico Fermi fuel pin matrix is made of uranium-molybdenum alloy (approximately 10 wt % molybdenum alloyed with uranium of 25.69 wt % ^{235}U enrichment). The fuel is metallurgically bonded to a zirconium tube that serves as cladding, resulting in no gap between cladding and fuel. Zirconium end pieces are fitted to the fuel rods, and 140 fuel rods plus four stainless steel connecting rods are installed in each fuel assembly. A short DOE standardized SNF canister contains two basket assemblies, with each basket assembly consisting of 12 nickel-gadolinium alloy tubes. Each tube contains an –01 aluminum canister, which itself contains an –04 aluminum canister holding 140 loose fuel pins from a single assembly. The space between the tubes is filled with $Fe/GdPO_4$ shot that is used as a neutron absorber.

- **Criticality Group 4: Highly Enriched Uranium Oxide Fuels**—Shippingport PWR fuel is the representative fuel type for the highly enriched uranium oxide fuel group. The Core 2 Seed 2 fuel cluster is used as the representative cluster because it has a higher ^{235}U loading per cluster than other types of fuel clusters. It is composed of four fuel subclusters arranged in a square array with spacing between them that accommodated a cruciform-shaped control rod during operation. Each subcluster contains 19 fuel and two neutron absorber plates. A fuel plate is formed by sandwiching UO_2–ZrO_2–CaO alloy wafers between two Zircaloy-4 cover plates and four side strips. The initial ^{235}U enrichment is 93.2 wt %. A long DOE standardized SNF canister contains a single square basket of stainless steel guide plates holding a single Shippingport PWR fuel cluster.

- **Criticality Group 5: ^{233}U/Th Oxide Fuels**—Shippingport Light Water Breeder Reactor fuel is the representative fuel type for the ^{233}U/Th oxide fuel group. The seed fuel is used as the representative assembly. It contains eight types of seed rods in four fuel regions, with a total of 619 cylindrical fuel rods in a triangular pitch array, supported by a hexagonal Zircaloy-4 outer shell. The fuel rods contain either thoria (ThO_2) or a mixture of thoria and UO_2. The uranium is 98.23 wt % ^{233}U. Two different enrichments (ratio of the mass of fissile isotopes to the total heavy metal mass) of the UO_2-ThO_2 matrix were used, a lower enrichment of 4.337 wt % and a higher enrichment of 5.202 wt %. A long DOE standardized SNF canister contains a rectangular stainless steel basket holding a single Light Water Breeder Reactor seed assembly. The space not occupied by the fuel assembly and basket is filled with aluminum shot containing $GdPO_4$ that is used as a neutron absorber.

- **Criticality Group 6: U/Th Carbide Fuels**—Fort St. Vrain fuel is the representative fuel type for the U/Th carbide fuel group. It consists of a mixture of small spheres on the order of 0.0450 to 0.0750 cm in diameter of uranium (enriched to 93.5 wt % ^{235}U) and thorium carbide. The individual spheres are coated with multiple thin layers of pyrolytic carbon and silicon carbide, which serve as tiny pressure vessels to contain fission products and the U/Th carbide matrix. In the fuel elements, the coated spheres are bound in a carbonized matrix to form fuel compacts that are loaded into drilled holes in a large hexagonal graphite prism that comprises one fuel element. Fuel holes containing the fuel compacts and coolant channels are distributed in a triangular array within the fuel element. A long DOE standardized SNF canister contains five hexagonal graphite fuel elements with no separate basket.

- **Criticality Group 7: UZrH Fuels**—TRIGA fuel is the representative fuel type for the UZrH fuel group. The highly enriched uranium fuel life improvement program variant is used in the analysis. A stainless steel fuel element contains 70 wt % enriched ^{235}U in a self-moderating zirconium-hydride matrix ($UZrH_{1.6}$). A short DOE standardized SNF canister contains three baskets, each holding 31 fuel elements, with basket tubes made of nickel-gadolinium alloy.

- **Criticality Group 8: Al-Based Fuels**—Advanced Test Reactor fuel is the representative fuel type for the Al-based fuel group. The Advanced Test Reactor fuel element consists of 19 curved aluminum clad uranium aluminide (UAl_x) plates containing highly enriched (93±1 wt % ^{235}U) uranium. The highest nominal fuel loading for a fresh fuel element is

1075 g of ^{235}U. A long DOE standardized SNF canister contains three baskets, each holding 10 fuel elements, with basket plates made of nickel-gadolinium alloy.

- **Criticality Group 9: Low Enriched Uranium Oxide Fuels**—Three Mile Island Unit 2 debris is the representative fuel type for the low enriched uranium oxide fuel group. The representative debris has the characteristics of a Babcock & Wilcox 15 × 15 fuel assembly with maximum enrichment of 2.96 wt % ^{235}U and a maximum beginning-of-life ^{235}U content of 13.72 kg. A long DOE standardized SNF canister contains one of three Three Mile Island Unit 2 canister types: a defueling canister holding debris large enough to grapple, a knockout canister holding vacuumed debris, or a filter canister holding debris caught in filters. The uranium loading for a single Three Mile Island Unit 2 canister ranges from zero to 441.9 ± 99.9 kg.

Consideration of waste form misload is not appropriate because the criticality safety analysis was performed only for representative DOE SNF fuel types and loading procedures for DOE standardized SNF canisters have not been established yet.

1.14.2.3.2.3.2 Moderation for DOE SNF

For DOE SNF, the analysis showed that moderation is a primary criticality control parameter (BSC 2008b, Section 2.3.2.5). Because moderator control is required to maintain subcriticality, event sequences related to introduction of moderator into breached DOE SNF canisters need to be identified (Section 1.6), and, if necessary, developed, quantified, and categorized (Section 1.7).

No Category 1 or Category 2 event sequences were identified that introduce moderator into breached DOE standardized SNF canisters during operations in the surface or subsurface facilities (Tables 1.7-11, 1.7-12, and 1.7-15 through 1.7-18). For operations with a single canister of the most reactive representative DOE SNF type, in the absence of moderation, subcriticality is maintained with a maximum k_{eff}, including calculational uncertainty, less than 0.75 (BSC 2008b, Section 2.3.2.3.2).

1.14.2.3.2.3.3 Fixed Neutron Absorber and Geometry for DOE SNF

The k_{eff} sensitivity calculations for DOE SNF examined the impact of fixed neutron absorber and geometry on system reactivity to determine the minimum neutron absorber characteristics (e.g., loading) and limiting geometry necessary to maintain subcriticality and the extent to which neutron absorber and geometry must be controlled. Fixed neutron absorber is present in DOE standardized SNF canisters containing Advanced Test Reactor, Shippingport Light Water Breeder Reactor, Fast Flux Test Facility, Enrico Fermi, and TRIGA SNF (BSC 2008b, Sections 2.3.2.3.3.1, 2.3.2.3.3.2, 2.3.2.3.3.5, 2.3.2.3.3.6, and 2.3.2.3.3.7, respectively). In the sensitivity calculations, these fixed neutron absorbers were modeled as being either present, or as partially or totally absent. In accordance with NUREG-1567 (NRC 2000, Section 8.4.1.1), only 75% credit was conservatively taken for the nominal quantity of fixed neutron absorber present.

In the sensitivity calculations, aspects of the geometry such as pin pitch, plate spacing, and fuel damage were varied between their nominal and most reactive or physically limiting states as idealized representations of potential off-normal conditions. Because geometry control inside several DOE standardized SNF canisters is provided by a gadolinium-bearing low-carbon high-nickel alloy, which also serves as the fixed neutron absorber, the k_{eff} sensitivity calculations examined the two parameters collectively.

The analysis showed that the need for fixed neutron absorber and geometry control depends upon the control of moderation. Because no Category 1 or Category 2 event sequences were identified that introduce moderator into any breached DOE standardized SNF canisters (Tables 1.7-11, 1.7-12, and 1.7-15 through 1.7-18), no initiating events in Section 1.6 or event sequences in Section 1.7 impacting fixed neutron absorber effectiveness or geometry need to be identified. In the absence of moderation, subcriticality is maintained even with nonphysical conservative geometrical rearrangement (BSC 2008b, Sections 2.3.2.3.3.1 through 2.3.2.3.3.7).

1.14.2.3.2.3.4 Interaction for DOE SNF

The k_{eff} sensitivity calculations bounded the effects of interaction of containers of DOE SNF with containers of the same or other waste forms on system reactivity and determined the extent to which interaction must be controlled.

The analysis showed that control of interaction is not necessary for DOE standardized SNF canisters while they are located in staging racks. Under the most reactive reflection conditions, subcriticality is maintained for DOE standardized SNF canisters in the staging racks with a surface-to-surface separation of canisters no less than 30 cm (BSC 2008b, Section 2.3.2.3.4).

Interaction is bounded for DOE SNF canisters containing Advanced Test Reactor, Shippingport Light Water Breeder Reactor, Shippingport PWR, and TRIGA SNF based on infinite hexagonal planar arrays of close-fitting canisters. The interstitial space between the canisters was analyzed as being filled with variable density water. The analysis demonstrated that the presence of moderator external to, and between, the DOE standardized SNF canisters results in a decrease in the system reactivity. Under the most reactive reflection conditions, subcriticality is maintained for those fuels with a maximum k_{eff}, including calculational uncertainty, less than 0.86 (BSC 2008b, Section 2.3.2.3.4).

However, the analysis indicated that to maintain subcriticality, interaction must be controlled for Fast Flux Test Facility, Enrico Fermi, and Fort St. Vrain DOE SNF canisters. For the most reactive DOE standardized SNF canister containing Fast Flux Test Facility SNF, no more than four canisters can be placed outside of the staging racks or codisposal waste packages. Because the fuel contained within a DOE standardized SNF canister is not obvious from visual inspection of a sealed canister, event sequences that result in placing more than four DOE standardized SNF canisters containing any DOE SNF type outside of their designated staging racks or a codisposal waste package must be identified (Section 1.6), and, if necessary, developed, quantified, and categorized (Section 1.7). No Category 1 or Category 2 event sequences were identified that result in placing more than four DOE standardized SNF canisters outside of their designated staging racks or codisposal waste packages (Tables 1.7-11 and 1.7-12) because the initiating event has been screened out (Table 1.7-1).

Interaction control is not necessary for codisposal waste packages containing any type of DOE standardized SNF canister. Under the most reactive reflection conditions, subcriticality is maintained for codisposal waste packages with a maximum k_{eff}, including calculational uncertainty, less than 0.65 for a normally loaded waste package and 0.80 for a misloaded waste package containing six DOE SNF canisters (BSC 2008b, Section 2.3.2.3.4).

Interaction between waste packages containing the same waste form in an emplacement drift is bounded by the use of mirror boundary conditions applied to the axial ends of close-fitting, radially reflected waste packages. There is no statistically significant difference between the k_{eff} for a single waste package and the k_{eff} for a waste package with mirror boundary conditions at the axial ends of the waste package. Thus, a waste package containing DOE SNF is always effectively infinite in length, so that its interaction with other waste packages (of commercial or DOE SNF) in an emplacement configuration is always bounded. In the absence of moderation, subcriticality is maintained for interaction of waste packages containing DOE SNF in an emplacement configuration with a maximum k_{eff}, including calculational uncertainty, less than 0.65 (BSC 2008b, Section 2.3.2.3.4).

1.14.2.3.2.3.5 Reflection for DOE SNF

The reflectors considered in the analysis of DOE SNF canisters and codisposal waste packages (Sections 1.14.2.3.2.3.2 through 1.14.2.3.2.3.4) include materials that could be present during surface and subsurface operations (i.e., stainless steel, concrete, lead, uranium, water, Alloy 22, HLW glass, titanium, and tuff). Reflector materials were modeled as close-fitting reflectors that are effectively full-thickness, that is, the thickness is greater than or equal to any dimension that may be encountered during surface and subsurface operations or the thickness is sufficient to be considered infinite (BSC 2008b, Section 2.3.2.3.5).

Therefore, reflection is considered bounded for DOE SNF and no initiating events in Section 1.6 or event sequences in Section 1.7 involving reflection need to be identified.

1.14.2.3.2.4 Criticality Control Parameters for HLW

The only criticality control parameter important for HLW is fissile isotope concentration (a waste form characteristic). The estimated quantities of fissile isotopes in HLW canisters are shown in Table 1.14-1, as well as the total fissile isotope concentration. The minimum subcritical limit for fissile solute (aqueous solution of fissile isotopes) from Table 1 of ANSI/ANS-8.1-1998, is 7.3 g/L for $^{239}Pu(NO_3)_4$. Because concentration limits for aqueous solutions are lower than those for other physical/chemical forms, the fact that the concentrations of fissile isotopes in Table 1.14-1 are approximately one order of magnitude less than 7.3 g/L demonstrates that HLW glass has a significant margin of subcriticality. The limits in Table 1 of ANSI/ANS-8.1-1998 assume a uniform homogeneous mixture. Because the glass canisters are poured as a melt, they are relatively homogeneous. Even if the glass is not completely homogeneous, the significant margin of subcriticality will compensate for any inhomogeneities in the glass. Therefore, individual HLW canisters and codisposal waste packages containing only HLW canisters are safely subcritical. No further analysis is required to demonstrate the subcriticality of individual HLW glass canisters and codisposal waste packages containing only HLW glass canisters (BSC 2008b, Section 2.3.1.1.3).

1.14.2.3.2.5 Summary of Criticality Control Parameters

A summary of the criticality control parameters is shown in Table 1.14-2. For dry operations with canistered commercial SNF, moderation is the primary criticality control parameter and must be controlled. Fixed neutron absorber and geometry need to be controlled only if moderator is present inside commercial SNF canisters. For wet operations with commercial SNF, soluble neutron absorber must be controlled. Fixed neutron absorber, geometry, and interaction need to be controlled only if there is a boron dilution event during wet operations that will lower the soluble boron concentration to a value that is insufficient to maintain subcriticality. For operations with DOE SNF, moderation and interaction between canisters are the primary criticality control parameters and they must be controlled. Fixed neutron absorber and geometry need to be controlled only if moderator is present inside DOE SNF canisters. There are no required criticality control parameters for HLW.

1.14.2.3.3 Facility Criticality Safety Evaluations

1.14.2.3.3.1 Initial Handling Facility Evaluation

The IHF provides the SSCs to handle a portion of the DOE-managed waste stream. The waste stream for the IHF is limited to naval SNF canisters and HLW canisters. Canisters received in transportation casks in the IHF are transferred directly into waste packages, which are welded closed and carried out of the IHF by the transport and emplacement vehicle for emplacement in the repository. The IHF is designed to receive naval SNF canisters by rail only, while HLW may arrive by rail or truck. Details of the operations performed in the IHF are described in Section 1.2.3.1.1.

There can be no criticality during normal operations with HLW because the concentration of fissile isotopes in an HLW canister is too low for criticality to be possible (Section 1.14.2.3.2.4). Therefore, no initiating events in Section 1.6 or event sequences important to criticality in Section 1.7 need to be identified for operations with HLW in the IHF.

The nuclear safety design bases for ITS features derived from the PCSA described in Sections 1.6 and 1.7 are presented in Section 1.9. There are no SSCs in the IHF that are classified as ITS for prevention of criticality in HLW (Table 1.9-2).

1.14.2.3.3.2 Receipt Facility Evaluation

The RF provides the SSCs that support receipt of transportation casks and canisters containing commercial SNF. It receives rail-based transportation casks containing TAD canisters and DPCs. TAD canisters and vertical DPCs are placed in aging overpacks and transferred to the Aging Facility or to a CRCF (TAD canisters) or the WHF (DPCs). Horizontal DPCs inside a transportation cask are placed on a transfer trailer and transferred to the Aging Facility for placement in a horizontal aging module. Details of the operations performed in the RF are described in Section 1.2.6.1.1.

Moderation is the primary criticality control parameter during normal operations in the RF (Table 1.14-2). Control of moderation is sufficient to maintain subcriticality during those operations with a maximum k_{eff}, including calculational uncertainty, less than 0.60 (BSC 2008b, Section 2.3.4.2). No Category 1 or Category 2 event sequences were identified that will introduce moderator into breached DPCs or TAD canisters during operations in the RF (Tables 1.7-9 and 1.7-10).

The nuclear safety design bases for ITS features derived from the PCSA described in Sections 1.6 and 1.7 are presented in Section 1.9. The SSCs in the RF that are classified as ITS for prevention of criticality (e.g., containment, which includes moderator control) are identified in Table 1.9-5.

1.14.2.3.3.3 Canister Receipt and Closure Facility Evaluation

A CRCF provides the SSCs that support receipt of transportation casks and canisters, transfer, and packaging of commercial SNF, DOE SNF, and HLW canisters. Naval SNF is handled only in the IHF. A CRCF receives truck- and rail-based transportation casks containing canistered waste forms. It also receives aging overpacks containing TAD canisters from the WHF and the Aging Facility. Canisters are placed in an aging overpack and transferred to the Aging Facility or sealed in a waste package and transferred to the subsurface facility for final emplacement. In addition, a CRCF provides separate staging racks for DOE SNF canisters and TAD canisters, which are described in Section 1.2.4.2.2.1.3. The racks provide a capacity to hold 10 HLW and DOE SNF canisters and two TAD canisters. Details of the operations performed in a CRCF are described in Section 1.2.4.1.1.

Moderation is the primary criticality control parameter during normal operations with commercial SNF in a CRCF (Table 1.14-2). Control of moderation is sufficient to maintain subcriticality during those operations with a maximum k_{eff}, including calculational uncertainty, less than 0.60 (BSC 2008b, Section 2.3.5.2). No Category 1 or Category 2 event sequences were identified that will introduce moderator into breached DPCs or TAD canisters during operations in a CRCF (Tables 1.7-11 and 1.7-12).

Moderation and interaction are the primary criticality control parameters during normal operations with DOE SNF in a CRCF (Table 1.14-2). Control of moderation and interaction during normal operations with DOE standardized SNF canisters is sufficient to maintain subcriticality during those operations with a maximum k_{eff}, including calculational uncertainty, less than 0.75 (BSC 2008b, Section 2.3.5.2). No Category 1 or Category 2 event sequences were identified that will introduce moderator into breached DOE standardized SNF canisters during operations in a CRCF (Tables 1.7-11 and 1.7-12).

As discussed in Section 1.14.2.3.2.3.4, for the most reactive DOE standardized SNF canister, no more than four canisters can be placed outside of the staging racks or codisposal waste packages. No Category 1 or Category 2 event sequences were identified that will result in placing more than four DOE standardized SNF canisters outside their designated staging racks or a codisposal waste package (Tables 1.7-11 and 1.7-12) because the initiating event has been screened out (Table 1.7-1).

There can be no criticality during operations with HLW because the concentration of fissile isotopes in an HLW canister is too low for criticality to be possible (Section 1.14.2.3.2.4).

The nuclear safety design bases for ITS features derived from the PCSA described in Sections 1.6 and 1.7 are presented in Section 1.9. The SSCs in a CRCF that are classified as ITS for prevention of criticality (e.g., moderator control; containment, which includes moderator control; or interaction control, which is implemented by maintaining separation of DOE SNF canisters) are identified in Table 1.9-3.

1.14.2.3.3.4 Wet Handling Facility Evaluation

The WHF provides the SSCs that support receipt of transportation casks and DPCs, transfer, and canisterization of commercial SNF. It receives truck- and rail-based transportation casks containing uncanistered fuel assemblies, and rail-based transportation casks containing DPCs. The WHF also receives aging overpacks containing vertical DPCs from the RF and Aging Facility, as well as shielded transfer casks containing horizontal DPCs from the Aging Facility. The commercial SNF in the transportation casks and DPCs is repackaged into TAD canisters, and the sealed TAD canisters are transported to either the Aging Facility or the CRCFs. In addition, the WHF provides staging racks in the pool for PWR and BWR fuel assemblies, which are described in Section 1.2.5.2.2.1.3. They provide fixed neutron absorber and geometry control (a stainless steel support structure). Details of the operations performed in the WHF are described in Section 1.2.5.1.1.

Moderation is the primary criticality control parameter during normal dry operations (outside the pool) (Table 1.14-2). Control of moderation is sufficient to maintain subcriticality during those operations with a maximum k_{eff}, including calculational uncertainty, less than 0.60 (BSC 2008b, Section 2.3.6.2). No Category 1 or Category 2 event sequences were identified that will introduce unborated moderator into breached or open transportation casks, DPCs, or TAD canisters during dry operations in the WHF (Tables 1.7-13 and 1.7-14).

The primary criticality control parameter for wet operations is soluble neutron absorber in the form of orthoboric acid, H_3BO_3, present in the WHF pool and in transportation cask/DPC fill water with a minimum required concentration of 2,500 mg/L of soluble boron enriched to 90 atom % ^{10}B (BSC 2008b, Section 2.3.6.3) (Table 1.14-2).

For normal operations involving the transfer of individual SNF assemblies from DPCs or transportation casks to the staging racks or into a TAD canister or from the staging racks to a TAD canister, subcriticality is maintained, crediting no more than 15% of the minimum required soluble boron concentration (BSC 2008b, Section 2.3.6.3).

For normal operations involving the transfer of shielded transfer casks containing either TAD canisters or DPCs into and out of the pool, subcriticality is maintained, crediting no more than 10% of the minimum required soluble boron concentration (BSC 2008b, Section 2.3.6.3).

For normal operations involving the staging racks, subcriticality is maintained, crediting no more than 15% of the minimum required soluble boron concentration, with no credit for the fixed neutron absorber (BSC 2008b, Section 2.3.6.3).

No Category 1 or Category 2 event sequences were identified that result in boron dilution that will lower the boron concentration to a value that is insufficient to maintain subcriticality (Tables 1.7-13 and 1.7-14) because boron dilution was screened out as an initiating event (Table 1.7-1). A procedural safety control is relied upon to ensure that a concentration of 2,500 mg/L of soluble boron (enriched to 90 atom % ^{10}B) is maintained in the WHF pool and the transportation cask/DPC fill water (Table 1.9-10). On this basis, subcriticality is maintained for the following types of event sequences:

- **Event Sequences That Impact Individual SNF Assemblies**—Subcriticality is maintained for the limiting PWR assembly with optimized pin pitch and the most limiting reflection conditions, crediting no more than 15% of the minimum required soluble boron concentration. No soluble neutron absorber is needed to maintain subcriticality for single BWR assemblies in the same configuration (BSC 2008b, Section 2.3.6.4.2).

- **Event Sequences That Impact Shielded Transfer Casks**—Subcriticality is maintained for any credible level of damage to dropped shielded transfer casks in any orientation containing either TAD canisters or DPCs including the bounding configurations in which the entire contents of the shielded transfer cask are rearranged outside of the confinement of the shielded transfer cask on the bottom of the pool, crediting 2,500 mg/L of soluble boron enriched to 90 atom % ^{10}B (BSC 2008b, Section 2.3.6.4.2).

- **Event Sequences That Impact the Staging Racks**—Subcriticality is maintained for any credible level of damage to the staging racks, crediting no more than 30% of the minimum required soluble boron concentration (BSC 2008b, Section 2.3.6.4.2). Note that the credible level of damage is limited by the requirement that the staging racks maintain confinement of the fuel assemblies within the staging rack fuel compartments, implementing the safety function "Protect against tipover of SNF" (Table 1.9-4).

The nuclear safety design bases for ITS features derived from the PCSA described in Sections 1.6 and 1.7 are presented in Section 1.9. The SSCs in the WHF that are classified as ITS for prevention of criticality (e.g., moderator control or containment, which includes moderator control) are identified in Table 1.9-4. A procedural safety control to maintain the minimum required soluble boron concentration of 2,500 mg/L during wet operations is specified in Table 1.9-10.

1.14.2.3.3.5 Intrasite Operations and Aging Facility Evaluation

Intrasite operations and the Aging Facility provide the SSCs to support movement of waste forms between surface facilities (e.g., WHF to CRCF), to and from the aging pads, and for aging of TAD canisters and DPCs. Transportation casks are moved from the receipt area to the buffer areas and subsequently to one of the surface facilities with the site prime mover. TAD canisters and vertical DPCs are moved to and from aging pads inside aging overpacks using a bottom-lift site transporter. Horizontal DPCs are moved to and from aging pads inside transportation casks or shielded transfer casks using a cask transfer trailer. TAD canisters and DPCs are aged in aging overpacks and horizontal aging modules. Section 1.2.1.3 includes an overview of intrasite operations. The details of operations performed in the Aging Facility are described in Section 1.2.7.1.1.

Moderation is the primary criticality control parameter during normal intrasite operations and aging (Table 1.14-2). Control of moderation is sufficient to maintain subcriticality during those operations with a maximum k_{eff}, including calculational uncertainty, less than 0.60 for commercial SNF and 0.75 for DOE SNF (BSC 2008b, Section 2.3.7.2). No Category 1 or Category 2 event sequences were identified that will introduce moderator into breached DOE SNF canisters, DPCs, TAD canisters, or transportation casks containing uncanistered commercial SNF during intrasite operations and aging (Tables 1.7-15 and 1.7-16).

The nuclear safety design bases for ITS features derived from the PCSA described in Sections 1.6 and 1.7 are presented in Section 1.9. The SSCs in intrasite operations and the Aging Facility that are classified as ITS for prevention of criticality (e.g., containment, which includes moderator control) are identified in Table 1.9-6.

1.14.2.3.3.6 Subsurface Facility Evaluation

The subsurface facility provides the SSCs for transfer to and operations in the underground, and locations for the emplacement of waste packages, as well as interfaces with the natural barrier. Sealed waste packages on pallets are moved from a surface facility to the underground on the waste package transport and emplacement vehicle, where they are unloaded to reside in their final locations. The details of the operations performed in the subsurface facility during the preclosure period are described in Section 1.3.1.2.

Moderation is the primary criticality control parameter during normal operations in the subsurface facility (Table 1.14-2). Control of moderation is sufficient to maintain subcriticality during those operations with a maximum k_{eff}, including calculational uncertainty, less than 0.50 for commercial SNF waste packages and less than 0.65 for DOE SNF codisposal waste packages (BSC 2008b, Section 2.3.9.2). No Category 1 or Category 2 event sequences were identified that will introduce moderator into breached TAD or DOE SNF canisters during operations in the subsurface facility (Tables 1.7-17 and 1.7-18).

The nuclear safety design bases for ITS features derived from the PCSA described in Sections 1.6 and 1.7 are presented in Section 1.9. The SSCs in the subsurface facility that are classified as ITS for prevention of criticality (e.g., containment, which includes moderator control) are identified in Table 1.9-7.

1.14.2.3.4 Criteria to Establish Subcriticality

A configuration is considered acceptably subcritical if (1) the maximum calculated effective neutron multiplication factor (k_{eff}) plus calculational uncertainties is less than or equal to the configuration-specific upper subcritical limit, or (2) it meets the single- or multiparameter limits established in Sections 5 and 6 of ANSI/ANS-8.1-1998. In equation notation, the use of the upper subcritical limit is:

$$k_{eff} + \Delta k_{eff} \leq \text{USL} \qquad \text{(Eq. 1.14-1)}$$

where

k_{eff}	=	calculated effective neutron multiplication factor for the system.
Δk_{eff}	=	an allowance for (1) statistical or convergence uncertainties, or both, in the computation of k_{eff} (Note: bounds for k_{eff} values are provided at the 95% confidence level.), (2) material and fabrication tolerances, and (3) uncertainties due to the geometric or material representations used in the computational method. (Note: allowance for items (2) and (3) can be obviated by using bounding representations.)
USL	=	an upper limit on k_{eff} characterized by statistical tolerance limits that accounts for (1) biases and uncertainties associated with the criticality code validation process, (2) any uncertainties due to extrapolation outside the range of experimental data, or limitations in the geometrical or material representations used in the computational method, and (3) a justified administrative margin to ensure subcriticality.

The upper subcritical limit is represented in equation form as:

$$\text{USL} = \text{LBTL} - \Delta k_{EROA} - \Delta k_m \qquad \text{(Eq. 1.14-2)}$$

where

LBTL	=	the lower-bound tolerance limit accounting for biases and bias uncertainties that cause the calculational results to deviate from the true value of k_{eff} for a critical experiment, as reflected over an appropriate set of critical experiments
Δk_{EROA}	=	penalty for extending the range of applicability
Δk_m	=	an administrative margin ensuring subcriticality.

The lowest (LBTL – Δk_{EROA}) values associated with the benchmarking and validation of the MCNP code and the associated cross section libraries for commercial SNF and DOE SNF configurations are 0.98 and 0.94, respectively (BSC 2008b, Section 2.3.10).

1.14.2.3.4.1 Administrative Margin and Justification

An administrative margin of 0.05 Δk is applied in the criticality safety analysis for all facilities, operations, and waste forms.

In accordance with the guidance provided in ANSI/ANS-8.24-2007, *American National Standard, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, Section 6.4, and "Justification for Minimum Margin of Subcriticality for Safety" (NRC 2006), the following considerations are taken into account in the justification of a conservative 0.05 Δk administrative margin for preclosure criticality safety analyses:

- Validation results
- Conservatisms in the calculational model
- Likelihood of off-normal conditions
- System sensitivity
- Knowledge of neutron physics.

The following subsections discuss these considerations for each of the evaluated waste forms.

Validation Results—Criticality calculational method validation in accordance with industry standards and guidance documents (e.g., ANSI/ANS-8.1-1998, including the exception taken in Regulatory Guide 3.71, and ANSI/ANS-8.24-2007) was performed. The lowest (LBTL – Δk_{EROA}) values for commercial SNF and DOE SNF (i.e., 0.98 and 0.94, respectively) are applied for normal operations and for end-states of Category 1 and Category 2 event sequences (BSC 2008b, Section 2.3.10.1).

Conservatisms in the Calculational Model—Preclosure criticality safety analyses for the various waste forms are based on conservative representations, including the following:

- Representation of all commercial SNF as fresh fuel (i.e., no burnup credit) with an enrichment of 5 wt % ^{235}U (maximum commercial SNF enrichment) and without credit for burnable poisons or the presence of ^{234}U and ^{236}U, which are neutron absorbers

- Evaluation of the most reactive fuel state for DOE SNF (i.e., fresh fuel for nonbreeder reactors, or calculated most reactive state for breeder reactor fuel)

- Demonstration of subcriticality without credit for fixed neutron absorbers, which is primarily based on moderator control for operations with canistered SNF and soluble boron for operations with uncanistered commercial SNF.

Likelihood of Off-Normal Conditions—Criticality safety design and operational criteria have been established such that normal operations and end-states of Category 1 and Category 2 event sequences are demonstrated to be subcritical. This goal is attained primarily on the basis of robust passive engineering controls with limited reliance on procedural safety controls. These controls include moderator control (for dry canister operations), and soluble neutron absorber control with a minimum concentration of 2,500 mg/L of boron (enriched to 90 atom % ^{10}B) during wet operations conducted in the WHF pool. Additional criticality control features that are not relied upon in the criticality safety analysis are provided by the following:

- Fixed neutron absorbers in TAD canisters and WHF pool staging racks

- Two types of neutron absorbers in canisters with highly enriched SNF, in the form of Ni-Gd alloy as well as gadolinium-bearing shot.

System Sensitivity—The results of the criticality analyses demonstrated that the waste forms are subcritical for normal operations and all Category 1 and Category 2 event sequences important to criticality. The criticality safety analysis (BSC 2008b) has demonstrated that the system sensitivity to perturbation in criticality control parameters is either bounded (e.g., reflection), or controlled with a margin. For example, even though moderator is controlled for dry operation such that no Category 1 or Category 2 event sequence results in introduction of moderator inside a breached canister, the criticality safety analysis has demonstrated that under conservative “nonmechanistic” geometric reconfigurations (e.g., complete flux trap gap collapse, optimized pin pitch, separation of fuel and basket, and fuel release) with reduced credit for neutron absorbers and bounding reflection conditions, subcriticality can be maintained with a substantial amount (i.e., over 100 liters) of water inside a breached commercial SNF canister (BSC 2008b).

Knowledge of Neutron Physics—Existing waste forms will be received in the GROA without the ability to alter their form to the extent that their neutron physics characteristics will be impacted. These waste forms are stable SNF from nuclear reactors whose neutronic characteristics have been well studied and extensively benchmarked.

The three fissile isotopes in commercial and DOE SNF considered in the preclosure criticality safety analysis are ^{233}U, ^{235}U, and ^{239}Pu at varying enrichments. The only neutron absorber relied upon in the criticality safety analysis is boron. The neutron moderators considered in the criticality safety analysis are hydrogen in water and in a zirconium hydride matrix as well as carbon in a limited number of canisters. The fissile isotopes at the varying enrichments, neutron absorbers, and moderators considered in the criticality safety analysis have been analyzed in numerous benchmarks with varying geometries and neutron spectra (NEA 2006).

Therefore, the neutron physics associated with the various waste forms and configurations considered in the criticality safety analysis are well behaved and well understood.

Upper Subcritical Limit Conclusion—Based on a conservative administrative margin of 0.05 Δk and the lowest (LBTL – Δk_{EROA}) values, the upper subcritical limits for normal operations and end-states of Category 1 and Category 2 event sequences for commercial SNF and DOE SNF are 0.93 and 0.89, respectively (BSC 2008b, Section 2.3.10.1).

1.14.2.3.5 Criticality Alarm Systems

The criticality evaluations described in Section 1.14.2.3.3 for the surface and subsurface facilities indicate that there are no Category 1 or Category 2 event sequences for which k_{eff} is greater than the upper subcritical limit in the surface and subsurface facilities during the preclosure period. Therefore, criticality alarms are not required to mitigate the consequences of Category 1 or Category 2 event sequences in order to meet the performance objectives of 10 CFR 63.111.

Based on the absence of specific criticality monitoring requirements in 10 CFR Part 63, prescriptive NRC regulations for similar applications (i.e., 10 CFR Part 50, 10 CFR Part 70, and

10 CFR Part 72) and guidance of ANSI/ANS-8.3-1997, a criticality alarm system is not required anywhere in the GROA. This conclusion is based on the following:

- As presented in Section 1.14.2.3.3, preclosure operations with fissile materials at the repository have been demonstrated to be subcritical for normal operations and for Category 1 and Category 2 event sequences.
- Given the conservatism in the preclosure criticality safety analysis (BSC 2008b), the probability of an actual criticality accident is less than 10^{-4} during the preclosure period.
- Canistered waste forms are packaged in transportable canisters (DPCs, TAD, naval SNF, and DOE canisters), for which criticality monitoring would not be expected to be required based on similar regulation.
- The radiation/radiological monitoring system in the surface and subsurface facilities (Section 1.4.2.2) is designed to detect radiological releases or extreme radiation levels regardless of the cause.

1.14.2.3.6 Offsite Operations

As discussed in Section 1.6.1, offsite operations are not considered as initiating events since they will be performed under an NRC-accepted Quality Assurance Program. In addition, all shipments to the repository must be loaded in accordance with the certificate of compliance for a specific transportation cask that is licensed under 10 CFR Part 71. Nonetheless, reliability of offsite operations and their impacts, if any, on the preclosure criticality safety control parameters are discussed in this section.

1.14.2.3.6.1 Waste Form Characteristics

1.14.2.3.6.1.1 Waste Form Characteristics for Commercial SNF

Because the waste form characteristics for commercial SNF used in the criticality safety analysis are considered bounding as described in Sections 1.14.2.3.2.1.1 and 1.14.2.3.2.2.1, there is no potential for commercial SNF misload.

1.14.2.3.6.1.2 Waste Form Characteristics for DOE SNF

The preclosure criticality safety analysis considers the nine representative DOE SNF types listed in Section 1.14.2.3.2.3.1. Consideration of waste form misload is not appropriate because the criticality safety analysis is only for representative DOE SNF fuel types and loading procedures for DOE standardized SNF canisters have not been established yet.

1.14.2.3.6.2 Moderation

1.14.2.3.6.2.1 Moderation and Commercial SNF

Prior to receipt at the GROA, commercial SNF canisters will have been dried using a process similar to the one described in NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*, Section 9.5.4.1 (NRC 2000), which states:

> …The staff has accepted the combination of a draining procedure and a vacuum drying procedure as providing adequate assurance that the gases in the cask meet the maximum oxidizing gas criteria. The vacuum drying procedure involves a vacuum test to demonstrate that there is no water in the cask or fuel. A cask that is evacuated to less than 3 torr and, after sealing, does not have a cask pressure which increases by 1 torr over 30 minutes is considered to be free of water…

10 CFR Part 71, Subpart E, Paragraphs 55(b) and 55(c) state:

> (b) Except as provided in paragraph (c) or (g) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained: (1) The most reactive credible configuration consistent with the chemical and physical form of the material; (2) Moderation by water to the most reactive credible extent; and (3) Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging. (c) The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.

Even for the low probability event of inadequate dewatering of commercial SNF canisters, there is no credible potential for criticality for the following reasons:

- Commercial SNF canisters will have to comply with 10 CFR Part 71, Subpart E, Paragraph 55(b) requirements as described above.

- For postclosure criticality control, commercial SNF canisters will contain basket designs with sufficient criticality control features that subcriticality is maintained under degraded flooded conditions.

- The criticality safety analysis demonstrates that under the worst damage conditions of complete flux trap gap collapse, fuel release, and maximum fuel pin pitch, subcriticality is still maintained with more than a hundred liters of water remaining in a commercial SNF canister (BSC 2008b, Section 2.3.12.2.1).

1.14.2.3.6.2.2 Moderation and DOE SNF

DOE standardized SNF canisters will have been dried as described in Section 1.14.2.3.6.2.1 prior to receipt at the GROA. Further, DOE SNF canisters will have to demonstrate compliance with 10 CFR Part 71, Subpart E, Paragraph 55(b) requirements as described in Section 1.14.2.3.6.2.1. Finally, for the purpose of postclosure criticality control, baskets and neutron absorbers in the form of plates and/or gadolinium-bearing shot are included in DOE standardized SNF canisters containing TRIGA, Fast Flux Test Facility, Enrico Fermi, Shippingport Light Water Breeder Reactor, and Advanced Test Reactor SNF to ensure subcriticality under fully flooded intact and degraded configurations. Therefore, even for the low probability event of inadequate dewatering of DOE SNF canisters, there is no credible potential for criticality.

1.14.2.3.6.3 Neutron Absorber

For dry operations, neutron absorber misload is inconsequential given that the preclosure criticality safety analysis does not credit neutron absorber in the absence of moderation as discussed in Section 1.14.2.3.3. For wet operations, the minimum required concentration of 2,500 mg/L of soluble boron (enriched to 90 atom % ^{10}B) in the WHF pool is sufficient to compensate for the complete omission of fixed neutron absorbers in the analyzed commercial SNF canister designs (Section 1.14.2.3.3.4).

1.14.2.3.6.4 Geometry

The criticality safety analysis considered a wide range of geometric reconfigurations (BSC 2008b, Section 2.3.12.4) that bound any reasonable potential reconfiguration due to undetected mishandling during offsite operations. Therefore, offsite operations have no additional impact on the preclosure criticality safety analysis from a geometry standpoint.

1.14.2.3.6.5 Interaction

Interaction is considered between separate units containing fissile material in the surface and subsurface facilities. Therefore, offsite operations have no impact on interaction.

1.14.2.3.6.6 Reflection

As discussed in Sections 1.14.2.3.2.1.5, 1.14.2.3.2.2.5, and 1.14.2.3.2.3.5, reflection is bounded in the preclosure criticality safety analysis, and potential reflectors that could be shipped are taken into account. Therefore, offsite operations have no additional impact on the preclosure criticality safety analysis from a reflection standpoint.

1.14.2.4 Example of the Criticality Safety Analysis

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.1: AC 1(8), (9); Section 2.1.1.7.3.3(I): AC 4(1), (2), (4), (5); Section 2.1.1.7.3.3(III): AC 1(2), (6)]

The detailed criticality safety analysis for the transfer of a PWR TAD canister from a transportation cask to an aging overpack, from a transportation cask to a waste package, or from an aging overpack to a waste package, using a canister transfer machine in a CRCF is described in this section to illustrate the application of the preclosure criticality analysis process. The categorization of the event sequence for this operation is presented in Section 1.7.5.

The design parameters used for the PWR TAD canister criticality safety model are summarized in Table 1.14-3. The table provides the design criteria from the TAD canister specification (DOE 2008) as well as additional parameters that are not given in that specification, but are necessary in order to determine the criticality potential of the TAD canister. The values assumed for these parameters reflect typical design practices for PWR commercial SNF transportation casks and are considered appropriate for this analysis. A cross-sectional view of the PWR TAD canister geometry model is shown in Figure 1.14-2.

Most of the parameters listed in Table 1.14-3 that do not have specific design criteria are those parameters relating to the thickness of the canister wall, base, and lid, and the canister height, which are inconsequential to the criticality safety analysis. For example, the examination in the criticality safety analyses of a wide variety of close-fitting reflectors that are effectively full-thickness, that is, the thickness is greater than or equal to any dimension that may be encountered during dry operations or the thickness is sufficient to be considered infinite, ensures that deviations between the "actual" and "modeled" TAD canister thickness are inconsequential. In addition, deviations between the "actual" and "modeled" canister height will not impact the conclusions of the analysis because the active fuel region which is modeled with a conservative length in the calculations is effectively infinite.

A few of the parameters listed in Table 1.14-3 that do not have specific design criteria are those parameters related to the fuel compartment width and wall design, in addition to the spacing between adjacent compartments. The values used for these parameters can influence the criticality safety performance of the TAD canister. In particular, the fuel compartment inner width and the spacing between adjacent compartments affect the criticality safety analysis for moderated conditions.

1.14.2.4.1 Sensitivity Study

In order to determine the criticality potential of this operation, a series of k_{eff} sensitivity calculations were performed. These calculations evaluated the impact on system reactivity of variations in each of the parameters important to criticality during the preclosure period, which are waste form characteristics, moderation, fixed neutron absorber, geometry, interaction, and reflection. The criticality sensitivity calculations determined the sensitivity of the effective neutron multiplication factor k_{eff} to variations in any of these parameters as a function of other relevant parameters. These criticality calculations demonstrated that each parameter:

- Does not need to be controlled because it is bounded (i.e., its analyzed value is greater than or equal to the design limit) or its effect is bounded,
- Needs to be controlled if another parameter is not controlled (conditional control), or
- Needs to be controlled because it is the primary criticality control parameter.

1.14.2.4.1.1 Waste Form Characteristics

For commercial SNF the analysis considered the bounding waste form parameters described in Section 1.14.2.3.2.1. Because these waste form parameters are considered bounding, no initiating events in Section 1.6 or event sequences in Section 1.7 involving misloads of commercial SNF need to be identified.

1.14.2.4.1.2 Moderation

Under normal conditions the TAD canisters are completely dry. Based on these dry (i.e., unmoderated) conditions, substantial margin exists (calculated peak k_{eff} is less than 0.50, as seen in Figure 1.14-3, for a single PWR TAD canister with various reflector materials). However, with introduction of moderator into the PWR TAD canister, the upper subcritical limit could be exceeded. Consequently, moderation control is essential for ensuring the subcriticality of PWR TAD canisters. Therefore, initiating events in Section 1.6, and, if necessary, event sequences in Section 1.7 associated with introduction of moderation into PWR TAD canisters during this operation, must be identified because moderator control is the primary criticality control parameter.

1.14.2.4.1.3 Neutron Absorber

Under normal conditions the TAD canisters are completely dry, which results in a hard neutron spectrum. Under these dry, unmoderated conditions, the borated stainless steel neutron absorber plates associated with the TAD canister basket structure provide limited neutron absorption due to the hard neutron spectrum, to the extent that their complete omission will not result in criticality potential (i.e., maximum k_{eff} is less than 0.50, as seen in Figure 1.14-3). However, with introduction of moderator into the PWR TAD canister, the neutron absorber in the basket is important. In this respect, the neutron absorber directly influences the established moderation limits tolerable for the PWR TAD canisters as illustrated in Figure 1.14-4.

Therefore, no initiating events in Section 1.6 or event sequences in Section 1.7 that impact neutron absorber during this operation need to be identified as long as the canister internals remain dry.

1.14.2.4.1.4 Geometry

Given the relatively low fissile enrichment of commercial SNF, any rearrangement of the SNF or the basket (without introduction of moderator) will not result in an a configuration that has potential for criticality. However, with introduction of moderator into the PWR TAD canister, the geometry of the commercial SNF and basket material is important. In this respect, the geometry of the canister basket and commercial SNF directly influence the established moderation limits tolerable for the PWR TAD canisters as illustrated in Figure 1.14-5.

Therefore, no initiating events in Section 1.6 or event sequences in Section 1.7 associated with geometry changes during this operation need to be identified as long as the canister internals remain dry.

1.14.2.4.1.5 Interaction (Neutronic Coupling)

In order to bound interaction or neutronic coupling with other TAD canisters, an infinite planar array configuration was modeled. These models include axial reflection conditions as described in Section 1.14.2.4.1.6. A periodic hexagonal boundary was modeled directly adjacent to the cylindrical surface of the canister to simulate an infinite planar array of canisters in a close-packed, triangular-pitched configuration, for which the results are presented in Figure 1.14-6. It is seen that under dry conditions, substantial subcriticality margin exists (i.e., maximum k_{eff} is less than 0.60).

The interstitial space between the TAD canisters was evaluated with variable density water. From the results presented in Figure 1.14-7, it is seen that the presence of moderator external to, and between, the TAD canisters results in a decrease in the system reactivity. Thus, interaction control between TAD canisters is not required for this operation. In addition, the analysis has shown that interaction between the most reactive DOE SNF canister and containers of commercial SNF does not need to be controlled (BSC 2008b, Section 2.3.2.3.4). Therefore, interaction control is not required for this operation, and no initiating events in Section 1.6 or event sequences in Section 1.7 involving interaction conditions need to be identified.

1.14.2.4.1.6 Reflection

The operations described in Section 1.14.2.4 result in positioning a PWR TAD canister in close proximity to, or in contact with, a wide variety of structures and components. To bound the wide range of reflection conditions that could exist, potential reflector materials (water, concrete, Alloy 22, stainless steel, lead, natural uranium metal, titanium, HLW glass, and tuff) were modeled as close-fitting reflectors that are effectively full-thickness, that is, the thickness is greater than or equal to any dimension that may be encountered during dry operations or the thickness is sufficient to be considered infinite. As seen in Figure 1.14-3, there is a significant margin of subcriticality (k_{eff} less than 0.50) for the most limiting reflection conditions, demonstrating that interaction is bounded for these operations.

Therefore, reflection control is not required for this operation, and no initiating events in Section 1.6 or event sequences in Section 1.7 involving reflection conditions need to be identified.

1.14.2.4.1.7 Summary of Sensitivity Study of Criticality Parameters

The criticality sensitivity calculations provided the following guidance on the criticality control parameters:

- Waste form characteristics are bounded and do not need to be controlled. Therefore, no event sequences need to be identified that are associated with waste form misload for these operations.

- Moderation is the primary criticality control parameter and event sequences associated with moderation inside a PWR TAD canister must be evaluated.

- Fixed neutron absorber needs to be controlled only if moderation is present. Only Category 1 and Category 2 event sequences that result in moderation inside a PWR TAD canister need to consider fixed neutron absorber.

- Geometry needs to be controlled only if moderation is present. Only Category 1 and Category 2 event sequences that result in moderation inside a PWR TAD canister need to consider geometry control.

- Interaction is bounded for these operations and does not need to be controlled. Therefore, no event sequences need to be identified that are associated with interaction for these operations.

- Reflection is bounded for these operations and does not need to be controlled. Therefore, no event sequences need to be identified that are associated with reflection for these operations.

1.14.2.4.2 Event Sequence Analysis

Figure 1.7-2 displays the event sequence diagram associated with structural challenges that may occur during the transfer of a TAD canister to or from staging, a transportation cask, a waste package, or an aging overpack, using a canister transfer machine in a CRCF.

The event sequences displayed in Figure 1.7-2 start with several possible initiating events, with each resulting in a structural challenge to the canister being transferred. These initiating events are grouped together, because they elicit the same pivotal events and lead to the same end states.

As discussed in Section 1.7.5, the event sequences shown in Figure 1.7-2 that result in moderation entering a breached TAD canister after a structural challenge are all beyond Category 2. Therefore, these event sequences are screened from further evaluation of criticality potential and need not include an examination of potential impacts on fixed neutron absorber and geometry.

1.14.3 Nuclear Criticality Safety Regulations, Codes, and Standards

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.3(I): AC 1]

The nuclear criticality safety program for the repository complies with 10 CFR Part 63 and with the applicable portions of Regulatory Guide 3.71. The information in this section is drawn principally from *Preclosure Criticality Analysis Process Report* (BSC 2008a, Section 2.1).

1.14.3.1 Applicable Standards Documents

[NUREG-1804, Section 2.1.1.6.3: AC 1(2)(f); Section 2.1.1.7.3.3(I): AC 1]

Regulatory Guide 3.71 endorses 15 ANSI/ANS-8 nuclear criticality safety standard documents (four with exceptions). Preclosure criticality safety analysis and repository design are performed in accordance with the following six applicable standards, except where noted:

- ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* (The exception is that a level of validation beyond that endorsed by the standard is required by Regulatory Guide 3.71.)

- ANSI/ANS-8.3-1997, *American National Standard Criticality Accident Alarm System* (The exception is that additional requirements beyond those in the standard are discussed in Regulatory Guide 3.71.)

- ANSI/ANS-8.14-2004 *American National Standard, Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors*

- ANSI/ANS-8.17-2004, *American National Standard, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors* (The exception taken by Regulatory Guide 3.71 is not applicable to preclosure.)

- ANSI/ANS-8.21-1995, *American National Standard for Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*

- ANSI/ANS-8.22-1997, *American National Standard for Nuclear Criticality Safety Based on Limiting and Controlling Moderators*.

Prior to beginning of operations, the following additional standards from Regulatory Guide 3.71 will be implemented:

- ANSI/ANS-8.19-2005, *American National Standard, Administrative Practices for Nuclear Criticality Safety*

- ANSI/ANS-8.20-1991, *American National Standard, Nuclear Criticality Safety Training*.

An additional standard developed since the publication of Regulatory Guide 3.71, that is used by the repository for nuclear criticality safety design, is the following standard which addresses validation of neutron transport methods used for criticality analysis:

- ANSI/ANS-8.24-2007, *American National Standard, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*.

1.14.4 General References

ANSI/ANS-8.1-1998. *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. La Grange Park, Illinois: American Nuclear Society. TIC: 242363.

ANSI/ANS-8.3-1997. 2003. *American National Standard Criticality Accident Alarm System*. La Grange Park, Illinois: American Nuclear Society. TIC: 258157.

ANSI/ANS-8.14-2004. *American National Standard, Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors*. La Grange Park, Illinois: American Nuclear Society. TIC: 258161.

ANSI/ANS-8.17-2004. *American National Standard, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*. La Grange Park, Illinois: American Nuclear Society. TIC: 257593.

ANSI/ANS-8.19-2005. *American National Standard, Administrative Practices for Nuclear Criticality Safety*. La Grange Park, Illinois: American Nuclear Society. TIC: 258037.

ANSI/ANS-8.20-1991. 2005. *American National Standard, Nuclear Criticality Safety Training*. La Grange Park, Illinois: American Nuclear Society. TIC: 258162.

ANSI/ANS-8.21-1995. 2001. *American National Standard for Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*. La Grange Park, Illinois: American Nuclear Society. TIC: 258163.

ANSI/ANS-8.22-1997. *American National Standard for Nuclear Criticality Safety Based on Limiting and Controlling Moderators*. La Grange Park, Illinois: American Nuclear Society. TIC: 235109.

ANSI/ANS-8.24-2007. *American National Standard, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations.* La Grange Park, Illinois: American Nuclear Society. TIC: 259483.

BSC (Bechtel SAIC Company) 2008a. *Preclosure Criticality Analysis Process Report*. TDR-DS0-NU-000001 REV 03. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080220.0001.

BSC 2008b. *Preclosure Criticality Safety Analysis*. TDR-MGR-NU-000002 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080307.0007.

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DOE (U.S. Department of Energy) 2008. *Transportation, Aging and Disposal Canister System Performance Specification*. WMO-TADCS-000001, Rev. 1 ICN 1. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20080331.0001.

NEA (Nuclear Energy Agency) 2006. *International Handbook of Evaluated Criticality Safety Benchmark Experiments*. September 2006 Edition. NEA/NSC/DOC(95)03. Paris, France: Nuclear Energy Agency, Organisation for Economic Co-Operation and Development. TIC: 259708.

NRC (U.S.Nuclear Regulatory Commission) 2000. *Standard Review Plan for Spent Fuel Dry Storage Facilities*. NUREG-1567. Washington, D.C.: U.S.Nuclear Regulatory Commission. TIC: 247929.

NRC 2006. "Justification for Minimum Margin of Subcriticality for Safety." Interim Staff Guidance FCSS-ISG-10. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20070306.0210.

Regulatory Guide 3.71, Rev. 1. 2005. *Nuclear Criticality Safety Standards for Fuels and Material Facilities*. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20060206.0325.

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Table 1.14-1. Fissile Isotopes in High-Level Radioactive Waste Glass Canisters

Fissile Isotope	Hanford Canister	Idaho National Laboratory Canister	Savannah River Site Canister	West Valley Demonstration Project Canister
^{233}U Mass (g)	0.217	6.29×10^{-4}	5.80	9.37
^{235}U Mass (g)	257	304	307	172
^{239}Pu Mass (g)	343	32.4	280	141
^{241}Pu Mass (g)	1.18	0.208	8.16	3.01
Total Fissile Isotope Mass (g)	601	337	601	325
Nominal Glass Volume (L)	1,080	625	670	665
Fissile Isotope Concentration (g/L)	0.557	0.539	0.897	0.489

Source: BSC 2008b, Table 3.

Table 1.14-2. Criticality Control Parameter Summary

Parameter/Operation	Canistered Commercial SNF (Dry Operations)	Commercial SNF (WHF Wet Operations)	DOE SNF	High-Level Radioactive Waste
Waste Form Characteristics	No[a]	No[a]	No[b]	No[c]
Moderation	Yes[d]	NA	Yes[d]	No
Interaction	No	Conditional[e]	Yes[f]	No
Geometry	Conditional[g]	Conditional[e]	Conditional[g]	No
Fixed Neutron Absorber	Conditional[g]	Conditional[e]	Conditional[g]	No
Soluble Neutron Absorber	NA	Yes[h]	NA	NA
Reflection	No	No	No	No

NOTE: [a]As described in Sections 1.14.2.3.2.1.1 and 1.14.2.3.2.2.1, the criticality safety analysis considers bounding waste form characteristics. Therefore, there is no potential for a waste form misload.
[b]As described in Section 1.14.2.3.2.3.1, the preclosure criticality safety analysis considers the nine representative DOE SNF types. Consideration of waste form misload is not appropriate because the criticality safety analysis is only for representative DOE SNF fuel types and loading procedures for DOE standardized SNF canisters have not been established yet.
[c]Criticality safety control features are not necessary for HLW canisters because criticality is not possible due to the low concentration of fissile isotopes in an HLW canister (Section 1.14.2.3.2.4).
[d]Moderation is the primary criticality control parameter.
[e]Needs to be controlled only if the soluble boron concentration in the pool and transportation cask/DPC fill water is less than the concentration required to maintain subcriticality.
[f]Placing more than four DOE standardized SNF canisters outside their designated staging racks or a codisposal waste package needs to be controlled.
[g]Needs to be controlled only if moderator is present.
[h]Minimum required soluble boron concentration in the pool is 2,500 mg/L boron enriched to 90 atom % ^{10}B.
NA = not applicable.

Source: BSC 2008b, Table 6.

Table 1.14-3. Design Parameters Evaluated for the Pressurized Water Reactor Transportation, Aging, and Disposal Canister MCNP Model

Design Parameter	MCNP Model		Design Criteria
TAD Canister Body			
Outer diameter of TAD canister	66.0 in.	167.64 cm	66.0 in. (min) 66.5 in. (max)
Inner diameter of TAD canister[a]	65.0 in.	165.10 cm	No specific criteria
Outer length/height of TAD canister	211.5 in.	537.21 cm	211.5 in. (min), 212.0 in. (max)
TAD canister spacer[a]	Not Modeled		Required for TAD canisters less than 211.5 in. in height
Inner length/height of TAD canister[a]	210.5 in.	534.67 cm	No specific criteria
TAD canister base thickness[a]	0.5 in.	1.27 cm	No specific criteria
TAD canister lid thickness[a]	0.5 in.	1.27 cm	No specific criteria
TAD Canister Basket Structure			
Number of fuel assembly compartments	21		21
Inner width of fuel assembly compartment[a]	9.0 in.	22.86 cm	No specific criteria
Compartment inner wall thickness[a]	0.1875 in.	0.48 cm	No specific criteria
Compartment borated stainless steel panel arrangement	Four panels around each compartment with a flux trap between		Panels must surround all four longitudinal sides of assemblies
Borated stainless steel panel thickness between adjacent fuel assemblies[b]	0.3150 in.	0.8 cm	6 mm remaining after 10,000 years with 250 nm/yr of corrosion for each surface
Basket height	Same as assembly height		The borated stainless steel plates are required to cover the entire active fuel region plus an allowance for any axial shift in the fuel assemblies
Compartment outer wall thickness[a]	0.1875 in.	0.48 cm	No specific criteria
Outer width of fuel assembly compartment[a]	10.38 in.	26.37 cm	No specific criteria
Spacing between compartments (surface-to-surface)[a]	0.0 – 0.91 in.	0 – 2.32 cm	No specific criteria
Axial placement of fuel/basket in TAD canister	Fuel/basket modeled to sit directly on the base of the TAD canister cavity		No specific criteria

NOTE: [a]These values are assumed.
[b]There are two 0.8-cm borated stainless steel panels between assemblies.

Source: BSC 2008b, Table 1.

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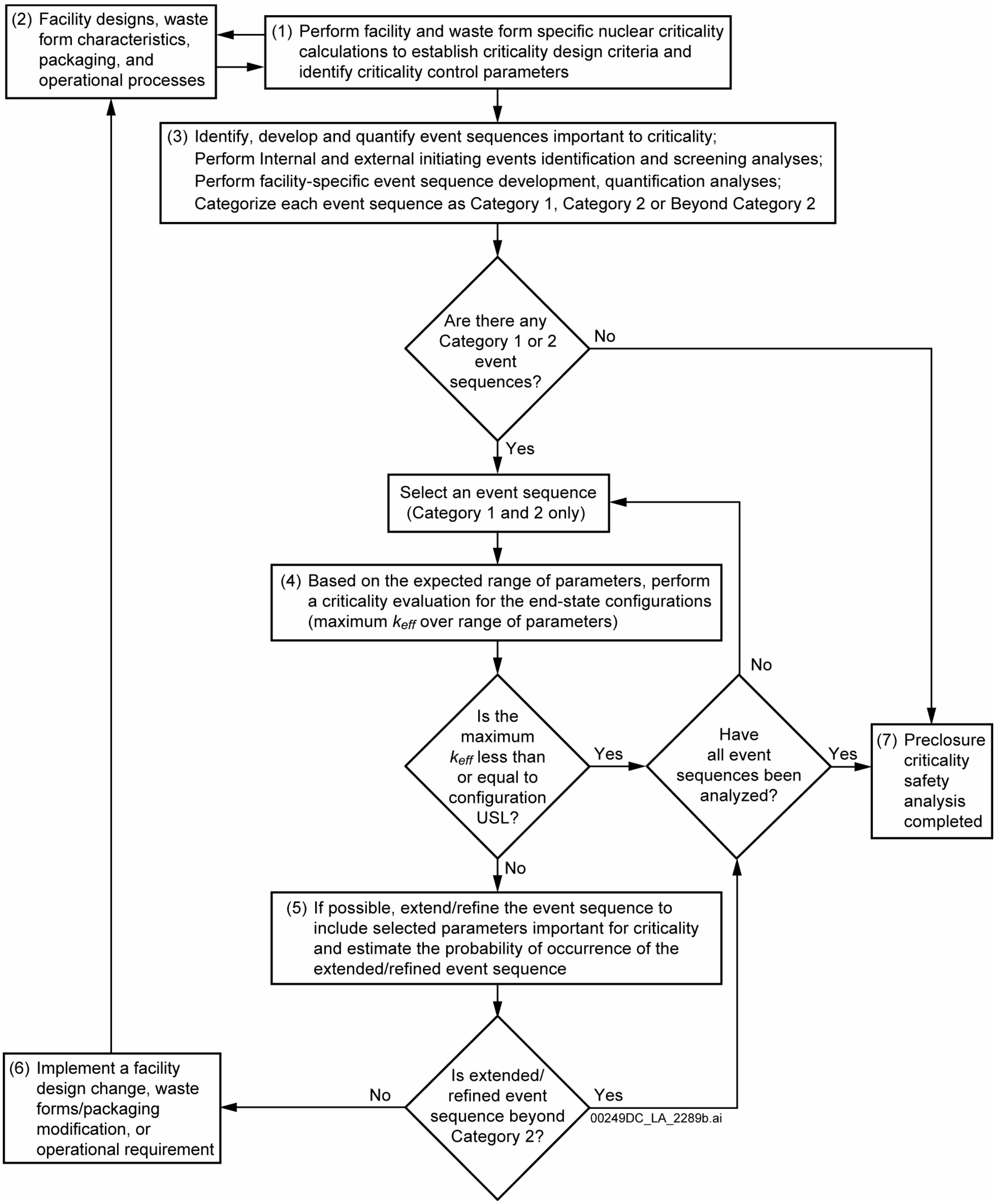

Figure 1.14-1. Overview of the Preclosure Criticality Analysis Process

NOTE: Step (1) may include evaluation against single- and multiparameter limits.
USL = upper subcritical limit.

Source: BSC 2008a, Figure 3-1.

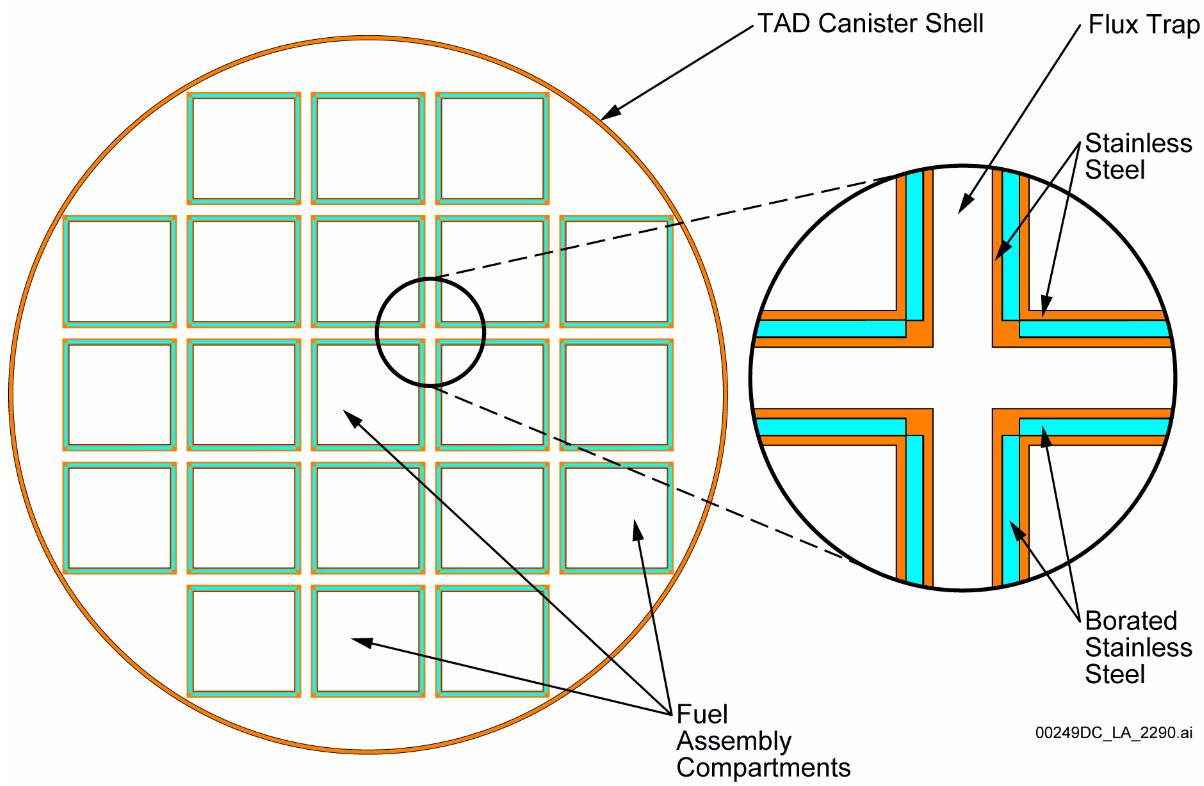

Figure 1.14-2. Radial Cross Section of the Pressurized Water Reactor Transportation, Aging, and Disposal Canister MCNP Model

Source: BSC 2008b, Figure 1.

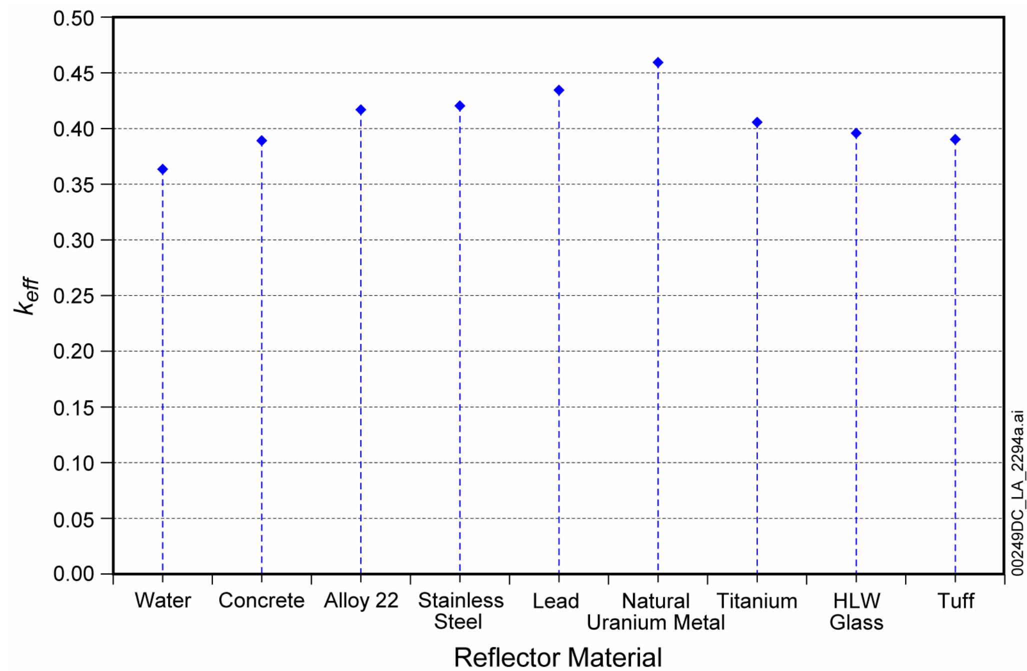

Figure 1.14-3. Results of Pressurized Water Reactor Transportation, Aging, and Disposal Canister Reflection Parameter Sensitivity Study

Source: BSC 2008b, Figure 19.

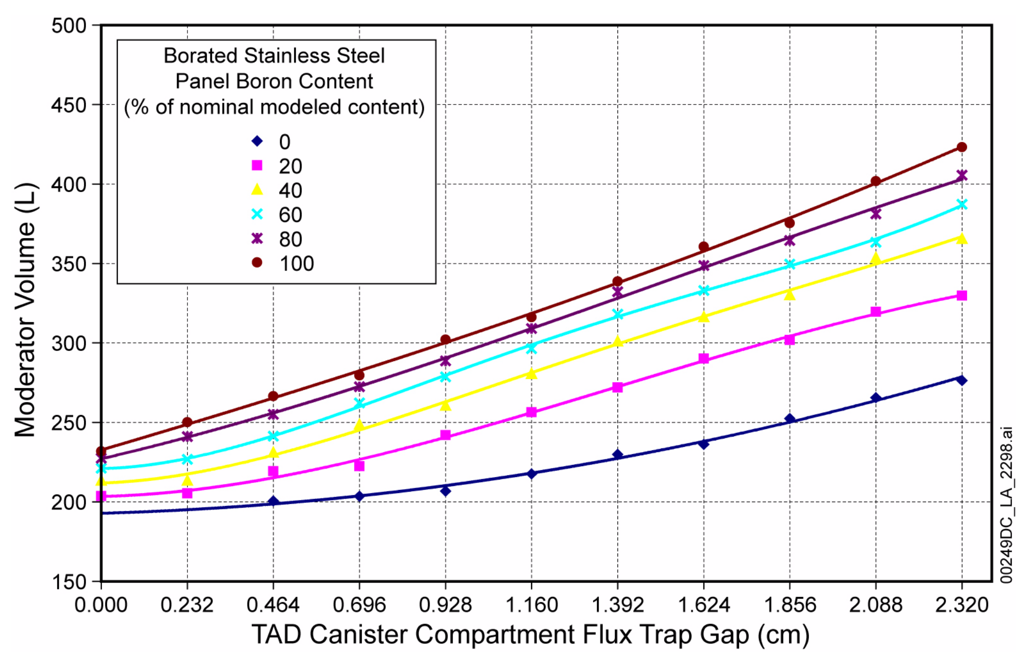

Figure 1.14-4. Maximum Safe Moderator Volume from Pressurized Water Reactor Transportation, Aging, and Disposal Canister Neutron Absorbers Sensitivity Study

Source: BSC 2008b, Figure 22.

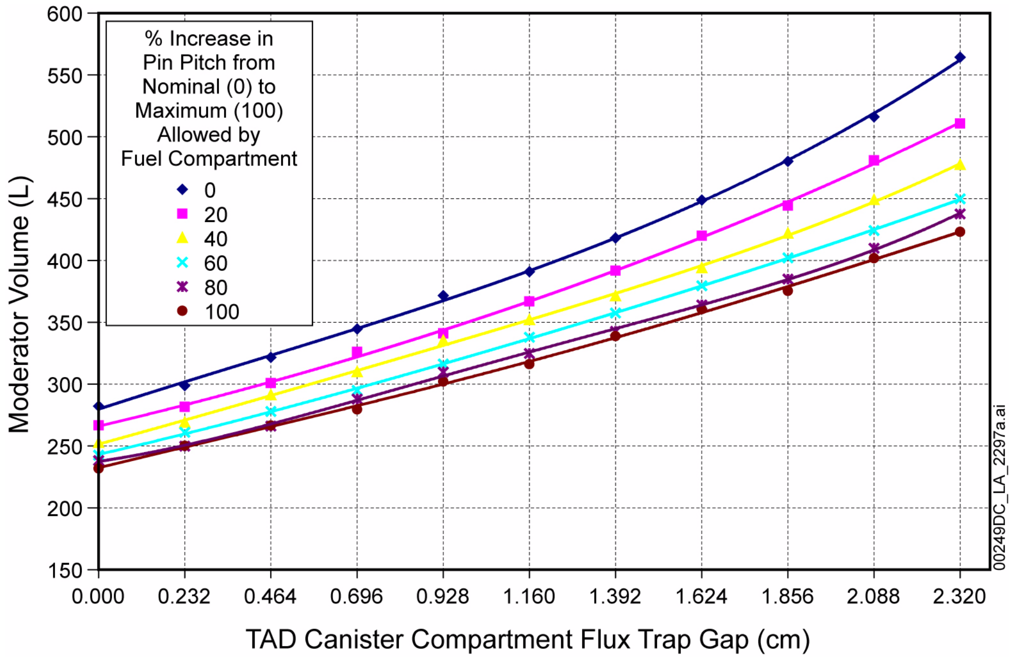

Figure 1.14-5. Maximum Safe Moderator Volume from Pressurized Water Reactor Transportation, Aging, and Disposal Canister Geometry Sensitivity Study

Source: BSC 2008b, Figure 20.

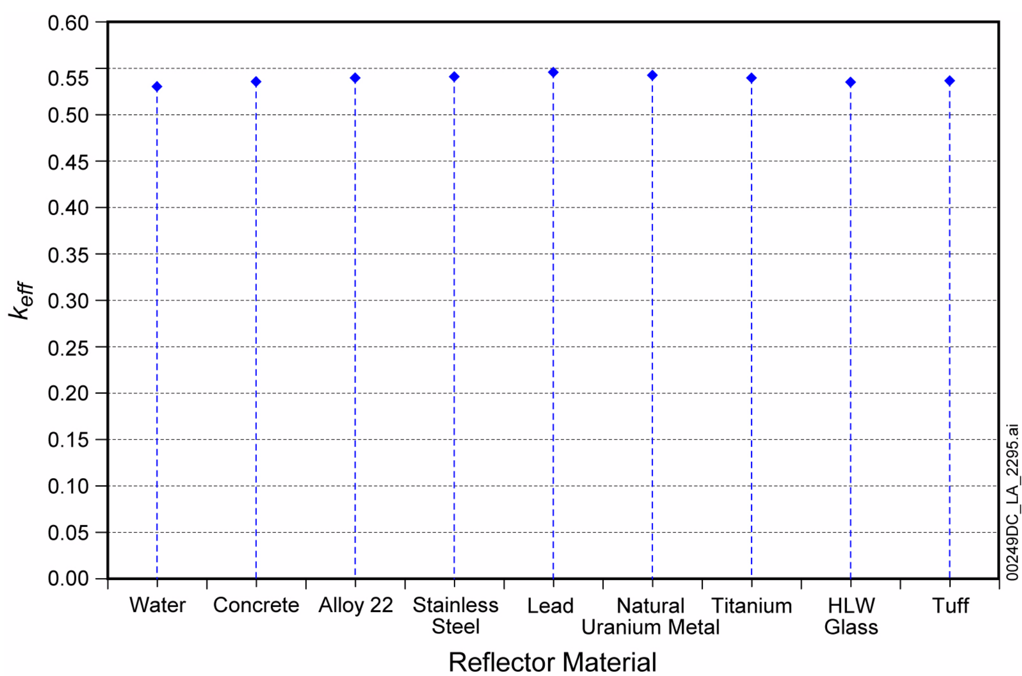

Figure 1.14-6. Results of Pressurized Water Reactor Transportation, Aging, and Disposal Canister Interaction Parameter Sensitivity Study

Source: BSC 2008b, Figure 24.

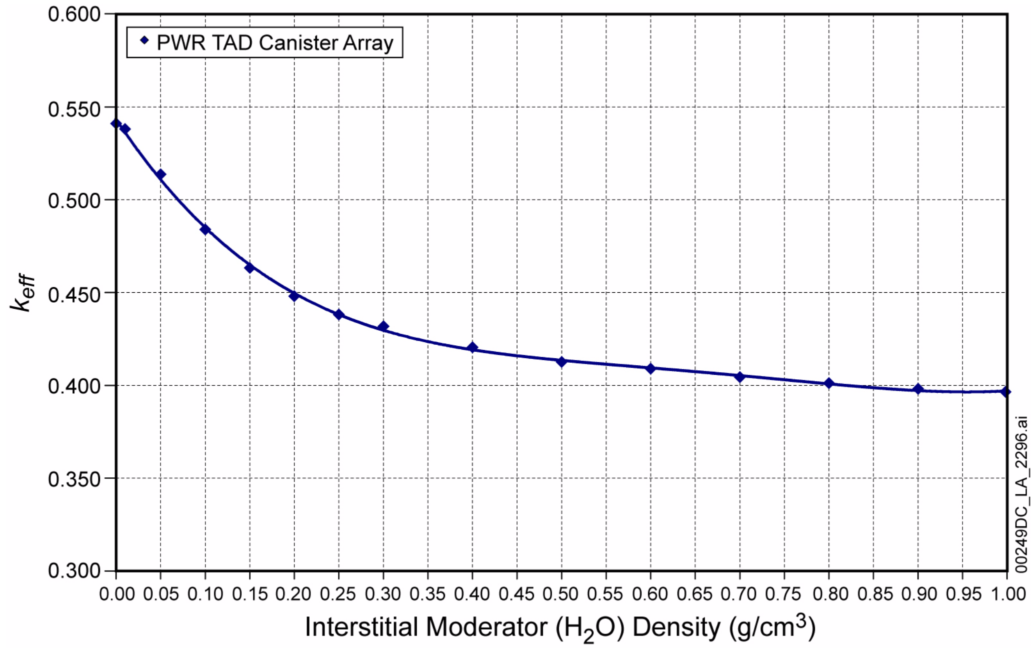

Figure 1.14-7. Results of Interaction with Interstitial Moderation Sensitivity Study

Source: BSC 2008b, Figure 25.

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