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1.5 WASTE FORM AND WASTE PACKAGE

Sections 1.5.1 and 1.5.2 describe the waste forms to be permanently disposed and the waste packages that contain the waste forms, respectively.

Specifically, the waste forms include commercial spent nuclear fuel (SNF), including mixed oxide; high-level radioactive waste (HLW), including plutonium arrayed in vitrified glass; U.S. Department of Energy SNF; and naval SNF, as further detailed in Sections 1.5.1.1 through 1.5.1.4, respectively. Section 1.5.1 characterizes the ranges of parameters that describe the SNF and the HLW. The U.S. Department of Energy SNF, HLW, and naval SNF are received at the repository in sealed canisters that are directly inserted into the waste packages. Additionally, commercial SNF will mostly be received in transportation, aging, and disposal (TAD) canisters from utility sites. These TAD canisters can also be directly inserted into the waste package or sent to the aging pad in aging overpacks if additional cooling of the SNF is required. The disposable canisters and their physical characteristics, functional features, and design are described in Section 1.5.1.

Some of the commercial SNF may also be sent to the repository as uncanistered SNF in a cask or in a dual-purpose canister. This commercial SNF will be removed from the cask or dual-purpose canister and placed in the TAD canister at the repository before it is placed into a waste package.

The waste package design and its various configurations are described in Section 1.5.2. There are six waste package configurations described that accommodate the physical, thermal, and neutronic characteristics of the SNF and HLW as further detailed in Section 1.5.2.1. Other waste package configurations will be designed and evaluated. The remaining Sections 1.5.2.2 through 1.5.2.9 address the principal characteristics of the waste package and its internal components, such as dimensions, weights, materials, fabrication, closure, and nondestructive examination.

The information presented in these sections addresses requirements contained in 10 CFR 63.21(c) by providing a description of the kind, amount, and specifications of radioactive material to be received at the geologic repository operations area and a general description of the structures, systems, and components and the operational process activities in the geologic repository operations area. The information presented in this section also addresses requirements for description of the repository safety analysis, as stated in 10 CFR 63.112. The following table lists each subsection of this section and the corresponding regulatory requirements and the acceptance criteria from NUREG-1804 that are addressed in that subsection.

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference (and Changes to NUREG-1804 from HLWRS ISGs)
1.5.1	Characteristics of Spent Nuclear Fuel and High-Level Radioactive Waste	63.21(c)(3) 63.21(c)(4) 63.112(a) 63.112(f)(2)	Section 2.1.1.2.3: Acceptance Criterion 4 Acceptance Criterion 5 Acceptance Criterion 6 Section 2.1.1.6.3: Acceptance Criterion 2 Section 2.1.1.7.3.1: Acceptance Criterion 1 Section 2.1.1.7.3.2: Acceptance Criterion 1 Section 2.1.1.7.3.3(III): Acceptance Criterion 1
1.5.2	Waste Packages and Their Components	63.21(c)(2) 63.21(c)(3) 63.112(a) 63.112(f)(2)	Section 2.1.1.2.3: Acceptance Criterion 5 Acceptance Criterion 6 Section 2.1.1.6.3: Acceptance Criterion 1 Acceptance Criterion 2 Section 2.1.1.7.3.1: Acceptance Criterion 1 Section 2.1.1.7.3.2: Acceptance Criterion 1 Section 2.1.1.7.3.3(III): Acceptance Criterion 1

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1.5.1 Characteristics of Spent Nuclear Fuel and High-Level Radioactive Waste

[NUREG-1804, Section 2.1.1.2.3: AC 4, AC 5, AC 6; Section 2.1.1.6.3: AC 2; Section 2.1.1.7.3.1: AC 1; Section 2.1.1.7.3.2: AC 1; Section 2.1.1.7.3.3(III): AC 1]

This section presents the range of parameters that describe the spent nuclear fuel (SNF) and high-level radioactive waste (HLW) waste forms that will be disposed in the repository. The designs of the canisters for commercial SNF, HLW, U.S. Department of Energy (DOE) SNF, and naval SNF are also described. The bases for the development of source-term data used in the preclosure safety analysis (PCSA), shielding analysis, postclosure performance assessments, and total system performance assessment (TSPA) are presented. The associated tables and figures provide more specific waste form parameters, classifications, and data.

In addition, this section references both the preclosure and postclosure safety analysis basis for the wastes and canisters that are planned for handling and disposal at the repository.

This section includes a description of commercial mixed oxide fuel for which DOE has not completed the necessary safety analyses. The commercial mixed oxide fuel has been included in the radionuclide inventory for the TSPA. It is the intent of the DOE to include this waste in future licensed operations at the repository. The process prescribed in 10 CFR 63.22 and 10 CFR 63.46 will be used, as appropriate, to obtain authorization to receive the waste once the analyses have been completed.

This section also includes a design description, the preclosure and postclosure nuclear safety design bases, and the design criteria for each canister type as needed to demonstrate that the safety analysis basis has been satisfied for each canister type. The N Reactor fuel and Shippingport pressurized water reactor (PWR) Core 2 blanket fuel are included in the radionuclide inventory to the TSPA and have been packaged in multicanister overpack (MCO) canisters; however, the PCSA basis for the MCO has not been completed. The MCO design has been deterministically analyzed and successful drop tests have been completed and are described in this section. Nonetheless, repository facility design details, probabilistic event sequence categorization analyses, and criticality analyses that are necessary to demonstrate compliance with 10 CFR Part 63 have not yet been completed for event sequences involving a low probability drop and breach of the MCO. The design approach for the MCO is to develop handling designs that, when evaluated with drop sequences, will result in low probability of canister breach such that consequence analyses are not required. When acceptable results from this approach are obtained, the basis for MCO acceptance and disposal will be included in an update of the license application. The MCO is included in this section to provide a description of the analyses that have been completed and to demonstrate the intent of DOE to complete the above analyses and include DOE SNF in MCOs in future licensed operations of the repository. The processes prescribed in 10 CFR 63.22 and 10 CFR 63.46 will be used, as appropriate, to obtain authorization to receive DOE SNF in MCOs once the safety analyses are completed.

The SNF and HLW waste forms to be received, staged, packaged, and emplaced are solid materials, such as metals, ceramics, and glass.

The waste forms to be disposed of are categorized as follows:

- Commercial SNF
- HLW
- DOE SNF
- Naval SNF.

Naval SNF, described in Section 1.5.1.4, is one of the 34 DOE SNF groups described in Section 1.5.1.3. However, for the purposes of SNF inventory, characterization, and analyses, the naval SNF is treated as a separate waste form in the PCSA.

The waste forms described above are received at the repository as canistered. However, a small fraction of commercial SNF is expected to be received as uncanistered in transportation casks. Commercial SNF that is received in dual-purpose canisters (DPCs) or as uncanistered SNF in a transportation cask is placed into a transportation, aging, and disposal (TAD) canister before being placed into an aging overpack for aging or into waste packages for disposal (BSC 2008a).

The geologic repository operations area (GROA) is designed to receive and package canistered and uncanistered commercial SNF, canistered HLW, canistered DOE SNF, uncanistered DOE SNF of commercial origin, and canistered naval SNF, and, after packaging, to emplace those radioactive wastes. The repository inventory is 70,000 MTHM (Nuclear Waste Policy Act of 1982), consisting of approximately 63,000 MTHM of commercial SNF and HLW of commercial origin and approximately 7,000 MTHM of DOE materials (including 65 MTHM of naval SNF). The DOE materials are about one-third SNF and about two-thirds HLW by quantity of MTHM. The project cites the MTHM value of commercial SNF in terms of initial MTHM in fresh fuel assemblies as reported by the utilities (Thorpe 2004, Section 3). For naval SNF and DOE SNF, the analysis is based on the final MTHM quantities of the SNF (DOE 2007, Section 3.2, Footnote b). For DOE HLW, emplacement limits are based on comparing the curie content of a typical DOE HLW canister with the curie content of a typical commercial HLW canister (Knecht et al. 1999, Section 2.2). Table 1.5.1-1 summarizes the allocation of these wastes and represents the basis for facility expected throughputs in Table 1.2.1-1, used for the PCSA, and the emplacement inventory in Table 2.2-12, used for the TSPA. Regarding throughput, the total number of waste packages to be emplaced at the repository, while accepting 70,000 MTHM, is not firmly established because of the variability of the waste stream. Accordingly, there exists a similar variability in the number of waste packages considered as part of the safety analyses. Additional information on variability of the waste stream is in Section 1.3.1.2.5. Analyses that evaluate repository performance that require assumptions regarding thermal and radionuclide decay of the waste stream generally assume initiation of waste receipt in calendar year 2017 (with closure in 2117), although the proposed repository schedule identifies operations beginning in calendar year 2020 (with closure in 2120). Drift by drift analyses prior to the emplacement of waste packages, as described in Section 1.3.1.2.5, will account for any such decay variances.

The waste forms are analyzed differently in the preclosure and the postclosure periods. The preclosure waste form analyses address the physical, thermal, and nuclear properties of the waste forms as they exist through repository closure. Following closure, the postclosure analyses address the same properties as well as the interactions of the waste within the repository as the radioactive components of the waste decay in the repository drifts. The radionuclide inventory is further discussed in Section 2.3.7.4. Waste form in-package chemistry (radiochemistry) is discussed in Section 2.3.7.5, which then factors into the degradation of commercial SNF, DOE SNF, and HLW, discussed in Sections 2.3.7.7 through 2.3.7.9 respectively, and into the dissolved radionuclide concentrations (discussed in Section 2.3.7.10). This section presents the characteristics for each of the four waste forms that have been considered part of the 70,000 MTHM (the current design and analyses are based on emplacement of 70,000 MTHM), including the radionuclide inventories, fuel enrichments, clad material, fuel geometry, canister materials, and other parameters unique to each waste form. This baseline of characteristics for each waste form is then considered in the development of appropriate input data for preclosure and postclosure analyses as discussed in detail in Sections 1.6 through 1.9 and 2.3.5 through 2.3.7, respectively. To the extent that more detailed information on a waste form than is in this section is needed for a specific preclosure or postclosure analysis, that information is presented in the section describing the specific analysis.

Starting with the allocation of wastes noted above, a radionuclide inventory (in grams per waste package) and associated uncertainty distributions have been developed for use in the TSPA in support of the license application. The radionuclide inventory for each waste form: (1) commercial SNF (including mixed-oxide), (2) DOE SNF, (3) HLW (including vitrified plutonium in lanthanide borosilicate glass matrix), and (4) naval SNF represents that inventory of radionuclides that has been identified as being important to dose. The quantity and activity of these radionuclides have been modeled and these data reported for the year of calculation: 2067 for commercial SNF (including 2035 for mixed-oxide), 2030 for DOE SNF and HLW (including 2003 for lanthanide borosilicate glass). The list of radionuclides is given in Table 2.3.7-2. The radionuclide inventory for a representative loaded naval SNF canister at 5 years after shutdown is provided in Section 1.5.1.4.3. The specific dates of the data were selected based on the calculation objectives and are further detailed in Section 2.3.7. The radionuclide inventories for the waste forms are then decayed to the year 2117 for TSPA calculations, which is the assumed beginning of the postclosure period (50 years following the end of the emplacement of the last waste package).

The radionuclide inventory analysis uses the inventory data and calculates the total grams for each of the radionuclides selected for the TSPA calculations. There are 32 radionuclides in the water transport scenarios and a subset of them (25) in the eruptive transport scenario. These radionuclides are included based on either the 10,000-year or the million-year screening analyses; however, they are tracked through the entire TSPA calculation independent of which time screen was the basis for their inclusion. Based on the configuration, quantity, and capacity of waste package types, the radionuclide inventory analysis provides the TSPA with the nominal grams per waste package of these radionuclides for each type of waste form. The total number of waste packages for each waste type is also provided as discussed in Section 2.3.7.4.1.2. In addition, the analysis provides uncertainty multipliers for inventory of radionuclides for each waste type.

Uncertainties applied to data considered in TSPA compliance calculations were developed from uncertainties commonly used in the nuclear industry. These uncertainties pertain to errors in records, errors in measured versus calculated data, and uncertainties related to the age and activity levels (burnup) of the waste and age-at-arrival using arrival scenarios. One or more of these uncertainty categories was applied to each type of waste form. Use of known error or uncertainty factors, calculation of averages, and developing ratios as factors to establish ranges of values are the methods used in the inventory uncertainty analysis. Section 2.3.7.4.2 has a detailed presentation of the treatment of inventory uncertainty for each type of waste form.

Postclosure criticality analyses are performed to demonstrate that the initial emplaced configuration of the waste form remains subcritical even under flooded conditions. This demonstration is performed for commercial SNF through the use of criticality loading curves. Loading curves, which are functions of burnup and enrichment, are the loci of values delineating the region of acceptable burnup/enrichment combinations for postclosure criticality control. The loading curves for canisters are based on a conservative (with respect to criticality) design basis postclosure configuration. In applying this methodology, the loading curve is generated once and the utility calculated burnup values of all assemblies being considered for loading into the TAD canister are compared directly against this loading curve. Assemblies having burnup values in the unacceptable range must be loaded into canisters with additional reactivity control mechanisms (e.g., disposal control rod assemblies). The key factors for consideration of design-basis configurations are geometry, materials, and inherent neutron spectrum. The process for developing the criticality loading curves and establishing design basis configurations is discussed in Section 2.2.1.4.1.1.2.1.

The information in this section comes primarily from databases maintained for the DOE. The commercial SNF data come from information submitted to and tabulated by the DOE Energy Information Administration. DOE SNF data come from documentation that references the DOE SNF database. The naval SNF data are supplied by the Naval Nuclear Propulsion Program, with more detail presented in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document that is part of the license application submitted under separate cover. Information about HLW is obtained from technical reports prepared by the sites generating HLW.

After receipt of a license to receive and possess SNF and HLW at the repository, information regarding radioactive materials at the waste generator sites will be provided by the records accompanying each shipment received.

The standard contract for disposal of commercial SNF or commercial HLW codified in 10 CFR Part 961 establishes the contractual terms and conditions under which the DOE will provide nuclear waste disposal services to the owners and generators of the commercial SNF and commercial HLW (the purchasers). Under the contract, DOE will take title to, transport, and dispose of commercial SNF and commercial HLW delivered to DOE at the purchaser’s site. Contracts with the individual purchasers, based on the standard contract, will be revised to permit the use of a TAD canister system when delivering commercial SNF to DOE.

Among a large number of items, the contract establishes the responsibilities of the parties, an order of waste acceptance, a scheduling process for waste acceptance, and waste acceptance criteria. Responsibilities assigned to the purchasers include (1) providing data on actual and projected discharges of commercial SNF from commercial nuclear power reactors; (2) scheduling deliveries of commercial SNF and commercial HLW to DOE; (3) providing a detailed description of the commercial SNF and commercial HLW scheduled for delivery; (4) canistering the commercial SNF and commercial HLW in U.S. Nuclear Regulatory Commission (NRC) certified transportation systems (or loading uncanistered commercial SNF into NRC-certified transportation casks) and preparing the loaded casks for transportation by DOE to the repository; and (5) transferring the title of the delivered commercial SNF or commercial HLW to the DOE. Responsibilities assigned to DOE include (1) accepting title to the commercial SNF and commercial HLW at the purchaser’s facility, (2) providing NRC-certified transportation systems suitable for use at the purchasers’ sites,

(3) transporting the commercial SNF and commercial HLW to the repository, and (4) disposing of the commercial SNF and commercial HLW.

Under the contract, DOE must accept all of the purchaser's commercial SNF. The contract identifies "standard fuel" and "other than standard fuel." Criteria for standard fuel include maximum dimensions, fuel condition, fuel type, and minimum cooling time. Standard fuel will be accepted on the schedule agreed to between the purchaser and DOE. For other than standard fuel, which includes nonstandard and failed fuel (later referred to as damage fuel), DOE may adjust the purchaser's proposed schedule for delivery if it is technically infeasible to accept that fuel on the normal schedule. Presently, multielement canisters, such as DPCs or transportable storage casks, are not acceptable under the contracts with the individual purchasers.

The purchaser is also responsible to accurately classify the commercial SNF and commercial HLW (Appendix E of the contract (10 CFR Part 961)) and to provide a detailed description of the commercial SNF and commercial HLW (in accordance with the provisions of Appendix F of the contract (10 CFR Part 961)) to be delivered. Because the physical characteristics of commercial SNF are well known, no additional characterization is required.

The contract also requires the purchasers to arrange for, and provide, all preparation, packaging, required inspections, and loading activities necessary to deliver commercial SNF or commercial HLW to DOE for transportation to Yucca Mountain. These preparatory activities by the purchaser will be conducted in accordance with all applicable laws and regulations. In addition, the purchaser's facility is licensed by the NRC under 10 CFR Part 50 (the GE Morris site is licensed under 10 CFR Part 70), and operations, such as canister and cask loading, are conducted under an NRC-approved quality assurance plan. Process records for the nuclear fuel are also maintained under the NRC-approved quality assurance plan. The NRC also conducts periodic surveillance and quality assurance audits related to the preparatory activities at the purchaser's site.

DOE has no regulatory authority at the purchasers' sites; however, as specified in the contract, DOE may observe, or designate a representative to observe, the preparatory activities. DOE does not currently intend to witness canister and cask loading activities on a routine basis. DOE may, as appropriate, audit the purchaser's implementation of its quality assurance plan as it affects the preparatory activities.

Canister and cask loading, as well as the description of the commercial SNF and commercial HLW in each shipping lot, is subject to verification by DOE prior to acceptance for transportation to the repository. The DOE plan for verifying delivery of commercial SNF at purchaser facilities is described in its SNF verification plan (DOE 1997).

The memorandum of agreement for acceptance of SNF and HLW between DOE Environmental Management and the Office of Civilian Radioactive Waste Management (Rispoli 2007) establishes policy for cooperation on current and future activities relating to acceptance of Environmental Management SNF and HLW. The memorandum of agreement applies only to SNF and HLW that are the responsibility of, or planned for the transfer of title to, Environmental Management. The Office of Civilian Radioactive Waste Management has executed a separate memorandum of agreement for disposal of naval SNF with the Naval Nuclear Propulsion Program (Carlson 2000). The Office of Civilian Radioactive Waste Management will not accept SNF or HLW from any federal entity unless it enters into a suitable agreement with the Office of Civilian Radioactive Waste Management.

Waste acceptance requirements applicable to Environmental Management SNF and HLW have been established. These technical requirements govern acceptance of Environmental Management SNF and HLW that derive from the agreements established in the memorandum of agreement. Commercial-origin SNF for which title has been transferred to Environmental Management and for which Environmental Management has accepted responsibility shall be accepted for transportation at Environmental Management facilities and disposed of by the Office of Civilian Radioactive Waste Management in accordance with the terms and conditions of the standard contract (10 CFR Part 961).

As part of the memorandum of agreement between the Office of Civilian Radioactive Waste Management and Environmental Management (Rispoli 2007), Environmental Management agrees to implement a quality assurance program that satisfies the applicable requirements of the Office of Civilian Radioactive Waste Management quality assurance requirements document described in Section 5.1. The quality assurance program requirements apply to federal waste custodians, the National Spent Nuclear Fuel Program, and principal contractors performing the following:

1. Activities related to a HLW waste form, from development through qualification, production, and acceptance

2. Activities related to the characterization of DOE SNF and its conditioning, treatment, and/or canisterization, through acceptance by the Office of Civilian Radioactive Waste Management.

The Office of Civilian Radioactive Waste Management and Environmental Management cooperate in conducting annual oversight audits of federal waste custodian, National Spent Nuclear Fuel Program, and principal contractor activities. In addition, Environmental Management implements the applicable requirements of 10 CFR Part 21 for components it, or its principal contractors, provided to be used or accepted at the repository.

Environmental Management is responsible for providing the Office of Civilian Radioactive Waste Management with characterization data, suitable for its intended use in accordance with the applicable quality assurance requirements, for Environmental Management SNF and HLW. The characterization data to be provided are those needed to support licensing of the repository by the NRC and certification by the NRC of transportation cask systems that will be used to transport canistered Environmental Management SNF, uncanistered Environmental Management SNF of commercial origin, and canistered HLW. Environmental Management is responsible to design, fabricate, and load SNF and HLW canisters. Environmental Management is also responsible to load the SNF and HLW canisters into NRC-certified transportation casks at Environmental Management facilities. The Office of Civilian Radioactive Waste Management is responsible to design, obtain NRC certificates of compliance for use, and fabricate the transportation systems for Environmental Management SNF and HLW and to conduct all transportation operations after acceptance at Environmental Management facilities.

Environmental Management is responsible for all required preparations, tests, and inspections during loading operations prior to delivery to the Office of Civilian Radioactive Waste Management for shipment to the repository. At its discretion, the Office of Civilian Radioactive Waste Management may observe any loading activities. Environmental Management is responsible to implement and maintain a quality assurance program that meets the applicable requirements of the the Office of Civilian Radioactive Waste Management quality assurance requirements document and to provide conforming documentation that demonstrates and certifies that SNF and HLW that it transfers to the Office of Civilian Radioactive Waste Management comply with all applicable waste acceptance criteria. In addition, Environmental Management is responsible to provide, at the time of waste acceptance, records packages for use by the Office of Civilian Radioactive Waste Management to verify the description and characteristics of the Environmental Management SNF and HLW being delivered.

The memorandum of agreement between the Naval Nuclear Propulsion Program and the Office of Civilian Radioactive Waste Management (Carlson 2000) establishes the terms and conditions under which the Office of Civilian Radioactive Waste Management will make available disposal services to the Naval Nuclear Propulsion Program for naval SNF. Naval SNF is defined as SNF that is generated and managed by the Naval Nuclear Propulsion Program. The Memorandum of Agreement specifies the responsibilities of the Naval Nuclear Propulsion Program and the Office of Civilian Radioactive Waste Management relative to transportation, storage (if needed), and disposal of naval SNF.

Waste acceptance requirements applicable to naval SNF have been developed. These technical requirements govern acceptance of naval SNF that derive from the Memorandum of Agreement (Carlson 2000).

Quality assurance for activities associated with disposal of naval SNF shall be accomplished as specified in Appendix E, “Coordination and Implementation of Quality Assurance Activities Associated with Naval Spent Nuclear Fuel,” of the memorandum of agreement (Carlson 2000). The Naval Nuclear Propulsion Program quality assurance program applies to all aspects of design, operation, construction, and maintenance of naval nuclear propulsion plants, including activities and components that support or are integral to emplacement and isolation of naval SNF in the repository. The memorandum of agreement specifies that, in accordance with the requirements of Executive Order 12344, the Naval Nuclear Propulsion Program quality assurance program shall be defined and administered solely by the Naval Nuclear Propulsion Program and that the Naval Nuclear Propulsion Program is responsible for conducting all oversight of the Naval Nuclear Propulsion Program activities related to the acceptance of naval SNF. Under the memorandum of agreement, the Office of Civilian Radioactive Waste Management is responsible for reviewing the Naval Nuclear Propulsion Program quality assurance practices regarding naval SNF and for determining the sufficiency of those practices. The memorandum of agreement provides for opportunities for the Office of Civilian Radioactive Waste Management to participate in observation activities and annual review of the quality assurance program activities.

The Naval Nuclear Propulsion Program is responsible for characterization of the naval SNF. The Naval Nuclear Propulsion Program is responsible to document that naval SNF is in compliance with all the Office of Civilian Radioactive Waste Management waste acceptance criteria. The Office of Civilian Radioactive Waste Management is responsible to define the contents of data records packages and develop a data needs schedule, and, the Naval Nuclear Propulsion Program is responsible to provide the naval SNF data, qualified for their intended use and in accordance with the data needs schedule.

The Naval Nuclear Propulsion Program is also responsible for the design and fabrication of the naval SNF canisters and transportation casks, including NRC certification of transportation casks. The Naval Nuclear Propulsion Program is responsible for loading naval SNF into naval SNF canisters and loading naval SNF canisters into NRC-certified naval M-290 transportation casks. The Naval Nuclear Propulsion Program is also responsible for all required preparations, tests, and inspections. At its discretion, the Office of Civilian Radioactive Waste Management may observe any loading activities. In addition, the Naval Nuclear Propulsion Program is responsible for transportation of naval SNF to the GROA where it is accepted by the Office of Civilian Radioactive Waste Management.

1.5.1.1 Commercial SNF

[NUREG-1804, Section 2.1.1.2.3: AC 4(1), AC 5(2), AC 6(1); Section 2.1.1.6.3: AC 2(1); Section 2.1.1.7.3.1: AC 1(1), (3), (4), (5), (8); Section 2.1.1.7.3.2: AC 1(1); Section 2.1.1.7.3.3(III): AC 1(1), (2), (3), (4), (5), (6), (8), (9), (10)]

The majority of commercial SNF assemblies will be shipped to the repository in TAD canisters. The TAD canisters are transferred directly into a waste package for disposal or into an aging overpack for aging. Commercial SNF assemblies that cannot be placed into TAD canisters at utility sites can be handled and shipped to the repository in transportation casks certified by the NRC or in DPCs. Commercial SNF assemblies shipped in a cask or DPC, once received at the repository, may be either loaded into an aging overpack and sent to the aging pad or opened and transferred into a TAD canister before being placed into a waste package. Failed-fuel assemblies, consolidated fuel rods, and nonfuel assembly hardware and structural parts of assemblies resulting from fuel assembly consolidation are packaged at the SNF sites in failed-fuel cans (later referred to as damaged fuel cans) or TAD canisters prior to shipment to the repository. Failed fuel is further discussed in Section 1.5.1.1.1.1. In each year of operation, the repository shall be capable of accepting, transporting, and disposing of commercial SNF where at least 90% is received in TAD canisters and no more than 10% is received as uncanistered assemblies.

Commercial SNF includes irradiated fuel discharged from PWRs and boiling water reactors (BWRs). The fuel used in these reactors consists of uranium dioxide pellets encased in zirconium alloy (Zircaloy) or stainless steel rods. The fuel assemblies vary in physical configuration, depending upon reactor type and manufacturer.

Commercial SNF assemblies are categorized by physical configuration (Reich et al. 1991, p. 4) into 22 classes: 16 PWR and 6 BWR fuel assembly classes. Commercial SNF data are collected by the Energy Information Administration for the Office of Civilian Radioactive Waste Management. Tables 1.5.1-2 and 1.5.1-3 present the assembly class, array size, fuel manufacturer, assembly version, assembly type code, length, width, and cladding material of commercial PWR SNF and commercial BWR SNF, respectively. Physical dimensions are those of unirradiated assemblies. Failed clad fractions in pre- and postclosure are identified in Sections 1.8.1.3.2 and 2.3.7.6.

Commercial SNF shall meet the acceptance requirements specified in 10 CFR Part 961, as modified by individual purchaser contracts. Commercial SNF may include uranium dioxide SNF and mixed oxide SNF from commercial power reactors and SNF from privately owned commercial research reactors (DOE 2008b, Section 4.1).

However, commercial mixed oxide assemblies (approximately 1,700), though evaluated for similarity to commercial SNF, cannot be compared to the existing analyses until the characteristics are known. Accordingly, the discussion of commercial SNF presented in this section will not include the mixed oxide assemblies.

1.5.1.1.1 Physical Characteristics of Commercial SNF and Canisters

1.5.1.1.1.1 Physical Characteristics of Commercial SNF

Assemblies of commercial SNF are categorized by physical configuration into assembly classes as noted in Tables 1.5.1-2 and 1.5.1-3. Physical configuration is determined by the physical dimensions of the fuel assemblies, the design of the lower and upper end fittings, and the design of the control elements. Within an assembly class, assembly types are of a similar size. There are 134 individual fuel assembly types in these classes.

Table 1.5.1-4 presents the manufacturer, initial uranium load, enrichment, and burnup characteristics of commercial SNF assembly types. A summary of initial uranium load, initial enrichment, and discharge burnup of commercial SNF inventory as of December 31, 2002, is presented in Table 1.5.1-5.

A source-term sensitivity study shows that, for PWR and BWR assemblies, the results are not sensitive to the selection of a particular fuel assembly to the resulting source term. Given these results, the Babcock and Wilcox 15 × 15 Mark B with 475 kg uranium loading and GE 2/3 8 × 8 with 200 kg uranium loading were selected as source term generation representative assemblies for PWRs and BWRs, respectively. These representative assemblies are discussed below.

Pressurized Water Reactor—For illustrative purposes, the arrangement and nomenclature for a typical PWR fuel assembly are presented in Figure 1.5.1-1. The Babcock & Wilcox 15 × 15 Mark B PWR fuel assembly is the representative PWR assembly. The characteristics of the average, and maximum burnup PWR assemblies used for shielding and radiological source-term generation follow:

- **Average PWR Assembly**—4.0% ^{235}U, initial uranium loading of 475 kg, 48 GWd/MTU, 25 years cooling time

- **Maximum PWR Assembly**—5.0% ^{235}U, initial uranium loading of 475 kg, 80 GWd/MTU, 5 years cooling time.

The average PWR and BWR assemblies represent the averaged characteristics of commercial SNF over the entire emplaced inventory and are used to develop the radionuclide inventory for use in the TSPA as discussed in Section 2.3.7.4.

The maximum PWR and BWR assemblies represent the combination of fuel parameters (initial enrichment, fuel loading, and fuel burnup) that produces a maximum source term for radiological and shielding analysis and represents the bounding values. Table 1.5.1-4 shows that no fuel assembly exists with all of these parameters simultaneously at the upper-limit conditions.

Figure 1.5.1-2 shows the evolution in time of ^{235}U enrichment and discharge burnup for PWR fuel assemblies. Both historical data and utility projections for the next five cycles are included and support the selection of the PWR bounding characteristics.

Approximately 1,700 PWR commercial SNF assemblies to be received at the repository will contain mixed oxide fuel. Thirty-four metric tons of weapons-grade plutonium will be disposed of by converting it to plutonium oxide and blending it with uranium dioxide from depleted uranium. The blend will vary, depending on the core design requirements of the reactor at which the fuel will be irradiated, but will contain no more than 6 wt % plutonium. The fuel assemblies will utilize a design of 264 fuel rods in a square 17 by 17 array and will be irradiated to an average burnup of 41 GWd/MTHM or maximum burnup of 45 GWd/MTHM (Duke Cogema Stone & Webster 2005, pp. 2, 4, 5, and 16). These assemblies will be stored in spent fuel pools for cooling for a minimum of 5 years following irradiation similar to commercial SNF without mixed oxide fuel, prior to shipment to the repository in accordance with 10 CFR 961, Appendix E, item B3. Mixed oxide SNF has been evaluated for its likely bounding contribution to the TSPA results. Analyses associated with preclosure handling and dose analyses and criticality analyses and detailed analysis of postclosure performance have not been performed. However, this waste form is identified in order to demonstrate a future intent to include it in licensed operations.

Boiling Water Reactor—For illustrative purposes, the arrangement and nomenclature for a typical BWR fuel assembly are presented in Figure 1.5.1-3. The General Electric 2/3 8 × 8 BWR fuel assembly is the representative BWR assembly. The characteristics of the average and bounding BWR assemblies used for source-term generation follow:

- **Average BWR Assembly**—3.5% ^{235}U, initial uranium loading of 200 kg, 40 GWd/MTU, 25 years cooling time.
- **Maximum BWR Assembly**—5.0% ^{235}U, initial uranium loading of 200 kg, 75 GWd/MTU, 5 years old. As noted in Section 1.5.1.1.4, the PWR assembly inventory and source intensity bound those of a BWR assembly.

Figure 1.5.1-4 shows the evolution in time of ^{235}U enrichment and discharge burnup for BWR fuel assemblies. Both historical data and utility projection for the next five cycles are included and support the selection of the BWR maximum (bounding) characteristics. In both the PWR and BWR cases noted above, historical utility data were used to extrapolate a bounding burnup used for the preclosure source-term development. Enrichment is based on what would be required to achieve the bounding burnup.

Chemical Composition—Commercial SNF consists of the uranium dioxide fuel itself, including fission products, cladding, and hardware. The vast majority of PWR and BWR fuels consist of uranium dioxide fuel pellets with a zirconium alloy cladding. Some assemblies, however, are clad in stainless steel.

Crud Deposits—The activity of the crud on the surface of the BWR and PWR assemblies at time of discharge is determined by multiplying the calculated assembly surface area exposed to coolant by the ^{60}Co and other corrosion product activity per unit area of surface. Crud is considered in design and PCSA; however, because the principal constituents are ^{60}Co with a half-life of 5.2 years and ^{55}Fe with a half-life of 2.7 years, crud is not modeled in the TSPA.

After decaying for 5 years, the principal radionuclide species in the crud are ^{55}Fe and ^{60}Co. Initial crud activities for commercial SNF at the time of discharge from a reactor are presented in Table 1.5.1-6. The values are based on analyses of measured crud activity data in NUREG-1567 (NRC 2000a) and NRC Interim Staff Guidance–5 (NRC 2003a).

Commercial SNF assemblies have the following values for surface area per fuel assembly:

- PWR = 449,003 (cm^2/fuel assembly)
- BWR = 168,148 (cm^2/fuel assembly).

These surface areas are bounding values, based on assemblies with the largest surface areas, a South Texas PWR assembly, and an Advanced Nuclear Fuels Corporation 9 × 9 JP–4,5 BWR assembly. The analysis of crud is further discussed in Section 1.8.1.3.1.

Extent of Failure—The repository uses the definitions and failed fuel classes of 10 CFR 961, Appendix E, when evaluating the handling options for commercial SNF shipped to the repository as assemblies in a cask or in a DPC. These failed fuel class assignments are made prior to shipment to the repository. The handling of the SNF assemblies depends upon the existence of adequate containment of the fuel so that thermal and criticality requirements can be met and to minimize contamination of the handling pool. SNF assemblies that may have experienced radioactive leakage during reactor operation (failed fuel—Class F-2) are handled as part of normal operations and may be transferred directly to the TAD canisters. Assemblies encapsulated by purchaser prior to classification in accordance with 10 CFR Part 961 shall be classified as failed fuel—Class F-3. Purchaser shall advise the DOE of the reason for the prior encapsulation of assemblies in sufficient detail so that the DOE may plan for appropriate subsequent handling. Assemblies shall be visually inspected by purchaser for evidence of structural deformity or damage to cladding or spacers which may require special handling. Assemblies that (1) are structurally deformed or have damaged cladding to the extent that special handling may be required or (2) for any reason cannot be handled with normal fuel handling equipment shall be classified as failed fuel—Class F-1.

The extent of fuel failure is considered in various other analyses. For example, in PCSA analyses, a damage ratio (equal to the fuel rod breakage fraction of 1%) is used in characterizing commercial SNF source terms for normal operation releases as discussed in further detail in Section 1.8.1.3.2. In postclosure, no credit is taken for commercial SNF cladding capability in the TSPA calculations. The cladding behavior in the repository is further discussed in Section 2.3.7.6.

1.5.1.1.1.2 Physical Characteristics of Commercial SNF Canisters

Commercial SNF may be stored and transported in DPCs and TAD canisters as described below.

1.5.1.1.1.2.1 Commercial SNF Canister Description

1.5.1.1.1.2.1.1 Damaged-Fuel Cans

While 10 CFR Part 961 has classified failed fuel as noted above, Revision 2 of Interim Staff Guidance–1 (NRC 2007) has subsequently expanded this classification of SNF as either (1) damaged, (2) undamaged, or (3) intact. Any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system related functions is classified as damaged SNF. This must be placed into a damaged fuel can prior to placement in a TAD canister prior to being loaded into a transportation cask, unless additional analyses can show that neither reconfiguration of the fuel assembly nor gross failure of the cladding will occur. The damaged fuel can is functionally the same as what was previously termed a failed-fuel can. These cans are metal enclosures sized to contain a single fuel assembly. The damaged fuel can must be individually removable from the cask using normal fuel handling methods (crane and grapple). The damaged fuel can may use a mesh screen to achieve gross particulate confinement but also allow water drainage during wet loading operations. The purpose of the damaged fuel can is to confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask to facilitate meeting criticality, shielding, and thermal requirements and permit normal handling and retrieval from the cask (NRC 2007).

A variety of damaged fuel cans exist, each specific to a particular fuel assembly design and transportation cask. Damaged fuel cans are approved as part of the transportation cask system they are used in.

Notification from the generator is required prior to shipping damaged fuel cans, and plans for handling damaged fuel cans will be developed on an individual basis. The intent is to evaluate the damaged fuel can and its contents prior to allowing shipment to the repository. If the encapsulated fuel can is acceptable, shipment will be allowed, and the damaged fuel can will be loaded into a TAD canister. DOE will propose an alternate shipping schedule that will allow time for the resolution of the issues that led to the determination of technical infeasibility.

1.5.1.1.1.2.1.2 Dual-Purpose Canisters

DPCs are currently used by several utilities to store and potentially ship commercial SNF. Currently licensed DPCs have not been shown to be suitable for disposal purposes. However, although not currently acceptable under the provisions of 10 CFR Part 961, the DOE may choose to receive DPCs at the repository and repackage the commercial SNF into a TAD canister for disposal after the execution of mutually agreeable amendments to the utilities disposal contract. Further, the DPC may be placed on the aging pad, within a properly designed overpack, while it awaits transfer of its contents or to cool the SNF. Accordingly, a brief description of DPCs is included as a discussion of canisters that may be received at the repository. Prior to the use of any DPC system (including associated overpacks) at the repository, analyses will be performed to demonstrate compliance with the Yucca Mountain repository specific criteria nuclear safety design bases. Analyses show that the local conditions at Yucca Mountain (e.g., temperature, rainfall, and tornado winds) are within those values specified in many certified DPC systems general licenses. Additional analyses, associated with structural analyses and criticality, to satisfy repository-specific PCSAs must be performed, once receipt of a DPC type is planned. To show likely capability of passing such structural evaluations, evaluations of generic canisters have been performed. Should a DPC system be shown to fall within this analyzed envelope, that analysis will be used for demonstration of the structural acceptability. Section 1.7 summarizes the methodology and provides results of such analyses.

The process of opening DPCs and loading the assemblies into TAD canisters at the repository is performed in the Wet Handling Facility (WHF). This is described in more detail in Section 1.2.5. DPC systems are licensed for storage at utility sites under 10 CFR Part 72 and for shipment under 10 CFR Part 71.

1.5.1.1.1.2.1.3 Transportation, Aging, and Disposal Canisters

In preclosure, the TAD canister forms the analytical basis for evaluations of the safety functions. The safety functions for the TAD canister are (1) providing containment, and (2) reducing the frequency of criticality. To ensure that these safety functions are maintained, the controlling parameters and values have been incorporated into the handling facilities of the TAD canister, which are the Canister Receipt and Closure Facility (CRCF), Receipt Facility (RF), and the WHF.

The TAD canister is loaded with commercial SNF and sealed at utilities (e.g., reactors) or the repository. The loaded TAD canister may be used for storage for a period of time at utilities; for this purpose, it must be approved for a storage system certified under 10 CFR Part 72. The loaded TAD canister may be delivered to DOE for transportation to the GROA, for which it would be listed as approved contents for packaging, including the transportation cask, certified under 10 CFR Part 71. At the GROA, a loaded TAD canister may also be handled using a shielded transfer cask or aged in an aging overpack and shall be disposed of in a waste package. All three of these functions performed at the repository will be covered by the repository license granted under 10 CFR Part 63 (DOE 2008c, Section 1.2). Procedures, developed in accordance with the *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008c), will control the loading of TAD canisters with uncanistered commercial SNF of 80 GWd/MTU or less. It is planned that this limit will be part of the certificate of compliance issued by the NRC when the TAD canister is approved and will provide control of loading operations both at the utility site and GROA.

For illustrative purposes, a conceptual depiction of a TAD canister is shown in Figure 1.5.1-5.

A TAD canister performance specification was developed for selected system components of the TAD canister-based system. The TAD canister discussions presented here are based on this TAD canister performance specification (DOE 2008c). The TAD canister design may change over time and the performance specification will be revised, if appropriate, to reflect this. The TAD canister, in conjunction with specialized overpacks, will accomplish a number of functions in the management and disposal of SNF. Some of these functions will be accomplished at the utilities where commercial SNF is stored, and some will be performed within the Office of Civilian Radioactive Waste Management transportation and disposal system. The performance specification contains only those requirements unique to applications within the DOE system. DOE recognizes that TAD canisters may have to perform similar functions at the utilities. Requirements to meet reactor functions, such as onsite dry storage, handling, and loading for transportation, are expected to be similar to commercially available canister-based systems (DOE 2008c, Section 1.1).

Additional TAD canister configurations may be developed for future applications to accommodate specific SNF or utility site characteristics. Additional analysis will be performed to account for these configuration differences over those provided herein.

When necessary, the following TAD canister system components shall work in conjunction with the TAD canister to meet the performance specification:

- Transportation cask
- Aging overpack
- Ancillary equipment
- Shielded transfer cask
- Site transporter.

The TAD canister will be part of an NRC-certified system, approved for confining commercial SNF during storage, transportation, aging, and disposal. However, analyses described in Sections 1.6 through 1.8 have resulted in classifying the TAD canister, the transportation cask, and the aging overpack as important to safety (ITS) as identified in the tables of Section 1.9. The TAD canister includes a canister shell, lid(s), and other required components (e.g., basket for holding fuel assemblies, thermal shunts, neutron absorbers) needed to perform its functions. In addition, the TAD canister shall have the general, containment, and material characteristics that follow.

1.5.1.1.1.2.1.4 TAD Canister General Characteristics

The TAD canisters have the following general characteristics:

1. The TAD canister shall be a right circular cylinder with a diameter of 66.5 in. (+0.0 in./–0.5 in.). The TAD canister height shall not be less than 186.0 in. and not greater than 212.0 in. including the lifting feature considering all relevant factors (e.g., tolerance stack-up, thermal expansion, internal pressure).

    a. For a TAD canister with a height less than the maximum, a TAD waste package spacer meeting requirements as stated in the performance specification shall be included. If required, the TAD waste package spacer shall have a diameter of 66.5 in. (+0.0 in./–0.5 in.) and length such that the combined height of the TAD waste package spacer and TAD canister shall be 212.0 in. (+0.0 in./–0.5 in.) considering all relevant factors (e.g., tolerance stack-up, thermal expansion, internal pressure).

    b. If required, the TAD waste package spacer shall be placed in a waste package prior to loading of the TAD canister for disposal. The TAD waste package spacer function is to restrict axial motion of the TAD canister within the waste package after emplacement.

2. The TAD canister loaded weight shall be consistent with the height determined in accordance with the above item. The combined weight of the loaded TAD canister and TAD waste package spacer shall not exceed 54.25 tons.

3. The capacity of the TAD canister shall be either 21-PWR spent fuel assemblies or 44-BWR spent fuel assemblies.

4. The loaded and closed TAD canister shall be capable of being reopened while submerged in a borated or an unborated pool.

5. A TAD canister for PWR assemblies shall be limited to accepting commercial SNF with characteristics less than 5% initial enrichment, less than 80 GWd/MTU burnup, and no less than 5 years out-of-reactor cooling time.

6. A TAD canister for BWR assemblies shall be limited to accepting commercial SNF with characteristics less than 5% initial enrichment, less than 75 GWd/MTU burnup, and no less than 5 years out-of-reactor cooling time.

7. A TAD canister shall be capable of being loaded with commercial SNF from one or more facilities that are licensed by the NRC and hold one or more contracts with the DOE for disposal of commercial SNF.

8. All external edges of the TAD canister shall have a minimum radius of curvature of 0.25 in.

9. To the extent practicable, projections or protuberances from reasonably smooth adjacent surfaces shall be avoided or smoothly blended into the adjacent smooth surfaces.

10. The TAD canister shall be designed to store vendor-defined design basis commercial SNF at utilities in accordance with 10 CFR Part 72 in either a horizontal or vertical orientation.

11. A TAD canister shall be designed to transport vendor-defined design basis commercial SNF to the GROA in a horizontal configuration.

12. A TAD canister shall be designed to dispose of vendor-defined design basis commercial SNF in a waste package in a horizontal configuration.

13. A TAD canister shall be designed to be handled at the GROA loaded with vendor-defined design basis commercial SNF in a vertical configuration.

14. A TAD canister shall be designed to age vendor-defined design basis commercial SNF in a vertical configuration.

15. At the time of delivery to the repository, a loaded TAD canister shall have a remaining service lifetime for aging of 50 years without maintenance.

16. The service lifetime environmental conditions shall be site appropriate for the period of deployment at reactors. Yucca Mountain environmental conditions apply for repository aging service.

17. TAD waste package spacer shall be constructed of materials specified in Section 1.5.1.1.1.2.7.

18. TAD waste package spacer shall be a right circular cylinder, either solid or hollow with sides and ends formed from plates at least 2 in. thick.

19. The TAD waste package spacer shall have an average mass density equal to or greater than that of the loaded TAD canister.

20. The TAD waste package spacer shall include four threaded holes in its top for the purpose of attaching temporary rigging, meeting the requirements of NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* (NRC 1980), to be used when inserting the TAD waste package spacer into an otherwise empty waste package.

1.5.1.1.1.2.2 TAD Canister Operational Processes

Operational processes will be developed to ensure the following TAD canister requirements are met:

1. The TAD canister lid shall be designed for handling under water with the TAD canister in a vertical orientation.

2. The TAD canister body and lid shall have features to center and seat the lid during submerged installation. The maximum off-center value is 0.5 in.

3. A feature for lifting a vertically oriented, loaded, and closed TAD canister from the lid shall be provided. The lifting feature may be integral with the lid or mechanically attached. The lifting feature shall be in place and ready for service prior to transport to the repository.

4. An open, empty, and vertically oriented TAD canister shall have integral lifting feature(s) provided to allow lifting by an overhead handling system.

5. The TAD canister shall be designed with features such that draining, drying, and backfill operations take advantage of as low as reasonably achievable principles, as detailed in Section 1.10.

1.5.1.1.1.2.2.1 TAD Canister Thermal Controls

TAD canister thermal controls shall be developed to ensure the thermal design criteria and design bases discussed in Section 1.5.1.1.1.2.5.3 are met.

1.5.1.1.1.2.2.2 TAD Canister Criticality Controls

In addition to the conformance to 10 CFR Part 63 for the overall TAD system as stated in Section 1.5.1.1.1.2.1.3, there are no specific preclosure criticality safety requirements beyond those of 10 CFR 71, Subpart E, Paragraph 55(b). Postclosure criticality control shall be maintained by employing either the items in (1) or the analysis in (2), as follows:

1. Include the following features in the TAD canister internals:

   a. Neutron absorber plates or tubes made from borated stainless steel produced by powder metallurgy and meeting ASTM A 887-89, *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application*, Grade “A” alloys.

   b. Minimum thickness of neutron absorber plates shall be 0.4375 in. Maximum and nominal thickness may be based on structural requirements. Multiple plates may be used if corrosion assumptions (250 nm/yr) are taken into account for all surfaces such that 6 mm remains after 10,000 years. For the license application, postclosure performance has been analyzed based on the use of single neutron absorber plates. If a cask vendor were to design a TAD canister using multiple neutron absorber plates, similar analyses based on the use of multiple neutron absorber plates would need to be performed.

   c. The neutron absorber plate shall have a boron content of 1.1 wt % to 1.2 wt %, a range that falls within the specification for Stainless Steel 304B4 (UNS S30464) as described in ASTM A 887-89, *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application*.

   d. Neutron absorber plates or tubes shall extend along the full length of the active fuel region inclusive of any axial shifting of the assemblies within the TAD canister.

   e. Neutron absorber plates or tubes must cover all four longitudinal sides of each fuel assembly.

   f. TAD canister designs for PWR fuel assemblies shall accommodate assemblies loaded with a disposal control rod assembly. A disposal control rod assembly is one potential method for acceptance of PWR commercial SNF with characteristics outside limits set in the postclosure criticality loading curves. Current postclosure criticality loading curves are provided in Attachment B of the performance specification (DOE 2008c). Updated postclosure criticality loading curves are provided in Figures 2.2-7 and 2.2-8 for the PWR and BWR TAD canisters, respectively.

2. Perform analyses of TAD canister-based systems to ensure the maximum calculated effective neutron multiplication factor (k_{eff}) for a TAD canister containing the most reactive commercial SNF for which the design is approved shall not exceed the critical limit for four postclosure archetypical proxy configurations.

The implementation of the TAD canister performance specification in the analyses of postclosure criticality is discussed in Section 2.2.1.4.1.1.2.

1.5.1.1.1.2.3 Structures, Systems, and Components Important to Safety and Important to Waste Isolation

Section 1.7.2.3 describes the determination of passive structure, system, or component reliability and discusses several types of failures for passive SSCs in the PCSA, including the structural challenge causing loss of containment (breach or fire) of a waste form container (e.g., TAD canister).

The TAD canister performs safety functions during preclosure handling, and the associated controlling parameters and values for operations can be found in Table 1.5.1-7.

Commercial SNF, as well as the TAD canister and canister internals (neutron absorbers) are classified as important to waste isolation (ITWI) as described in Table 1.9-8. The commercial SNF cladding is classified as non-ITWI. Derived internal constraints that control the TAD canister analysis basis for TSPA are identified in Table 1.5.1-8.

Section 1.7.2.3 describes the determination of passive structure, system, or component reliability and discusses several types of failures for passive SSCs in the PCSA, including the structural challenge causing loss of containment (breach or fire) of a waste form container (e.g., DPC).

The DPC design description is discussed in Section 1.5.1.1.1.2.1.2. The DPC performs ITS functions during preclosure handling, and the associated controlling parameters and values can be found in Table 1.5.1-9. The evaluation of DPCs has been performed generically, and the criteria associated with the structural envelope evaluations are provided.

The DPC has no postclosure functions, and it is classified as not ITWI.

1.5.1.1.1.2.4 Procedural Safety Controls to Prevent Event Sequences and Mitigate Their Effects

Characteristics of commercial SNF shipped to the repository are within the following parameters (PSC-20 per Table 1.9-10):

- 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.

The above parameters are part of the physical characteristics of commercial SNF, as discussed in Section 1.5.1.1.1.1. The individual radionuclide inventories per assembly for commercial SNF are limited to the values presented in the consequence analysis.

The radioactive material shipping and receipt procedure will specify restrictions and safety controls to preclude the receipt of TAD canisters, DPCs, and commercial SNF that are not within the source term limits specified within the safety analyses. This procedure will require a review of shipping records to verify that the received SNF is within the specified limits.

1.5.1.1.1.2.5 Design Criteria and Design Bases

1.5.1.1.1.2.5.1 Structural Design Criteria and Design Basis

The relationship between TAD canister structure, system, or component (SSC) characteristics and the postclosure nuclear safety design bases can be seen in Table 1.5.1-8.

1.5.1.1.1.2.5.2 Criticality Safety Design Criteria and Design Basis

The TAD canister is designed to meet the requirements of 10 CFR Part 72 for storage, 10 CFR Part 71 for transportation, and 10 CFR Part 63 for repository disposal.

For the preclosure period, the SNF and canister designs, in conjunction with the facility systems, structures, and components, 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.

The principal design basis for criticality safety during the preclosure period is moderator control. Moderator control as part of the canister and surface facility design renders the accumulation of moderator inside the commercial SNF canister following a canister breach (i.e., drop), a beyond Category 2 event sequence. Criticality analysis methods and calculated results for commercial SNF for the preclosure period are described in Section 1.14.

For the postclosure period, criticality analyses are performed for configurations and conditions determined by probabilistic analyses. These evaluations use qualified models with conservative biases and uncertainties and demonstrate that the probability for criticality is low enough to be screened out and not be included in the performance assessment. The bases for postclosure criticality analysis are the features, events, and processes (FEPs) that affect SNF. The methodology and limits used for performing the probabilistic criticality analyses, as well as the factors that affect the configurations of SNF following closure of the repository and calculated results, are further detailed in Section 2.2.1.4.

1.5.1.1.1.2.5.3 Thermal Design Criteria and Design Basis

The thermal design criteria for the commercial SNF waste form are the following:

- The commercial SNF cladding thermal limits prevent fission product and actinide release from within the commercial SNF cladding. These commercial SNF cladding thermal limits are based on accumulated time at each temperature.

- In this regard, commercial SNF cladding temperature in TAD canisters shall not exceed 752°F during normal operations. Normal operations include storage at purchaser sites, transportation from purchasers to the GROA, and handling at the GROA.

- Commercial SNF cladding temperature shall not exceed 1058°F during draining, drying, and backfill operations following TAD canister loading.

- The maximum leakage rate of a TAD canister shall be 9.3×10^{-10} fraction of canister-free volume per second (off-normal) after a fully engulfing fire characterized by an average flame temperature of 1720°F and lasting 30 minutes. During this event, the TAD canister is in either a closed vendor-defined transportation cask (with or without impact limiters) or an open vendor-defined transportation cask without impact limiters. For this event, canister design codes may be exceeded (i.e., vendor may rely on capacity in excess of code allowances).

- The TAD canister cooling features and mechanisms shall be designed to be passive.

- To ensure adequate thermal performance of the TAD canister when emplaced in the waste package, the peak cladding temperature shall be less than 662°F for each set of conditions detailed in the performance specification.

1.5.1.1.1.2.6 Design Methodologies

1.5.1.1.1.2.6.1 Structural Design

1.5.1.1.1.2.6.1.1 TAD Canister Structural Characteristics

The TAD canister has the following structural characteristics:

1. For each of the following design basis seismic events and configurations, the TAD canister shall meet the performance specifications. Seismic vertical and horizontal spectral accelerations are detailed in Attachment A of the performance specification (DOE 2008c).

   a. Following a 2,000-year seismic return period event, a TAD canister shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal), maximum cladding temperature of 752°F (normal), and remain within design codes while in the configurations described below:

      - While suspended by a crane inside an ASTM A 36/A 36M-04 cylindrical steel cavity with an inner diameter of 72.5 in. with a 12-in.-thick wall.

      - While contained in a vendor-defined transportation cask (with impact limiters) described in the performance specification.

      - While contained in a vendor-defined transportation cask (without impact limiters) described in the performance specification that is constrained in an upright position. A constrained transportation cask is one properly secured into the GROA transfer trolley and restrained from tipover in a seismic event.

- While contained in a vendor-defined aging overpack as described in the performance specification.

b. Following a 10,000-year seismic return period event, a TAD canister shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal), cladding temperature limit of 1058°F (off-normal), and remain within design codes while in the configurations described below.

- While suspended by a crane inside an ASTM A 36/A 36M-04 cylindrical steel cavity with an inner diameter of 72.5 in. with a 12-in.-thick wall.

- While contained in a vendor-defined transportation cask (with impact limiters) described in the performance specification.

- While contained in a vendor-defined transportation cask (without impact limiters) described in the performance specification that is constrained in an upright position. A constrained transportation cask is one properly secured into GROA transfer trolley and restrained from tipover in a seismic event.

- While contained in a vendor-defined aging overpack as described in the performance specification.

c. Following a seismic event characterized by horizontal and vertical peak ground accelerations of 96.52 ft/s^2 (3 g), a TAD canister shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) while in the configurations described below. For this initiating event, canister design codes may be exceeded (i.e., a vendor may rely on capacity in excess of code allowances).

- A TAD canister in a vendor-defined transportation cask that drops 10 ft onto an unyielding surface in the most damaging orientation. The transportation cask configuration shall be with or without impact limiters.

- While contained in a vendor-defined transportation overpack (without impact limiters) that is constrained in an upright position. A constrained transportation overpack is one properly secured into the GROA transfer trolley and restrained from tipover in a seismic event.

- While contained in a vendor-defined aging overpack.

2. A TAD canister in a vendor-defined aging overpack shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) and cladding temperature limits during and following exposure to the environmental conditions listed

below (for a through e, the cladding temperature limits are 752°F and 1058°F for "normal" and "off-normal" limits, respectively).

a. These environmental conditions are not cumulative but occur independently:

- Outdoor average daily temperature range of 2°F to 116°F with insulation as specified in 10 CFR Part 71 (normal).
- An extreme wind gust of 120 mph for 3 seconds (normal).
- Maximum tornado wind speed of 189 mph with a corresponding pressure drop of 0.81 lb/in.^{2} and a rate of pressure drop of 0.30 lb/in.^{2}/s (off-normal). The spectrum of missiles from the maximum tornado is provided in the performance specification.

b. Annual precipitation of 20 in./yr (normal). The spectrum of rainfall is provided in the performance specification.

c. Maximum daily snowfall of 6.0 in. (normal).

d. Maximum monthly snowfall of 6.6 in. (normal).

e. A lightning strike with a peak current of 250 kiloamps over a period of 260 microseconds and continuous current of 2 kiloamps for 2 seconds (off-normal).

3. A TAD canister in a transportation overpack (with impact limiters) shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (off-normal) and cladding temperature limits during and following exposure to the environmental conditions listed below (for a through e, the cladding temperature limits are 752°F and 1058°F for "normal" and "off-normal" limits, respectively).

a. These environmental conditions are not cumulative but occur independently:

- Outdoor average daily temperature range of 2°F to 116°F with insulation as specified in 10 CFR Part 71 (normal).
- An extreme wind gust of 120 mph for 3 seconds (normal).
- Maximum tornado wind speed of 189 mph with a corresponding pressure drop of 0.81 lb/in.^{2} and a rate of pressure drop of 0.30 lb/in.^{2}/s (off-normal). The spectrum of missiles from the maximum tornado is provided in the performance specification.

b. Annual precipitation of 20 in./yr (normal).

c. Maximum daily snowfall of 6.0 in. (normal).

d. Maximum monthly snowfall of 6.6 in. (normal).

e. A lightning strike with a peak current of 250 kiloamps over a period of 260 microseconds and continuous current of 2 kiloamps for 2 seconds (off-normal).

4. The TAD canister should have a flat bottom.

1.5.1.1.1.2.6.1.2 TAD Canister Containment Characteristics

1. The TAD canister design shall meet either of the requirements below.

   a. The qualification of the TAD canister final closure welds shall meet the requirements of NRC Interim Staff Guidance–18, "Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation" (NRC 2003b), for ensuring no credible leakage for containment and confinement.

   b. The TAD canister shall be designed to facilitate helium leak testing of closure features using methods that can demonstrate the defined leak-tight requirements have been met. Leak testing shall be performed in accordance with ANSI N14.5-97, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment*. Note that NRC Interim Staff Guidance–18 (NRC 2003b) is an alternate method created by the NRC to meet ANSI N14.5-97.

2. Helium shall be the only gas used for final backfill operations.

3. TAD canister shell and lid shall be designed and fabricated in accordance with the *2004 ASME Boiler and Pressure Vessel Code*, Section III, Division 1, Subsection NB, for Class 1 components. Vendor shall identify applicable exceptions, clarifications, interpretations, and code cases.

4. In accordance with industry standards and regulatory guidance, the TAD canister shall be designed to facilitate the following:

   a. Draining and drying to remove water vapor and oxidizing material shall be carried out in accordance with NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems Plants* (NRC 1997).

   b. Filling with helium to atmospheric pressure or greater as required to meet leak test procedural requirements.

   c. Sampling of the gas space to verify helium purity.

d. Limiting maximum allowable oxidizing gas concentration within the loaded and sealed TAD canister to 0.20% of the free volume in the TAD canister at atmospheric pressure.

5. A loaded TAD canister shall maintain a leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) and cladding temperature below 752°F (normal) following a 12-in. vertical flat-bottom drop. The impacted surface is a solid carbon steel plate.

1.5.1.1.1.2.6.1.3 TAD Canister Characteristics as Developed in the SAR or at Vendor

The relationship between the TAD canister characteristics and how they are used in other parts of the SAR or intended to be used by vendors is noted in Table 1.5.1-10.

1.5.1.1.1.2.6.2 Thermal Design

The thermal design of the TAD canister will ensure the thermal canister controls are not exceeded as discussed in Section 1.5.1.1.1.2.5.3.

1.5.1.1.1.2.6.3 Nuclear Criticality Safety of Commercial SNF

Subcriticality during the preclosure period is demonstrated for TAD canisters and DPCs based on moderator control for sealed canisters and neutron absorbers for uncanistered fuel. The analysis of criticality safety for commercial SNF is discussed in Section 1.14.

1.5.1.1.1.2.6.4 TAD Canister Dose and Shielding Characteristics

Gamma and neutron sources for maximum commercial SNF for shielding are provided in Section 1.10, Table 1.10-18.

1. For GROA operations, the combined neutron and gamma integrated average dose rate over the top surface of a loaded TAD canister shall not exceed 800 mrem/hr on contact.

2. For GROA operations, the combined contact neutron and gamma maximum dose rate at any point on the top surface of the TAD canister shall not exceed 1,000 mrem/hr.

3. The TAD canister shall be designed such that contamination on an accessible external surface shall be removable to:

   a. 1,000 dpm/100 cm^2–beta-gamma with a wipe efficiency of 0.1.
   b. 20 dpm/100 cm^2–alpha with a wipe efficiency of 0.1.

1.5.1.1.1.2.7 Consistency of Materials with Design Basis

The TAD canister system materials are selected because of their resistance to degradation in the repository environment as follows:

1. Required Materials—Except for thermal shunts and criticality control materials, the TAD canister and structural internals (i.e., basket) shall be constructed of a Type 300-series stainless steel (such as UNS S31603, which may also be designated as Type 316L) as listed in ASTM A 276-06, *Standard Specification for Stainless Steel Bars and Shapes*.

2. The TAD canister and its basket materials shall be designed to be compatible with either borated or unborated repository pool water as defined in the performance specification.

3. Prohibited or Restricted Materials

    a. The TAD canister shall not have organic, hydrocarbon-based materials of construction.

    b. All metal surfaces shall meet surface cleanliness classification C requirement defined in ASME NQA-1-2000, Subpart 2.1, *Quality Assurance Requirements for Nuclear Facility Applications*.

    c. The TAD canister shall not be constructed of pyrophoric materials.

    d. The TAD canister, including the steel matrix, gaskets, seals, adhesives and solder, shall not be constructed with materials that would be regulated as hazardous wastes and prohibited from land disposal under the Resource Conservation and Recovery Act of 1976 if declared to be waste.

1.5.1.1.1.2.8 Design Codes and Standards

The materials, design, fabrication, testing, examination, and shipping of the TAD canister shall meet the requirements of the following codes and standards:

- "Design, Fabrication and Construction of Freight Cars." Section C, Part II of *Manual of Standards and Recommended Practices* (AAR 1993)

- *Manual of Standards and Recommended Practices*, Association of American Railroads (AAR 2004)

- *A Policy on Geometric Design of Highways and Streets* (AASHTO 2004)

- ANSI/ANS-57.7-1988, *Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)*

- ANSI N14.5-97, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment*
- ANSI/ANS-57.9-1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*
- ASCE 7-98, *Minimum Design Loads for Buildings and Other Structures*
- *2004 ASME Boiler and Pressure Vessel Code* (ASME 2004)
- ASTM A 276-06, *Standard Specification for Stainless Steel Bars and Shapes*
- ASTM A 887-89, *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application*
- ISO 1161:1984/Cor.1:1990(E), *Series 1 Freight Containers—Corner Fittings—Specification*
- SEI/ASCE 7-02, *Minimum Design Loads for Buildings and Other Structures*
- IEEE/ASTM SI 10-1997, *Standard for Use of the International System of Units (SI): The Modern Metric System*.

1.5.1.1.2 Thermal Characteristics of Commercial SNF

The thermal characteristics of the waste stream are dependent on the TAD system design and therefore a range of thermal characteristics for the TAD system is considered in the design and analysis. Several characteristics associated with the acceptability limits of commercial SNF and the ongoing development of transportation components lead to this uncertainty. These characteristics include:

- The thermal characteristics of the TAD canister, expected to be used for the majority of commercial SNF waste form transportation, are at the discretion of the vendor and have yet to be determined.
- Limited staging capacity is available in the WHF, which limits the ability to significantly blend to control the thermal loading of a TAD canister following receipt of uncanistered commercial SNF.
- Each commercial utility has the ability to load any SNF, within the acceptability limits under 10 CFR Part 961 and within the technical specifications of an NRC-approved certificate of compliance for the TAD canister, uncanistered commercial SNF transportation cask, or DPC.
- The sequence of pickup of commercial SNF is dependent on the contractual arrangements between the DOE and the various commercial utilities.

Thermal power of the canisters/waste packages is important in two ways. First, since they must be handled during preclosure operations, adequate cooling of the canisters/waste packages must be provided for normal and off-normal situations. Accordingly, limiting values on power output have been chosen and analyzed for the various components and facilities. Second, since the decay heat of the waste occurs throughout the postclosure period, it is necessary to emplace canisters/waste packages in a sequence that satisfies postclosure analyses (BSC 2007a, Sections 6.1.1 and 6.2). Section 1.3.1.2.5 describes these limits and emplacement sequence limits.

The thermal power (heat generation rate) of the average and bounding SNF assemblies is presented in Table 1.5.1-11 for both PWR and BWR fuel assemblies. The thermal power for 5 years and 25 years after discharge from the reactor is provided. Figure 1.5.1-6 compares the evolution in time of the thermal power by the same fuel assemblies.

1.5.1.1.3 Nuclear Characteristics of Commercial SNF

The commercial SNF radionuclide inventories are a function of the fuel enrichment, fuel compound, cladding type, moderator type, cooling time after discharge from the reactor, and reactor operating history. The SAS2H sequence in SCALE V4.4 (CRWMS M&O 2000) is used to calculate PWR and BWR radionuclide inventories for SNF assemblies as a function of assembly burnup, enrichment, and cooling time after discharge from a reactor. The ORIGEN-S code (ORNL 1991) is a functional module in the SCALE system and is one of the modules invoked in the SAS2H control module, or it may be applied as a stand alone program. This code performs fuel depletion, actinide transmutation, fission product buildup, and decay, and generates associated radiation source terms for a selected fuel-type with user-specified irradiation conditions and decay times (Hermann and Westfall 2000, p. F7.iii and Section F7).

The radionuclide inventories of the representative PWR and BWR assemblies (Babcock & Wilcox 15 × 15 Mark B and General Electric BWR 2/3 8 × 8, respectively) are presented in Table 1.5.1-12 for the average and maximum (bounding) burnup cases. The inventory includes radionuclides from the fuel section, top and bottom end fittings, fuel plenum, and crud, including those radionuclides important for the PCSA and TSPA. Radionuclides used in the PCSA are selected based on the selection criteria in NUREG-1567 (NRC 2000a, p. 9-11) and Interim Staff Guidance–5 (NRC 2003a, Attachment, Section V.3) as discussed in Section 1.8.1.3.1. Radionuclides used in the TSPA are selected based on their potential contribution to the dose at the accessible environment as discussed in Section 2.3.7.4. Figure 1.5.1-7 compares the evolution in time of the total radioactivity after discharge of the average and maximum (bounding) PWR and BWR fuel assemblies.

1.5.1.1.4 Source-Term Characteristics of Commercial SNF

Source terms for PWR and BWR SNF assemblies with four different combinations of initial enrichment, burnup, and decay time are considered in the PCSA. For Category 1 and Category 2 event sequences, both maximum (bounding) PWR and maximum (bounding) BWR assemblies are considered to calculate doses. For normal operation releases, both representative annual average PWR and BWR assemblies are considered to calculate doses as discussed is Section 1.8.2.2.1.

The PWR SNF is utilized as the bounding commercial SNF source term for repository shielding design (e.g., worker dose assessments, process facility design, as low as is reasonably achievable) and consequence analyses. The PWR assembly inventory and source intensity bound those of a BWR assembly for equivalent burnup, initial enrichment, and cooling time.

The capacity of a TAD canister is either 21 PWR spent fuel assemblies or 44 BWR spent fuel assemblies (Section 1.5.1.1.1.2.1.4). For TAD canister repository shielding design, only the 21 PWR SNF case is considered, because evaluated dose rates for a 21 PWR SNF TAD canister bound those for a 44 BWR SNF TAD canister.

Gamma and neutron sources for maximum and design basis commercial SNF for shielding are provided in Tables 1.10-18 and 1.10-19. Maximum source terms represent the bounding fuel assembly in the inventory to be received and are used for permanent structural shielding designs, including walls, floors, and shield doors. Design basis source terms cover a minimum of 95% of the historical and projected fuel inventory and are used for various transfer shield designs on a case-by-case basis in order to limit shield weight. Shielding design is discussed in Section 1.10.3.

In preclosure analyses, the source term released during normal operations or Category 2 event sequences is discussed in Section 1.8.1.3.1. Treatment of source-term modeling (e.g., for Engineered Barrier System transport and colloid source terms) for waste form degradation and transport as part of the TSPA is discussed at length in Section 2.3.7 and determination of the radionuclide inventory for use in the TSPA is specifically discussed in Section 2.3.7.4. A summary of the commercial SNF analysis basis in the SAR is provided in Table 1.5.1-13.

1.5.1.1.5 TAD Canister and DPC Conformance of Design to Criteria and Bases

[NUREG-1804, Section 2.1.1.7.3.2: AC 1(1)]

The nuclear safety design bases for ITS and ITWI SSCs and features are derived from the PCSA presented in Sections 1.6 through 1.9 and the postclosure performance assessment presented in Sections 2.1 through 2.4. The nuclear safety design bases identify the safety function to be performed and the controlling parameters with values or ranges of values that bound the design.

The quantitative assessment of event sequences, including the evaluation of component reliability and the effects of operator action, is developed in Section 1.7. Any SSC or procedural safety control appearing in an event sequence with a prevention or mitigation safety function is described in the applicable design section of the SAR.

Section 1.9 describes the methodology for safety classification of SSCs and features of the repository. The tables in Section 1.9 present the safety classification of the SSCs and features, including those items that are non-ITS or non-ITWI. These tables also list the preclosure and postclosure nuclear safety design bases for each structure, system, or major component.

The design criteria are specific descriptions of the SSCs or features (e.g., configuration, layout, size, efficiency, materials, dimensions, and codes and standards) that are utilized to implement the assigned safety functions. Tables 1.5.1-7 and 1.5.1-9 present the nuclear safety design bases and design criteria for the TAD canisters and DPCs, respectively. Table 1.5.1-8 presents the derived requirements and associated design solutions for the ITWI function of the TAD canisters.

1.5.1.2 High-Level Radioactive Waste

[NUREG-1804, Section 2.1.1.2.3: AC 4(2), AC 5(2), AC 6(1); Section 2.1.1.6.3: AC 2(1); Section 2.1.1.7.3.1: AC 1(1), (3), (4), (5), (8); Section 2.1.1.7.3.2: AC 1(1); Section 2.1.1.7.3.3(III): AC 1(1), (2), (3), (4), (5), (6), (8), (10)]

HLW is generated by the reprocessing of SNF. HLW is mixed with a combination of silica sand and other constituents or with glass-forming chemicals that are melted together and poured into stainless steel canisters. Once the material solidifies, the canister is sealed.

The repository shall only accept HLW that is not subject to regulation as hazardous waste under the Resource Conservation and Recovery Act of 1976, Subtitle C, for disposal in the geologic repository licensed by NRC under the Nuclear Waste Policy Act of 1982. Further information on the acceptance requirements of HLW are detailed in the *Waste Acceptance System Requirements Document* (DOE 2008b).

1.5.1.2.1 Physical Characteristics of HLW and HLW Canisters

1.5.1.2.1.1 Physical Characteristics of HLW

The repository is designed to accommodate HLW from the Hanford Waste Treatment and Immobilization Plant, the Defense Waste Processing Facility at the Savannah River Site, the Idaho National Laboratory, and the West Valley Demonstration Project.

Chemical Composition—Table 1.5.1-14 provides the chemical compositions of vitrified HLW to be shipped from the various HLW production facilities. The chemical composition of the Hanford glass given in Table 1.5.1-14 bounds the expected range of HLW canisters to be produced by the Hanford Waste Treatment and Immobilization Plant. The chemical composition of the West Valley Demonstration Project is from the compositions for the canistered waste form.

A total of approximately 900 of the HLW canisters received from the Savannah River Site may also contain cans of vitrified glass containing plutonium arrayed within the vitrified HLW. This is the method currently proposed for disposal of 13 metric tons of weapons-usable plutonium. A vitrification technology utilizing a lanthanide borosilicate glass appears to be a viable option for dispositioning excess weapons-usable plutonium that is not suitable for processing into mixed oxide fuel. Nominally 8.4 wt % plutonium is vitrified in lanthanide borosilicate in cans with an outer diameter of 2.88 in. and a length of 19.25 in. Four cans, each containing about 920 g of bulk plutonium-containing material, will be loaded into a magazine, and seven magazines will be loaded into an HLW canister that is then filled with vitrified HLW. The resultant product is referred to as the vitrified plutonium waste form (Marra et al. 2005, Sections 3.2 and 3.7). The vitrified plutonium waste form has been evaluated for its likely bounding contribution to the TSPA results. The inventory characteristics of this waste form are discussed further in Section 2.3.7.4.1.1. Analyses associated with preclosure handling and dose analyses and detailed analysis of postclosure criticality screening have not been performed; therefore, this waste form is not fully analyzed and not acceptable for disposal at the repository at this time. However, this waste form is identified in order to demonstrate a future intent to include it in licensed operations.

Mass and Volume—Table 1.5.1-15 provides the approximate mass of HLW per waste canister for each HLW originating site. For design purposes, the DOE has used an estimate of 0.5 MTHM per canister equivalence for DOE HLW to determine the number of HLW canisters that can be accepted within the planned DOE material allocation. The HLW producer is required to provide the weight of each chemical element contained in each canister. While the sum total of HLW canister production from the DOE sites is expected to far exceed the 4,667 MTHM allowed total, an integrated acceptance schedule will be developed to control shipments from DOE and commercial sites.

The HLW from West Valley is classified as commercial HLW and not DOE HLW (Knecht et al. 1999, p. 1). West Valley has 275 canisters resulting from reprocessing 640 MTHM of SNF, and DOE uses an estimate of 2.3 MTHM per canister equivalence for commercial HLW.

1.5.1.2.1.2 Physical Characteristics of HLW Canister

The loaded, sealed HLW canister and its contents constitute the complete canistered waste form. HLW is shipped in cylindrical canisters with physical parameters as presented in Table 1.5.1-16.

The estimated total mass of vitrified HLW to be generated at the Hanford site could require as many as 13,205 canisters. The canisters have a nominal outside diameter of 2 ft and a nominal height of 15 ft.

The Savannah River Site will generate an estimated 6,833 canisters of vitrified HLW. The canisters have a nominal outside diameter of 2 ft and a nominal height of 10 ft.

The Idaho National Laboratory will generate about 1,190 canisters of HLW. The canisters have a nominal outside diameter of 2 ft and a nominal height of 10 ft.

The West Valley Demonstration Project produced 275 canisters of HLW. These canisters have a nominal outside diameter of 2 ft and a nominal height of 10 ft.

1.5.1.2.1.2.1 HLW Canister Description

Figure 1.5.1-8 shows a representation of the HLW canisters.

1.5.1.2.1.2.2 HLW Canister Operational Processes

Prior to shipment to the repository, HLW canisters are filled with HLW in molten glass, sealed, inspected, and stored at the facility producing the HLW.

Waste acceptance procedures will be developed and implemented in accordance with Section 5.6 to ensure that the shipper has loaded each HLW canister such that the thermal, shielding, and criticality characteristics of the received waste are within acceptable limits for the certificate of compliance and the repository.

1.5.1.2.1.2.3 Canister Thermal Controls

All systems designed to handle HLW canisters during normal operations shall ensure that the maximum temperature of the vitrified glass does not exceed 400°C (DOE 2008d, Section 13). This temperature limit is sufficient to preserve the properties of the vitrified HLW. There are no design constraints imposed on the canister because of decay heat generated by the contained HLW. Total heat generation rate for canisters containing HLW shall not exceed 1,500 W (5,120 BTU/hr) per canister at the year of shipment (DOE 2008b, Section 4.8.13).

1.5.1.2.1.2.4 Canister Criticality Controls

HLW canister criticality controls for normal repository operations and waste emplacement are unnecessary because of the low concentrations of fissile radionuclides in each HLW canister as discussed in greater detail in Section 1.14.2.3.2.4.

1.5.1.2.1.3 Structures, Systems, and Components Important to Safety and Important to Waste Isolation

Section 1.7.2.3 describes the determination of passive structure, system, or component reliability and discusses several types of failures for passive SSCs in the PCSA, including thermal challenge causing loss of containment (breach or fire) of a waste form container (e.g., the HLW canister).

The HLW canister is classified as ITS and performs safety functions during preclosure handling, and the controlling parameters and values of the nuclear safety design bases and operations can be found in Table 1.5.1-17.

The HLW glass is classified as ITWI as described in Table 1.9-8. However, the HLW canister is not ITWI. No derived internal requirements are identified for the TSPA analyses for the HLW canister.

1.5.1.2.1.4 Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effect

The individual radionuclide inventories per HLW canister are limited to the values presented in the Section 1.8 consequence analysis.

The radioactive material shipping and receipt procedure will specify restrictions and safety controls (PSC-21 per Table 1.9-10) to preclude the receipt of HLW canisters that are not within the source term limits specified within the safety analyses. This procedure will require a review of shipping records to verify that the received HLW canisters are within the specified limits.

1.5.1.2.1.5 Design Criteria and Design Bases

The HLW canisters are classified as ITS for their resistance to the maximum credible fire. The HLW canisters are designed to handle the temperatures and stresses during the canister handling, glass pouring, canister closure, and transfer operation steps at production. The HLW canisters were designed prior to being classified as ITS, and the design considered neither fire nor design basis ground motion–2 for the repository. The analysis basis for loss of containment for the HLW canister is discussed in Sections 1.7.2.3.1 and 1.7.2.3.3.1.

1.5.1.2.1.6 Consistency of Materials with Design Methodologies

The canister materials are compatible with the waste package inner shell and divider plate material and do not introduce mechanisms for unacceptable performance of either the canister or the waste package, nor do they induce waste package internal corrosion.

1.5.1.2.1.7 Design Codes and Standards

The design codes and standards related to HLW canister materials of construction and canister welding are described in Table 1.5.1-18.

1.5.1.2.2 Thermal Characteristics of HLW

The thermal output, both for total watts per site and for average watts per canister from each site, is calculated from the radionuclide inventory. Table 1.5.1-19 provides the results.

1.5.1.2.3 Nuclear Characteristics of HLW

Table 1.5.1-20 lists the radionuclide (total curie) distributions of HLW from each generating site, in the year 2017, while Table 1.5.1-21 lists the radionuclide distributions (maximum per canister) from each generating site in the year 2017.

1.5.1.2.4 Source-Term Characteristics of HLW

There are no normal operation releases from HLW as discussed in Section 1.8.2.2.1. There are no Category 1 event sequences involving HLW as discussed in Section 1.8.2.3. Potential doses to members of the public from Category 2 event sequences, including those involving HLW, are discussed in Section 1.8.3.2.2. Potential doses to workers from normal operations including those involving HLW, are discussed in Section 1.8.4. For PCSA modeling, the HLW source-term characteristics are the maximum per canister inventories provided in Table 1.5.1-21. Characteristics of the Savannah River Site HLW canister represent the bounding glass compositions from a dose perspective (Section 1.8). The radionuclide inventory (Table 1.8-5) of the Savannah River Site HLW canister is the bounding case used in the PCSA consequence evaluation presented in Section 1.8. For TSPA modeling, the HLW source-term characteristics are taken from the HLW inventories as provided in the *Initial Radionuclide Inventories* (SNL 2007a) as discussed in Section 2.3.7.4.3. Table 1.5.1-22 provides the HLW disposal analysis basis.

1.5.1.2.5 Conformance of Design to Criteria and Bases

Facility and equipment shielding evaluations are based on bounding source terms, including SNF and HLW. Savannah River Site HLW and Hanford HLW canister source terms (as appropriate) are utilized in the shielding evaluation. See Section 1.10.3 for details.

The nuclear safety design bases for ITS and ITWI SSCs and features are derived from the PCSA presented in Sections 1.6 through 1.9 and the postclosure performance assessment presented in Sections 2.1 through 2.4. The nuclear safety design bases identify the safety function to be performed and the controlling parameters with values or ranges of values that bound the design.

The quantitative assessment of event sequences, including the evaluation of component reliability and the effects of operator action, is developed in Section 1.7. Any SSC or procedural safety control appearing in an event sequence with a prevention or mitigation safety function is described in the applicable design section of the SAR.

Section 1.9 describes the methodology for safety classification of SSCs and features of the repository. The tables in Section 1.9 present the safety classification of the SSCs and features, including those items that are non-ITS or non-ITWI. These tables also list the preclosure and postclosure nuclear safety design bases for each structure, system, or major component.

The design criteria are specific descriptions of the SSCs or features (e.g., configuration, layout, size, efficiency, materials, dimensions, and codes and standards) that are utilized to implement the assigned safety functions. Table 1.5.1-17 presents the nuclear safety design bases and design criteria for the HLW canisters. There are no derived requirements and associated design solutions for the ITWI function of the HLW canisters.

1.5.1.3 DOE SNF

[NUREG-1804, Section 2.1.1.2.3: AC 4(1), AC 5(2), AC 6(1); Section 2.1.1.6.3: AC 2(1); Section 2.1.1.7.3.1: AC 1(1), (3), (4), (5), (8); Section 2.1.1.7.3.2: AC 1(1); Section 2.1.1.7.3.3(III): AC 1(1), (2), (3), (4), (5), (6), (8), (9), 10)]

DOE SNF is primarily generated by DOE production reactors, demonstration commercial power reactors, and domestic and foreign research and training reactors. DOE SNF includes some commercial SNF not in the possession of NRC-licensed commercial utilities. Although naval SNF is included as a DOE SNF group, a separate description and analyses are provided in Section 1.5.1.4 for naval SNF.

Over the past several decades, since the inception of nuclear reactors, the DOE and its predecessor agencies operated or sponsored a variety of research, test, training, and other experimental reactors with different characteristics from the commercial power reactors of today. DOE SNF generated in production reactors supported weapons and other isotope production programs. An example of SNF existing today from production reactors is the N Reactor fuel stored at Hanford. Some SNF from commercial power reactors, such as Shippingport, Peach Bottom, and Fort St. Vrain, is stored within the DOE complex. This SNF was generated for commercial power demonstration purposes or as part of research projects. Also, the Three Mile Island Unit 2 fuel debris is stored at the Idaho National Laboratory.

DOE has sponsored nuclear research activities in the United States and overseas. There are numerous university and government research reactor sites within the United States. SNF from research reactors is stored primarily at the Idaho National Laboratory and Savannah River Site. Examples of research reactor SNF being stored within the DOE complex include the High-Flux Beam Reactor fuel stored at the Savannah River Site; the Fast Flux Test Facility SNF stored at

Hanford and the Idaho National Laboratory; TRIGA SNF stored at Hanford and the Idaho National Laboratory; and the Advanced Test Reactor SNF stored at the Idaho National Laboratory.

Additional research reactor SNF is being returned to the United States from foreign research reactors as part of the DOE Foreign Research Reactor Spent Nuclear Fuel Return Program.

All DOE SNF, except some uncanistered DOE SNF of commercial origin, shall be placed in a sealed disposable canister compatible with all applicable requirements detailed in the *Waste Acceptance System Requirements Document* (DOE 2008b), before acceptance into the repository.

DOE SNF of commercial origin having handling features interchangeable with either BWR or PWR fuel assemblies and known to have no defects may be handled in the same manner as commercial SNF as specified in 10 CFR Part 961. All DOE SNF of commercial origin that (1) cannot be shown to have handling interfaces functionally interchangeable with those of an intact assembly from either a commercial BWR or PWR, or (2) has known or suspected defects (to either structural components or to cladding beyond hairline cracks or pinhole leaks), such that the SNF requires isolation or special handling, shall be placed in a DOE standardized canister before acceptance into the repository.

DOE SNF of commercial origin delivered uncanistered shall be classified using Appendix E of 10 CFR Part 961.

Only a range of canister counts can be cited since little of the DOE SNF has been packaged for final disposal and packaging efficiencies can only be estimated. Depending on packaging efficiencies, Appendix F of *Source Term Estimates for DOE Spent Nuclear Fuels* (DOE 2004a,Appendix F) estimated that the DOE canisters can range from a minimum of 2,500 to a maximum of 5,000 with a point estimate of 3,500 canisters.

1.5.1.3.1 Physical Characteristics of DOE SNF and Disposal Canisters

Most DOE SNF (approximately 98% of the heavy metal) is shipped to and handled at the repository in sealed canisters that are suitable for codisposal in waste packages with HLW without being opened. A small quantity of DOE SNF (approximately 2% of the heavy metal) in the possession of the DOE is intact SNF of commercial origin having no known defects and having handling features interchangeable with either BWR or PWR fuel assemblies (DOE 2007, Section 3.2). This commercial SNF in the possession of the DOE may be transported and handled as uncanistered assemblies and can be shipped to the repository in a transportation cask and placed in a TAD canister at the repository (DOE 2008b, Section 4.2.3).

1.5.1.3.1.1 Physical Characteristics of DOE SNF

A large and varied number of SNF is currently in the possession of the DOE. These fuels come from a wide range of reactor types, such as light- and heavy-water-moderated reactors, graphite-moderated reactors, and breeder reactors, with various cladding materials and enrichments, varying from depleted uranium to over 93% enriched ^{235}U. Many of these reactors, now decommissioned, had unique design features, such as core configuration, fuel element and assembly geometry, moderator and coolant materials, operational characteristics, and neutron spatial and spectral properties (DOE 2004).

As described below, there is a large diversity of reactor and fuel designs. In addition, there is a relatively large number (over 200,000) of fuel pieces or assemblies, which range from a large number of pieces for some reactors (N Reactor) to a few individual pieces for other unique reactors (Chicago Pile-5 converter cylinders) (DOE 2007).

The receipt and handling approach to be analyzed for DOE SNF is that it is to be packaged in sealed canisters that are designed so that breach of a canister upon a drop is a beyond Category 2 event sequence. The canister shell provides the criticality safety design control feature of moderator control as described in Section 1.14.2.3.1.5. Based on the receipt and handling approach, release of radionuclides from a DOE SNF canister is a beyond Category 2 event and dose calculations will not be relied upon to demonstrate regulatory compliance. For postclosure, a surrogate fuel is used to model radionuclide releases. This surrogate fuel is based on an estimate of the total DOE SNF activity, averaged over the estimated number of waste packages for DOE SNF.

The following fuel descriptions provide an overview of the DOE SNF based upon process knowledge and best available information, giving selected examples of fuels and providing a nominal range of the fuel characteristics.

Moderator—The reactors generating the DOE SNF have used a variety of moderators, including the following (DOE 2007, Section 3.1):

- Graphite
- Heavy water
- Light water
- Uranium zirconium-hydride
- Organics.

Coolant—The reactors generating the DOE SNF have used a variety of coolants, including the following (DOE 2007, Section 3.1):

- Heavy water
- Light water
- Nitrogen
- Organics
- Air
- Sodium
- Helium.

Mission—The reactors generating the DOE SNF have been operated for a variety of purposes, including the following (DOE 2007, Section 3.1):

- Defense power
- Development power
- Experimental power reactors

- Material production
- Process development
- Testing, research, and education.

Fuel Manufacturers—DOE SNF has been manufactured by numerous suppliers. Some of these suppliers are no longer producing reactor fuels or no longer exist. The following list includes representative commercial vendors, government suppliers, and foreign suppliers that supported the development of experimental fuels for the DOE laboratories (DOE 2007, Section 3.2):

- Aerojet General Nuclear
- ALCO Products, Inc.
- Advanced Nuclear Fuels Corporation Framatome
- Atomics International
- Babcock & Wilcox
- Battelle Columbus D.
- Belgonucleaire S.A.
- Combustion Engineering
- Curtis-Wright Corporation
- D.E. Makepeace
- DOE laboratories (Argonne National Laboratory, Idaho National Laboratory, Los Alamos National Laboratory, Pacific Northwest National Laboratory)
- Euratom
- General Atomics
- General Electric Company
- Great Lakes Carbon Company
- Gulf United Nuclear
- Martin Nuclear
- McDermott Company
- Nuclear Components

- Texas Instruments

- Westinghouse.

Compound—The fuel compounds in DOE SNF include the following: (DOE 2007, Section 3.2):

- Americium oxide
- Plutonium oxide
- Plutonium-uranium alloy
- Plutonium-uranium carbide
- Plutonium-uranium nitride
- Plutonium-uranium oxide
- Thorium-uranium metal
- Thorium-uranium carbide
- Thorium-uranium oxide
- Uranium-zirconium hydride
- Uranium alloy
- Uranium carbide
- Uranium metal
- Uranium oxide
- Uranium silicide.

Matrix—The fuel matrices include aluminum, carbon (graphite), stainless steel, nichrome, zirconium oxide–calcium oxide, beryllium oxide, and zirconium oxide (DOE 2007, Section 3.2).

Cross Section—The cross section of DOE SNF assemblies includes the following (DOE 2007, Section 3.2):

- Annular
- Circular
- Hexagonal
- Rectangular
- Rhomboid
- Square
- Trapezoidal
- Triangular.

Configuration and Size—The configuration of DOE SNF varies from intact assemblies, plates, and rods to cans of debris. The sizes of these configurations are highly variable and range from about 0.2 in. to 22.3 in. across and 0.3 ft to 14.7 ft in length (DOE 2007, Table 1).

Cladding—Cladding of DOE SNF varies in composition and integrity. The cladding materials used for DOE SNF include the following (DOE 2007, Section 3.2):

- Aluminum
- Tristructural isotropic, buffered isotropic, and monopyrolytic carbon coatings
- Hastelloy

- Incoloy
- Nichrome
- Stainless steel
- Zirc, which includes both zirconium and Zircaloy.

The condition of the cladding of DOE SNF varies from intact to degraded. Some fuels have been declad or have undergone destructive examination. While intact cladding is considered a primary isolation barrier, no credit is taken in the preclosure and postclosure analysis (DOE 1999a; SNL 2008a).

The following criteria are used for categorizing the cladding condition:

- Good—No known or suspected through-cladding defects
- Fair—Known or suspected defects are limited to hairline cracks or pinhole leaks
- Poor—Known or suspected defects are greater than hairline cracks or pinhole leaks
- None—Declad or unclad SNF.

History—The burnup of DOE SNF ranges from slightly irradiated to over 500,000 MWd/MTU. For some DOE SNF, burnup is recorded in terms of ^{235}U burnup percent consumed, rather than gigawatt days per metric ton of uranium. The burnup for these fuels ranges from slightly irradiated to over 80% consumed of the initial ^{235}U to over 70% of the initial heavy metal. (DOE 2007, Section 3.2).

The cooling times for the SNF range from the minimum time required to meet transportation cask decay heat limits to over 40 years. The majority of DOE SNF has been in wet storage. Much of the SNF has been moved to dry storage. Except for commercial SNF with a good or fair cladding condition, the DOE SNF is dried and placed in sealed, inerted canisters prior to shipment to the repository for disposal. Some DOE SNF has been modified, including being disassembled or cut.

Fissile Material—The fissile material in the DOE SNF includes ^{233}U, ^{235}U, the various nuclides of plutonium, and other transuranics. The ^{235}U enrichment ranges from depleted uranium to over 93%. The effective enrichment for DOE SNF is defined as the ratio of the fissile mass to the sum of the total U plus total Pu expressed as a percentage. Fissile mass (kg) here includes ^{233}U, ^{235}U, ^{239}U, and ^{241}Pu. Total U (kg) is the amount of all isotopes of uranium (atomic number 92) in kilograms, and total Pu (kg) is the amount of all isotopes of plutonium (atomic number 94) in kilograms.

For the purpose of grouping, the SNF enrichment is categorized as high-enriched uranium, medium-enriched uranium, and low-enriched uranium. High-enriched uranium fuel has been defined as fuel with enrichment of greater than or equal to 20%. Medium-enriched uranium fuel has been defined as fuel with enrichment of greater than or equal to 5% but less than 20%. Low-enriched uranium fuel has been defined as fuel with enrichment of less than 5% (DOE 2007, Section 3.2). The effective end of life enrichments and end of life MTHM quantities are reported as part of the ranges of nominal DOE SNF properties.

DOE SNF Groups—There are several hundred distinct types of DOE SNF, and it is not practical to attempt to determine the impact of each individual type on repository performance. The DOE

SNF inventory was first reduced to 34 DOE SNF groups based on fuel matrix, cladding, cladding condition, and enrichment. These parameters are the fuel characteristics that were determined to have major impacts on the release of radionuclides from the DOE SNF and contributed to nuclear criticality scenarios (DOE 2000a, Section 5).

Separate groups were further refined for the purposes of criticality, design basis events, and TSPA based on key parameters such as fuel matrix, cladding, and fuel condition, as well as fissile species and enrichment, and reactor and fuel design (DOE 2000a, Section 5.1). For criticality nine DOE SNF groups have been identified and are presented in Table 1.5.1-23. The nine criticality groups and the representative fuel types considered in the criticality analysis for each group are summarized in Section 1.5.1.3.1.1.3. See Section 1.14 and 2.2.1.4.1 for the preclosure and postclosure criticality evaluations, respectively. For design basis events, six groups were developed to represent the variability for the entire inventory in these parameters (DOE 2000a, Section 7.1). For TSPA, only the fuel matrix parameter was determined to be of significance, and 11 groups were developed to represent the entire inventory for TSPA modeling (DOE 2000a, Section 8).

1.5.1.3.1.1.1 First Tier Grouping of DOE SNF

The following discussions of each of the 34 groupings are presented in this section (DOE 2007, Section 4). These 34 groups are then used as the basis for further grouping to support development of radionuclide source terms and fuel reactivity. The following discussions of each of the 34 groups provide a description of the fuel group and an example of fuel that makes up the group. When appropriate, a more detailed description of a fuel with the largest percentage of MTHM within each group is provided. This discussion is not intended to address each fuel in the group. Table 1.5.1-24 describes the typical ranges of the nominal properties for DOE SNF in the 34 groups, and Table 1.5.1-23 maps each SNF database record to one of the 34 fuel groups and describes the disposal analysis basis for each of the 34 fuel groups.

Intact fuel is made up of fuel from the good or fair cladding condition categories. Nonintact fuel is composed of fuel from the poor or none cladding categories.

Group 1: U Metal, Zirc Clad, Low-Enriched Uranium—This group contains a low-enriched uranium-metal compound SNF with zirconium cladding (accounting for approximately 86% of the DOE SNF inventory by mass). Greater than 99% of the MTHM of SNF in this group is N Reactor SNF. The N Reactor was used for both material and power production. N Reactor fuel consists of two concentric tubes about 2.4 in. in diameter and typically 2 ft long. N Reactor SNF has a nominal enrichment of about 1% and a typical burnup of about 2.4 GWd/MTU. The cladding condition of the N Reactor SNF is fair to poor.

Group 2: U Metal, Nonzirc Clad, Low-Enriched Uranium—This group contains a low-enriched uranium-metal compound SNF with nonzirc cladding. The largest single source of SNF in this group (over 40% of the MTHM) is from the Single-Pass Reactor, which was used for material production. The Single-Pass Reactor SNF consists of circular tubes roughly 1.5 in. in diameter and 0.66 ft long. The Single-Pass Reactor SNF has a nominal enrichment of about 1% and an average burnup of about 3 GWd/MTU. The cladding condition of the Single-Pass Reactor SNF is generally poor.

Group 3: U-Zirc—This group contains uranium-zirc compound SNF. Greater than 99% of the MTHM of fuel in this group is from the Heavy Water Components Test Reactor. Heavy Water Components Test Reactor semi-production run SNF is the dominant SNF in this group (67% of the MTHM). Heavy Water Components Test Reactor semi-production run SNF consists of circular tubes about 2.1 in. in diameter and 11 ft long. The Heavy Water Components Test Reactor semi-production run SNF is about 0.6% enriched. The condition of the Heavy Water Components Test Reactor semi-production run SNF cladding is fair.

Group 4: U-Mo—This group contains a uranium-molybdenum alloy compound SNF. More than 99% of the MTHM of the SNF in this group is from the Enrico Fermi Atomic Power Plant, and the majority (over 90% of the MTHM) of the SNF in this group consists of Fermi standard fuel subassemblies. Fermi was a sodium-cooled fast neutron spectrum power reactor. Fermi driver fuel consists of rods roughly 0.16 in. in diameter and 2.7 ft long. The Fermi standard fuel subassembly SNF has an enrichment of about 26% and an average burnup of about 1.6 GWd/MTU. The condition of the cladding for the SNF in this group ranges from good to none.

Group 5: U Oxide, Zirc Clad, Intact, High-Enriched Uranium—This group contains a high-enriched uranium oxide SNF with intact zirc cladding. Greater than 90% of the MTHM of the SNF in this group consists of Shippingport PWR Core 2 blanket SNF, which is a uranium oxide compound dispersed in a zirconium-oxide (Seed 1) or zirconium-oxide calcium-oxide (Seed 2) matrix. Shippingport PWR was a light-water-moderated and cooled power reactor. Shippingport PWR fuel assemblies consist of 19 flat plates; the assemblies are 7.4 in. square and about 8.7 ft long. The Shippingport PWR Core 2 SNF has an enrichment of about 69% to 81% and a burnup of roughly 38% of the initial fissile mass. The condition of the Shippingport PWR Core 2 blanket fuel cladding is good.

Group 6: U Oxide, Zirc Clad, Intact, Medium-Enriched Uranium—This group contains medium-enriched uranium oxide SNF with intact zirc cladding. Greater than 80% of the MTHM in this group consists of Experimental Boiling Water Reactor SNF. The Experimental Boiling Water Reactor was a DOE light-water-cooled and moderated experimental power reactor. Experimental Boiling Water Reactor SNF consists of plate-type assemblies, roughly 3.75 in. square and 5.2 ft long. Experimental Boiling Water Reactor SNF has an enrichment of 6% and a maximum burnup of 1.6 GWd/MTU. The cladding condition of the Experimental Boiling Water Reactor SNF is fair.

Group 7: U Oxide, Zirc Clad, Intact, Low-Enriched Uranium—This group contains low-enriched uranium oxide with intact zirc cladding. The majority (75% of the MTHM) of the SNF in this group was generated by typical commercial power reactors, such as the Robert E. Ginna, Calvert Cliffs, Big Rock Point, Surry, and Turkey Point reactors. The commercial power reactor SNF configuration includes intact rod arrays. The commercial power reactor SNF in this group has enrichments ranging from 0.6% to 2.9%. The average burnup of the commercial power reactor SNF in this group ranges from about 1.6 GWd/MTU for some Big Rock Point SNF to about 43 GWd/MTU for the Calvert Cliffs 1 SNF. The cladding condition of the commercial power reactor SNF in this group is good.

Group 8: U Oxide, SST/Hastelloy Clad, Intact, High-Enriched Uranium—This group contains high-enriched uranium oxide with intact stainless steel or Hastelloy cladding. About 40%

of the MTHM of the SNF in this group was generated by superheaters for the Pathfinder Atomic Power Plant, a power reactor, and the Boiling Reactor Experiment V, a test, research, and education reactor. The Pathfinder SNF consists of rods 0.9 in. in diameter and 6.5 ft long. The Boiling Reactor Experiment V SNF consists of flat plate assemblies 3.7 in. wide and 2.1 ft long. The SNF in this group has an enrichment of roughly 93%. The Pathfinder and Boiling Reactor Experiment V SNF in this group have a burnup of less than 6% of the initial fissile mass, and the cladding condition is good to fair.

Group 9: U Oxide, SST Clad, Intact, Medium-Enriched Uranium—This group contains medium-enriched uranium oxide SNF with intact stainless steel cladding. Greater than 80% of the MTHM of the SNF in this group was driver fuel for the Power Burst Facility, which was a test reactor designed to investigate fuel performance during accident conditions. Power Burst Facility SNF consists of rods measuring 0.75 in. in diameter and 4 ft long. Power Burst Facility SNF has an enrichment of about 18% and an average burnup of about 0.5 GWd/MTU. The Power Burst Facility cladding condition is good.

Group 10: U Oxide, SST Clad, Intact, Low-Enriched Uranium—This group contains low-enriched uranium oxide SNF with intact stainless steel cladding. This group contains a small amount of material, over 40% of which by MTHM was generated by Connecticut Yankee reactors. The Connecticut Yankee SNF is typical commercial power reactor SNF, except that it has stainless steel cladding. The Connecticut Yankee SNF has an enrichment of 1.9%. The Connecticut Yankee SNF has a burnup of about 32 GWd/MTU. The cladding condition of the Connecticut Yankee SNF is good.

Group 11: U Oxide, Nonalum Clad, Nonintact or Declad, High-Enriched Uranium—This group contains high-enriched uranium oxide SNF with nonaluminum cladding that is not intact or that has been removed. About 60% of the MTHM of the SNF in this group is generated from medical isotope production targets from foreign research reactors in Canada. The Canadian foreign research reactor targets have an enrichment of about 50%. As there is no cladding on the Canadian foreign research reactor targets, the fuel cladding is categorized as none.

Group 12: U Oxide, Nonalum Clad, Nonintact or Declad, Medium-Enriched Uranium—This group contains medium-enriched uranium oxide SNF with failed nonaluminum cladding or no cladding. Virtually all of this SNF was generated as a result of severe-condition fuel experiments. These experiments generally involved segments of previously irradiated fuel rods that were sectioned and placed into experiment capsules for further irradiation under extremely high temperatures. The SNF in this group has enrichments ranging from 5% to nearly 20%. The cladding condition of the SNF in this group is either poor or none (the cladding has been removed).

Group 13: U Oxide, Nonalum Clad, Nonintact or Declad, Low-Enriched Uranium—This group contains low-enriched uranium oxide SNF with failed nonaluminum cladding or no cladding. 99% of the MTHM of the SNF in this group is core debris from the Three Mile Island Unit 2 reactor accident. The Three Mile Island Unit 2 fuel has an enrichment of about 2.4% and a burnup of about 3.2 GWd/MTU. The cladding condition of the Three Mile Island Unit 2 SNF is poor.

Group 14: U Oxide, Alum Clad, High-Enriched Uranium—This group contains high-enriched uranium oxide SNF with aluminum cladding. Greater than 80% of the MTHM of the SNF in this group is High-Flux Isotope Reactor SNF. The High-Flux Isotope Reactor is a DOE test reactor. High-Flux Isotope Reactor SNF consists of two concentric assemblies consisting of curved involute plates that are separated for disposal. The outer assemblies are about 17 in. in diameter and 2.6 ft long, and the inner assemblies are about 12 in. in diameter and 2.5 ft long. High-Flux Isotope Reactor SNF has an enrichment of about 87%. High-Flux Isotope Reactor SNF has an average burnup of about 230 GWd/MTU. The cladding condition of High-Flux Isotope Reactor SNF is good.

Group 15: U Oxide, Alum Clad, Medium-Enriched Uranium, Low-Enriched Uranium—This group contains medium-enriched uranium oxide SNF with aluminum cladding. Nearly all of the SNF in this group was generated from a number of foreign research reactors. The largest single source (56% of the MTHM) is the G.A. Siwabessy RSG-GAS-30 reactor in Indonesia. This Indonesian foreign research reactor SNF consists of square assembly plate-type fuel with a typical width of 3 in. and a length of about 2.9 ft. This Indonesian research reactor SNF has an enrichment of about 10% and a burnup of about 50% of the initial fissile mass. The cladding condition of most of the Indonesian research reactor SNF in this group is good.

Group 16: U-Al$_x$, Al-Clad High-Enriched Uranium—This group contains high-enriched uranium aluminide SNF. The SNF in this group is generated from domestic and foreign test, research, and education reactors. The Advanced Test Reactor is the largest single source of SNF in this group, accounting for 67% of the MTHM. The Advanced Test Reactor SNF consists of curved plate assemblies about 4.2 in. wide, 2.6 in. high, and 5.5 ft long, before being cropped to about 4.1 ft for storage. The Advanced Test Reactor SNF has a typical enrichment of about 80% with an average burnup of about 250 GWd/MTU. The cladding condition of Advanced Test Reactor SNF is good.

Group 17: U-Al$_x$, Al-Clad Medium-Enriched Uranium—This group contains medium-enriched uranium aluminide SNF. The SNF in this group is generated from numerous domestic and foreign test, research, and education reactors. The largest single source of SNF in this group (30% of the MTHM) is the R-2 reactor in Sweden. The R-2 SNF is a square assembly of plate-type fuel about 3 in. wide and about 2.9 ft long. The R-2 SNF has an enrichment of about 9% and a burnup of 60% of the initial fissile mass. The cladding condition of the SNF in this group is generally good.

Group 18: U_3Si_2—This group contains uranium-silicide SNF. The SNF in this group is generated from numerous domestic and foreign test, research, and education reactors. About 45% of the MTHM in this group consists of foreign research reactor multipin clusters generated by the National Research Universal reactor in Canada. The National Research Universal reactor is heavy water moderated and cooled. National Research Universal SNF has a typical enrichment of about 5.6% and a burnup of about 76% of the initial fissile mass. The cladding condition of National Research Universal SNF is good.

Group 19: Th/U Carbide, TRISO- or BISO-Coated Particles in Graphite—This group contains thorium-carbide and uranium-carbide SNF with tristructural isotropic- or buffered isotropic-coated particles embedded in a graphite matrix. About 95% of the MTHM of the SNF in this group was generated from the Fort St. Vrain reactor. The Fort St. Vrain SNF consists of hexagonal graphite blocks about 14 in. wide by 2.6 ft long, containing tristructural isotropic-coated (i.e., inner pyrocarbon, silicon carbide, and outer pyrocarbon coatings) particles. The Fort St. Vrain SNF has an enrichment of about 80% and burnups of about 45% of the initial fissile mass. The particle coating condition of the Fort St. Vrain SNF is good.

Group 20: Th/U Carbide, Monopyrolytic Carbon-Coated Particles in Graphite—This group contains thorium-carbide and uranium-carbide SNF with monopyrolytic carbon-coated particles in a graphite matrix. The coated particles are embedded in a graphite matrix. Nearly all (greater than 99%) of the SNF in this group is Peach Bottom Unit 1 reactor core 1 fuel. The Peach Bottom Unit 1 reactor was a helium-cooled, graphite-moderated, electric power reactor. The Peach Bottom Unit 1 SNF is about 3.5 in. wide and 12 ft long. The Peach Bottom Unit 1 core 1 SNF has a typical enrichment of about 86% and a burnup of about 30% of the initial fissile mass. The particle coating condition of the Peach Bottom Unit 1 core 1 SNF is poor.

Group 21: Pu/U Carbide, Nongraphite Clad, Not Sodium Bonded—This group contains a small quantity of plutonium/uranium-carbide SNF with nongraphite cladding and no sodium bonding. This SNF was generated primarily by the Fast Flux Test Facility and has stainless steel cladding. The Fast Flux Test Facility reactor was a sodium-cooled DOE test and research reactor. About 56% of the MTHM in this group is the Fast Flux Test Facility test fuel assembly TFA-FC-1. The Fast Flux Test Facility TFA-FC-1 assembly cross section is a hexagon about 4.6 in. across the flats, 5.2 in. across the points, and the SNF is 12 ft long. The Fast Flux Test Facility TFA-FC-1 SNF is about 21% enriched and has a burnup of about 60 GWd/MTU. The Fast Flux Test Facility TFA-FC-1 cladding condition is good.

Group 22: Mixed Oxide, Zirc Clad—This group contains a small quantity of mixed oxide, uranium-oxide, and plutonium-oxide SNF with zirconium cladding. About 60% of the MTHM in this group is Experimental Boiling Water Reactor SNF, which experimented with the recycling of plutonium. The Experimental Boiling Water Reactor SNF has an enrichment of 1.6% and a burnup of 3% of the initial fissile mass. The Experimental Boiling Water Reactor SNF cladding condition is fair.

Group 23: Mixed Oxide, SST Clad—This group contains mixed oxide, uranium-oxide, and plutonium-oxide SNF with stainless steel cladding. About 80% of the MTHM of this group is Fast Flux Test Facility reactor driver fuel assemblies and test driver fuel assemblies. The Fast Flux Test Facility driver and test driver fuel assembly cross section is a hexagon about 4.6 in. across the flats and 5.2 in. across the points, and the SNF is 12 ft long. The Fast Flux Test Facility driver fuel assembly and test driver fuel assembly SNF have enrichments of about 24% and an average burnup of about 70 GWd/MTU. The cladding condition of the SNF in this group is poor to good.

Group 24: Mixed Oxide, Non-SST/Nonzirc Clad—This group contains a small quantity of mixed oxide (uranium-oxide and plutonium-oxide, mixed oxide) SNF that does not have stainless steel or zirconium cladding. The SNF in this group is mostly the residue from hot cells and small experiments and does not have intact cladding. The majority of the SNF in this group (97% of the MTHM) is mixed-oxide scrap with an enrichment of about 15%. The cladding condition of the SNF in this group is either poor or none.

Group 25: Th/U Oxide, Zirc Clad—This group contains thorium-oxide and uranium-oxide SNF with zirconium cladding. The SNF in this group was generated by the Shippingport Atomic Power Station with the Light Water Breeder Reactor core. The Shippingport Light Water Breeder Reactor was a power reactor that converted fertile ^{232}Th to fissile ^{233}U. About 27% of the MTHM in this group is Shippingport Light Water Breeder Reactor reflector IV SNF. Shippingport Light Water Breeder Reactor reflector IV assemblies are rods in a rectangular array about 17.1 in. by 13.8 in. and 11.8 ft long. The Shippingport Light Water Breeder Reactor reflector IV SNF has an enrichment of about 98% and a burnup of about 2 GWd/MTU. The cladding condition of the Shippingport Light Water Breeder Reactor reflector IV SNF is generally good.

Group 26: Th/U Oxide, SST Clad—This group contains thorium-oxide and uranium-oxide SNF with stainless steel cladding. About 66% of the MTHM of the SNF in this group was generated from the Elk River Reactor, a light water power reactor. Elk River Reactor assemblies are rods in square arrays that are about 1.4 in. wide and high and 5.3 ft long. Elk River Reactor SNF has an enrichment of 96%. Elk River Reactor SNF has a typical burnup of about 5.4 GWd/MTU. The cladding condition of the Elk River Reactor SNF is generally fair.

Group 27: U-Zirc Hydride, SST/Incoloy Clad, High-Enriched Uranium—This group contains high-enriched, uranium-zirc hydride SNF with stainless steel or Incoloy cladding. Most of the SNF in this group was generated from numerous domestic and foreign TRIGA research reactors, with no dominant single generator. The TRIGA SNF in this group is generally of the fuel life improvement program design. TRIGA fuel life improvement program rods are typically 1.5 in. in diameter and 2.4 ft long. The enrichment of the TRIGA fuel life improvement program SNF in this group has a range from about 60% to 70%, and the burnup ranges from about 9,400 MWd/MTU to over 300 GWd/MTU. The cladding condition of the TRIGA fuel life improvement program SNF is generally good.

Group 28: U-Zirc Hydride, SST/Incoloy Clad, Medium-Enriched Uranium—This group contains medium-enriched uranium-zirconium hydride SNF with stainless steel or Incoloy cladding. The SNF in this group was generated from numerous domestic and foreign TRIGA research reactors, with no dominant single generator. TRIGA rods in this group are typically 1.5 in. in diameter and 2.4 to 3.8 ft long. The TRIGA SNF in this group has enrichments ranging from about 12% to 20% with burnups ranging from slight irradiation to nearly 95 GWd/MTU. The cladding condition of the SNF in this group is generally good.

Group 29: U-Zirc Hydride, Alum Clad, Medium-Enriched Uranium—This group contains medium-enriched uranium-zirconium hydride SNF with aluminum cladding. The SNF in this group was generated from numerous domestic and foreign TRIGA research reactors, with no dominant single generator. The TRIGA rods in this group are typically 1.5 in. in diameter and 2.4 ft long. The TRIGA SNF in this group has enrichments ranging from about 17% to 20%. The SNF in this group has highly variable burnups, ranging from slightly irradiated to about 37 GWd/MTU. The cladding condition of the SNF in this group is generally good.

Group 30: U-Zirc Hydride, Declad—This group contains uranium-zirconium hydride SNF that has been declad. The SNF in this group was generated from the System for Nuclear Auxiliary Power program, which was an experimental power program that involved five different reactors. The System for Nuclear Auxiliary Power rods are about 1.2 in. in diameter and 1.2 ft long. The

System for Nuclear Auxiliary Power SNF has an enrichment of about 90%. The cladding has been removed, so the cladding condition is none.

Group 31: Metallic Sodium Bonded—This group contains a wide variety of SNF that has the common attribute of containing metallic-sodium bonding between the fuel matrix and the cladding. The disposition of metallic-sodium-bonded SNF is not included in the inventory of SNF to be disposed of at the repository.

Group 32: Naval—Naval SNF is addressed in Section 1.5.1.4.1.1.

Group 33: Canyon Stabilization—This SNF is being treated in the Savannah River Site canyons and will be disposed of as HLW; therefore, this SNF is not addressed in this section.

Group 34: Miscellaneous—This group contains SNF that does not fit into other groups. The SNF in this group was generated from numerous reactors of different types. The dominant source is the Keuring van Electrotechnische Materialen SNF from the Aqueous Homogeneous Suspension Reactor, an experimental power reactor. Keuring van Electrotechnische Materialen SNF consists of canisters of thorium-oxide and uranium-oxide scrap. Keuring van Electrotechnische Materialen SNF has an enrichment of about 90%. Keuring van Electrotechnische Materialen SNF does not have cladding, so the condition is none.

1.5.1.3.1.1.2 Preclosure Releases Grouping of DOE SNF

For design basis events, two parameters (fuel matrix and fuel condition) were determined to be significant for fuel grouping, and six groups were developed to represent the variability for the entire inventory in these parameters. Nonetheless, fuel Groups 1 through 30 (excluding spent fuel in MCOs) are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required. Analyses involving determination of potential event sequences involving a drop and breach of an MCO with DOE SNF has not been completed. See Section 1.8 for detailed preclosure release evaluations.

1.5.1.3.1.1.3 Criticality Evaluation Grouping of DOE SNF

Within each of the nine DOE SNF criticality groups, a single fuel design was selected as being representative of the remaining fuel within each group. The term representative means that all fuels would perform similarly regarding chemical interactions within the waste package and basket, and that canister loading limits from the representative fuel (ranges of key parameters important to criticality such as linear fissile loading and total fissile mass) are established, for which other fuels within the group can be shown to not exceed. Waste forms within a single criticality group that have configurations or key criticality parameters outside the range of applicability of the representative fuel will require supplemental analysis and/or additional reactivity control mechanisms. The nine criticality groups and the representative fuel types considered in the criticality analysis for each group are summarized below. See Sections 1.14 and 2.2.1.4.1 for the preclosure and postclosure criticality evaluations, respectively.

- **Criticality Group 1**: **U-Metal**—N Reactor SNF is the representative fuel type for this fuel group.
- **Criticality Group 2**: **Mixed-Oxide Fuels**—Fast Flux Test Facility SNF is the representative fuel type for this fuel group.
- **Criticality Group 3**: **U-Mo/U-Zr Alloy Fuels**—Enrico Fermi SNF is the representative fuel type for this fuel group.
- **Criticality Group 4**: **Highly Enriched Uranium Oxide Fuels**—Shippingport PWR Core 2 blanket SNF is the representative fuel type for this fuel group.
- **Criticality Group 5**: **^{233}U/Th Oxide Fuels**—Shippingport Light Water Breeder Reactor SNF is the representative fuel type for this fuel group.
- **Criticality Group 6**: **U/Th Carbide Fuels**—Fort St. Vrain SNF is the representative fuel type for this fuel group.
- **Criticality Group 7**: **U-ZrHx Fuels**—TRIGA SNF is the representative fuel type for this fuel group.
- **Criticality Group 8**: **Al-Based Fuels**—Advanced Test Reactor SNF is the representative fuel type for this fuel group.
- **Criticality Group 9**: **Low Enriched Uranium Oxide Fuels**—Three Mile Island Unit 2 debris is the representative fuel type for this fuel group.

1.5.1.3.1.1.4 TSPA Grouping of DOE SNF

For TSPA, only the fuel matrix parameter was determined to be of significance for fuel grouping, and 11 groups were developed to represent the entire inventory for TSPA modeling. Nonetheless, fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. See Sections 2.3.7.4.1.1, 2.3.7.4.2.2, and 2.3.7.8.1 for detailed TSPA evaluations.

1.5.1.3.1.2 Physical Characteristics of DOE SNF Canisters

1.5.1.3.1.2.1 Canister Description

DOE SNF is received at the repository in disposable canisters. There are two types of DOE SNF canisters: standardized canisters and MCOs (BSC 2004a, Section 1). The standardized canisters and MCOs are functionally similar, although there are differences in their design features and details. There are four configurations of standardized canisters and two configurations of the MCO (BSC 2004a, Section 1).

The number of DOE SNF canisters to be shipped to the repository is estimated to be 3,500, about 3,100 of which are standardized canisters and about 400 of which are MCOs. The uncertainty in the number of DOE SNF canisters results from uncertainties in future generations of fuels, receipts from foreign countries, treatment options of some fuels, basket designs, and canister size selection (DOE 2004a, Volume 1, Appendix F, Table F-1).

1.5.1.3.1.2.1.1 Standardized Canister Shell

This section is based on *Preliminary Design Specification for Department of Energy Standardized Spent Nuclear Fuel Canisters* (DOE 1999a, Volume I, Section 3, Appendix A). There are four configurations of DOE SNF standardized canisters, but the functions and requirements associated with each are the same. A cutaway perspective of a representative small-diameter standardized canister is shown in Figure 1.5.1-9.

The standardized canister design includes an integral, energy-absorbing skirt. The skirt reduces drop-induced damage to the standardized canister containment barrier because the skirt deforms on impact. The standardized canisters have a lifting ring integral within each skirt to provide a grappling capability. Because the lifting rings are inside the skirts, they do not affect the standardized canister external dimensions. The lifting rings also provide a beneficial stiffening and energy-absorbing effect during a drop event. The standardized canister design includes 2-in.-thick internal impact plates to protect the dished heads from internal impacts and punctures (DOE 1999a, Section 3 and Appendix A).

Incorporated into the standardized canister design is the option of a plug (threaded or welded) in the top and bottom heads. If the plug is included as part of a canister, then the plug is seal welded prior to shipment. These plugs can be used, when necessary, for a number of functions, including draining, inerting, leak testing, venting, and remote inspection. Installation or removal of the plugs is performed only at the owner's site while the standardized canister is inside a hot cell since the containment feature of the canister depends upon the proper installation of the plug (DOE 1999b, Section 4.9).

Geometry—The large-diameter standardized canister has a nominal outer diameter of 24 in. and a nominal wall thickness of 0.5 in. The small diameter standardized canister has a nominal outer diameter of 18 in. and a nominal wall thickness of 0.375 in. Both the large- and small-diameter standardized canisters are designed for two nominal overall lengths of approximately 10 and 15 ft (DOE 1999a, Section 3).

Material—The standardized canisters are fabricated from SA-312 (welded or seamless pipe) Stainless Steel Type 316L (UNS S31603) for the shell. SA-240 (plate) Stainless Steel Type 316L is used for the heads and lifting rings. The optional plugs are SA-479 (bar) Stainless Steel Type 316L. The stainless steel materials are annealed and pickled. This low-carbon austenitic alloy is chosen for its corrosion resistance, American Society of Mechanical Engineers code-approved mechanical properties (ASME 2001, Section III, Division 3), and ductility over a wide range of temperatures (DOE 1999a, Section 3).

Weight—The maximum total allowable weight of each standardized canister plus its contents is approximately (DOE 1999a, Table 3-2):

- 10,000 lb for the 24-in.-diameter 15 ft standardized canister
- 9,000 lb for the 24-in.-diameter 10 ft standardized canister
- 6,000 lb for the 18-in.-diameter 15 ft standardized canister
- 5,000 lb for the 18-in.-diameter 10 ft standardized canister.

1.5.1.3.1.2.1.2 Standardized Canister Internals

The internal basket assemblies within standardized canisters have several functions. These functions are to facilitate loading of DOE SNF and to provide structural support of the DOE SNF during operations. The standardized canister internals also may serve a criticality control function in both the pre- and postclosure time periods (Montierth 2003, Section 1.2), as discussed later in this section. The internals are not classified ITWI.

The DOE fuel assemblies to be loaded into a canister set the pattern for the arrangement of the basket configurations within the standardized canister. The basket for each configuration is customized to meet physical dimensions, type, and number of fuel assemblies to be packaged in a standardized canister. Some baskets are preinstalled in the standardized canister prior to loading and final closure. Other baskets are preloaded with fuels, and the loaded basket is placed into the standardized canisters. Each DOE SNF type has been assigned to one of the nine criticality analysis groups (eight for standardized canisters and one for MCOs). Each criticality analysis group has a corresponding basket design that was used for the representative DOE SNF type from that group. The following summarizes basket designs for standardized canisters.

FFTF-Mixed Oxide Basket—A spoked-wheel basket contains five hexagonally shaped Fast Flux Test Facility standard driver fuel assemblies and one Ident-69 fuel pin container in the center; only five of the six basket compartments will be utilized for any fully loaded canister. The Ident-69 container will contain partially disassembled Fast Flux Test Facility assemblies and individual fuel pins from assemblies that had undergone postirradiation examination (DOE 2004b, Section 3.2.3). The length of the Fast Flux Test Facility fuels requires the use of the 15-ft-long, 18-in.-diameter standardized canister. This basket is shown in Figure 1.5.1-10.

Shippingport LWBR Basket—A rectangular basket contains one Shippingport Light Water Breeder Reactor seed assembly. The length of this fuel (11.66 ft) requires the use of the 15-ft, 18-in.-diameter standardized canister. The rectangular basket contains a box made of 3/8 in. plate and a 3/16 in. thick sleeve. This basket is shown in Figure 1.5.1-11.

Shippingport PWR Basket—A square basket contains one Shippingport PWR Core 2 single fuel assembly. The length of this fuel (approximately 104.5 in.) requires it to be placed within a 15-ft-long, 18-in.-diameter standardized canister. The square basket includes a box made of 3/8-in. plate and a 3/16-in.-thick sleeve. This basket is shown in Figure 1.5.1-12.

Enrico Fermi Basket—A large pipe bundle basket will hold Fermi fuel pins. Two large pipe bundle baskets will be placed inside a 10-ft-long, 18-in.-diameter standardized canister. The large pipe bundle basket includes eleven 4-in. tubes just over 43 in. long resting on a base plate. The basket design includes a 3/16-in.-thick sleeve which retains poison beads for long-term criticality control. This basket is shown in Figure 1.5.1-13.

TRIGA Basket—A small pipe bundle basket will hold TRIGA fuel. Three small pipe bundle baskets could be placed inside of a 10-ft-long, 18-in.-diameter standardized canister. The small pipe bundle basket includes thirty-one 2-in. pipes, 33 in. long, resting on a base plate. The basket may also include an optional 3/16-in.-thick sleeve. This basket is shown in Figure 1.5.1-14.

Fort St. Vrain Basket—The Fort St. Vrain fuel assembly is hexagonal in shape and 31.22 in. long. Five of these assemblies could be placed within a 15-ft-long, 18-in.-diameter standardized canister. This basket is shown in Figure 1.5.1-15.

Three Mile Island Unit 2 Basket—The Three Mile Island Unit 2 canisters are 150 in. long and fit within a cylinder of 14.25 in. in diameter. Each Three Mile Island Unit 2 canister is to be placed within a 15-ft-long, 18-in.-diameter standardized canister. The basket is a pipe basket consisting of a 16-5/8-in. by 3/16-in.-thick sleeve with four 3/8-in. spacer pipes (or rods), within which the Three Mile Island Unit 2 canister is placed. This basket is shown in Figure 1.5.1-16.

Advanced Test Reactor Basket—A rectangular grid basket will hold the following aluminum fuels: Advanced Test Reactor, Oak Ridge Research Reactor, Massachusetts Institute of Technology, University of Missouri Research Reactor, and Peach Bottom. This rectangular grid basket will be used in 10-ft- and 15-ft-long (depending on fuel and loading) 18-in.-diameter standardized canisters. The basket consists of several 3/8-in. plates welded together to form a rectangular grid with an integral 0.062-in.-thick sleeve. This basket is shown in Figure 1.5.1-17.

Further details for the evaluations of the above baskets are found in the appendices of *Design Consideration for the Standardized DOE SNF Canister Internals* (DOE 2006, Appendices B through I).

Structural Support—The standardized canister fuel basket is designed to remain intact and to provide relative geometry control of the fuel debris that might be formed from a drop event or other handling operations (Montierth 2003, Forward).

Criticality Control—The canister internal fuel basket sets the number of assemblies that can be loaded, which controls the amount of fissile materials in a canister. As required to provide criticality control, supplemental neutron absorber materials are added to the internal basket design. Basket materials may include either stainless steel baskets with or without supplemental neutron-absorbing materials and Ni/Gd alloy material with or without supplemental neutron-absorbing materials (DOE 2004b, Sections 3.1.4 and 3.1.5).

1.5.1.3.1.2.1.3 Multicanister Overpack Shell

The MCO is a canister used for SNF (N Reactor, Shippingport PWR Core 2 blanket, and Single Pass Reactor) at the Hanford site. There are expected to be approximately 400 MCOs (DOE 2004a, Volume 1, Appendix F, Table F-1). Most of the MCOs have been fabricated and loaded. The MCO is designed to allow loading and stacking of five or six N Reactor fuel baskets within its cavity, depending upon the fuel type. Under normal conditions, there is a 1 in. nominal distance between the underside of the shield plug and the fuel of the top basket (Garvin 2002, Sections 1.2 and 1.3).

The MCO shell, which provides moderator control, is a cylindrical vessel with access at the top end that is closed with a shield plug. The shell bottom assembly is flat and includes an internal liquid collection sump at the centerline, used during the drying process. The shell bottom assembly is welded to the cylindrical shell wall. The welds joining the cylindrical shell to the bottom plate of the MCO are full-penetration circumferential welds and are examined by radiography and dye penetrant. MCO welds are performed and examined in accordance with the American Society of Mechanical Engineers code (ASME 1998), Section III, Division 1, Subsection NB. The MCO is shown in Figure 1.5.1-18.

The MCO top closure assembly includes a shield plug, locking ring, and cover cap. The shield plug protects workers from ionizing radiation and confines radioactive material during the drying and closure process. A threaded locking ring is screwed into the MCO collar to hold the shield plug in place. Set screws in the locking ring are tightened to hold a silver-clad seal between the MCO shell and the shield plug. The shield plug has four processing ports used for the drying process. Two of the ports are connected to internal high-efficiency particulate air filters for filtering the exhaust. After drying, a cover cap is welded over the shield plug and locking ring for final sealing. The MCO handling interface is a lifting ring on the cover cap. The cover cap has an integral centering-backing ring built into the cap design, so the placement and centering of the cover cap on the MCO collar is a simple process. When the cover cap is on the MCO, the cover cap penetration allows helium to be put under the cover cap so the attachment weld may be leak-rate tested using a test collar on the outside and the flow pumped to a helium mass spectrometer (this is an essential function of the cover cap penetration) (Garvin 2002, Sections 1.2 and 1.3).

The MCO design for the Hanford Shippingport Core 2 blanket SNF is identical to the N Reactor MCO, except that there is only one port in the shield plug rather than four, the shield plug is thicker, and the internal filtration mechanism has been removed to provide additional cavity space to accommodate the assembly length (Fluor Hanford 2003, Section 1). Less than 5% of the MCOs contain Hanford Shippingport Core 2 blanket SNF (DOE 2004a, Volume 1, Appendix F, Table F-1). Additionally, the MCOs for the Hanford Shippingport Core 2 blanket SNF use a cruciform-like insert for positioning the fuel. The design of the Hanford Shippingport MCO conforms to the performance specification for Shippingport SNF canisters (Fluor Hanford 2003). Unless specifically noted, discussions of MCOs apply to both the N Reactor MCOs and the Hanford Shippingport MCOs.

Geometry—Each MCO is a stainless steel canister having a shell diameter of approximately 24 in. and a closure diameter at the widest point of 25.51 in. and approximately 166 in. long. The MCO shell is a cylindrical vessel fabricated from 0.5-in. stainless steel welded to a 2-in. bottom plate assembly (Garvin 2002, Section 1.2; DOE 2008d, Figure C-5, Notes 5 and 6).

Material—Materials are specified as American Society of Mechanical Engineers (SA-182) or American Society for Testing and Materials (A) materials. The locking ring is made from Stainless Steel Type 304H or 304N (Garvin 2002, Section 3.1.10.1). The MCO shell, collar, bottom, and shield plug are Stainless Steel Type 304/304L dual-certified material. This low-carbon austenitic alloy was chosen for its corrosion resistance, American Society of Mechanical Engineers code-approved mechanical properties, and excellent ductility over a wide range of temperatures. No ferritic materials are used in the design (Garvin 2002, Section 1.3).

Weight—The maximum dry weight of an N Reactor MCO, including the heaviest fuel arrangement, is 20,220 lb (Garvin 2002, Table 3-6). The weight for an MCO containing Shippingport Core 2 blanket SNF is about 9,525 lb (Fluor Hanford 2003, Section 4.9.3).

1.5.1.3.1.2.1.4 Multicanister Overpack Internals

The internal basket configuration for MCOs provides the same functions as the internal baskets for DOE SNF standardized canisters. The MCO basket array involves two design variants to accommodate the Mark IA and Mark IV N Reactor fuel and one to accommodate Shippingport PWR Core 2 blanket assemblies. The MCO internals are based on the physical dimensions of each of these types of DOE SNF:

- **Mark 1A Basket**—The basket design for the Mark 1A (Figure 1.5.1-19) fuels uses a six-high, stacked-basket design inside the MCO. The MCO basket designs also include a basket for fuel debris that can be installed in the top position in the MCO basket stack. A second basket for fuel debris can be placed in the bottom position in the MCO basket stack. Each Mark IA basket contains up to 48 N Reactor assemblies, and a loaded MCO can contain as many as 288 assemblies (DOE 2004b, Section 3.2.2.1). Loading of the basket for fuel debris varied based on a variety of factors, including fuel surface area and fissile material content (Fluor Hanford 2005). The Mark 1A fuel baskets were modified to contain an insert to permit single pass reactor fuel to be stacked two to three elements high to ensure efficient packing densities and permit all of this fuel to fit into a single MCO (Garvin 2002, Section 2.1). The single pass reactor basket, as modified from the Mark 1A basket, is shown in Figure 1.5.1-20.

- **Mark IV Basket**—The Mark IV basket, shown in Figure 1.5.1-21, uses a variant of the Mark IA basket design in that it is taller, so it can only be stacked five high in the MCO. The MCO basket designs also include a basket for fuel debris that can be installed in the top position in the MCO basket stack. A second basket for fuel debris can be placed in the bottom position in the MCO basket stack. Each Mark IV fuel basket contains up to 54 assemblies, and a loaded MCO with only fuel baskets can contain as many as 270 assemblies (DOE 2004b, Section 3.2.2.1.1). Loading of the basket for fuel debris varied based on a variety of factors, including fuel surface area and fissile material content (Fluor Hanford 2005).

- **Shippingport Basket**—A cruciform-like insert is used to position four Shippingport PWR Core 2 blanket assemblies in an MCO. The basket allows for 0.5 in. of SNF bowing in any direction (Fluor Hanford 2003, Section 4.12). The Shippingport blanket insert is shown in Figure 1.5.1-22.

Structural Support—There is no requirement for the MCO fuel baskets to provide geometry control of the fuel.

Criticality Control—Prior to receipt and acceptance of MCOs, criticality safety analyses of MCOs containing SNF will be performed to demonstrate compliance with the criticality safety requirements in Section 1.14.2.1.

1.5.1.3.1.2.2 Operational Processes

Waste acceptance procedures will be developed and implemented to ensure that the shipper loads each DOE SNF canister such that the thermal, shielding, and criticality characteristics of the received waste are within acceptable limits for the repository.

1.5.1.3.1.2.2.1 Standardized Canisters

Mixing of SNF—Different types of fuel may be mixed within a standardized canister. As with all DOE SNF (including mixed fuel), the basis shall be provided for ensuring subcriticality at the time of delivery to the geologic repository and during all subsequent handling operations (DOE 2008b, Section 4.3.8). Many fuels by themselves will only result in a partial canister fill. Therefore, other fuels within the same fuel group category and those that fall within the space constraints may be mixed in to fill those basket positions. Canisters containing mixed fuel types from different fuel group categories will be qualified by analysis for intact fuel and basket combinations at the time of loading (DOE 2004b, Summary, pg. 8 of 122).

Thermal Control—Section 1.5.1.3.2 presents the thermal characteristics of DOE SNF. All systems designed to handle DOE standardized canisters during normal operations shall ensure that canister wall temperatures do not exceed 316°C in enclosed environments and 149°C in open (air) environments. The thermal loading of standardized canisters is within the waste package limits; therefore, control of waste package loading to meet thermal limits is not necessary (DOE 2008d, Section 10.1.3).

Criticality Control—Operational processes and procedures ensure that the appropriate amount and type of fuel is loaded into the correct basket configuration and that the SNF in the canister is dried inside, filled with an inert gas, and sealed. These processes will ensure that the canisters are loaded in a configuration that has been analyzed and accepted for criticality safety. The operational processes follow standard industry practices. The canister criticality analysis process is summarized in Section 1.14.2.2.

Drying—A pressure rebound test or the equivalent is performed prior to inerting and sealing. A pressure rebound test involves maintaining a constant pressure over a period of 30 minutes without vacuum pump operation. This test is consistent with industry standards (BSC 2004a, Section 4.4).

Inerting—After loading and drying the SNF, the standardized canisters are filled with an inert gas, such as helium, prior to sealing (DOE 1999a, Section 3.2.5).

Sealing—The standardized canister boundary components are joined with full-penetration welds that meet the requirements of the *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001), Section III, Division 3, Subsections WA and WB. The closure weld attaches the standardized canister top head to the main body and is applied after the DOE SNF has been loaded. If needed, a clamping device is used to minimize ovalization of the standardized canister shell to ensure proper head fit. The top head closure weld, which is a simple butt weld, is made using a vessel head that has a permanent backing ring attached. The backing ring helps with weld fit and also protects the standardized canister contents during welding. Volumetric inspection of the closure weld is achieved using ultrasonic testing. Additionally, the standardized canister final closure weld is implemented using an American Society of Mechanical Engineers-acceptable welding procedure. Prior to transportation to the repository, any threaded plugs are installed and seal welded in place in order to establish an American Society of Mechanical Engineers-acceptable containment boundary (BSC 2004a, Section 3.1).

Leak Test—To demonstrate leak tightness, the standardized canister is helium leak-tested in accordance with *2001 ASME Boiler and Pressure Vessel Code*, Section V, Article 10, Appendix IV (ASME 2001; DOE 1999a, Section 3.2.5).

1.5.1.3.1.2.2.2 Multicanister Overpack

Compliance with MCO interface requirements will be achievable for any SSC that relies on the performance of the canister pressure vessel boundary. This includes activities from the loading of Environmental Management SNF into disposable canisters through final closure of the waste package.

All systems designed to handle the MCOs during normal operations shall ensure that canister wall temperatures do not exceed 132°C in either enclosed or open (air) environments (DOE 2008d, Section 10.2.3).

Thermal Control—Section 1.5.1.3.2 presents the thermal characteristics of DOE SNF. The thermal loading of the MCOs is within the waste package limits; therefore, control of waste package loading to meet thermal limits is not necessary.

Criticality Control—Operational processes and the physical design of the fuel basket ensure that the appropriate amount and type of fuel is loaded into the correct basket configuration and that the canister is dried inside, filled with an inert gas, and sealed. These processes will ensure that the canisters are loaded in a configuration that has been analyzed and accepted for criticality safety. Because of the low enrichment of the N Reactor, Single Pass Reactor, and Shippingport Core 2 Blanket fuels, no supplemental neutron absorber material is required in the basket designs for criticality control (DOE 2004b; Fluor Hanford 2003, Sections 4.0 and 4.12). Loading plans are generated, checked, and approved prior to loading operations being performed, and independent reviews of the loading operations are performed.

Drying—N Reactor MCOs are loaded underwater, drained, and vacuum dried at the Hanford site. The drying and verification process for the N Reactor MCOs is a four-step process consisting of (1) draining and cold vacuum drying, (2) initial pressure rebound test, (3) pressure rise proof test, and (4) final pressure rebound test (BSC 2004a, Section 3.2). Details of the drying process are provided in the SNF product specification (Fluor Hanford 2005). The N Reactor MCO drying process ensures that the remaining free water is less than 200 g (Garvin 2002, Section 9.1).

The Shippingport Core 2 blanket SNF is loaded dry and vacuum dried. A pressure rebound test (vacuum hold test) is used to verify dryness, consistent with NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997; BSC 2004a, Section 4.4).

Inerting—After loading and drying the SNF, the MCOs are filled with an inert gas, such as helium, prior to sealing (BSC 2004a, Section 3.2).

Sealing—The MCO cover cap is placed on the collar. The inert gas is introduced underneath the cover cap through the penetration. Welding begins with an autogenous root pass. The complete autogenous root pass is dye penetrant examined. Multiple weld passes are laid down, and another dye-penetrant examination may be performed at the midpoint of the weld. Additional weld passes are then laid to complete the welding. Following the final weld pass, the newly created cavity is vacuum pumped to ensure dryness under the cover cap, and the chamber is refilled with helium. The test plug is installed, and the weld and plug are leak rate tested. The weld is subjected to a final dye-penetrant examination in accordance with American Society of Mechanical Engineers Code Case N-595 (ASME 1998). The penetration is permanently closed with a 4 in. diameter cover plate that is welded into the cap penetration hole (Garvin 2002, Section 1.3.2).

1.5.1.3.1.2.3 SSCs Important to Safety and Important to Waste Isolation

Section 1.7.2.3 describes the determination of passive structure, system, or component reliability and discusses several types of failures for passive SSCs in the PCSA, including the structural challenge causing loss of containment (breach or fire) of a waste form container (e.g., DOE SNF canister).

The DOE SNF canister that performs a safety function during preclosure handling, and the associated controlling parameters and values for operations, are addressed in Table 1.5.1-25.

The DOE canister internals (neutron absorbers) are classified as ITWI as described in Table 1.9-8. The DOE SNF cladding and canister are classified as non-ITWI components and there are no postclosure nuclear safety design bases requirements identified for the TSPA analyses.

The DOE SNF canisters provide containment of radioactive materials during repository waste handling activities from the point that the transportation cask is opened through the closure of the waste package. Due to this preclosure containment safety role, the DOE SNF canisters are classified as ITS. Event sequences involving DOE SNF canisters are shown to have a likelihood of radioactivity release that is less than one in 10,000 during the preclosure operational period, making the breach of a DOE SNF canister a beyond Category 2 event sequence (BSC 2004a, Section 1).

No credit is taken for the DOE SNF canister shell integrity after the SNF is sealed inside a waste package. The DOE SNF canister is not relied on for postclosure barrier performance.

1.5.1.3.1.2.4 Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

There are no procedural safety controls for DOE SNF canisters.

1.5.1.3.1.2.5 Design Criteria and Design Bases

Design bases for the DOE SNF canisters are presented in Table 1.9-4. This section identifies the design criteria for the DOE SNF canisters. For preclosure, the DOE SNF design criteria are provided in the DOE SNF preclosure nuclear safety design basis, Table 1.5.1-25.

1.5.1.3.1.2.5.1 Structural Design Criteria

ITS—The integrity of the DOE SNF canister is relied upon for safety during the preclosure period to maintain containment for a spectrum of drop accident sequences during waste handling operations (BSC 2004a, Section 1). The DOE SNF canisters are designed and analyzed to demonstrate that they can withstand drops from the design basis conditions identified in Table 1.9-4 with the mean conditional probability of breaching being a beyond Category 2 event sequence.

ITWI—No credit for containment is taken for the DOE SNF canister shells after SNF is sealed inside the waste package.

1.5.1.3.1.2.5.2 Criticality Control Design Criteria

ITS—During the preclosure period, the DOE standardized canister and MCO shells are considered ITS because they provide moderator control for prevention of criticality and containment for prevention of radionuclide release during normal operations (Table 1.9-4).

ITWI—For a criticality to occur during postclosure, moderator must be introduced into an already breached canister. In this case, waste isolation has already failed and therefore there is no context for an ITWI categorization.

1.5.1.3.1.2.6 Design Methodologies

1.5.1.3.1.2.6.1 Structural Design

The DOE SNF canisters are designed to the *1998 ASME Boiler and Pressure Vessel Code* (ASME 1998) for expected conditions, but the American Society of Mechanical Engineers code does not address drop conditions. NRC interim staff guidance on alternatives to the American Society of Mechanical Engineers code (NRC 2000b) specifies that use of the American Society of Mechanical Engineers code for dry SNF storage systems may be implemented with allowance for some alternatives to its requirements. The American Society of Mechanical Engineers code alternatives utilized in the canister design, fabrication, and closure include allowing some field operations (e.g., final closure welds) rather than shop fabrication, N-stamping prior to SNF loading, use of helium leak tests in lieu of pressure tests, and the absence of pressure relief devices. Because the American Society of Mechanical Engineers code does not address drop events, alternative methods are used to demonstrate DOE SNF canister survival from drops. The alternate methods used to address drop events for DOE SNF canisters include a combination of analyses and drop tests (BSC 2004a, Section 5.2).

Structural response analyses of the DOE SNF canisters are performed for standardized canisters (Blandford 2003) and for MCOs (Snow 2003; Snow 2004). These analyses are performed using ABAQUS/Explicit, which is a nonlinear, finite-element software package widely used in many industries (Blandford 2003; Snow 2003; Snow 2004). ABAQUS/Explicit, Version 6.3-3 will be used to determine the structural response for the 24 in. diameter standardized canister. Version 6.3-3 is an updated version of the software used to evaluate the 18 in. diameter canister (Version 5.8-1) and is the current NSNFP validated version. Modeling methodology previously used in the 18-inch diameter canister analytical evaluation will be used except where changes are required to comply with Version 6.3-3 and computer program validation requirements.

The analysis methodology to be used for the comparative analytical evaluations will be similar to that used for the 1999 drop test effort. A solid model is first developed using appropriate software. The actual ABAQUS/Explicit FE model is generated and then subjected to rigorous checks to assess adequacy before any actual analysis is performed. This rigorous checking process eliminates the need to control or validate the solid modeling software (Blandford 2003, Section 2). Structural analyses of the DOE standard canister and MCO were performed assuming a normal temperature of 70°F (Blandford 2003; Snow 2003). The drop tests performed at ambient temperature not only demonstrated canister performance but also served to validate the analytical models used to calculate strains for repository-defined drop scenarios at maximum temperature. These analyses have shown that the resulting strains are well below values where failure would be expected.

A breach (through-wall fracture) of a containment boundary made of highly ductile steel, such as Stainless Steel Type 304 or Stainless Steel Type 316, is characterized by tearing of material, accompanied by appreciable gross plastic deformation and expenditure of considerable energy. The likelihood of ductile tearing is limited for the highly ductile plates subjected to a displacement-controlled bending, which is the case for the DOE SNF canisters. The tensile tearing is governed primarily by the through-wall (membrane) strains (BSC 2004a, Section 5.2.1). For the standardized canisters, a conservative through-wall strain limit (i.e., the average strain across the wall thickness) of 48% is used to evaluate standardized canister containment capability (Blandford 2003, Sections 8.1 and 9.0).

For the MCOs, a conservative through-wall strain limit of 47% is used to evaluate MCO containment capability (Snow 2003, Sections 8.2.3, 8.3.3, and 8.4.3). The minimum elongation for the MCO shell is 59%, based on tests of two MCO main shells; therefore, the strain limit used to evaluate MCO survival for puncture drops is 59% (Snow 2004, Section 8.4).

To demonstrate that the standardized canisters can survive Category 2 drop event sequences and to validate the analytical approach, a series of standardized canister drop tests have been performed. These drop tests were performed at Sandia National Laboratories. Full-scale 18 in. diameter standardized canisters have been tested for the relevant events identified in Table 1.5.1-26. The tests are summarized in Section 1.5.1.3.1.2.9 (Morton et al. 2002, Part I). The deformation patterns predicted in the analyses are consistent with those of the test canisters, and the magnitudes of the deformations in the analyses are consistent with those of the tests. Figures 1.5.1-23 to 1.5.1-28 show the predicted and actual test deformations for three drop events of an 18-in.-diameter standardized canister.

1.5.1.3.1.2.6.2 Criticality Design

During the preclosure period, the DOE standardized canister and MCO shells are considered ITS because they provide moderator control. Criticality safety requirements and criteria are discussed in Section 1.14.2.1.

For postclosure analysis of DOE SNF, analyses have been performed by following the methodology documented in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003), as further discussed in Section 2.2.1.4.1.1. Due to the variety of DOE SNF, DOE designated nine representative fuel groups for disposal criticality analyses discussed in Section 1.5.1.3.1.1 and identified in Table 1.5.1-23. For each representative waste form, a comprehensive evaluation of various states of degradation from fully intact to fully degraded have been evaluated (BSC 2004b; BSC 2004c) with criticality control design limits set based on maintaining subcriticality for the most restrictive degraded scenario, for each criticality group. Waste forms within a single criticality group that have configurations or key criticality parameters outside the range of applicability of the representative fuel will require supplemental analysis and/or additional reactivity control mechanisms.

1.5.1.3.1.2.7 Consistency of Materials with Design Methodologies

The stainless steel materials of construction for the DOE SNF canisters are selected because of their resistance to degradation in the disposal environment. Chemical, galvanic, and other reactions are considered in the material selection process, consistent with NRC guidance (NRC 2001, Section X.5.3.1; DOE 1999b, Section 4.7.2.

In addition to selection of degradation-resistant materials, DOE SNF canister contents are verified to be dry to ensure that material interactions do not degrade the DOE SNF canisters. After loading, the DOE SNF canisters are filled with an inert gas, such as helium (BSC 2004a, Section 5.4).

DOE SNF canister degradation is negligible as a result of the use of degradation-resistant materials, drying and verification, and inerting. The DOE SNF canisters perform consistent with their design bases (BSC 2004a, Section 5.4).

Supplemental neutron absorber materials may be included as part of the design of the internal components of the DOE SNF canister if analyses of the specific waste loading requires it. These materials are intended to prevent in-package criticality in the unlikely event that a breached waste package becomes flooded at some time after closure.

1.5.1.3.1.2.8 Design Codes and Standards

The standardized canisters are N-stamped, demonstrating compliance with the *1995 ASME Boiler and Pressure Vessel Code* (ASME 1995). The editions of the American Society of Mechanical Engineers code used for the preliminary design of the standardized canisters are specified in *Preliminary Design Specification for Department of Energy Standardized Spent Nuclear Fuel Canisters* (DOE 1999a, Volume I, Section 2). The MCOs are N-stamped, demonstrating compliance with the *1998 ASME Boiler and Pressure Vessel Code* (ASME 1998). The editions and specific portions of the American Society of Mechanical Engineers code applicable to the MCOs are specified in topical reports (Garvin 2002, Sections 1.2 and 1.3; BSC 2004a, Section 3). American Society of Mechanical Engineers code compliance includes design, materials, and fabrication, including welding, examination, and testing.

1.5.1.3.1.2.8.1 Standardized Canisters

The code requirements applicable to the standardized canisters include the following (DOE 1999a, Sections 2, 3.2.1, and 3.2.5):

- *ASME Boiler and Pressure Vessel Code*, Section III, Division 3, 1997 Edition for design, fabrication, and examination

- *1995 ASME Boiler and Pressure Vessel Code* (ASME 1995), Section V, Article 10, Appendix IV, 1995 Edition, 1997 Addenda for leak testing.

The references to other code versions for the standardized canister above are from a preliminary design specification which provided the applicable codes used for the prototype canisters and specified that the most-current approved code would be used for actual canister design. Subsequent canister design has been completed using the 1998 version, as specified in Section 1.5.1.3.1.2.6.1.

1.5.1.3.1.2.8.2 Multicanister Overpacks

The code requirements applicable to the MCO from the MCO topical report include the following (Garvin 2002, Sections 1.2 and 1.3.2):

- *1998 ASME Boiler and Pressure Vessel Code* (ASME 1998), Section III, Division 1, Subsection NB, 1998 Edition for design, fabrication, inspection, and examination, with ASME Code Case -595, Revision 3, invoked for the final closure welds

- *1998 ASME Boiler and Pressure Vessel Code* (ASME 1998), Section III, Division 1, Subsection NG, is applied to the design of the baskets.

In addition, the code requirement for MCO leak testing in the MCO fabrication specification is as follows (Lucas 2002, Section 7.4.1): *1998 ASME Boiler and Pressure Vessel Code* (ASME 1998), Section V, Article 10, Appendix V.

1.5.1.3.1.2.9 Design Load Combinations

The DOE SNF canisters are designed for combinations of temperature, pressure, content, and drop loads. The drop loads present the greatest challenge to the integrity of the DOE SNF canisters. Analyses and tests have been performed to evaluate the structural response of the DOE SNF canister to drop events. At the time of acceptance into the Civilian Radioactive Waste Management System, disposable multielement canisters shall be capable of sustaining a flat-bottom drop from a height of 23 ft and a drop in any orientation from a height of 2 ft (individually—not both in sequence) onto an essentially unyielding surface, without releasing radioactivity exceeding the applicable limits (DOE 2008b, Section 4.3.5). Table 1.5.1-26 identifies the drop events evaluated for each DOE SNF canister configuration. The results of these analyses are summarized here and reported in detail in

Structural Response Evaluation of the 24-Inch Diameter DOE Standardized Spent Nuclear Fuel Canister (Blandford 2003) for 18-in. and 24-in. standardized canisters and in *Analytical Evaluation of the MCO for Repository-Defined and Other Related Drop Events* (Snow 2003) and *Analytical Evaluation of the MCO for Puncture Drop Events* (Snow 2004) for MCOs. Table 1.5.1-27 presents the peak equivalent plastic strains for each of the analyses. Unless otherwise noted, strain values reported for DOE SNF canisters are peak equivalent plastic strains. The values presented in Table 1.5.1-27 are the maximum strains occurring anywhere in the containment boundary of the standardized canisters and MCOs. Strains in the energy-absorbing skirt of the standardized canisters may be substantially greater than the strains in the containment boundary.

As shown in Table 1.5.1-27, the standardized canister through-wall strains for the 2-ft drop, the 23-ft drop, and the puncture events do not exceed the 48% through-wall strain limit for 316L stainless steel. The midplane strains are less than half the 48% limit for all drop events (BSC 2004a, Section 5.2). Therefore, the 18-in.-diameter standardized canister and 24-in.-diameter standardized canister containment boundaries remain intact for the drop events that have been analyzed deterministically (BSC 2004a, Section 5.2).

The analyzed strains for the MCO flat surface drop event do not exceed the 47% through-wall strain limit. The midplane strains are less than half the 47% limit for the identified drop events (BSC 2004a, Section 5.2). For the 23 ft edge-to-collar drop, with the MCO falling and its collar catching on the upper edge of the waste package, the peak surface strain of 130% exceeds the 47% value in a small region of the outer surface. This value indicates a large, localized plastic distortion of the outside surface in the region of a structural discontinuity. The MCO wall in this collar region is about double the nominal wall thickness for the MCO. The strain for the nominal wall thickness, meaning the midpoint to the inner surface, is less than or equal to 17%, well below the 47% through-wall strain limit. Therefore, the MCO containment boundary is predicted to remain intact for this edge-to-collar drop event as analyzed deterministically (BSC 2004a, Section 5.2).

The conclusion drawn from the deterministic analyses results presented in Table 1.5.1-27 is that, while deformation may occur, the standardized canisters and MCOs survive the repository facility drops. This conclusion is drawn because the through-wall strains are well below material through-wall strain limits: 48% for the standardized canisters and 47% for the MCOs (BSC 2004a, Section 5.2).

The conclusion of standardized canister survivability deterministic analyses for the drop events is confirmed by the tests performed at Sandia National Laboratories. The actual field drop tests on the 18-in. and 24-in. diameter standardized canisters and MCOs are documented in *FY1999 Drop Testing Report for the Standardized 18-Inch DOE SNF Canister* (Morton et al. 2002), *Drop Testing Representative 24-Inch Diameter Idaho Spent Fuel Project Canisters* (Morton and Snow 2005a), and *Drop Testing Representative Multi-Canister Overpacks* (Morton and Snow 2005b), respectively. These tests show that all the canisters survived the events identified in Table 1.5.1-26 with deformation but without a loss of confinement. Pressure tests (50 psig) and helium leak tests were performed to confirm the containment boundary remained intact (BSC 2004a, Section 5.2.2).

However, to demonstrate compliance with 10 CFR Part 63, probabilistic analyses are needed for all canisters handled at the repository, in order to demonstrate their safety function to provide containment in the event of a vertical drop on an unyielding surface. The probabilistic analyses of the mean frequency of a breach resulting from a potential drop while removing an MCO from the transportation cask or loading an MCO into the DOE codisposal waste package may require design details that are not yet available. Details such as controls for lift and alignment of the MCO during transfer operations to limit the drop of an MCO, and/or the incorporation of energy absorbing materials to mitigate the impacts of a drop of an MCO in either the waste package or transportation cask will be evaluated as the designs determined to be necessary become available. MCOs will be accepted for disposal at the repository when the design details, event sequence, and reliability analyses needed to determine the nuclear safety design bases for the MCOs are completed and establish that the MCOs can be safely received and handled at the repository during the preclosure period. The processes prescribed in 10 CFR 63.22 and 10 CFR 63.46 will be used, as appropriate, to obtain authorization to receive DOE SNF in MCOs once these safety analyses are completed.

The postclosure analyses have assumed the N Reactor fuel will be disposed of in the repository in codisposal waste packages with a configuration of 2 MCOs and 2 DHLW canisters in each waste package. The N Reactor fuel waste form, although only comprising about 200 waste packages (about 5% of the codisposal waste packages), constitutes the vast majority of the DOE SNF MTHM mass allocation in the repository (DOE 2007, Table 5) and therefore is potentially significant to postclosure performance. This configuration has been addressed in the postclosure performance assessment by analyses of specific excluded FEPs related to DOE SNF as presented in Section 2.2.1 and by the inclusion of this DOE SNF waste form in the assessment of radionuclide inventory, in package chemistry, and DOE SNF waste form degradation in the codisposal waste packages as presented in Sections 2.3.7.4, 2.3.7.5, and 2.3.7.8, respectively.

1.5.1.3.1.2.10 Weld and Material Flaws and Degraded Canisters

Undetected flaws in a canister that might lead to a canister breach in a drop event are highly unlikely due to the materials used, as well as the fabrication processes, controls, and examinations (BSC 2004a, Section 5.3.3). The closure welds for the standardized canisters and MCOs include multiple-pass welds with multiple examinations, thereby making undetected flaws greater than one weld bead deep highly improbable. This approach is similar to the approach accepted by the NRC in Interim Staff Guidance–18 (NRC 2003b). The standardized canister invokes a volumetric inspection of the closure weld performed by ultrasonic inspection.

Therefore, undetected flaws in welds in either standardized canisters or MCOs are not a significant contributor to the likelihood of canister breach. The ultrasonic examination of each weld pass makes flaws especially improbable for the standardized canisters. In addition, the mechanical seal below the closure weld provides defense in depth in the improbable event of an MCO closure weld failure (BSC 2004a, Section 5.3.3).

After drying, degradation of standardized canisters and MCOs will be negligible. The drying process for the standardized canisters and Shippingport MCO will include at least cold vacuum drying with a single pressure rebound test or equivalent (BSC 2004a, Section 5.4). The drying process for the K Basin MCOs includes alternating vacuum and pressure cycles followed by an initial pressure rebound test and a pressure rise proof test (BSC 2004a, Section 3.2).

1.5.1.3.2 Thermal Characteristics of DOE SNF

The decay heat from DOE SNF is based on calculated radionuclide inventories. The radionuclide inventory is multiplied by the appropriate curie-to-watt conversion factor to obtain the decay heat for each radionuclide. The total decay heat is calculated by summing the decay heat for each radionuclide. Table 1.5.1-28 provides the estimated total thermal power in 2010 and 2030 for the DOE SNF to be disposed at Yucca Mountain. The total heat generation rate in DOE SNF canisters shall be less than 1,970 W (per canister) (DOE 2008b, Section 4.3.9).

1.5.1.3.3 Nuclear Characteristics of DOE SNF

Process knowledge and the best available information regarding fuel fabrication, operations, and storage for DOE SNF is used to develop a conservative source-term estimate. The DOE SNF characterization process relies on precalculated results that provide radionuclide inventories for typical SNF at a range of decay times. These results are used as templates that are scaled to estimate radionuclide inventories for other similar fuels.

The templates are generated using ORIGEN-based computational techniques described in *Methodologies for Calculating DOE Spent Nuclear Fuel Source Terms* (DOE 2000b), which includes discussion about and references to relevant experimental data and validation studies. A process for creating a conservative estimate of these source terms was developed by a team of experts representing each of the DOE SNF storage sites. The process relies on precalculated results that provide radionuclide inventories for typical SNFs at a range of decay times. These results are used as templates that are scaled to estimate radionuclide inventories for other similar fuels. The templates were generated using ORIGEN-based calculational techniques described in *Methodologies for Calculating DOE Spent Nuclear Fuel Source Terms* (DOE 2000b), which includes discussion and references to relevant experimental data and validation studies. Additional validation studies have been performed that further demonstrate the validity of the model and underlying codes (DOE 2004a, p. 14). One or more templates are developed for each of the following: Fast Flux Test Facility, Fermi, Fort St. Vrain, N Reactor, High-Flux Beam Reactor, Advanced Test Reactor, Pathfinder, Shippingport Light Water Breeder Reactor, commercial PWR, TRIGA, and a bounding composite. By modeling various combinations of reactor moderator, fuel enrichment, fuel compound, and fuel cladding, templates have been developed to reasonably model a broad range of DOE SNF. A template contains precalculated (i.e., ORIGEN output) radionuclide inventories at each of 10 specified decay periods, ranging from 5 to 100 years following irradiation. Templates include 145 radionuclides that account for over 99.9% of the total curie inventory. To conservatively estimate source terms for fuels that do not fit well within the precalculated templates or when sufficient information is not available to determine the appropriate template, a bounding composite template is used (DOE 2004a, Sections 2 and 5).

To estimate an SNF source term, the appropriate template is selected to model the production of activation products and transuranics by matching the reactor moderator and fuel cladding, constituents, and beginning-of-life enrichment. Precalculated radionuclide inventories are extracted from the appropriate template at the desired decay period and then scaled to account for differences in fuel mass and specific burnup (DOE 2004a, Section 6). Table 1.5.1-29 lists the projected radionuclide inventory of DOE SNF for the nominal and bounding cases as of 2010.

The radionuclide inventory of DOE SNF is not used in the PCSA because there are no normal operations or event sequences that result in a release from DOE SNF canisters. Therefore, preclosure consequence analyses for DOE SNF are not performed.

1.5.1.3.4 Source-Term Characteristics of DOE SNF

The shielding and TSPA source-term characteristics for DOE SNF are discussed in greater detail in Sections 1.10 and 2.3.7, respectively. Since the approach to the breach of the DOE SNF canister is determined to be a beyond Category 2 event sequence, no source term has been developed for the purpose of calculating preclosure onsite or offsite doses due to releases from the canisters.

1.5.1.3.5 Conformance of Design to Criteria and Bases

The nuclear safety design bases for ITS and ITWI SSCs and features are derived from the PCSA presented in Sections 1.6 through 1.9 and the postclosure performance assessment presented in Sections 2.1 through 2.4. The nuclear safety design bases identify the safety function to be performed and the controlling parameters with values or ranges of values that bound the design.

The quantitative assessment of event sequences, including the evaluation of component reliability and the effects of operator action, is developed in Section 1.7. Any SSC or procedural safety control appearing in an event sequence with a prevention or mitigation safety function is described in the applicable design section of the SAR.

Section 1.9 describes the methodology for safety classification of SSCs and features of the repository. The tables in Section 1.9 present the safety classification of the SSCs and features, including those items that are non-ITS or non-ITWI. These tables also list the preclosure and postclosure nuclear safety design bases for each structure, system, or major component.

The design criteria are specific descriptions of the SSCs or features (e.g., configuration, layout, size, efficiency, materials, dimensions, and codes and standards) that are utilized to implement the assigned safety functions. Table 1.5.1-25 presents the nuclear safety design bases and design criteria for the DOE SNF canisters. There are no derived requirements and associated design solutions for the ITWI function of the DOE SNF canisters.

1.5.1.4 Naval SNF

[NUREG-1804, Section 2.1.1.2.3: AC 4(1), AC 5(2), AC 6(1); Section 2.1.1.6.3: AC 2(1); Section 2.1.1.7.3.1: AC 1(1), (3), (4), (5), (8); Section 2.1.1.7.3.2: AC 1(1); Section 2.1.1.7.3.3(III): AC 1(1), (2), (3), (4), (5), (6), (8), (9), (10)]

The loaded naval SNF canister includes naval SNF, control rods or installed neutron poison assemblies, retention hardware for the control rods or installed neutron poison assemblies, and the naval SNF baskets and basket spacers. SNF from the Naval Nuclear Propulsion Program is temporarily stored at the Idaho National Laboratory. It is prepared for disposal and loaded into naval SNF canisters before being shipped to the repository. The Naval Nuclear Propulsion Program is responsible for preparing and loading naval SNF canisters. The Naval Nuclear Propulsion Program ships each loaded naval SNF canister to the repository in a naval M-290 transportation cask. Loaded naval SNF canisters arriving at the repository will comply with repository waste acceptance requirements (e.g., weight, thermal output, dose rate) detailed in the *Waste Acceptance System Requirements Document* (DOE 2008b).

Classified details regarding naval SNF and the loaded naval SNF canister are presented in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

1.5.1.4.1 Physical Characteristics of Naval SNF and Canisters

This section describes the physical characteristics of the naval SNF and the loaded naval SNF canister. This section also describes processes and structural, thermal, and criticality safety design criteria and methodologies for the naval SNF and the loaded naval SNF canister.

1.5.1.4.1.1 Physical Characteristics of Naval SNF

Naval SNF consists of solid metal and metallic components that are nonflammable, highly corrosion-resistant, and neither pyrophoric, explosive, combustible, chemically reactive, nor subject to gas generation by chemical reaction or off-gassing. The Naval Nuclear Propulsion Program has been allocated 65 MTHM for the emplacement of naval SNF in the repository. Naval SNF to be emplaced in the repository is from pressurized water reactors (PWRs), with the exception of one design operated in sodium-cooled reactors. A small amount of the naval SNF from the sodium-cooled reactors remains (approximately 0.0036% of naval SNF allocation). Residual sodium has been cleaned from this naval SNF.

Enrichment and Chemical Composition—Naval nuclear fuel is highly enriched (approximately 93 wt % to 97 wt %) in the isotope ^{235}U. As a result of the high initial uranium enrichment, very small amounts of transuranics are generated by end of life when compared to commercial SNF. The cladding of naval nuclear fuel provides primary containment for the radioactive fission products. Structural components, made of Alloy 600 (UNS N06600), Alloy 625 (UNS N06625), Alloy X-750 (UNS N07750), or Stainless Steel Type 304 (UNS S30400), are attached to the naval fuel assemblies to provide support to the fuel assemblies in the reactor. In cases where it is advantageous to remove some of this structural material to make packaging more efficient, a specified amount is removed. In other cases, it is more efficient to package the naval SNF in the condition it was in when removed from the reactor plant, and portions of the structural components remain attached to the naval SNF assemblies. For additional information on the chemical composition and materials present in naval SNF, see Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

Condition of Naval SNF—Development of naval fuel systems has been supported by a long-standing program of examination of irradiated test specimens and naval SNF after removal from prototype reactor plants and operating ships. These examinations are conducted at the Idaho National Laboratory. As part of the examinations, some of the naval SNF assemblies and test specimens are disassembled. In most cases, the parts produced when naval SNF assemblies are disassembled for examination have intact cladding and no exposed actinides or fission products. In a few cases, destructive evaluations of disassembled components result in nonintact cladding and exposed fission products and actinides; some test specimens have nonintact cladding because they were intentionally tested until the cladding failed.

The following definitions are used to categorize the cladding condition of naval SNF:

- **Intact**—Cladding is undamaged but may have hairline cracks or pinhole leaks in very few cases. Cladding with hairline cracks or pinhole leaks is not "damaged fuel" as defined in the U.S. Nuclear Regulatory Commission's (NRC) Spent Fuel Project Office, *Interim Staff Guidance–1, Revision 2, Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function* (NRC 2007).

- **Nonintact—**Cladding has either been intentionally removed to expose fuel for examination during material testing or tested to failure. Less than 2% of the approximately 400 loaded naval SNF canisters will contain nonintact naval SNF.

Additional classified details regarding the design of the naval fuel system, along with a description of key characteristics of the naval fuel system related to the performance of naval SNF assemblies in the repository, are provided in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

1.5.1.4.1.2 Physical Characteristics of Naval SNF Canisters

This section describes the naval SNF canister system, operational processes used to ensure the loaded naval SNF canister will meet its design criteria, and the design criteria and design bases under which the naval SNF canister system is used. The naval SNF canister system is used in conjunction with the naval M-290 transportation cask to transport all naval SNF from the Naval Reactors Facility on the Idaho National Laboratory to the repository. The naval SNF canister system includes naval SNF canisters, naval SNF baskets, and naval SNF basket spacers.

1.5.1.4.1.2.1 Naval SNF Canister Description

To accommodate different naval fuel assembly designs, naval SNF is shipped to the repository in either a naval short SNF canister or a naval long SNF canister designed to fit within a waste package. Approximately 400 loaded naval SNF canisters will be shipped to the repository for disposal—310 naval long SNF canisters, and 90 naval short SNF canisters (DOE 2008e, Section 3.1.2.J). The naval SNF canister consists of a right circular cylinder with a bottom plate. The naval SNF canister is fabricated from stainless steel that meets the requirements of Stainless Steel Types 316 and 316L (Stainless Steel Type 316/316L). The naval short SNF canister is 185.5 in. (nominal) in length (187 in. maximum), and the naval long SNF canister is 210.5 in. (nominal) in length (212 in. maximum). With the exception of length, the other characteristics of naval SNF canisters are identical. Except for the top section of the canister, which has features to support retention of the shield plug, the canister walls are 1 in. thick. The bottom plate is 3.5 in. thick, and the top shield plug is 15 in. thick. The outer diameter of the naval SNF canister is 66 in. nominal (66.5 in. maximum). The maximum external dimensions ensure naval SNF canisters fit into the waste packages. Figure 1.5.1-29 shows the typical naval SNF canister.

The closure system for the naval SNF canister is shown in Figure 1.5.1-30. The shield plug for the naval SNF canister is held in place with a shear ring that is seal-welded to the naval SNF canister shell and to the shield plug. The outer seal plate, which forms a redundant seal over the cavity containing the shear ring and its welds, is also welded to the shield plug and naval SNF canister shell. Penetrations in the seal plate (one) and in the shield plug (two), which are used for leak testing, evacuation, and backfill with helium, are closed with a welded seal plug and threaded pipe plugs, respectively. The naval SNF canister shield plug contains six 3-in.-diameter threaded holes for lifting the naval SNF canister (NNPP 2006). The maximum design weight of the loaded long or short naval SNF canister is 98,000 lb. For the purposes of establishing margin in crane capability, a maximum weight of 108,500 lb has been assigned (NNPP 2006, Section 24).

A naval SNF canister contains one or more baskets of naval SNF, stacked axially. Naval SNF baskets are used to provide separation of naval SNF assemblies during the loading of naval SNF assemblies into naval SNF canisters and during repository disposal. Naval SNF baskets vary in design and in the number of naval SNF assemblies they contain, depending on the design of the naval SNF assemblies. Naval SNF basket spacers are included in the naval SNF canister to fill space not occupied by naval SNF baskets.

There are three different methods for packaging naval SNF into naval SNF canisters: Packaging Methods A, B, and C; however, the design of the naval SNF canister is the same irrespective of packaging method. These packaging methods are based on the type of naval SNF assemblies and whether the naval SNF cladding is intact or nonintact. Designs for Packaging Method A are either completed or in development. Designs for Packaging Method B and Packaging Method C are still conceptual in nature. The variations in the packaging methods, naval SNF types, the configuration of naval SNF for disposal, and components used to package naval SNF (e.g., naval SNF baskets, basket spacers, hafnium control rods, control rod retention hardware, and installed neutron poison assemblies) are described below:

- Packaging Method A uses naval SNF baskets designed for specific naval SNF assemblies from the most common naval reactor designs. The naval SNF baskets are made from corrosion-resistant materials (e.g., Alloy 22 (UNS N06022) and Stainless Steel Type 316/316L). The number and dimensions of fuel ports differ in Packaging Method A baskets to accommodate specific naval SNF assembly designs. Naval SNF assemblies packaged using Packaging Method A have intact cladding.

  Packaging Method A naval SNF assemblies use hafnium control rods or installed neutron poison assemblies to reduce the reactivity of the naval SNF assemblies under moderated conditions. In many cases, pins, which are cylindrical rods inserted into holes drilled through the structural portion of the naval SNF assembly, affix the hafnium control rod within the naval SNF assembly. Retention hardware (e.g., retention pins) for structurally affixed hafnium control rods is made from zirconium alloy. A typical control rod retention pin is shown in Figure 1.5.1-31. The retention pins or other control rod retention hardware have features that prevent them from being removed once installed (e.g., extensions compressed during pin installation that spring back to normal position when the pins are completely inserted). When control rods are not installed in naval SNF assemblies, neutron poison assemblies are placed in the control rod channel instead. The installed neutron poison assemblies can be held in place with retention features similar to those used for control rods.

  In some cases, hafnium control rods or installed neutron poison assemblies are not affixed to the naval SNF assemblies. When criticality analyses show acceptable performance of the control rods or installed neutron poison assemblies within the limits of travel permitted by the naval SNF canister system, procedures allow for insertion of the control rods or neutron poison assemblies without affixing them to the naval SNF assemblies. The hafnium control rods and installed neutron poison assemblies remain in place in the naval SNF assemblies because they are retained within the basket height envelope by the adjacent basket or the naval SNF canister shield plug.

Figure 1.5.1-32 shows a typical Packaging Method A naval SNF basket design. Classified details pertaining to the design of naval SNF baskets, installed neutron poison assemblies, and control rod and installed neutron poison assembly retention hardware for Packaging Method A are discussed in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

- Packaging Method B uses naval SNF baskets made from corrosion-resistant material such as Stainless Steel Type 316/316L and Alloy 22. Packaging Method B naval SNF baskets contain partial naval SNF assemblies that result from post-operational naval SNF examinations, or naval SNF assemblies from less common core designs. Packaging Method B naval SNF baskets have capped sleeves located in the fuel ports in the support plates. These sleeves retain the naval SNF components, which vary in size and shape, in a fixed-space envelope. Perforations in the sleeves allow the sleeves to drain and dry when the naval SNF canister is drained and dried. The number and dimensions of fuel ports differ in Packaging Method B baskets to accommodate different complete or partial naval SNF assemblies. Figure 1.5.1-33 shows the conceptual Packaging Method B naval SNF basket design. Naval SNF assemblies packaged using Packaging Method B have intact cladding. Many of these naval SNF assemblies do not contain control rods. Neutron poison assemblies will be inserted into the sleeves when necessary to reduce the reactivity of the naval SNF.

- Packaging Method C uses naval corrosion-resistant cans to package pieces, parts, and fines. Pieces, parts, and fines of naval SNF include portions of naval SNF assemblies, and small punchings, chips, and grinding residue that result from destructive examinations of naval SNF. Pieces also include small test specimens. The pieces, parts, and fines may have intact or nonintact cladding. Figure 1.5.1-34 shows the conceptual corrosion-resistant can design for Packaging Method C. These naval corrosion-resistant cans will be made from Alloy 22 and are designed to be loaded into a Packaging Method B naval SNF basket. When necessary to reduce the reactivity of the naval SNF, neutron poison assemblies will be inserted into the corrosion-resistant cans.

1.5.1.4.1.2.2 Operational Processes for Naval SNF Canisters

Each naval SNF canister will be loaded such that thermal, shielding, criticality, and other characteristics of the received waste are within repository waste acceptance requirement limits established in the *Waste Acceptance System Requirements Document* (DOE 2008b).

1.5.1.4.1.2.2.1 Naval SNF Canister Thermal Controls

The following thermal controls are applied to the naval SNF canister and its contents:

- The loaded naval SNF canister will not be shipped to the repository until the decay heat at time of acceptance at the repository is less than or equal to 11.8 kW (DOE 2008b, Section 4.4.9). The decay heat limit of 11.8 kW for each naval waste package is sufficiently low such that no aging is required before repository emplacement.

- Initial Handling Facility (IHF) design and operational controls (such as limiting combustion sources, natural or artificial cooling, and proximity to other heat sources) will be established to ensure that the thermal performance criteria for naval SNF are met (DOE 2008d).

- The subsurface drift design and operational controls (such as ventilation system monitoring) will be established to ensure that the thermal performance criteria for naval SNF are met.

For the design criteria and design basis associated with these thermal controls, see Section 1.5.1.4.1.2.5.3.

1.5.1.4.1.2.2.2 Naval SNF Canister Criticality Controls

For the preclosure period, criticality is controlled by a breach of the naval SNF canister being beyond Category 2. Because breach of the naval SNF canister is beyond Category 2, introduction of moderator into naval SNF canisters is also beyond Category 2.

For the postclosure period, criticality control of naval SNF (i.e., assurance of a low probability that criticality involving naval SNF could occur) is provided by controlling one or more of the following characteristics of the loaded naval SNF canister: the amount of fissile material; the materials used for naval SNF canisters, baskets, spacers, naval corrosion-resistant cans, control rods, and installed neutron poison assemblies and their retention hardware; and geometric separation of naval SNF assemblies.

Fissile Material—To control fissile material loading, the number, type, and identity of naval SNF components (naval SNF assemblies, partial SNF assemblies, fines) for each port in each specific naval SNF basket or corrosion-resistant can is controlled.

Materials—The naval SNF basket and basket spacers are made of Stainless Steel Type 316/316L, except that Alloy 22 is used for tie rods for some naval SNF baskets. The naval SNF canister is made from Stainless Steel Type 316/316L. The naval corrosion-resistant cans are made of Alloy 22. The neutron absorbing portions of control rods and the neutron poison assemblies are made from hafnium; structural portions of control rods and neutron poison assemblies and the retention hardware for control rods and neutron poison assemblies are made from zirconium alloy.

Geometric Separation—The following aspects of component designs are controlled:

- Thickness of the naval SNF canister bottom plate and shield plug, components of naval SNF basket and spacers, and naval corrosion-resistant cans.
- The number and arrangement of ports in naval SNF baskets.
- The diameter of the naval corrosion-resistant cans used in Packaging Method C.

Installation of Control Rods, Neutron Poison Assemblies, Control Rod Retention Hardware —Control rods from reactor operation or neutron poison assemblies are required in some cases to provide criticality control for moderated conditions. When criticality analyses demonstrate that these components are necessary for disposal, procedural controls are established to ensure that they, and any required retention hardware, are present in naval SNF that is shipped to the repository for disposal.

1.5.1.4.1.2.3 Structures, Systems, and Components Important to Safety and Important to Waste Isolation

Section 1.7.2.3 describes the determination of passive structure, system, or component reliability and discusses several types of failures for passive SSCs in the PCSA, including the structural challenge causing loss of containment (breach) of a waste form container (e.g., naval SNF canister).

The naval SNF canister system performs safety functions during preclosure handling and the associated controlling parameters and values for operations can be found in Table 1.5.1-30. The analysis results of the PCSA (as applicable to the naval SNF canister) are discussed in Section 1.5.1.4.1.2.6.1. The naval SNF canister performs functions that affect the performance of the ITWI Engineered Barrier System; these functions and the associated controlling parameters are presented in Table 1.5.1-31.

1.5.1.4.1.2.4 Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

Procedural safety controls for naval SNF canisters are provided in Table 1.9-10.

1.5.1.4.1.2.5 Design Criteria and Design Bases

1.5.1.4.1.2.5.1 Structural Design Criteria and Design Bases

The naval SNF canister system is used for dry storage at the Naval Nuclear Propulsion Program Naval Reactors Facility in Idaho, transportation to the repository, and, when placed into a waste package, emplacement in the repository. The design of the naval SNF canister system began before repository conditions were defined. Therefore, the Naval Nuclear Propulsion Program used design criteria for dry storage and transportation, and the available information about the potential conditions in the repository, to develop design criteria for the naval SNF canister system and its transportation cask. As the repository conditions were developed, the existing naval SNF canister system was evaluated under repository conditions to ensure that the requirements of 10 CFR Part 63 are met.

Naval SNF is shipped to the repository in naval M-290 transportation casks. The design criteria for the loaded naval M-290 transportation cask is acceptable performance (no radionuclides released, and no moderator enters the naval SNF canister) for the normal conditions of transport specified in 10 CFR 71.71, the hypothetical accident conditions of transportation specified in 10 CFR 71.73, and the submergence requirements of 10 CFR 71.61. These conditions include a thirty-foot drop onto an unyielding surface in the worst orientation, a 40-in. drop onto a 6-in.-diameter pin in the worst orientation, a 30-min engulfing fire at 1,475°F, and an immersion in 3 ft of water, all in sequential order. The naval SNF canister is also designed and analyzed to demonstrate that it can withstand an immersion in 600 ft of water. The integrity of the loaded naval M-290 transportation cask is relied upon to maintain containment for normal and hypothetical accident conditions during transportation.

The design criteria below apply to the naval SNF canister system, corrosion-resistant cans, control rods, neutron poison assemblies, and their retention hardware for the design basis conditions:

- Naval SNF assemblies must remain supported by all naval SNF basket support plates (accounting for the worst-case tolerance stack-up of all cargo within the naval SNF canister and the naval SNF canister itself).

- Naval SNF and corrosion-resistant cans must remain within their naval SNF basket ports, and control rods and installed neutron poison assemblies must remain in their design location, to the extent necessary to demonstrate compliance with the criticality safety design criteria. For additional information on postclosure criticality analyses, see Section 2.2.1.4.1 of the Naval Nuclear Propulsion Program Technical Support Document.

- The naval corrosion-resistant can shall remain in its basket port and retain all fuel-bearing items larger than 0.06 in. in diameter for an explosion of a hydrogen-oxygen gas mixture within the corrosion-resistant can. This criterion ensures that, if water seeps into the naval corrosion-resistant can and hydrogen is generated and explodes, almost all fissile material will remain in the naval corrosion-resistant cans.

Section 1.5.1.4.1.2.5.1 of the Naval Nuclear Propulsion Program Technical Support Document provides additional details on the development of the design criteria for the naval SNF canister system.

1.5.1.4.1.2.5.2 Criticality Safety Design Criteria and Design Bases

The criticality safety design criterion for the preclosure period is that naval SNF must be subcritical for configurations resulting from conditions that have at least one chance in 10,000 of occurring before permanent repository closure. To provide assurance of subcriticality, the methodology used to determine k_{eff} includes a 5% administrative margin (Δk_m) and accounts for the biases and uncertainties in both the calculations and experimental data used in the development of k_{eff}.

The design basis condition for criticality safety during the preclosure period is that naval SNF canisters remain unmoderated. Naval SNF remains subcritical when unmoderated as shown by meeting the above design criterion (Section 1.14 of the Naval Nuclear Propulsion Program Technical Support Document). Because breach of naval SNF canister is beyond Category 2 (Section 1.7.5.1), moderating materials cannot enter the naval SNF canister, and moderation of naval SNF is beyond Category 2. Therefore, naval SNF remains subcritical for the preclosure period. Additional information on configurations analyzed for preclosure nuclear safety is provided in Section 1.5.1.4.1.2.6.3.

For the postclosure period, the criticality safety design criterion is that the probability of criticality involving naval SNF will not cause the total probability of criticality to exceed the FEPs screening criterion (1 chance in 10,000 for the first 10,000 years) for all waste forms. The design basis conditions evaluated to determine the probability of criticality are the configurations that result from the postclosure structural and thermal analysis, which include consideration of the factors that could affect the reactivity of naval SNF in the postclosure environment (e.g., human errors in emplacing naval waste packages and degradation of naval SNF). Section 2.2.1.4.1 of the Naval Nuclear Propulsion Program Technical Support Document describes the process by which the criticality potential for naval SNF emplaced in the repository during the postclosure period is assessed.

1.5.1.4.1.2.5.3 Thermal Design Criteria and Design Bases

The principal thermal design criterion for naval SNF for disposal is that naval SNF cladding will not fail due to thermal damage before permanent repository closure. This condition is imposed for the preclosure period as a condition for permanent disposal; because breach of the naval SNF canister is beyond Category 2, naval SNF cladding integrity is not necessary to retain radionuclides and meet preclosure safety criteria. The Naval Nuclear Propulsion Program calculates cladding temperature using naval SNF canister surface temperatures as boundary conditions. The IHF design and emplacement operational controls will be established to ensure that the analyzed naval SNF canister surface temperature will remain below 400°F from the time of detensioning the transportation cask closure until completion of emplacement of the naval SNF waste package in the emplacement drift, and the overall duration of these handling operations shall not exceed 30 days (DOE 2008d, Section 10.3.2.2). This temperature limit does not apply in the unlikely event of a fire to which a naval SNF canister may be exposed (BSC 2008j, Section 3.2.1.9.4). Analyses of performance of a naval SNF canister during a fire event are discussed in Sections 1.5.1.4.1.2.6.1 and 1.7.2.3.3.1 and Table 1.7-7.

For preclosure analysis of naval SNF in the emplacement drifts, the naval SNF canister external surface temperature is calculated using the worst-case heat flux profiles for naval SNF and repository thermal boundary conditions (BSC 2006).

The principal thermal design criterion and the associated design bases for naval SNF for the postclosure period is that naval SNF cladding will not fail due to thermal damage for the early failure scenario class, drip shield early failure modeling case. This criterion supports representation of waste packages of naval SNF by an equal number of waste packages of commercial SNF in the TSPA by limiting the release of radionuclides from naval SNF in this scenario class to levels where the corresponding release of radionuclides important to dose from a commercial SNF waste package are larger than those from a naval SNF waste package. Naval SNF cladding is not required to remain intact for the thermal conditions imposed by scenario classes other than the early failure scenario class, drip shield early failure modeling case, because cladding integrity in other scenario classes is not necessary to demonstrate that naval SNF waste packages can be represented by an equal number of commercial SNF waste packages. For the postclosure thermal analyses, the naval SNF canister surface temperature is calculated using a range of heat flux profiles for the surface of the naval SNF canister and subsurface thermal conditions after closure of the repository (BSC 2006).

The decay heat limit of 11.8 kW for each naval waste package is sufficiently low that no aging is required before repository emplacement.

1.5.1.4.1.2.6 Design Methodologies

1.5.1.4.1.2.6.1 Structural Design

For lifting operations of the naval SNF canister once received at the repository, the naval SNF canister is designed to meet or exceed the requirements of ANSI N14.6 for critical loads. The maximum weight of a fully loaded naval SNF canister used for this analysis is 98,000 lb (NNPP 2006).

The PCSA evaluates representative containers within a class of containers that encompass TAD canisters, naval SNF canisters, and a variety of DPCs for the probability of a breach of the representative canister due to:

- **Structural Challenges**—The structural challenges considered are: flat bottom drop of the representative container, collision of the representative container with an object or structure (which, for example, could occur while the container is on a conveyance that derails or when the container is handled by a crane), and drop of an object onto the representative container (Section 1.7.2.3.1). According to Section 1.7.5.1, a breach of the naval SNF canister due to these structural challenges is beyond Category 2.

- **Fire**—The maximum temperature reached by a representative container is characterized with a probability distribution. To determine whether the temperature reached by a representative container is sufficient to cause the container to fail, the probability of a breach of the container as a function of its temperature is evaluated (Section 1.7.2.3.3.1). According to Section 1.7.5.1, a breach of the naval SNF canister (in the M-290 transportation cask, canister transfer machine shield bell, or waste package) due to a fire is beyond Category 2.

- **An Increase in Temperature Resulting from a Loss of Heating, Ventilation, and Air-Conditioning Cooling Inside a Surface Facility**—The calculations show that the calculated maximum temperatures for the representative container from a loss of heating, ventilation, and air-conditioning cooling inside a surface facility are significantly lower than the failure threshold for the representative container (Section 1.7.2.3.3.2).

- **Seismic Event**—The seismic event sequence analysis is conducted in four stages (Section 1.7.1.4). In the first stage, seismic event sequences are developed. In the second stage, a seismic hazard curve is developed. In the third stage, seismic fragility evaluations are performed where the fragility curve provides the mean probability of unacceptable performance of a waste form container as a function of a ground motion parameter. In the fourth stage, event sequences are quantified. According to Table 6.4-2 of *Seismic Event Sequence Quantification and Categorization* (BSC 2008b), a breach of the naval SNF canister from a seismic event is beyond Category 2.

Structural analysis of naval SNF, naval SNF baskets, naval corrosion-resistant cans, and other internal components of the loaded naval SNF canister are performed for the postclosure period to determine the condition of the loaded naval SNF canister as it applies to postclosure criticality evaluations (as discussed in Section 2.2.1.4.1 of the Naval Nuclear Propulsion Program Technical Support Document). The bottom plate and shield plug of the naval SNF canister do not need to remain attached to the naval SNF canister shell to restrict the motion of naval SNF, control rods, and installed neutron poison assemblies to the extent that they are credited in postclosure analyses (see Section 2.2.1.4.1 of the Naval Nuclear Propulsion Program Technical Support Document). The naval SNF canister is also not relied upon for containment or mechanical support. Therefore, no structural analysis of the naval SNF canister is performed for postclosure scenarios.

Structural analyses for naval SNF, naval SNF baskets, naval corrosion-resistant cans, and other internal components of the loaded naval SNF canister are performed for the postclosure period for the following loads and conditions:

- Lateral acceleration of 114 g of a loaded naval SNF canister inside a waste package
- Axial impact of a loaded naval SNF canister with a flat unyielding surface at 6.5 m/s.

The loads and conditions are evaluated with 0.1 in. of material removed from all stainless steel surfaces for general corrosion. The general corrosion allowance is specified to account for general corrosion for 10,000 years after closure of the repository. These loads and conditions are derived from kinematic analyses of repository conditions occurring within the probability threshold for FEPs that must be included in the TSPA in accordance with 10 CFR Part 63 (e.g., magnitudes of seismic events and amount of material removed from Stainless Steel Type 316/316L by general corrosion). Section 1.5.1.4.1.2.6.1 of the Naval Nuclear Propulsion Program Technical Support Document provides additional details on the development of the loads and conditions evaluated for the postclosure period.

For components of the loaded naval SNF canister designed for the repository, loads and conditions exceeding those specified above are modeled to cause failure (e.g., structural collapse of the naval SNF basket) of the component. Naval fuel assemblies, designed for shipboard requirements (e.g., combat shock loads), are typically capable of withstanding greater loads than those specified above; however, for conservatism, this excess capacity is not credited. Section 1.5.1.4.1.2.6.1 of the Naval Nuclear Propulsion Program Technical Support Document discusses the evaluation of naval SNF assemblies, baskets, installed neutron poison assemblies, and control rod and neutron poison assembly retention hardware during postclosure seismic conditions.

1.5.1.4.1.2.6.2 Thermal Design

The evaluation of naval SNF cladding integrity is performed using a methodology developed expressly for naval SNF. The methodology depends on a combination of time, temperature, and fuel characteristics. The naval SNF cladding integrity thermal limit is dependent on specific characteristics for each naval SNF type. A time-at-temperature profile is developed for dry operations. A cumulative time-at-temperature calculation is performed to ensure thermal limits are not violated, thereby preventing fission product and actinide release.

Thermal analyses to define maximum temperature for an operation include modeling heat transfer by conduction, radiation, and convection to ambient. The analysis models generally do not include internal convection heat transfer. This is conservative because convection is an effective heat removal mechanism at the temperature magnitudes for the times being considered. Conservative decay heat rates are applied to configurations of the naval SNF assemblies in the loaded naval SNF canister as a function of time after reactor shutdown.

The dimensionality of the model depends on the extent of precision required to achieve the desired results; one-dimensional models are the most conservative and three-dimensional models are the most precise. The one-dimensional models use spreadsheets to determine which fuel type has the least margin in regard to naval SNF cladding thermal performance limits, while the two- and three-dimensional models use well-established computer programs such as ANSYS (version 10) or ABAQUS/Standard (version 6.7) to determine the temperatures to use in naval cladding thermal analyses and in structural and radionuclide release source term analyses. The two-dimensional model is chosen if the analysis using that model shows that the thermal design criteria are met; otherwise a three-dimensional model is used to incorporate the axial profile of the decay heat and heat transfer in the axial direction. Qualification of the thermal models is accomplished by comparison against some combination of hand calculations, independent models, previous analyses, and thermal tests.

The determination of the decay heat content of a naval SNF canister is discussed in Section 1.5.1.4.2.

For the preclosure period, the design and operating conditions will be established such that the thermal analyses for the naval SNF canister surface temperature shall not exceed 400°F during handling operations from the time of detensioning of the transportation cask closure until emplacement operations are complete (DOE 2008d, Section 10.3.2.2). The duration of these handling operations shall not exceed 30 days. The IHF analysis will show (Section 1.2.3.4) that the naval SNF canister surface temperature limit will not be exceeded. The design of the TEV and its operational controls will also ensure the naval SNF canister surface temperature is not exceeded during emplacement operations (Section 1.3.3.5.1). For the subsurface drift, thermal analyses show the naval SNF canister external surface temperature inside a waste package will not exceed the envelope defined by the time temperature plot shown in Figure 1.3.1-8, which includes consideration of a 30-day loss of ventilation. Under these boundary conditions for the external surface temperature of the naval SNF canister, naval SNF cladding does not fail.

For postclosure thermal analyses, the method used to determine the temperatures of naval SNF and naval SNF canister system components was developed to decouple the thermal analysis of the naval

SNF and naval SNF canister system components from the large-scale thermal analysis of the repository drifts. In the first phase of this analysis method, the possible heat flux profiles on the surface of the naval SNF canister and the corresponding temperatures at the same locations are determined.

A set of possible heat flux distributions at the surface of the naval SNF canister has been developed based on (1) an initial decay heat production rate of 11.8 kW at time of acceptance at the repository, and the decay of this heat over time; (2) the distribution of decay heat in the naval SNF assemblies; and (3) thermal analyses that demonstrate the effects of heat transfer within the naval SNF canister on the heat flux distribution at the surface of the naval SNF canister.

A set of naval SNF canister surface temperature profiles, as functions of time after emplacement, is developed for each heat flux distribution using time-dependent repository thermal boundary conditions. In the second phase of the analysis methodology, the time-dependent temperature of naval SNF and components of the naval SNF canister system are determined from the time-dependent decay heat production of the naval SNF and the naval SNF canister surface temperature profiles that correspond to that decay heat production. These naval SNF temperatures are then used to determine corrosion rates of Zircaloy and hafnium in the postclosure radionuclide release source term analysis discussed in Section 2.3.7 of the Naval Nuclear Propulsion Program Technical Support Document. Naval SNF temperatures are also used to determine if thermal conditions will cause cladding failure. Additional details pertaining to thermal design and analyses are provided in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

1.5.1.4.1.2.6.3 Criticality Safety Design

During the preclosure period, assurance that naval SNF remains subcritical relies on breach of the naval SNF canister being beyond Category 2. In addition, the control of moderating materials is an IHF design requirement (Section 1.2.3).

To determine the criticality potential of naval SNF during the preclosure period, Monte Carlo transport theory calculations are performed for unmoderated configurations of naval SNF. The Monte Carlo codes are qualified and verified, having been benchmarked against measured moderated and unmoderated critical configurations. Since a breach of the naval SNF canister is beyond Category 2, moderator entering the naval SNF canister has less than one chance in 10,000 of occurring over the preclosure period. Therefore, the configurations modeled to demonstrate subcriticality for preclosure do not include moderator. The following conservatisms are used in the Monte Carlo models for preclosure criticality analyses:

- Fuel depletion is not included.
- For each core type assessed, only the most reactive naval SNF assembly type is modeled.
- Naval SNF and surrounding reflector materials are rearranged in the most reactive configurations.
- The presence of some high-worth neutron poisons is not included.

- Results are adjusted by conservative estimates of model biases and uncertainties.
- An administrative margin of 5% is included.

The evaluations are performed for several Packaging Method A and Packaging Method B naval SNF types (including those cases that contain the most fissile material and that are expected to be the most reactive) and for Packaging Method C naval SNF. Evaluations of naval SNF in the IHF are provided in Section 1.14 of the Naval Nuclear Propulsion Program Technical Support Document.

Criticality analyses performed for the preclosure period and experimental evidence confirm that when naval SNF is unmoderated its reactivity level is low and criticality is not possible. This is true even when the geometric separation provided by naval SNF baskets is not credited, allowing close-packed arrangements to form. In addition, such close-packed, unmoderated configurations are more subcritical than the moderated configurations that include the geometric separation in the as-loaded condition. Therefore, the fact that the naval SNF remains subcritical during underwater loading operations (in unborated water) provides substantive additional evidence that the naval SNF will remain subcritical during the preclosure period.

The postclosure criticality evaluation includes a probabilistic evaluation to determine what configurations must be analyzed to determine the reactivity of the configurations, and then those configurations are analyzed to determine if they are critical. The methodology starts by determining what factors could influence the reactivity of naval SNF, proceeds by assigning probabilities to those factors that have an impact on reactivity, and is completed by determining which combinations of factors are credible and have the potential to significantly affect the total probability of criticality for naval SNF. The resulting configurations are either assumed to be critical (if the probabilities of achieving those configurations are low enough that they would not cause an unacceptable probability of criticality) or the configurations are analyzed to determine their reactivity and the conditional probability that they are critical, once they occur. The postclosure criticality analyses use qualified models with conservative biases and uncertainties. The overall postclosure criticality methodology is described in Section 2.2.1.4.1 of the Naval Nuclear Propulsion Program Technical Support Document.

1.5.1.4.1.2.6.4 Radiation Sources

The gamma and neutron radiation fluxes at the surface of the naval SNF canister are used to develop the radiation sources for use in designing the IHF. At the time of shipment to the repository, the surface gamma and neutron fluxes do not exceed those provided for use in design of the IHF (McKenzie 2007). The maximum on-contact total (gamma and neutron) radiation level at the top of the naval SNF canister will not exceed 100 millirem per hour (DOE 2008d, Figure C-6, Note 11).

The energy-dependent gamma and neutron fluxes at the surface of the loaded naval SNF canister are calculated in gammas or neutrons per centimeter squared-second, using version 2.92 of the publicly available PARTISN computer program. For this analysis, PARTISN solves the transport equation in two-dimensional cylindrical geometry using a finite-difference, discrete-ordinates method. PARTISN is qualified and verified for use in the Naval Nuclear Propulsion Program for gamma and neutron radiation shielding design.

There are several possible configurations for naval SNF assemblies inside a naval SNF canister. Fluxes from various combinations of naval SNF assemblies are calculated. To provide a bounding source for all possible configurations, the maximum gamma and neutron fluxes from the various configurations are selected for the bottom and side of the loaded naval SNF canister as well as for three specific locations on top of the naval SNF canister.

The gamma and neutron radiation sources are developed for five years after reactor shutdown, since this is the earliest anticipated time that a naval SNF canister will be shipped to the repository. The gamma and neutron fluxes also contain a deliberate 30% added conservatism to maximize calculated personnel radiation exposure.

Gamma Flux—Gamma flux calculations are performed for a 27-group energy structure. An energy structure takes a continuous energy spectrum of the gamma photons present and subdivides it into groups. The gamma radiation source term used in the calculations includes the gammas from both fission products and activated structural components in naval SNF assemblies. Contributions due to crud and transuranic decay are negligible and are not included.

The fission product gamma radiation source values (provided in Table 1.10-21) are obtained using depletion computer codes developed by the Naval Nuclear Propulsion Program that solve for the change in isotopic inventories with fuel assembly burnup or depletion. These codes are qualified and verified for use with naval SNF in repository applications. The activated structural component portion of the gamma radiation source term is calculated using two separate computer programs: the publicly available ORIGEN-S computer program, qualified and verified for use with naval SNF in repository applications, and the computer program developed by the Naval Nuclear Propulsion Program used to obtain the fission product radiation source term. ORIGEN-S is qualified for activation calculations above and below the power-generating portions of the reactor core while the other computer program developed by the Naval Nuclear Propulsion Program is qualified for activation calculations in the power-generating area of the reactor core only.

Additional details on the development of the gamma flux for naval SNF is described in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

Neutron Flux—Neutron flux calculations are performed for a 15-group energy structure. The neutron radiation source term used in the calculations includes the neutrons from spontaneous fission and (α, n) reactions of transuranic nuclides formed in the fission process. Subcritical multiplication in naval SNF (source neutrons from spontaneous fission and (α, n) reactions induce fission in the residual uranium atoms) is also included in the neutron source term. Three radionuclides, ^{238}Pu, ^{242}Cm, and ^{244}Cm, yield more than 98% of the total naval SNF neutron radiation source.

The neutron radiation source term (provided in Table 1.10-22) is obtained using depletion codes developed by the Naval Nuclear Propulsion Program that solve for the change in isotopic inventories with fuel assembly burnup or depletion. These Naval Nuclear Propulsion Program depletion codes are qualified and verified for use with naval SNF in repository applications.

Additional details on the development of the neutron flux for naval SNF is described in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

1.5.1.4.1.2.7 Consistency of Materials with Design Methodologies

Materials of the naval SNF, naval SNF canister system, control rods, and neutron poison assemblies resist degradation in the repository environment. Materials for the naval SNF canister are compatible with the waste package inner vessel and outer corrosion-resistant barrier materials, and interactions among these materials will not be detrimental to the stability of naval SNF. Cleanliness requirements of the *Waste Acceptance System Requirements Document* (DOE 2008b, Section 4.4.11) require avoiding introducing foreign materials into the loaded naval SNF canister.

1.5.1.4.1.2.8 Design Codes and Standards

The materials, design, fabrication, testing, examination, and transportation of the naval canister system and neutron poison assemblies meet the requirements of the following codes and standards:

Naval SNF Canister:

- *1998 ASME Boiler and Pressure Vessel Code*, Section III, Division 1, Subsection NB, for normal and accident conditions of storage and transportation.
- For the lifting features of the naval SNF canister, ANSI N14.6-1993 structural limits for normal handling operations at the repository surface facilities.
- For leak-testing of the naval SNF canister, ANSI N14.5-1997, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment*, at a sensitivity of less than or equal to 6.10×10^{-6} in.^{3}/s (1×10^{-4} ref-cc/s) for the outermost closure at the time of naval SNF canister closure.

Naval SNF Baskets and SNF Basket Spacers:

- *1998 ASME Boiler and Pressure Vessel Code*, Section III, Division 1, Subsections NF (for items not important to criticality) and NG (for items important to criticality), for normal and accident conditions of storage and transportation.

Neutron Poison Assemblies:

- For hafnium components: *1998 ASME Boiler and Pressure Vessel Code*, Section III, Division 1, Subsection NG (for parts important to criticality) and Subsection NF (for other parts) for normal and accident conditions of storage and transportation.
- For zirconium alloy components: fabrication to Naval Nuclear Propulsion Program procurement specification requirements since *1998 ASME Boiler and Pressure Vessel Code*, Section III, Division 1, Subsections NF and NG do not address zirconium alloys.

Control Rod Retention Hardware:

- Fabrication to Naval Nuclear Propulsion Program procurement specification requirements since *1998 ASME Boiler and Pressure Vessel Code*, Section III, Division 1, Subsections NF and NG do not address zirconium alloys.

1.5.1.4.2 Thermal Characteristics of Naval SNF

Naval fuel assemblies are composed of materials that keep temperatures low enough to maintain integrity of the cladding. The heat transfer characteristics of Zircaloy and hafnium are provided in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document. The thermal properties of Stainless Steel Type 316/316L and Alloy 22 are provided in *2004 ASME Boiler and Pressure Vessel Code* (ASME 2004, Section II) and Hastelloy C-22 Alloy (Haynes International 2002, p. 13), respectively. Standard properties for air are used in both preclosure and postclosure thermal calculations.

The decay heat in naval SNF originates from fission product and actinide decay, and decreases exponentially over time based on the effective decay constant for the particular radionuclides. The decay heat load for the loaded naval SNF canister is calculated by Naval Nuclear Propulsion Program codes using *American National Standard for Decay Heat Power in Light Water Reactors* (ANSI/ANS-5.1-1994) for exponential fits of decay heat with time, or by converting the activities for the radionuclide inventory in the naval SNF canister to a heat generation rate. The decay heat powers from contributing radionuclides are calculated and summed using ORIGEN-S or other codes developed by the Naval Nuclear Propulsion Program. The codes sum the decay heat powers from contributing radionuclides to calculate the decay heat load.

The thermal power for naval SNF canisters containing two naval SNF types at five years after reactor shutdown is given in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

1.5.1.4.3 Nuclear Characteristics of Naval SNF

The actual radionuclide inventory varies depending on naval SNF type, naval SNF canister size, naval SNF basket design, and packaging method. In addition, within each naval SNF type, there are variations related to operational history and time after shutdown. As a result, a radionuclide inventory for a representative naval SNF canister is developed for use in the postclosure radionuclide release source term analysis.

The radionuclide inventory for a representative naval SNF canister is developed based on detailed core depletion calculations. The radionuclide inventory accounts for fission products, actinides, Zircaloy cladding, hafnium control rods, activated structural components, and crud. Additional description of the methodology used to create a radionuclide inventory for a representative naval SNF canister is provided in Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.

Depletion codes developed by the Naval Nuclear Propulsion Program, in conjunction with the publicly available ORIGEN-S computer program, solve for the change in radionuclide inventories in a fuel assembly. The depletion codes developed by the Naval Nuclear Propulsion Program and ORIGEN-S are qualified and verified for use in developing radionuclide inventories.

The Naval Nuclear Propulsion Program uses fine-mesh three-dimensional diffusion theory calculations to determine the radionuclide inventories of a reactor core throughout core life on a very detailed spatial basis for a given set of reactor core parameters including operating power level, temperature, control rod positions, and a given set of operating time steps. Results from these detailed calculations are edited to develop average fuel assembly neutron fluxes and cross sections that can be used in subsequent depletion analyses. Some of these depletion analyses are performed with codes developed by the Naval Nuclear Propulsion Program. Alternatively, the operating time-step fluxes and cross sections from the Naval Nuclear Propulsion Program codes are passed to the ORIGEN-S computer program (or an equivalent, verified code developed by the Naval Nuclear Propulsion Program) where the radionuclide inventory is calculated through end of reactor core life. The radionuclide inventories from the depletion analyses are then decayed to five years after reactor shutdown for the preclosure source term. The initial radionuclide inventory of a representative naval SNF canister at five years after reactor shutdown is provided in Table 1.5.1-32.

1.5.1.4.4 Source Term Characteristics of Naval SNF

The gamma and neutron radiation shielding source term characteristics for naval SNF are discussed in Section 1.10 and Section 1.5.1.4.1.2.6.4. The thermal (decay heat content) source term characteristics for naval SNF are described in Section 1.5.1.4.2.

The postclosure radionuclide release source term for naval SNF is discussed in Section 2.3.7 of the Naval Nuclear Propulsion Program Technical Support Document. For the TSPA, a naval waste package is represented by a waste package of commercial SNF. Special case analyses are conducted to demonstrate that a waste package containing naval SNF can be represented by a waste package containing commercial SNF in the TSPA (SNL 2008b).

1.5.1.4.5 Naval SNF Canister Conformance of Design to Criteria and Bases

The nuclear safety design bases for ITS and ITWI SSCs and features are derived from the PCSA presented in Sections 1.6 through 1.9 and the postclosure performance assessment presented in Sections 2.1 through 2.4. The nuclear safety design bases identify the safety function to be performed and the controlling parameters with values or ranges of values that bound the design.

The quantitative assessment of event sequences, including the evaluation of component reliability and the effects of operator action, is developed in Section 1.7. Any SSC or procedural safety control appearing in an event sequence with a prevention or mitigation safety function is described in the applicable design section of the SAR.

Section 1.9 describes the methodology for safety classification of SSCs and features of the repository. The tables in Section 1.9 present the safety classification of the SSCs and features, including those items that are non-ITS or non-ITWI. These tables also list the preclosure and postclosure nuclear safety design bases for each structure, system, or major component.

The design criteria are specific descriptions of the SSCs or features (e.g., configuration, layout, size, efficiency, materials, dimensions, and codes and standards) that are utilized to implement the assigned safety functions. Table 1.5.1-30 presents the nuclear safety design bases and design criteria for the naval SNF canisters.

1.5.1.5 General References

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AASHTO (American Association of State Highway and Transportation Officials) 2004. *A Policy on Geometric Design of Highways and Streets*. 5th Edition. Washington, D.C.: American Association of State Highway and Transportation Officials. TIC: 257443.

ANSI N14.5-1997. 1998. *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment*. New York, New York: American National Standards Institute. TIC: 247029.

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Table 1.5.1-1. Summary of Repository Inventory

Type of Waste	Estimated Number of Canisters	Metric Tons of Heavy Metal
Commercial SNF and HLW (West Valley)	~221,000 assemblies ~7,500 TAD canisters 275 HLW canisters	63,000
HLW	~9,300 canisters	4,667
DOE SNF	~2,500 to ~5,000 canisters	2,268
Naval SNF	~400 canisters	65
Total	—	70,000

NOTE: The estimated number of HLW canisters represents the canisters corresponding to the allotment of 4,667 MTHM and not the total number of canisters to be produced at the originating sites.

Table 1.5.1-2. Physical Characteristics of Pressurized Water Reactor Assembly Classes

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
B&W 15 × 15	15 × 15	B&W	B&W Mark B	B1515B	165.7	8.54	Zircaloy-4
			B&W Mark B10	B1515B10	165.7	8.54	Zircaloy-4
			B&W Mark B3	B1515B3	165.7	8.54	Zircaloy-4
			B&W Mark B4	B1515B4	165.7	8.54	Zircaloy-4
			B&W Mark B4Z	B1515B4Z	165.7	8.54	Zircaloy-4
			B&W Mark B5	B1515B5	165.7	8.54	Zircaloy-4
			B&W Mark B5Z	B1515B5Z	165.7	8.54	Zircaloy-4
			B&W Mark B6	B1515B6	165.7	8.54	Zircaloy-4
			B&W Mark B7	B1515B7	165.7	8.54	Zircaloy-4
			B&W Mark B8	B1515B8	165.7	8.54	Zircaloy-4
			B&W Mark B9	B1515B9	165.7	8.54	Zircaloy-4
			B&W Mark BGD	B1515BGD	165.7	8.54	Zircaloy-4
			B&W Mark BZ	B1515BZ	165.7	8.54	Zircaloy-4
		WE	WE	B1515W	165.7	8.54	not available
B&W 17 × 17	17 × 17	B&W	B&W Mark C	B1717B	165.7	8.54	Zircaloy-4
CE 14 × 14	14 × 14	ANF	ANF	C1414A	157.0	8.10	Zircaloy-4
		CE	CE	C1414C	157.0	8.10	Zircaloy-4
		WE	WE	C1414W	157.0	8.10	Zircaloy-4
CE 16 × 16	16 × 16	CE	CE	C1616CSD	176.8	8.10	Zircaloy-4
CE System 80	16 × 16	CE	CE System 80	C8016C	178.3	8.10	Zircaloy-4
WE 14 × 14	14 × 14	ANF	ANF	W1414A	159.8	7.76	Zircaloy-4
		ANF	ANF Top Rod	W1414ATR	159.8	7.76	Zircaloy-4
		B&W	B&W	W1414B	159.8	7.76	not available
		WE	WE LOPAR	W1414WL	159.8	7.76	Zircaloy-4
		WE	WE OFA	W1414WO	159.8	7.76	Zircaloy-4
		WE	WE Std	W1414W	159.8	7.76	Zircaloy-4

Table 1.5.1-2. Physical Characteristics of Pressurized Water Reactor Assembly Classes (Continued)

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
WE 15 × 15	15 × 15	ANF	ANF	W1515A	159.8	8.44	Zircaloy-4
			ANF HT	W1515AHT	159.8	8.44	not available
			ANF Part Length	W1515APL	159.8	8.44	not available
		WE	LOPAR	W1515WL	159.8	8.44	Zircaloy-4
			OFA	W1515WO	159.8	8.44	Zircaloy-4
			WE Std	W1515W	159.8	8.44	Zircaloy
			WE Vantage 5	W1515WV5	159.8	8.44	not available
WE 17 × 17	17 × 17	ANF	ANF	W1717A	159.8	8.44	Zircaloy-4
		B&W	B&W Mark B	W1717B	159.8	8.44	not available
		WE	WE	W1717WRF	159.8	8.44	not available
			WE	W1717WVJ	159.8	8.44	not available
			WE LOPAR	W1717WL	159.8	8.44	Zircaloy-4
			WE OFA	W1717WO	159.8	8.44	Zircaloy-4
			WE Pressurized	W1717WP	159.8	8.44	not available
			WE Vantage	W1717WV	159.8	8.44	not available
			WE Vantage +	W1717WV+	159.8	8.44	ZIRLO
			WE Vantage 5	W1717WV5	159.8	8.44	Zircaloy-4
			WE Vantage 5H	W1717WVH	159.8	8.44	not available
South Texas	17 × 17	WE	WE	WST17W	199.0	8.43	Zircaloy-4
Ft. Calhoun	14 × 14	ANF	ANF	XFC14A	146.0	8.10	not available
		CE	CE	XFC14C	146.0	8.10	Zircaloy-4
		WE	WE	XFC14W	146.0	8.10	not available

Table 1.5.1-2. Physical Characteristics of Pressurized Water Reactor Assembly Classes (Continued)

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
Haddam Neck	15 × 15	B&W	B&W SS	XHN15B	137.1	8.42	SS-304
			B&W Zir	XHN15BZ	137.1	8.42	Zircaloy
		GA	Gulf SS	XHN15HS	137.1	8.42	SS
			Gulf Zir	XHN15HZ	137.1	8.42	Zircaloy
		NU	NUM SS	XHN15MS	137.1	8.42	SS
			NUM Zir	XHN15MZ	137.1	8.42	Zircaloy
		WE	WE	XHN15W	137.1	8.42	SS-304
			WE Zir	XHN15WZ	137.1	8.42	not available
Indian Point-1	13 × 14	WE	WE	XIP14W	138.8	6.27	SS
Palisades	15 × 15	ANF	ANF	XPA15A	147.5	8.20	Zircaloy-4
		CE	CE	XPA15C	147.5	8.20	Zircaloy-4
St. Lucie-2	16 × 16	CE	CE	XSL16C	158.2	8.10	Zircaloy-4
San Onofre-1	14 × 14	WE	WE	XSO14W	137.1	7.76	SS-304
			WE D	XSO14WD	137.1	7.76	not available
			WE M	XSO14WM	137.1	7.76	not available
Yankee Rowe	15 × 16	ANF	ANF	XYR16A	111.8	7.62	Zircaloy-4
		CE	CE	XYR16C	111.8	7.62	Zircaloy-4
		UNC	UNC	XYR16U	111.8	7.62	not available
	17 × 18	WE	WE	XYR18W	111.8	7.62	SS

NOTE: Some characteristics of more recently discharged fuel (post-1999) have not yet been provided.

Table 1.5.1-3. Physical Characteristics of Boiling Water Reactor Assembly Classes

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
GE BWR/2,3	7 × 7	ANF	ANF	G2307A	171.2	5.44	Zircaloy-2
	8 × 8	ANF	ANF	G2308A	171.2	5.44	Zircaloy-2
	9 × 9	ANF	ANF	G2309A	171.2	5.44	Zircaloy-2
			ANF IX	G2309AIX	171.2	5.44	Zircaloy-2
	8 × 8	ANF	ANF Pressurized	G2308AP	171.2	5.44	Zircaloy-2
		GE	GE-10	G2308G10	171.2	5.44	Zircaloy-2
	9 × 9	GE	GE-11	G2309G11	171.2	5.44	Zircaloy-2
	7 × 7	GE	GE-2a	G2307G2A	171.2	5.44	Zircaloy-2
			GE-2b	G2307G2B	171.2	5.44	Zircaloy-2
			GE-3	G2307G3	171.2	5.44	Zircaloy-2
	8 × 8	GE	GE-4	G2308G4	171.2	5.44	Zircaloy-2
			GE-5	G2308G5	171.2	5.44	Zircaloy-2
			GE-7	G2308G7	171.2	5.44	NA
			GE-8a	G2308G8A	171.2	5.44	Zircaloy-2
			GE-8b	G2308G8B	171.2	5.44	Zircaloy-2
			GE-9	G2308G9	171.2	5.44	Zircaloy-2
			GE-Barrier	G2308GB	171.2	5.44	Zircaloy-2
			GE-Pressurized	G2308GP	171.2	5.44	Zircaloy-2
	NA	NA	NA	9X9IXQFA	171.2	5.44	NA
GE BWR/4-6	9 × 9	ANF	ANF	G4609A	176.2	5.44	Zircaloy-2
	10 × 10	ANF	ANF	G4610A	176.2	5.44	NA
	9 × 9	ANF	ANF 9-5	G4609A5	176.2	5.44	Zircaloy-2
			ANF 9X	G4609A9X	176.2	5.44	Zircaloy-2
			ANF IX	G4609AIX	176.2	5.44	Zircaloy-2
	10 × 10	ANF	ANF IX	G4610AIX	176.2	5.44	NA
	9 × 9	ANF	ANF X+	G4609AX+	176.2	5.44	NA
	8 × 8	ANF	ANF-Pressurized	G4608AP	176.2	5.44	Zircaloy-2
	NA	AREVA	NA	ATRIUM10	176.2	5.44	Zircaloy-2[a]

Table 1.5.1-3. Physical Characteristics of Boiling Water Reactor Assembly Classes (Continued)

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
GE BWR/4-6 (Continued)	10 × 10	ABB	CE	G4610C	176.2	5.44	NA
	8 × 8	GE	GE-10	G4608G10	176.2	5.44	Zircaloy-2
			GE-11	G4608G11	176.2	5.44	NA
	9 × 9	GE	GE-11	G4609G11	176.2	5.44	Zircaloy-2
	8 × 8	GE	GE-12	G4608G12	176.2	5.44	NA
	10 × 10	GE	GE-12	G4610G12	176.2	5.44	Zircaloy-2
	9 × 9	GE	GE-13	G4609G13	176.2	5.44	Zircaloy-2
	10 × 10	GE	GE-14	G4610G14	176.2	5.44	NA
	7 × 7	GE	GE-2	G4607G2	176.2	5.44	Zircaloy-2
			GE-3a	G4607G3A	176.2	5.44	Zircaloy-2
			GE-3b	G4607G3B	176.2	5.44	Zircaloy-2
	8 × 8	GE	GE-4a	G4608G4A	176.2	5.44	Zircaloy-2
			GE-4b	G4608G4B	176.2	5.44	Zircaloy-2
			GE-5	G4608G5	176.2	5.44	Zircaloy-2
			GE-8	G4608G8	176.2	5.44	Zircaloy-2
			GE-9	G4608G9	176.2	5.44	Zircaloy-2
			GE-Barrier	G4608GB	176.2	5.44	Zircaloy-2
			GE-Pressurized	G4608GP	176.2	5.44	Zircaloy-2
		WE	WE	G4608W	176.2	5.44	Zircaloy-2
Big Rock Point	9 × 9	ANF	ANF	XBR09A	84	6.52	Zircaloy-2
	11 × 11	ANF	ANF	XBR11A	84	6.52	Zircaloy-2
	7 × 7	GE	GE	XBR07G	84	6.52	NA
	8 × 8	GE	GE	XBR08G	84	6.52	NA
	9 × 9	GE	GE	XBR09G	84	6.52	Zircaloy-2
	11 × 11	GE	GE	XBR11G	84	6.52	Zircaloy-2
		NFS	NFS	XBR11N	84	6.52	NA

Table 1.5.1-3. Physical Characteristics of Boiling Water Reactor Assembly Classes (Continued)

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
Dresden-1	6 × 6	ANF	ANF	XDR06A	134.4	4.28	Zircaloy-2
		GE	GE	XDR06G	134.4	4.28	Zircaloy-2
	7 × 7	GE	GE SA-1	XDR07GS	134.4	4.28	NA
	8 × 8	GE	GE PF Fuels	XDR08G	134.4	4.28	NA
	6 × 6	GE	GE Type III-B	XDR06G3B	134.4	4.28	NA
			GE Type III-F	XDR06G3F	134.4	4.28	NA
			GE Type V	XDR06G5	134.4	4.28	NA
		UNC	UNC	XDR06U	134.4	4.28	NA
Humboldt Bay	6 × 6	ANF	6 × 6 ANF	XHB06A	95	4.67	Zircaloy
		GE	GE	XHB06G	95	4.67	Zircaloy-2
	7 × 7	GE	GE Type II	XHB07G2	95	4.67	Zircaloy
LaCrosse	10 × 10	AC	AC	XLC10L	102.5	5.62	SS348H
		ANF	ANF	XLC10A	102.5	5.62	SS348H

NOTE: Some characteristics of more recently discharged fuel (post-1999) have not yet been provided.
NA = not applicable.

Table 1.5.1-4. Assembly Types and Their Main Characteristics as of December 31, 2002

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (^{235}U wt %)			Burnup (MWd/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
BWR	not available	9X9IXQFA	170.713	170.800	3.25	3.25	3.25	39,166	39,248
BWR	AC	XLC10L	120.160	121.034	3.63	3.77	3.94	14,419	21,532
BWR	ANF	G2307A	181.574	183.797	2.56	2.64	2.65	24,256	27,826
BWR	ANF	G2308A	174.624	184.355	2.39	2.66	3.13	28,814	36,826
BWR	ANF	G2308AP	172.753	173.132	2.82	2.83	2.83	34,366	34,826
BWR	ANF	G2309A	168.097	169.520	2.78	3.10	3.15	35,941	40,818
BWR	ANF	G2309AIX	169.185	170.059	3.25	3.31	3.82	39,151	43,778
BWR	ANF	G4608AP	176.175	176.800	2.62	2.88	3.40	31,248	35,518
BWR	ANF	G4609A	172.970	174.700	0.72	3.42	3.73	36,933	47,000
BWR	ANF	G4609A5	176.147	177.000	2.90	3.28	3.55	36,536	43,555
BWR	ANF	G4609A9X	169.155	176.800	2.53	2.87	3.11	36,880	43,330
BWR	ANF	G4609AIX	174.788	177.000	3.00	3.58	3.94	24,156	36,777
BWR	ANF	G4609AX+	167.264	167.277	3.13	3.14	3.15	39,239	40,457
BWR	ANF	G4610A	176.900	176.900	3.94	3.94	3.94	38,207	39,000
BWR	ANF	G4610AIX	175.000	175.000	3.39	3.39	3.39	37,706	38,009
BWR	ANF	XBR09A	127.687	131.406	3.45	3.48	3.52	20,981	22,811
BWR	ANF	XBR11A	130.237	133.174	3.13	3.42	3.82	22,716	34,212
BWR	ANF	XDR06A	95.206	95.478	2.23	2.23	2.24	4,907	5,742
BWR	ANF	XHB06A	69.734	73.800	2.35	2.40	2.41	9,037	22,377
BWR	ANF	XLC10A	108.657	109.609	3.68	3.69	3.71	15,017	20,126
BWR	AREVA	ATRIUM10	176.900	176.900	3.94	3.94	3.94	38,406	39,000
BWR	ABB	G4610C	175.683	176.300	2.51	3.29	3.62	38,133	42,640
BWR	GE	G2307G2A	194.902	197.604	2.07	2.10	2.11	16,775	24,902
BWR	GE	G2307G2B	193.203	197.400	1.65	2.15	2.62	16,384	29,728
BWR	GE	G2307G3	187.419	189.105	1.96	2.41	2.60	25,420	38,861
BWR	GE	G2308G10	172.225	173.512	3.10	3.25	3.56	33,988	43,977
BWR	GE	G2308G4	183.991	185.496	2.19	2.51	2.76	26,087	40,523

Table 1.5.1-4. Assembly Types and Their Main Characteristics as of December 31, 2002 (Continued)

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (^{235}U wt %)			Burnup (MWd/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
BWR	GE	G2308G5	176.971	177.628	2.39	2.66	2.82	29,009	33,597
BWR	GE	G2308G7	178.520	179.400	2.96	2.97	2.99	31,570	35,894
BWR	GE	G2308G8A	175.695	179.584	2.55	3.09	3.40	34,848	44,933
BWR	GE	G2308G8B	172.590	178.000	2.96	3.19	3.39	36,400	42,518
BWR	GE	G2308G9	172.017	173.108	2.85	3.18	3.48	37,268	42,295
BWR	GE	G2308GB	177.983	180.060	2.62	2.80	3.39	32,014	43,381
BWR	GE	G2308GP	177.145	179.200	2.08	2.77	3.01	29,317	38,139
BWR	GE	G2309G11	165.650	169.500	3.10	3.56	3.78	40,522	45,117
BWR	GE	G4607G2	194.729	197.334	1.09	1.56	2.50	9,362	11,829
BWR	GE	G4607G3A	187.455	189.141	1.10	2.33	2.51	21,058	32,188
BWR	GE	G4607G3B	189.925	191.542	1.10	2.31	2.51	21,948	30,831
BWR	GE	G4608G10	177.778	186.094	2.63	3.24	3.70	36,695	44,343
BWR	GE	G4608G11	170.786	171.000	3.38	3.38	3.38	35,194	42,551
BWR	GE	G4608G12	180.873	181.484	3.69	3.71	3.99	32,069	34,462
BWR	GE	G4608G4A	183.931	185.221	2.19	2.62	2.99	24,931	43,430
BWR	GE	G4608G4B	186.709	187.900	2.10	2.31	2.76	21,362	32,941
BWR	GE	G4608G5	183.007	185.366	0.70	2.36	3.01	23,964	38,224
BWR	GE	G4608G8	179.801	185.854	2.95	3.19	3.40	34,905	44,640
BWR	GE	G4608G9	177.738	185.789	1.51	3.23	3.88	36,492	47,062
BWR	GE	G4608GB	184.636	186.653	0.71	2.53	3.25	26,297	45,986
BWR	GE	G4608GP	183.195	186.888	0.70	2.38	3.27	23,112	42,428
BWR	GE	G4609G11	170.123	178.136	1.46	3.56	4.14	40,351	65,149
BWR	GE	G4609G13	171.417	172.912	3.24	3.85	4.17	42,045	53,636
BWR	GE	G4610G12	176.100	182.141	3.12	3.98	4.20	44,175	52,735
BWR	GE	G4610G14	179.127	180.402	4.01	4.11	4.24	5,868	8,915
BWR	GE	XBR07G	131.500	133.000	2.88	2.88	2.88	1,643	1,690
BWR	GE	XBR08G	112.500	113.000	2.85	2.85	2.85	4,546	7,027

Table 1.5.1-4. Assembly Types and Their Main Characteristics as of December 31, 2002 (Continued)

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (^{235}U wt %)			Burnup (MWd/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
BWR	GE	XBR09G	137.088	141.000	3.51	3.58	3.62	15,092	22,083
BWR	GE	XBR11G	124.500	132.000	3.11	3.46	3.63	22,802	24,997
BWR	GE	XDR06G	111.352	111.352	1.47	1.47	1.47	23,522	23,522
BWR	GE	XDR06G3B	101.610	102.520	1.83	1.83	1.83	18,632	27,106
BWR	GE	XDR06G3F	102.049	102.876	2.25	2.25	2.25	22,132	28,138
BWR	GE	XDR06G5	105.857	112.257	2.26	2.26	2.26	21,095	25,886
BWR	GE	XDR07GS	59.000	59.000	3.10	3.10	3.10	29,000	29,000
BWR	GE	XDR08G	99.714	99.714	1.95	1.95	1.95	25,287	25,287
BWR	GE	XHB06G	76.355	77.000	2.35	2.43	2.52	17,170	22,876
BWR	GE	XHB07G2	76.325	77.100	2.08	2.11	2.31	18,187	20,770
BWR	NFS	XBR11N	128.991	134.414	2.16	2.83	3.51	18,940	21,850
BWR	UNC	XDR06U	102.021	103.441	1.83	2.24	2.26	17,685	26,396
BWR	WE	G4608W	156.696	171.403	2.69	2.85	3.01	28,041	33,140
PWR	ANF	C1414A	380.870	400.000	0.30	3.50	4.32	38,899	50,871
PWR	ANF	W1414A	378.274	406.840	0.71	3.42	4.50	37,500	56,328
PWR	ANF	W1414ATR	362.788	368.011	2.39	3.38	3.57	38,168	46,000
PWR	ANF	W1515A	428.888	434.792	2.01	3.00	3.60	33,344	49,859
PWR	ANF	W1515AHT	434.546	438.074	3.51	4.08	4.59	45,441	56,922
PWR	ANF	W1515APL	307.361	310.073	1.23	1.55	1.88	27,971	37,770
PWR	ANF	W1717A	413.845	460.540	2.43	4.19	4.77	45,291	53,958
PWR	ANF	XFC14A	353.345	358.811	3.50	3.57	3.80	37,205	46,048
PWR	ANF	XPA15A	396.674	408.040	1.50	3.17	4.05	34,362	51,486
PWR	ANF	XYR16A	233.555	237.300	3.49	3.78	4.02	29,034	35,088
PWR	B&W	B1515B	463.398	465.480	2.74	3.57	3.62	40,407	50,128
PWR	B&W	B1515B10	476.778	489.299	3.24	3.90	4.73	44,417	56,880
PWR	B&W	B1515B3	463.845	465.830	1.08	2.42	2.84	21,036	32,267
PWR	B&W	B1515B4	464.285	474.853	0.90	2.91	4.06	29,534	57,000

Table 1.5.1-4. Assembly Types and Their Main Characteristics as of December 31, 2002 (Continued)

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (^{235}U wt %)			Burnup (MWd/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
PWR	B&W	B1515B4Z	463.735	466.305	3.22	3.84	3.95	39,253	51,660
PWR	B&W	B1515B5	468.250	468.250	3.13	3.13	3.13	38,017	39,000
PWR	B&W	B1515B5Z	464.421	465.176	3.20	3.22	3.23	36,016	42,328
PWR	B&W	B1515B6	462.495	464.403	3.22	3.47	3.66	41,790	49,383
PWR	B&W	B1515B7	463.244	464.513	3.48	3.51	3.55	42,059	48,738
PWR	B&W	B1515B8	464.864	468.560	3.29	3.65	4.01	42,692	54,000
PWR	B&W	B1515B9	463.566	467.566	3.29	3.96	4.76	44,097	53,952
PWR	B&W	B1515BGD	429.552	430.255	3.92	3.92	3.92	49,027	58,310
PWR	B&W	B1515BZ	463.410	466.279	3.05	3.47	4.68	37,441	54,023
PWR	B&W	B1717B	456.722	457.929	2.64	2.84	3.04	29,517	33,904
PWR	B&W	W1414B	383.157	383.157	3.22	3.22	3.22	24,398	24,465
PWR	B&W	W1717B	455.799	466.688	2.00	3.84	4.60	40,741	54,014
PWR	B&W	XHN15B	409.913	415.060	3.00	3.99	4.02	33,776	37,833
PWR	B&W	XHN15BZ	363.921	368.072	3.40	3.80	3.91	34,278	42,956
PWR	CE	C1414C	382.437	408.508	1.03	3.20	4.48	33,597	56,000
PWR	CE	C1616CSD	413.912	442.986	1.87	3.62	4.63	37,916	63,328
PWR	CE	C8016C	421.468	442.000	1.92	3.57	4.27	38,490	56,312
PWR	CE	XFC14C	362.313	376.842	1.39	2.96	3.95	32,130	52,125
PWR	CE	XPA15C	412.442	416.780	1.65	2.47	3.06	16,020	33,630
PWR	CE	XSL16C	381.018	394.400	1.72	3.44	4.28	38,807	54,838
PWR	CE	XYR16C	228.766	233.400	3.51	3.80	3.92	24,282	35,999
PWR	GA	XHN15HS	406.163	406.163	3.99	3.99	3.99	32,151	32,151
PWR	GA	XHN15HZ	362.863	362.863	3.26	3.26	3.26	18,546	18,546
PWR	NU	XHN15MS	405.979	406.992	3.66	3.66	3.66	28,324	28,324
PWR	NU	XHN15MZ	370.776	371.039	2.95	2.95	2.95	25,643	25,643
PWR	UNC	XYR16U	238.573	241.300	3.96	3.99	4.02	27,461	31,986
PWR	WE	B1515W	461.819	464.763	3.90	4.06	4.22	36,993	49,075

Table 1.5.1-4. Assembly Types and Their Main Characteristics as of December 31, 2002 (Continued)

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (^{235}U wt %)			Burnup (MWd/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
PWR	WE	C1414W	403.483	411.719	2.70	3.15	3.76	30,039	37,781
PWR	WE	W1414W	393.896	403.683	2.26	3.04	3.47	27,315	39,723
PWR	WE	W1414WL	399.092	405.809	2.27	3.07	3.41	31,940	47,932
PWR	WE	W1414WO	355.724	369.265	0.99	3.92	4.95	44,730	69,452
PWR	WE	W1515W	451.193	458.091	2.21	3.00	3.35	29,324	41,806
PWR	WE	W1515WL	455.236	465.600	1.85	2.98	3.80	30,874	55,385
PWR	WE	W1515WO	460.764	465.747	1.91	3.53	4.60	39,071	56,138
PWR	WE	W1515WV5	457.793	462.934	2.99	3.92	4.80	37,556	53,056
PWR	WE	W1717WL	461.323	469.200	1.60	3.12	4.40	32,340	58,417
PWR	WE	W1717WO	425.107	459.433	1.60	3.05	4.02	32,690	53,000
PWR	WE	W1717WP	417.069	417.878	3.73	4.59	4.81	50,707	58,237
PWR	WE	W1717WRF	455.497	456.735	4.00	4.18	4.42	45,530	48,037
PWR	WE	W1717WV	425.399	426.042	4.21	4.38	4.41	44,263	48,385
PWR	WE	W1717WV+	424.010	465.469	1.61	4.16	4.66	45,430	61,685
PWR	WE	W1717WV5	424.269	430.925	1.49	4.01	4.95	43,872	56,570
PWR	WE	W1717WVH	461.954	473.962	2.11	3.87	4.95	41,081	55,496
PWR	WE	W1717WVJ	461.518	465.200	3.71	3.99	4.40	43,922	46,847
PWR	WE	WST17W	540.480	546.600	1.51	3.38	4.41	35,926	54,399
PWR	WE	XFC14W	374.055	376.000	0.27	3.75	4.25	38,521	51,971
PWR	WE	XHN15W	415.557	421.227	3.02	3.59	4.00	27,922	35,196
PWR	WE	XHN15WZ	384.894	386.689	4.20	4.39	4.60	14,321	19,376
PWR	WE	XIP14W	191.152	200.467	2.83	4.12	4.36	16,471	27,048
PWR	WE	XSO14W	368.153	374.885	3.16	3.87	4.02	27,232	39,275
PWR	WE	XSO14WD	373.323	373.643	4.01	4.01	4.02	18,259	18,424
PWR	WE	XSO14WM	311.225	311.225	0.71	0.71	0.71	19,307	19,636
PWR	WE	XYR18W	273.350	274.100	4.94	4.94	4.94	25,484	31,755

Table 1.5.1-5. Summary of Commercial SNF Characteristics as of December 31, 2002

	Initial Uranium Loading		Initial Enrichment		Discharge Burnup	
	Average (kg per assembly)	**Maximum (kg per assembly)**	**Average (wt % ^{235}U)**	**Maximum (wt % ^{235}U)**	**Average (MWd/MTU)**	**Maximum (MWd/MTU)**
PWR	431.0	546.6	3.45	4.95	36,242	69,452
BWR	179.0	197.6	2.77	4.24	28,619	65,149

Table 1.5.1-6. Commercial SNF Fuel Assembly Initial Crud Activities

Radionuclide	PWR (μCi/cm^2)	BWR (μCi/cm^2)
^{60}Co	140	1,254
^{55}Fe	5,902	7,415

NOTE: Crud activities are bounding estimated based on analysis of measured crud activity data.

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister

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	Design Criteria
DOE and Commercial Waste Package System (DS)	Canistered Spent Nuclear Fuel	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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the canister capacity curve.
				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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
Mechanical Handling System (H)	Waste Transfer/ Canister Transfer	TAD Staging Racks (and Thermal Barrier) (060-HTC0-RK-00011/00012)	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.	The thermal barrier (around the staging racks) is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the reliabilities of each component.)

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister (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	Design Criteria
DOE and Commercial Waste Package System (DS)	Canistered Spent Nuclear Fuel	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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the canister capacity curve.
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the canister capacity curve.
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
		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.	The aging overpack and canister are required to be designed such that the canister maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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	The aging overpack and canister are required to be designed such that the canister maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The horizontal aging module and canister are required to be designed such that the canister maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The horizontal aging module and canister are required to be designed such that the canister maximum effective plastic strain from low speed impact or collisions meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The cask and horizontal shielded transfer cask are required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The horizontal aging module is required to be designed such that the thermal penetration of the horizontal aging module meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-7. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the TAD Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

NOTE: "Protect against" in this table means either "reduce the probability of" or "reduce the frequency of."
For casks, canisters, and associated handling equipment that were previously designed, the component design will be evaluated to confirm that the controlling parameters and values are met.
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.
The numbers appearing in parentheses in the third column are component numbers.
Facility Codes: CR: Canister Receipt and Closure Facility; RF: Receipt Facility; SB: Balance of Plant; WHF: Wet Handling Facility.
System Codes: DS: DOE and Commercial Waste Package System; H: Mechanical Handling System.
Subsystem Codes: HTC: Canister Transfer.

Table 1.5.1-8. Summary of Conformance of Commercial SNF to Postclosure Control Parameters

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
Waste Package	03-09 Waste Package Worst-Case Dose Rate	The waste package containing the TAD canister with 21 PWR fuel assemblies shall represent the worst-case dose rate (80 GWd/MTU burnup, 5% ^{235}U enrichment and 5 years decay).	No	NA (Background Information: Gamma and neutron sources for maximum commercial SNF for shielding are provided in Section 1.10.)	Procedures will be developed that will limit the receipt of TAD canisters per the subject constraint. The acceptance process described in Section 1.5.1 describes the bases on which DOE will receive commercial SNF from the vendors. Procedures will control these receipts, and the loading of the subject TAD canisters into waste packages will satisfy this constraint.
Waste Form TAD	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	NA	Procedures will be developed for handling process in accordance with the guidelines of Section 5.6. Adherence to those guidelines requires consideration of industry applications and lessons learned during the procedure preparation. For those processes precedented in the nuclear industry, those applications and lessons learned will be incorporated. Activities will be performed by an operator and an independent observer and records demonstrating compliance will be prepared.
Waste Form	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	NA	Procedures will be developed for handling uncanistered SNF in accordance with the guidelines of Section 5.6. Adherence to those guidelines requires consideration of industry applications and lessons learned during the procedure preparation. For those processes precedented in the nuclear industry, those applications and lessons learned will be incorporated.

Table 1.5.1-8. Summary of Conformance of Commercial SNF to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter		Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
	Parameter Number and Title	**Values, Ranges of Values or Constraints**			
Waste Form and TAD Canister	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 PWR burnup) (NRC 1997, Section 8.V.1).	No	NA	Procedures will be developed in accordance with the *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008c) (Section 1.5.1.1.1.2.1.4), that will control the loading of TAD canisters with uncanistered commercial SNF of 80 GWd/MTU or less. It is planned that this limit for commercial SNF packaged in TAD canisters at the generator's site will conform to the 10 CFR Part 71 and 10 CFR Part 72 certificate of compliance issued by the NRC when the TAD canister is approved.
Waste Form and TAD Canister	04-04 Waste From Moisture Removal and Inerting	TAD canisters shall be dried and backfilled with helium in a manner consistent with that described in *Standard Review Plan for Dry Cask Storage Systems* (NUREG 1536) (NRC 1997), Section 8.V.1.[a]	Yes	NA	Procedures will be developed based upon the vendor's recommendation for drying and backfilling canisters once canisters have been developed in accordance with the *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008c) (Section 1.5.1.1.1.2.6.1.2). Operations in the WHF, when loading and inerting TAD canisters at the repository, will be controlled by procedures per the subject constraint. Operations at the commercial SNF generator's site will be controlled by the 10 CFR Part 71 and 10 CFR Part 72 certificate of compliance issued by the NRC when the TAD canister is approved, and will provide control of loading operations both at the utility site and GROA.

Table 1.5.1-8. Summary of Conformance of Commercial SNF to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
Waste Form	04-08 Handling of Waste Forms	Waste form handling operations shall be performed in a standard industry fashion to limit damage. An operator and an independent checker shall perform the operations.	No	NA	Procedures will be developed for handling waste forms in accordance with the guidelines of Section 5.6. Adherence to those guidelines requires consideration of industry applications and lessons learned during the procedure preparation. For those processes precedented in the nuclear industry, those applications and lessons learned will be incorporated. Activities will be performed by an operator and an independent observer and records demonstrating compliance will be prepared.
Waste Form and TAD Canister	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	NA	At the GROA, the procedures for TAD canister loading and canister transfer procedures will specifically identify the items that are allowed to be placed inside a TAD canister (PWR or BWR SNF assemblies or damaged fuel cans) or waste package (TAD, naval SNF canister, DOE SNF canister, or HLW canister). The procedures will also identify steps to exclude foreign material. The loading plans prepared before a particular waste package or TAD canister is loaded will uniquely identify the items to be placed in the waste package or TAD canister by the canister, TAD canister, or commercial assembly unique identifiers. Controls and accountability logs combined with closeout inspections, as appropriate, will be established to limit unauthorized materials entry into the canister or waste package. Loading of the TAD canister at off-GROA locations will be controlled by the certificate of compliance issued by the NRC as part of the TAD certification for 10 CFR Part 71 and 10 CFR Part 72.

NOTE: [a]For postclosure moisture removal purposes, NUREG-1536 (NRC 1997) and NUREG-1567 (NRC 2000a) are equivalent in their requirements for moisture removal.
NA = not applicable.

Table 1.5.1-9. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Dual-Purpose Canister

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
DOE and Commercial Waste Package System (DS)	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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the canister capacity curve.
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-9. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Dual-Purpose Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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)	Provide containment (Continued)	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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister are required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the canister capacity curve.
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-9. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Dual-Purpose Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curve.
				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.	The canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the canister capacity curve.

Table 1.5.1-9. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Dual-Purpose Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced hazard curve when convolved with the fire hazard meets the required reliability. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
		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.	The aging overpack and canister are required to be designed such that the canister maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-9. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Dual-Purpose Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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 (Both Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	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.	The aging overpack and canister are required to be designed such that the canister maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The horizontal aging module and canister are required to be designed such that the canister maximum effective plastic strain from a drop meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The horizontal aging module and canister are required to be designed such that the canister maximum effective plastic strain from low speed impact or collisions meets the required reliability when evaluated against the canister capacity curves. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The cask and horizontal shielded transfer cask are required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-9. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Dual-Purpose Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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 (Both Analyzed as a Representative Canister) (Continued)	Provide containment (Continued)	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.	The horizontal aging module is required to be designed such that the thermal penetration of the horizontal aging module meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The aging overpack is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

NOTE: For casks, canisters, and associated handling equipment that were previously designed, the component design will be evaluated to confirm that the controlling parameters and values are met.
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.
Facility Codes: CR: Canister Receipt and Closure Facility; RF: Receipt Facility; SB: Balance of Plant; WH: Wet Handling Facility.
System Codes: DS: DOE and Commercial Waste Package.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(1)a Structural	TAD Canister/ Transportation Cask/Aging Overpack	Leakage Rate/ Temperature/ Design Code Compliance	a. Following a 2,000-year seismic return period event, a TAD canister shall maintain a maximum leakage rate of 1.5 × 10^{-12} fraction of canister free volume per second (normal), maximum cladding temperature of 752°F (normal) and remain within design codes while in the configurations described below: • While suspended by a crane inside an ASTM A 36/A 36M-04 cylindrical steel cavity with an inner diameter of 72.5 in. with 12-in.-thick wall. • While contained in a vendor-defined transportation cask (with impact limiters) described in Section 3.2 of the TAD canister performance specification (DOE 2008c). • While contained in a vendor-defined transportation cask (without impact limiters), described in Section 3.2 of the TAD canister performance specification (DOE 2008c) that is constrained in an upright position. A constrained transportation cask is one properly secured into GROA transfer trolley and restrained from tipover in a seismic event. • While contained in a vendor-defined aging overpack as described in Section 3.3 of the TAD canister performance specification (DOE 2008c).	1.7	Cask vendor to provide both canister leakage rate and cladding temperature analyses for the TAD canister unaffected by failure of other SSCs *Seismic Event Sequence Quantification and Categorization* (BSC 2008b)	Tables 1.7-10, 1.7-12, and 1.7-16 identify the seismic event sequences that could affect a TAD canister in the RF, the CRCF, and the Intrasite, respectively. Each of the seismic event sequences are categorized as beyond Category 2. Accordingly, no consequence analyses are performed for these seismic event sequences.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(1)b Structural	TAD Canister/ Transportation Cask/Aging Overpack	Leakage Rate/ Temperature/ Design Code Compliance	b. Following a 10,000-year seismic return period event, a TAD canister shall maintain a maximum leakage rate of 1.5 × 10^{-12} fraction of canister free volume per second (normal), cladding temperature of 1,058°F (off-normal), and remain within design codes while in the configurations described below: • While suspended by a crane inside an ASTM A 36/A 36M-04 cylindrical steel cavity with an inner diameter of 72.5 in. with 12-in.-thick wall. • While contained in a vendor-defined transportation cask (with impact limiters) described in Section 3.2 of the TAD canister performance specification (DOE 2008c). • While contained in a vendor-defined transportation cask (without impact limiters) described in Section 3.2 of the TAD canister performance specification (DOE 2008c) that is constrained in an upright position. A constrained transportation cask is one properly secured into GROA transfer trolley and restrained from tipover in a seismic event. • While contained in a vendor-defined aging overpack as described in Section 3.3 of the TAD canister performance specification (DOE 2008c).	1.7	Cask vendor to provide both canister leakage rate and cladding temperature analyses for the TAD canister unaffected by failure of other SSCs *Seismic Event Sequence Quantification and Categorization* (BSC 2008b)	Tables 1.7-10, 1.7-12, and 1.7-16 identify the seismic event sequences that could affect a TAD canister in the RF, the CRCF, and the Intrasite, respectively. Each of the seismic event sequences are categorized as beyond Category 2. Accordingly, no consequence analyses are performed for these seismic event sequences.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(1)c Structural	TAD Canister/ Transportation Cask/Aging Overpack	Leakage Rate	c. Following a seismic event characterized by horizontal and vertical peak ground accelerations of 96.52 ft/sec^2 (3 g), a TAD canister shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) while in the configurations described below. For this initiating event, canister design codes may be exceeded (i.e., vendor may rely on capacity in excess of code allowances). • A TAD canister in a vendor-defined transportation cask described in Section 3.2 of the TAD canister performance specification (DOE 2008c) that drops 10 ft onto an unyielding surface in the most damaging orientation. The transportation cask configuration shall be with or without impact limiters. • While contained in a vendor-defined transportation cask (without impact limiters) described in Section 3.2 of the TAD canister performance specification (DOE 2008c) that is constrained in an upright position. A constrained transportation cask is one properly secured into GROA transfer trolley and restrained from tipover in a seismic event.	1.7	Cask vendor to provide canister leakage analyses for the TAD canister unaffected by failure of other SSCs *Seismic Event Sequence Quantification and Categorization* (BSC 2008b)	Tables 1.7-10, 1.7-12, and 1.7-16. identify the seismic event sequences that could affect a TAD canister in the RF, the CRCF, and the Intrasite, respectively. Each of the seismic event sequences are categorized as beyond Category 2. Accordingly, no consequence analyses are performed for these seismic event sequences.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(1)c Structural (Continued)			• While contained in a vendor-defined aging overpack as described in Section 3.3 of the TAD canister performance specification (DOE 2008c).			
3.1.2(2) Structural	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature	A TAD canister in a vendor-defined aging overpack shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) and cladding temperature limits of 752°F and 1058°F for “normal” and “off-normal” limits, respectively, during and following exposure to the environmental conditions listed below. a. These environmental conditions are not cumulative but occur independently: • Outdoor average temperature range of 2°F to 116°F with insulation as specified in 10 CFR Part 71 (normal) • An extreme wind gust of 120 mph for 3 sec (normal) • Maximum tornado wind speed of 189 mph with a corresponding pressure drop of 0.81 lb/in.^{2} and a rate of pressure drop of 0.30 lb/in.^{2}/sec (off-normal). The spectrum of missiles from the maximum tornado is provided below (off-normal):	1.1.3.2 1.1.3.6.1 1.6.3.4.4	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(2) Structural (Continued)			**Spectrum of Missiles** Type of Missile: Mass (lb) Dimensions (ft) Horizontal velocity (ft/s) Wood Plank: 114.6 0.301 × 0.948 × 12 190.2 6-in. Sch. 40 pipe: 286.6 0.551D × 15.02 32.8 1-in. steel rod: 8.8 0.0833D × 3 26.3 Utility pole: 1,124 1.125D × 35.04 85.3 12-in. Sch. 40 pipe: 749.6 1.05D × 15.02 23.0			

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(2) Structural (Continued)			b. Annual precipitation of 20 in./yr (normal). The spectrum of rainfall is provided below (normal): **Spectrum of Rainfall** Parameter (Frequency) – Nominal Estimate – Upper Bound 90% Confidence Interval[a] Max. 24-hour precipitation (50-year return period) – 2.79 in./day – 3.30 in./day Max. 24-hour precipitation (100-year return period) – 3.23 in./day – 3.84 in./day Max. 24-hour precipitation (500-year return period) – 4.37 in./day – 5.25 in./day Precipitation 1-hour intensity (50-year return period) – 1.35 in./hr – 1.72 in./hr Precipitation 1-hour intensity (100-year return period) – 1.68 in./hr – 2.15 in./hr	1.1.3.2 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(2) Structural (Continued)			c. Maximum daily snowfall of 6.0 in. (normal)	1.1.3.6.4 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.
			d. Maximum monthly snowfall of 6.6 in. (normal)	1.1.3.6.4 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(2) Structural (Continued)			e. A lightning strike with a peak current of 250 kiloamps over a period of 260 microseconds and continuous current of 2 kiloamps for 2 seconds (off-normal)	1.1.3.6.2 1.6.3.4.6	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.
3.1.2(3) Structural	TAD Canister/ Transportation Cask	Leakage Rate/ Temperature	A TAD canister in a transportation cask (with impact limiters) shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (off-normal) and cladding temperature limits of 752°F and 1058°F for “normal” and “off-normal,” respectively, during and following exposure to the environmental conditions listed below. a. These environmental conditions are not cumulative but occur independently: • Outdoor average temperature range of 2°F to 116°F with insulation as specified in 10 CFR Part 71 (normal) • An extreme wind gust of 120 mph for 3 sec (normal)	1.1.3.2 1.1.3.6.1 1.2.8.4.5 1.6.3.4.4	Cask vendor to provide canister leakage and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(3) Structural (Continued)			• Maximum tornado wind speed of 189 mph with a corresponding pressure drop of 0.81 lb/in.^{2} and a rate of pressure drop of 0.30 lb/in.^{2}/sec (off-normal). The spectrum of missiles from the maximum tornado is provided below (off-normal): **Spectrum of Missiles** Type of Missile: Mass (lb) Dimensions (ft) Horizontal Velocity (ft/s). Wood Plank: 114.6 0.301 × 0.948 × 12 190.2 6 in. Sch. 40 Pipe: 286.6 0.551D × 15.02 32.8 1 in. Steel Rod: 8.8 0.0833D × 3 26.3 Utility Pole: 1,124 1.125D × 35.04 85.3 12 in. Sch. 40 Pipe: 749.6 1.05D × 15.02 23.0			

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(3) Structural (Continued)			b. Annual precipitation of 20 in./yr (normal). The spectrum of rainfall is provided below (normal): **Spectrum of Rainfall** Parameter (Frequency) – Nominal Estimate – Upper Bound 90% Confidence Interval[a] Max. 24-hr precipitation (50-year return period) – 2.79 in./day – 3.30 in./day Max. 24-hr precipitation (100-year return period) – 3.23 in./day – 3.84 in./day Max. 24-hr precipitation (500-year return period) – 4.37 in./day – 5.25 in./day Precipitation 1-hr intensity (50-year return period) – 1.35 in./hr – 1.72 in./hr Precipitation 1-hr intensity (100-year return period) – 1.68 in./hr – 2.15 in./hr	1.1.3.2 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(3) Structural (Continued)			c. Maximum daily snowfall of 6.0 in. (normal)	1.1.3.6.4 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.
			d. Maximum monthly snowfall of 6.6 in. (normal)	1.1.3.6.4 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.2(3) Structural (Continued)			e. A lightning strike with a peak current of 250 kiloamps over a period of 260 microseconds and continuous current of 2 kiloamps for 2 seconds (off-normal)	1.1.3.6.2 1.6.3.4.6	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.
3.1.3(1) Thermal	TAD Canister/ Aging Overpack	Temperature	Except as noted in Section 3.1.3(2) of the TAD canister performance specification (DOE 2008c), commercial SNF cladding temperature in TAD canisters shall not exceed 752°F during normal operations. Normal operations include storage at purchaser sites, transportation from purchasers to the GROA, and handling at the GROA (e.g., aging, storage, onsite transfer).		Cask vendor to provide cladding temperature analyses. *Thermal Loading Study of the TAD Waste Package* (BSC 2008d)	The cladding temperature requirement in the TAD canister performance specifications bounds the calculated temperatures for the TAD canister during normal operations in the GROA. Thermal performance calculations for normal operations at the aging facilities will be performed by the vendor.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.3(2) Thermal	TAD Canister	Temperature	Commercial SNF cladding temperature shall not exceed 1,058°F during draining, drying, and backfill operations following TAD canister loading.	1.2.4 1.2.5		Based on the thermal characteristics of the commercial SNF assemblies when received at the WHF, commercial SNF assemblies will be loaded in conformance with vendor thermal loading limits (similar to 10 CFR Part 71 and 10 CFR Part 72 certificate of compliance loading limits) so that the cladding temperature does not exceed the thermal limit.
3.1.3(3) Thermal	TAD Canister/ Transportation Cask	Leakage Rate/ Temperature	The maximum leakage rate of a TAD canister shall be 9.3×10^{-10} fraction of canister free volume per second (off-normal) after a fully engulfing fire characterized by an average flame temperature of 1,720°F and lasting 30 minutes. During this event, the TAD canister is in either a closed vendor-defined transportation cask (with or without impact limiters) or an open vendor-defined transportation cask without impact limiters. For this event, canister design codes may be exceeded (i.e., vendor may rely on capacity in excess of code allowances).	1.6.3.4.10 1.6.3.5 1.7.1 1.7.2.3.3	Cask vendor to provide canister leakage rate analyses *External Initiating Events Screening Analysis* (BSC 2008c) *Construction Hazards Screening Analysis* (BSC 2008e)	External fire event sequences and construction hazard fire event sequences for the TAD canister were determined to be beyond Category 2. Large facility fires and localized fires affecting TAD canisters in the CRCF and WHF are categorized as Category 2 in Tables 1.7-11 and 1.7-13, respectively. The TAD canister performance specification requirement bounds the fire requirements for the waste handling facilities (1472°F for 30 minutes) in the GROA.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.3(5) Thermal	TAD Canister	Temperature	To ensure adequate thermal performance of the TAD canister when emplaced in the waste package, the peak cladding temperature shall be less than 662°F for each set of the following conditions. **Thermal Conditions for Cladding Temperature Determination** Thermal Output (kW) – Canister surface temperature boundary conditions (°F) 11.8 kW 525°F 18 kW 450°F 25 kW 358°F	1.3.1	*Thermal Loading Study of the TAD Waste Package* (BSC 2008d)	The thermal loading study demonstrates that the waste package containing a TAD canister will meet the peak cladding thermal limits.
3.1.4(1) Dose and Shielding	TAD Canister	Dose Rate	For GROA operations, the combined neutron and gamma integrated dose rate over the top surface of a loaded TAD canister shall not exceed 800 mrem/hr on contact.	1.10	Cask vendor to provide shielding/dose rate calculation	The limit is bounded by 1 rem/hr as analyzed in *Shielding Calculation for Canister Receipt and Closure Facility 1 and Receipt Facility* (BSC 2008f).
3.1.4(2) Dose and Shielding	TAD Canister	Dose Rate	For GROA operations, the combined neutron and gamma maximum dose rate at any point on the top surface of a loaded TAD canister shall not exceed 1,000 mrem/hr.	1.10	Cask vendor to provide shielding/dose rate calculation	The limit is bounded by 1 rem/hr as analyzed in *Shielding Calculation for Canister Receipt and Closure Facility 1 and Receipt Facility* (BSC 2008f).

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.5(1) Criticality	TAD Canister	Reactivity (k_{eff})	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 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 (10 CFR 71, Subpart E, Paragraph 55(b)).	1.14.2 2.2.1	*Preclosure Criticality Analysis Process Report* (BSC 2008g)	Subcriticality is maintained based on moderator control for sealed TAD canisters and soluble boron for TAD canisters loaded in the WHF pool.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.5(2) Criticality	TAD Canister	Reactivity (k_{eff})	Postclosure criticality control shall be maintained by employing either the items in (a) or the analysis in (b), as follows: a. Include the following features in the TAD canister internals: 1. Neutron absorber plates or tubes made from borated stainless steel produced by powder metallurgy and meeting ASTM A 887-89, *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application,* Grade “A” alloys. 2. Minimum thickness of neutron absorber plates shall be 0.4375 in. Maximum and nominal thickness may be based on structural requirements. Multiple plates may be used if corrosion assumptions (250 nm/yr) are taken into account for all surfaces such that 6 mm remains after 10,000 years. To date, postclosure performance has been analyzed based on the use of single neutron absorber plates. If a cask vendor were to design a TAD canister using multiple neutron absorber plates, similar analyses based on the use of multiple neutron absorber plates would need to be performed.	2.2.1.4.1	Cask vendor	Nuclear criticality during the postclosure is screened from the postclosure performance assessment based on low probability and conformance to this design.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.5(2) Criticality (Continued)			3. The neutron absorber plate shall have a boron content of 1.1 wt % to 1.2 wt %, a range that falls within the specification for Stainless Steel Type 304B4 (UNS S30464) as described in ASTM A 887-89, *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application*. 4. Neutron absorber plates or tubes shall extend along the full length of the active fuel region inclusive of any axial shifting of the assemblies within the TAD canister. 5. Neutron absorber plates or tubes must cover all four longitudinal sides of each fuel assembly.			

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.5(2) Criticality (Continued)			6. TAD canister designs for PWR fuel assemblies shall accommodate assemblies loaded with a disposal control rod assembly. A disposal control rod assembly is intended for acceptance of PWR commercial SNF with characteristics outside limits set in the postclosure criticality loading curves. Current postclosure criticality loading curves are shown in Attachment B of the TAD canister performance specification (DOE 2008c). Updated postclosure criticality loading curves that represent a PWR TAD canister with features described in Items 1 through 5 above may be provided at a later date.			

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.5(2) Criticality (Continued)			b. Perform analyses of TAD canister-based systems to ensure the maximum calculated effective neutron multiplication factor (k_{eff})[b] for a TAD canister containing the most reactive commercial SNF for which the design is approved shall not exceed the critical limit[c] for the following four postclosure archetypical proxy configurations: 1. Nominal case, basket assembly degraded, commercial SNF intact. 2. Seismic case-I, basket assembly intact, commercial SNF degraded. 3. Seismic case-II, basket assembly degraded, commercial SNF degraded. 4. Igneous intrusion case, basket assembly degraded, commercial SNF degraded, waste package and TAD canister structural deformation.	2.2.1.4.1.1	*Disposal Criticality Analysis Methodology Topical Report* (YMP 2003) *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008c) *CSNF Loading Curve Sensitivity Analysis* (SNL 2008d)	Analyses were performed using the standard established computer codes MCNP and SCALE for criticality analyses and SCALE/SAS2H for depletion calculations. Postclosure criticality events were screened out based on these analyses.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.1.6(5) Containment	TAD Canister	Leakage Rate	A loaded TAD canister shall maintain a leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) and cladding temperature below 752°F (normal) following a 12-in. vertical flat-bottom drop. The impacted surface is a solid carbon steel plate, simply supported as shown in Figure 3.1-1 of the TAD canister performance specification (DOE 2008c). The material conforms to ASTM A 36/A 36M-04, *Standard Specification for Carbon Structural Steel.* Centerline of the TAD canister may be offset from centerline of the plate by as much as 3 in.	1.7	Cask vendor to provide both canister leakage rate and cladding temperature analyses; *Canister Receipt and Closure Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008h) *Wet Handling Facility Reliability and Event Sequence Categorization Analysis* (BSC 2008i)	Tables 1.7-11 and 1.7-13 identify the drop event sequences for the TAD canister in the CRCF and the WHF. This 12-in. vertical flat-bottom drop onto a solid carbon steel plate corresponds to a TAD canister drop inside the canister transfer machine in the CRCF or WHF. For both facilities, the drop events are categorized as beyond Category 2 and no further consequence assessments are necessary.
3.2.3 Thermal	TAD Canister/ Transportation Cask	Temperature	During normal operations, the commercial SNF cladding temperature in the TAD canister (contained within a transportation cask) shall not exceed 752°F. Normal operations include transportation from purchaser sites to the GROA.		Cask vendor to provide cladding temperature analyses. *Thermal Loading Study of the TAD Waste Package* (BSC 2008d)	The cladding temperature requirement in the TAD canister performance specifications exceed the calculated temperatures for the TAD canister during normal operations in the GROA. Normal operations at the aging facilities will be performed by the vendor.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(1)a Structural	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature/ Design Code Compliance/ Structural Stability	Following a 2,000-year seismic return period event: • TAD canister in an aging overpack shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal). • Maintain a maximum cladding temperature of 752°F (normal). • Canister design codes shall not be exceeded. • The aging overpack shall remain upright and freestanding.	1.7	Cask vendor to provide canister leakage rate, cladding temperature, and aging overpack stability analyses *Seismic Event Sequence Quantification and Categorization* (BSC 2008b)	Tables 1.7-10, 1.7-12, and 1.7-16. identify the seismic event sequences that could affect a TAD canister in the RF, the CRCF, and the Intrasite, respectively. Each of the seismic event sequences are categorized as beyond Category 2. Therefore, no consequence analyses are necessary for these seismic event sequences.
3.3.2(1)b Structural	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature/ Design Code Compliance/ Structural Stability	Following a 10,000-year seismic return event: • TAD canister in an aging overpack shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal). • Maintain a maximum cladding temperature of 1058°F (off-normal). • Canister design codes shall not be exceeded. • The aging overpack shall remain upright and free standing.	1.7	Cask vendor to provide canister leakage rate, cladding temperature, and aging overpack stability analyses *Seismic Event Sequence Quantification and Categorization* (BSC 2008b)	Tables 1.7-10, 1.7-12, and 1.7-16. identify the seismic event sequences that could affect a TAD canister in the RF, the CRCF, and the Intrasite, respectively. Each of the seismic event sequences are categorized as beyond Category 2. Therefore, no consequence analyses are necessary for these seismic event sequences.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(1)c Structural	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature/ Structural Stability	Following a seismic event characterized by horizontal and vertical peak ground accelerations of 96.52 ft/sec^2 (3 g): • TAD canister in an aging overpack, shall maintain a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal). • Canister design codes may be exceeded (i.e., vendor may rely on capacity in excess of code allowances). • The aging overpack shall remain upright and free standing.	1.8 1.7	Cask vendor to provide canister leakage rate and aging overpack stability analyses *Seismic Event Sequence Quantification and Categorization* (BSC 2008b)	Tables 1.7-10, 1.7-12, and 1.7-16. identify the seismic event sequences that could affect a TAD canister in the RF, the CRCF, and the Intrasite, respectively. Each of the seismic event sequences are categorized as beyond Category 2. Therefore, no consequence analyses are necessary for these seismic event sequences.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(2) Structural	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature	During GROA operations, aging overpack shall be designed to maintain a maximum TAD canister leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) and cladding temperature limits of 752°F and 1058°F for “normal” and “off-normal” limits, respectively, during and following exposure to the environmental conditions listed below (which are not cumulative but occur independently): a. These environmental conditions are not cumulative but occur independently: • Outdoor average daily temperature range of 2°F to 116°F with insulation as specified in 10 CFR Part 71 (normal) • An extreme wind gust of 120 mph for 3 sec (normal) • Maximum tornado wind speed of 189 mph with a corresponding pressure drop of 0.81 lb/in.^{2} and a rate of pressure drop of 0.30 lb/in.^{2}/sec (off-normal). The spectrum of missiles from the maximum tornado is provided below (off-normal)	1.1.3.2 1.1.3.6.1 1.6.3.4.4	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(2) Structural (Continued)			**Spectrum of Missiles** Missile: Mass (lb) Dimensions (ft) Horizontal Velocity (ft/s). Wood Plank: 114.6 0.301 × 0.948 × 12 190.2 6-in. Sch. 40 Pipe: 286.6 0.551D × 15.02 32.8 1-in. Steel Rod: 8.8 0.0833D × 3 26.3 Utility Pole: 1,124 1.125D × 35.04 85.3 12-in. Sch. 40 Pipe: 749.6 1.05D × 15.02 23.0			

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(2) Structural (Continued)			b. Annual precipitation of 20 in./yr (normal). The spectrum of rainfall is provided below (normal): **Spectrum of Rainfall** Parameter (Frequency) – Nominal Estimate – Upper Bound 90% Confidence Interval[a] Max. 24-hr precipitation (50-year return period) – 2.79 in./day – 3.30 in./day Max. 24-hr precipitation (100-year return period) – 3.23 in./day – 3.84 in./day Max. 24-hr precipitation (500-year return period) – 4.37 in./day – 5.25 in./day Precipitation 1-hr intensity (50-year return period) – 1.35 in./hr – 1.72 in./hr Precipitation 1-hr intensity (100-year return period) – 1.68 in./hr – 2.15 in./hr	1.1.3.2 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(2) Structural (Continued)			c. Maximum daily snowfall of 6.0 in. (normal)	1.1.3.6.4 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.
			d. Maximum monthly snowfall of 6.6 in. (normal)	1.1.3.6.4 1.6.3.4.5	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.2(2) Structural (Continued)			e. A lightning strike with a peak current of 250 kiloamps over a period of 260 microseconds and continuous current of 2 kiloamps for 2 seconds (off-normal)	1.1.3.6.2 1.6.3.4.6	Cask vendor to provide canister leakage rate and cladding temperature analyses under these site-specific environmental conditions *External Events Hazards Screening Analysis* (BSC 2008c)	No Category 1 or Category 2 event sequences were identified for extreme site environmental conditions. Therefore, cask vendor calculations bound the analyses needed for the preclosure safety demonstration.
3.3.2(4) Structural	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature	The TAD canister in an aging overpack shall be designed to a maximum leakage rate of 1.5×10^{-12} fraction of canister free volume per second (normal) and maximum cladding temperature of 1058°F (off-normal) following 4 in. of volcanic ash accumulation. The aging overpack may be on a site transporter. The ash fall loads are estimated at 21 lb/ft^2 with a thermal conductivity of 0.11 BTU/hr-ft-°F.	1.6.3.4.3	Cask vendor to provide canister leakage rate and cladding temperature analyses	The probability of a volcanic activity event is 5×10^{-9} per year and therefore beyond category 2. Consequently, the volcanic activity event at the surface, including ashfall on the TAD aging overpacks, is screened out as a beyond Category 2 event sequence.
3.3.2(5) Structural	TAD Canister/ Aging Overpack	Structural Stability	The aging overpack shall retain the TAD canister following a drop and/or tipover event.	1.6	Cask vendor to provide aging overpack retention of canister analyses	

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.3(2) Thermal	TAD Canister/ Aging Overpack	Leakage Rate/ Temperature	A loaded aging overpack shall be capable of withstanding a fully engulfing fire without the TAD canister exceeding a leak rate of 9.3 × 10^{-10} fraction of canister free volume per second (off-normal) and maximum cladding temperature of 1058°F (off-normal) under the conditions below. b. The fire described in 10 CFR 71.73c(4) requirements as modified below. 1. The 30-minute period shall be replaced by a period to be determined by calculation of a pool spill fire formed by 100 gal of diesel fuel. 2. Additionally, a surrogate fully engulfing fire of duration twice the duration of the pool fire, which starts simultaneously with the pool fire and with a steady-state heat release rate of 10 MW, shall be used to model the burning rate of all other solid and liquid combustible materials. For this purpose, assume the heat transfer conditions specified in 10 CFR 71.73c(4). Temperature conditions from this fire shall be consistent with a totally engulfing black body emitting from the 10-MW requirement.	1.6.3.4.10 1.6.3.5 1.7.1 1.7.2.3.3	Cask vendor to provide canister leakage rate and cladding temperature analyses *External Events Hazards Screening Analysis* (BSC 2008c) *Construction Hazards Screening Analysis* (BSC 2008e)	External fire event sequences and construction fire sequences for the TAD canister in the aging overpack (Table 1.7-10) are categorized as beyond Category 2. No further consequence assessments are necessary.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.3(2) Thermal (Continued)			c. A loaded aging overpack shall withstand a deflagration blast wave, fuel tank projectiles, and incident thermal radiation resulting from the worst-case engulfing fire determined in the previous fire protection requirement without the TAD canister exceeding a leakage rate of 9.3×10^{-10} fraction of canister free volume per second (off-normal) and maximum cladding temperature of 1058°F (off-normal).	1.7.2.3.3	Cask vendor to provide canister leakage rate and cladding temperature analyses. *Intrasite Operations and Balance of Plant BOP Reliability and Event Sequence Categorization Analysis* (BSC 2007b).	
3.3.4 Dose and Shielding	TAD Canister/ Aging Overpack	Dose Rate	When the loaded aging overpack is on the aging pad with its vertical axis in its normal orientation, then combined neutron and gamma contact dose rate on any accessible surface (excluding the underside of the aging overpack) shall not exceed 40 mrem per hour at any location. This is inclusive of air circulation ducts, penetrations, and other potential streaming paths on the overpack surface.	1.2.7.6.3	Cask vendor to provide canister surface dose rate/shielding analyses.	Dose rates considered as part of total worker dose reported in Section 1.8.

Table 1.5.1-10. Demonstration of Transportation, Aging, and Disposal Canister System Compliance with Transportation, Aging, and Disposal Canister Performance Specification Requirements (Continued)

TAD Canister Performance Specification Section	TAD Canister Component(s) Involved	Parameter(s) Specified	TAD Canister Performance Specification/Requirement	Applicable SAR Section(s)	Supporting SAR Methodology/ Calculation(s)	TAD Canister System Compliance Determination
3.3.6 Containment	TAD Canister/ Aging Overpack	Leakage Rate	The aging overpack shall be designed such that following a 3-ft vertical drop or tipover from a 3-ft-high site transporter, the TAD canister maximum leakage rate is 9.3×10^{-10} fraction of canister free volume per second (off-normal) under applicable repository environmental conditions. The impacted surface characteristics are as follows: 1. 5,000 psi concrete with a minimum thickness of 3 ft with a broom finish. 2. Reinforcing steel shall be #11s on 8 in. centers, each direction, top and bottom, standard cover top and bottom, with #5 ties spaced at 2 ft –0 in. On the perimeter there are #5 ties spaced at 8 in. with 2 #11s spaced at 10 in. on the vertical face of the foundation. 3. Soil data are in Attachment E and F of the TAD canister performance specification (DOE 2008c).	1.7 1.7.2.3.1	Cask vendor to provide canister leakage rate calculation	Table 1.7-15 identifies the drop and tipover event sequences for the TAD canister in an aging overpack. Based on event sequence analysis of the drop and tipover events, these events are categorized as beyond Category 2 and no further consequence assessments are necessary.

NOTE: [a]Use the values for upper bound 90% confidence interval.
[b]The maximum k_{eff} for a configuration is the value at the upper limit of a two-sided 95% confidence interval.
[c]The critical limit is the value of k_{eff} at which a configuration is considered potentially critical including biases and uncertainties.

Source: DOE 2008c.

Table 1.5.1-11. Thermal Power of the Average and Bounding Pressurized Water Reactor and Boiling Water Reactor Fuel Assemblies

Source of Thermal Power	Thermal Power (W per assembly)			
	PWR Average (25 years)	PWR Maximum (5 years)	BWR Average (25 years)	BWR Maximum (5 years)
Activation products	5	93	1	14
Fission products	389	1,610	133	540
Actinides and daughters	207	772	53	255
TOTAL	601	2,475	186	779

NOTE: Times given are aging times after discharge from the reactor.

Table 1.5.1-12. Nuclide Radioactivity of the Average and Bounding Pressurized Water Reactor and Boiling Water Reactor Fuel Assemblies

Nuclide	Radioactivity (curies per assembly)			
	PWR		BWR	
	PWR Average 25 years	PWR Bounding 5 years	BWR Average 25 years	BWR Bounding 5 years
	48 GWd/MTU	80 GWd/MTU	40 GWd/MTU	75 GWd/MTU
^{227}Ac	1.61×10^{-5}	—	—	—
^{241}Am	1.98×10^{3}	8.79×10^{2}	5.58×10^{2}	2.66×10^{2}
^{242m}Am	6.39×10^{0}	1.02×10^{1}	2.17×10^{0}	3.40×10^{0}
^{242}Am	6.36×10^{0}	1.01×10^{1}	2.16×10^{0}	3.39×10^{0}
^{243}Am	2.20×10^{1}	6.00×10^{1}	5.35×10^{0}	1.93×10^{1}
^{137m}Ba	3.88×10^{4}	9.89×10^{4}	1.31×10^{4}	3.65×10^{4}
^{14}C	3.32×10^{-1}	5.35×10^{-1}	1.75×10^{-1}	3.16×10^{-1}
^{113m}Cd	7.66×10^{0}	4.31×10^{1}	2.26×10^{0}	1.39×10^{1}
^{144}Ce	1.18×10^{-4}	5.80×10^{3}	2.89×10^{-5}	1.38×10^{3}
^{249}Cf	7.75×10^{-5}	3.90×10^{-3}	—	4.73×10^{-4}
^{251}Cf	—	1.96×10^{-4}	—	2.29×10^{-5}
^{36}Cl	6.80×10^{-3}	1.05×10^{-2}	2.93×10^{-3}	4.99×10^{-3}
^{242}Cm	5.27×10^{0}	3.56×10^{1}	1.79×10^{0}	1.13×10^{1}
^{243}Cm	1.03×10^{1}	4.19×10^{1}	2.48×10^{0}	1.12×10^{1}
^{244}Cm	1.36×10^{3}	1.40×10^{4}	2.56×10^{2}	3.95×10^{3}
^{245}Cm	3.07×10^{-1}	1.79×10^{0}	4.04×10^{-2}	3.54×10^{-1}
^{246}Cm	1.04×10^{-1}	1.21×10^{0}	1.45×10^{-2}	2.97×10^{-1}
^{248}Cm	—	1.40×10^{-4}	—	2.38×10^{-5}
^{60}Co	3.14×10^{2}	6.00×10^{3}	4.71×10^{1}	8.97×10^{2}
^{134}Cs	2.52×10^{1}	4.05×10^{4}	6.32×10^{0}	1.16×10^{4}
^{135}Cs	3.50×10^{-1}	6.34×10^{-1}	1.39×10^{-1}	2.82×10^{-1}
^{137}Cs	4.11×10^{4}	1.05×10^{5}	1.39×10^{4}	3.87×10^{4}
^{152}Eu	1.31×10^{0}	4.54×10^{0}	5.29×10^{-1}	1.69×10^{0}
^{154}Eu	6.71×10^{2}	6.15×10^{3}	1.80×10^{2}	1.83×10^{3}
^{155}Eu	5.16×10^{1}	1.80×10^{3}	1.64×10^{1}	6.37×10^{2}

Table 1.5.1-12. Nuclide Radioactivity of the Average and Bounding Pressurized Water Reactor and Boiling Water Reactor Fuel Assemblies (Continued)

Nuclide	Radioactivity (curies per assembly)			
	PWR		BWR	
	PWR Average 25 years	PWR Bounding 5 years	BWR Average 25 years	BWR Bounding 5 years
	48 GWd/MTU	80 GWd/MTU	40 GWd/MTU	75 GWd/MTU
^{55}Fe	6.94×10^{0}	1.28×10^{3}	3.27×10^{0}	5.85×10^{2}
^{3}H	1.14×10^{2}	4.95×10^{2}	3.95×10^{1}	1.77×10^{2}
^{129}I	2.20×10^{-2}	3.60×10^{-2}	7.43×10^{-3}	1.36×10^{-2}
^{85}Kr	1.13×10^{3}	5.79×10^{3}	3.81×10^{2}	2.03×10^{3}
^{93m}Nb	1.29×10^{1}	4.87×10^{1}	4.74×10^{-1}	1.22×10^{0}
^{94}Nb	8.40×10^{-1}	1.37×10^{0}	1.87×10^{-2}	3.39×10^{-2}
^{59}Ni	2.09×10^{0}	2.96×10^{0}	5.03×10^{-1}	7.80×10^{-1}
^{63}Ni	2.52×10^{2}	4.52×10^{2}	5.87×10^{1}	1.16×10^{2}
^{236}Np	—	3.45×10^{-5}	—	—
^{237}Np	2.47×10^{-1}	4.01×10^{-1}	6.89×10^{-2}	1.33×10^{-1}
^{238}Np	2.87×10^{-2}	4.58×10^{-2}	9.75×10^{-3}	1.53×10^{-2}
^{239}Np	2.20×10^{1}	6.00×10^{1}	5.35×10^{0}	1.93×10^{1}
^{231}Pa	2.97×10^{-5}	4.18×10^{-5}	1.39×10^{-5}	2.94×10^{-5}
^{107}Pd	8.41×10^{-2}	1.60×10^{-1}	2.65×10^{-2}	5.70×10^{-2}
^{147}Pm	1.19×10^{2}	2.29×10^{4}	3.98×10^{1}	7.46×10^{3}
^{144}Pr	1.18×10^{-4}	5.80×10^{3}	2.89×10^{-5}	1.38×10^{3}
^{236}Pu	1.01×10^{-3}	3.46×10^{-1}	1.67×10^{-4}	6.96×10^{-2}
^{237}Pu	—	1.03×10^{-11}	—	1.64×10^{-12}
^{238}Pu	2.29×10^{3}	6.80×10^{3}	5.85×10^{2}	2.11×10^{3}
^{239}Pu	1.77×10^{2}	1.83×10^{2}	5.35×10^{1}	5.36×10^{1}
^{240}Pu	3.18×10^{2}	4.01×10^{2}	1.14×10^{2}	1.48×10^{2}
^{241}Pu	2.47×10^{4}	8.00×10^{4}	6.78×10^{3}	2.25×10^{4}
^{242}Pu	1.64×10^{0}	3.34×10^{0}	5.09×10^{-1}	1.26×10^{0}
^{106}Rh	1.23×10^{-2}	1.33×10^{4}	3.00×10^{-3}	3.29×10^{3}
^{106}Ru	1.23×10^{-2}	1.33×10^{4}	3.00×10^{-3}	3.29×10^{3}

Table 1.5.1-12. Nuclide Radioactivity of the Average and Bounding Pressurized Water Reactor and Boiling Water Reactor Fuel Assemblies (Continued)

Nuclide	Radioactivity (curies per assembly)			
	PWR		BWR	
	PWR Average 25 years	PWR Bounding 5 years	BWR Average 25 years	BWR Bounding 5 years
	48 GWd/MTU	80 GWd/MTU	40 GWd/MTU	75 GWd/MTU
^{125}Sb	9.71×10^{0}	2.14×10^{3}	2.89×10^{0}	6.21×10^{2}
^{126}Sb	5.39×10^{-2}	9.57×10^{-2}	1.78×10^{-2}	3.53×10^{-2}
^{126m}Sb	3.85×10^{-1}	6.83×10^{-1}	1.27×10^{-1}	2.52×10^{-1}
^{79}Se	4.57×10^{-2}	7.35×10^{-2}	1.59×10^{-2}	2.89×10^{-2}
^{151}Sm	2.11×10^{2}	3.19×10^{2}	5.39×10^{1}	8.22×10^{1}
^{121m}Sn	1.59×10^{0}	3.58×10^{0}	5.99×10^{-1}	1.48×10^{0}
^{126}Sn	3.85×10^{-1}	6.83×10^{-1}	1.27×10^{-1}	2.52×10^{-1}
^{90}Sr	2.72×10^{4}	6.52×10^{4}	9.54×10^{3}	2.52×10^{4}
^{99}Tc	8.99×10^{0}	1.34×10^{1}	3.20×10^{0}	5.35×10^{0}
^{125m}Te	2.38×10^{0}	5.21×10^{2}	7.06×10^{-1}	1.52×10^{2}
^{230}Th	1.48×10^{-4}	3.33×10^{-5}	6.09×10^{-5}	2.05×10^{-5}
^{208}Tl	7.56×10^{-3}	1.64×10^{-2}	1.72×10^{-3}	6.10×10^{-3}
^{232}U	2.05×10^{-2}	5.97×10^{-2}	4.64×10^{-3}	2.00×10^{-2}
^{233}U	4.07×10^{-5}	2.42×10^{-5}	1.14×10^{-5}	—
^{234}U	6.77×10^{-1}	5.21×10^{-1}	2.49×10^{-1}	2.26×10^{-1}
^{235}U	7.36×10^{-3}	3.28×10^{-3}	2.62×10^{-3}	9.40×10^{-4}
^{236}U	1.72×10^{-1}	2.23×10^{-1}	6.26×10^{-2}	9.55×10^{-2}
^{237}U	5.90×10^{-1}	1.91×10^{0}	1.62×10^{-1}	5.40×10^{-1}
^{238}U	1.48×10^{-1}	1.42×10^{-1}	6.32×10^{-2}	6.07×10^{-2}
^{90}Y	2.72×10^{4}	6.53×10^{4}	9.54×10^{3}	2.52×10^{4}
^{93}Zr	8.94×10^{-1}	1.41×10^{0}	3.39×10^{-1}	6.03×10^{-1}
Sum	$\mathbf{1.68 \times 10^{5}}$	$\mathbf{5.64 \times 10^{5}}$	$\mathbf{5.53 \times 10^{4}}$	$\mathbf{1.90 \times 10^{5}}$

NOTE: The radionuclide activities not provided in table were eliminated by the ORIGEN-S code; therefore, not reported in the output files.

Table 1.5.1-13. Commercial SNF Analysis Basis

Fuel Form	Preclosure Releases	Criticality	TSPA	Postclosure Criticality
BWR commercial SNF, PWR commercial SNF	Analyses of the releases for PWR/BWR commercial SNF (either uncanistered or in TAD canister or DPC handling operations) for bounding Category 2 event sequences are provided in Section 1.8.3.2.2.	Criticality of canistered or uncanistered commercial SNF is a beyond Category 2 event sequence based on the results of the analyses described in Sections 1.14, 1.6, and 1.7.	The TSPA model uses commercial SNF with an average burnup of 38 GWd/MTHM, based on initial shipment of 10-year-old fuel first, as discussed in Section 2.3.7.4.2.1.	The postclosure criticality evaluation of BWR/PWR commercial SNF in TAD canisters is provided in Section 2.2.1.4.1.

Table 1.5.1-14. Chemical Composition (wt %) of HLW Glasses

Compound or Metal	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory (Idaho Nuclear Technology and Engineering Center)
Al_2O_3	8.28	7.08	6.04	7.11
AgO	0.05	—	—	—
As_2O_5	0.03	—	—	—
$N_3H_{12}PMo_{12}O_{40}$	—	—	—	1.40
B_2O_3	6.16	6.94	12.97	10.94
BaO	0.12	0.12	0.16	—
BeO	0.01	—	—	—
Bi_2O_3	0.01	—	—	—
CaF_2	—	—	—	7.75
CaO	0.53	1.05	0.48	0.22
CdO	1.20	—	—	—
CeO_2	0.16	—	—	—
Ce_2O_3	—	—	0.31	—
Co_2O_3	0.01	—	—	—
Cr_2O_3	1.92	0.09	0.14	—
Cs_2O	—	0.07	—	0.01
CuO	0.05	0.25	—	—
Fe_2O_3	19.53	7.38	12.09	0.04
K_2O	0.45	2.14	5.03	—
La_2O_3	0.64	0.09	—	—
Li_2O	2.33	4.62	3.73	—
MgO	0.28	1.45	0.90	—
MnO	—	2.07	0.82	—
MnO_2	0.25	—	—	—
MoO_3	0.44	—	—	—
Na_2O	15.72	8.24	8.05	13.48

Table 1.5.1-14. Chemical Composition (wt %) of HLW Glasses (Continued)

Compound or Metal	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory (Idaho Nuclear Technology and Engineering Center)
Nd_2O_3	0.51	—	0.14	—
NiO	0.97	0.40	0.25	—
P_2O_5	0.11	0.05	1.21	0.05
PbO	0.05	0.01	—	—
PbS	—	0.06	—	—
PdO	0.00	—	—	—
Pr_2O_3	0.08	—	—	—
PuO_2	—	0.06	—	—
Rb_2O	0.01	—	—	—
Rh_2O_3	0.04	—	—	—
RuO_2	—	—	0.08	—
Ru_2O_3	0.26	—	—	—
Sb_2O_3	0.00	—	—	—
SeO_2	0.06	—	—	—
SiO_2	31.63	54.39	41.22	54.87
SO_3	0.49	—	—	—
SO_4	—	0.14	—	—
SrO	0.07	0.01	0.02	—
Ta_2O_5	0.00	—	—	—
TeO_2	0.02	—	—	—
ThO_2	0.05	0.55	3.58	—
TiO_2	0.02	0.55	0.80	—
Tl_2O_3	0.00	—	—	—
UO_3	—	—	0.63	—
U_3O_8	1.40	1.01	—	—
V_2O_5	0.01	—	—	—
WO_3	0.00	—	—	—

Table 1.5.1-14. Chemical Composition (wt %) of HLW Glasses (Continued)

Compound or Metal	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory (Idaho Nuclear Technology and Engineering Center)
Y_2O_3	0.00	0.04	—	—
ZnO	0.03	0.02	0.02	—
ZrO_2	5.92	0.37	1.33	0.93
$(R.E.)_2O_3$[a]	—	0.63	—	—
Cd	—	—	—	2.27
Cl	0.01	—	—	—
Cr	—	—	—	0.73
F	0.08	—	—	—
Hg	—	—	—	0.01
Ni	—	—	—	0.08
Pb	—	—	—	0.10
Pd	—	0.03	—	—
Rh	—	0.02	—	—
Ru	—	0.08	—	—

NOTE: [a]$(R.E.)_2O_3$ represents the total wt % of the oxides of Pr, Ce, Nd, Sm, and Eu estimated from isotopes.

Table 1.5.1-15. Approximate Mass of HLW per Canister

Originating Site	Estimated Mass (kg) per Canister
Hanford	3,360
Savannah River Site	1,795
Idaho National Laboratory	1,560
West Valley Demonstration Project	2,000

Table 1.5.1-16. Nominal Values of Physical Parameters of the Standard Canisters for HLW

	Length (cm)	Nominal Outer Diameter (cm)	Thickness (cm)	Material	Empty Canister Weight (kg)	Available Volume (m^3)	Nominal Percent Fill Height (%)	Glass Volume (m^3)
Hanford	450	61	0.95	Stainless Steel 304L	715	1.19	87 95 100	1.04 1.14 1.19
Savannah River Site, Idaho National Laboratory	300	61	0.95	Stainless Steel Type 304L	500	0.736	90	0.66
West Valley Demonstration Project	300	61	0.34	Stainless Steel Type 304L	181.4	0.83	91	0.76

Table 1.5.1-17. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the HLW Canister

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	**Controlling Parameters and Values**	
DOE and Commercial Waste Package System (DS)	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.	The HLW canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the HLW canister capacity curve.
				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.	The HLW canister is required to be designed such that the maximum effective plastic strain from low speed impact or collisions meets the required reliability when evaluated against the HLW canister capacity curve.
				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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The HLW canister is required to be designed such that the fire-induced hazard curve failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires The HLW canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-17. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the HLW Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	**Controlling Parameters and Values**	
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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The HLW canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The HLW canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the HLW canister capacity curve.
				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.	The HLW canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the HLW canister capacity curve.
				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.	The HLW canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the HLW canister capacity curve.
				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.	The HLW canister is required to be designed such that the maximum effective plastic strain from low speed impact or collisions meets the required reliability when evaluated against the HLW canister capacity curve.

Table 1.5.1-17. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the HLW Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
DOE and Commercial Waste Package System (DS) (Continued)	HLW (Continued)	HLW Canister (Continued)	Provide containment (Continued)	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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The HLW canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
	High-Level Waste/DOE SNF Codisposal	HLW Canister	Provide containment	DS.CR.16. The mean conditional probability of breach of an HLW canister within a cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The HLW canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The HLW canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The HLW canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the HLW canister capacity curve.

Table 1.5.1-17. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the HLW Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The HLW canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the HLW canister capacity curve.
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The HLW canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

NOTE: For casks, canisters, and associated handling equipment that were previously designed, the component design will be evaluated to confirm that the controlling parameters and values are met.
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.
Facility Codes: CR: Canister Receipt and Closure Facility; IHF: Initial Handling Facility; SB: Balance of Plant.
System Codes: DS: DOE and Commercial Waste Package.

Table 1.5.1-18. Design Codes and Standards Related to HLW Canisters

HLW Canister Design	Hanford[a]	Idaho National Laboratory[b]	Savannah River Site	West Valley
Material of construction	Austenitic stainless steel per nationally recognized code	Austenitic stainless steel per nationally recognized code	Austenitic stainless steel 304L	Austenitic stainless steel 304L
Canister welding	*2001 ASME Boiler and Pressure Vessel Code*, Section IX (ASME 2001) (or equal)	*2001 ASME Boiler and Pressure Vessel Code*, Section IX (ASME 2001) (or equal)	*2001 ASME Boiler and Pressure Vessel Code*, Section IX (ASME 2001)	*2001 ASME Boiler and Pressure Vessel Code*, Section IX (ASME 2001)
Canister weld nondestructive examination	Radiographic examination of all full penetration butt welds per *2001 ASME Boiler and Pressure Vessel Code*, Section V (ASME 2001) (or equal)	Radiographic examination of all full penetration butt welds per *2001 ASME Boiler and Pressure Vessel Code*, Section V (ASME 2001) (or equal)	Radiographic examination of all full penetration butt welds per *2001 ASME Boiler and Pressure Vessel Code*, Section V (ASME 2001) (or equal)	Dye penetrant examination of all fabrication welds per *2001 ASME Boiler and Pressure Vessel Code*, Section V (ASME 2001) (or equal)
Final testing prior to HLW producer acceptance	Pass both vendor pressure and helium leak tests	Pass both vendor pressure and helium leak tests	Pass both vendor pressure and helium leak tests	Pass both vendor pressure and helium leak tests

NOTE: [a]Because the HLW canisters for the Hanford site have not been fabricated, the HLW canister designs are assumed to be comparable to the Savannah River Site and West Valley canisters.
[b]Because the HLW canisters for the Idaho National Laboratory site have not been fabricated, the HLW canister designs are assumed to be comparable to the Savannah River Site and West Valley canisters.

Source: Brosnee 1999, Part 4; Cadoff 1995, Section 2; ASME 2001.

Table 1.5.1-19. HLW Waste Thermal Information at 2017

Thermal Output	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory[a]	Total
Thermal Output (W/canister)	7.20×10^2	4.46×10^2	2.25×10^2	1.40×10^2	—
Total Thermal Output (W)	3.91×10^5	3.01×10^6	4.42×10^4	7.94×10^4	3.52×10^6

NOTE: [a]Heat output for Idaho National Laboratory HLW canister provided only for year 2035.

Table 1.5.1-20. Total Radionuclide Inventory for each HLW Glass Type at 2017

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{225}Ac	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{227}Ac	1.33×10^{2}	1.41×10^{-4}	1.22×10^{1}	1.66×10^{-14}	1.45×10^{2}
^{228}Ac	1.36×10^{1}	6.69	1.64	4.64×10^{-12}	2.19×10^{1}
^{241}Am	1.42×10^{5}	2.22×10^{6}	5.30×10^{4}	1.27×10^{4}	2.43×10^{6}
^{242}Am	—	4.97×10^{2}	2.59×10^{2}	1.01×10^{-2}	7.56×10^{2}
^{242m}Am	—	5.00×10^{2}	2.61×10^{2}	—	7.61×10^{2}
^{243}Am	1.50×10^{1}	9.24×10^{3}	3.46×10^{2}	1.39×10^{-2}	9.60×10^{3}
^{217}At	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{137m}Ba	2.99×10^{7}	2.81×10^{8}	3.66×10^{6}	5.62×10^{6}	3.20×10^{8}
^{210}Bi	1.75×10^{-2}	4.06×10^{-5}	1.42×10^{-4}	1.59×10^{-10}	1.77×10^{-2}
^{211}Bi	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{212}Bi	4.95×10^{1}	7.26	7.53	6.93×10^{-9}	6.43×10^{1}
^{213}Bi	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{214}Bi	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{14}C	—	—	1.37×10^{2}	2.78×10^{-2}	1.37×10^{2}
^{113}Cd	6.37×10^{-15}	1.77×10^{-7}	2.19×10^{-15}	—	1.77×10^{-7}
^{113m}Cd	7.30×10^{3}	—	5.70×10^{2}	—	7.87×10^{3}
^{144}Ce	—	3.20	2.45×10^{-11}	—	3.20
^{249}Cf	—	1.55×10^{2}	—	—	1.55×10^{2}
^{251}Cf	—	1.24×10^{2}	—	—	1.24×10^{2}
^{242}Cm	—	4.12×10^{2}	2.15×10^{2}	1.24×10^{-2}	6.27×10^{2}
^{243}Cm	9.28	2.24×10^{3}	6.96×10^{1}	4.70×10^{-4}	2.32×10^{3}
^{244}Cm	1.60×10^{2}	2.00×10^{6}	2.72×10^{3}	1.03×10^{-2}	2.00×10^{6}
^{245}Cm	—	1.63×10^{2}	8.79×10^{-1}	3.69×10^{-6}	1.64×10^{2}
^{246}Cm	—	1.96×10^{2}	1.01×10^{-1}	8.66×10^{-8}	1.96×10^{2}
^{247}Cm	—	1.48×10^{2}	—	3.09×10^{-14}	1.48×10^{2}
^{248}Cm	—	—	—	9.35×10^{-15}	9.35×10^{-15}
^{60}Co	9.88×10^{2}	3.33×10^{5}	2.20×10^{1}	3.21×10^{1}	3.34×10^{5}

Table 1.5.1-20. Total Radionuclide Inventory for each HLW Glass Type at 2017 (Continued)

Nuclide	Radioactivity (Ci)				
	HS	**SRS**	**WVDP**	**INL[a]**	**Total[a]**
^{134}Cs	8.47×10^{1}	4.39×10^{4}	5.90×10^{-1}	3.28×10^{-2}	4.40×10^{4}
^{135}Cs		1.46×10^{3}	1.61×10^{2}	1.63×10^{2}	1.78×10^{3}
^{137}Cs	3.16×10^{7}	2.98×10^{8}	3.87×10^{6}	5.95×10^{6}	3.39×10^{8}
^{152}Eu	7.16×10^{2}	—	9.02×10^{1}	—	8.06×10^{2}
^{154}Eu	3.80×10^{4}	1.25×10^{6}	1.09×10^{4}	5.98×10^{3}	1.30×10^{6}
^{155}Eu	8.58×10^{2}	1.03×10^{3}	4.59×10^{2}	7.55	2.35×10^{3}
^{55}Fe	—	—	6.86×10^{-1}	—	6.86×10^{-1}
^{221}Fr	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{223}Fr	1.83	1.95×10^{-6}	1.69×10^{-1}	2.29×10^{-16}	2.00
^{152}Gd	1.08×10^{-11}	—	6.16×10^{-12}	—	1.70×10^{-11}
^{3}H	—	—	1.80×10^{1}	3.56×10^{3}	3.58×10^{3}
^{129}I	4.80×10^{1}	2.18	2.10×10^{-1}	5.64	5.60×10^{1}
^{40}K	—	—	—	—	—
^{138}La	—	—	—	—	—
^{93m}Nb	3.21×10^{3}	1.57×10^{3}	2.46×10^{2}	4.74×10^{2}	5.50×10^{3}
^{94}Nb	—	—	—	5.36×10^{-3}	5.36×10^{-3}
^{144}Nd	—	9.66×10^{-12}	—	—	9.66×10^{-12}
^{59}Ni	1.37×10^{3}	5.71×10^{3}	1.06×10^{2}	—	7.19×10^{3}
^{63}Ni	1.14×10^{5}	5.04×10^{5}	7.06×10^{3}	—	6.25×10^{5}
^{236}Np	—	—	9.47	—	9.47
^{237}Np	1.41×10^{2}	2.01×10^{2}	2.39×10^{1}	6.26	3.72×10^{2}
^{238}Np	—	2.25	1.17	—	3.42
^{239}Np	1.50×10^{1}	9.24×10^{3}	3.46×10^{2}	1.47×10^{-3}	9.60×10^{3}
^{231}Pa	2.72×10^{2}	9.69×10^{-4}	1.52×10^{1}	1.48×10^{-9}	2.87×10^{2}
^{233}Pa	1.41×10^{2}	2.01×10^{2}	2.39×10^{1}	6.06×10^{-2}	3.66×10^{2}
^{234}Pa	2.59×10^{-1}	4.19×10^{-1}	1.11×10^{-3}	1.46×10^{-7}	6.79×10^{-1}
^{234m}Pa	1.99×10^{2}	3.22×10^{2}	8.54×10^{-1}	3.19×10^{-4}	5.22×10^{2}
^{209}Pb	2.31	9.37×10^{-1}	2.32×10^{-1}	2.80×10^{-15}	3.48

Table 1.5.1-20. Total Radionuclide Inventory for each HLW Glass Type at 2017 (Continued)

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{210}Pb	1.75×10^{-2}	4.06×10^{-5}	1.42×10^{-4}	9.16×10^{-9}	1.77×10^{-2}
^{211}Pb	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{212}Pb	4.95×10^{1}	7.26	7.53	1.05×10^{-8}	6.43×10^{1}
^{214}Pb	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{107}Pd	—	8.84	1.10×10^{1}	—	1.98×10^{1}
^{146}Pm	—	—	3.67×10^{-1}	—	3.67×10^{-1}
^{147}Pm	—	1.03×10^{6}	7.01×10^{1}	2.67×10^{1}	1.03×10^{6}
^{210}Po	1.61×10^{-2}	3.58×10^{-5}	1.34×10^{-4}	7.59×10^{-14}	1.63×10^{-2}
^{211}Po	3.66×10^{-1}	3.88×10^{-7}	3.37×10^{-2}	4.18×10^{-21}	4.00×10^{-1}
^{212}Po	3.17×10^{1}	4.65	4.83	4.44×10^{-9}	4.12×10^{1}
^{213}Po	2.26	9.17×10^{-1}	2.27×10^{-1}	9.15×10^{-15}	3.40
^{214}Po	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.43×10^{-4}	9.12×10^{-2}
^{215}Po	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{216}Po	4.95×10^{1}	7.26	7.53	6.11×10^{-8}	6.43×10^{1}
^{218}Po	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{144}Pr	—	3.20	2.45×10^{-11}	—	3.20
^{144m}Pr	—	4.49×10^{-2}	3.43×10^{-13}	—	4.49×10^{-2}
^{236}Pu	—	—	8.43×10^{-1}	—	8.43×10^{-1}
^{238}Pu	4.31×10^{3}	6.14×10^{6}	6.85×10^{3}	8.98×10^{4}	6.24×10^{6}
^{239}Pu	6.91×10^{4}	1.18×10^{5}	1.65×10^{3}	1.81×10^{3}	1.91×10^{5}
^{240}Pu	1.23×10^{4}	5.94×10^{4}	1.23×10^{3}	1.57×10^{3}	7.45×10^{4}
^{241}Pu	5.78×10^{4}	3.50×10^{5}	2.22×10^{4}	1.93×10^{4}	4.49×10^{5}
^{242}Pu	1.00	1.44×10^{2}	1.65	3.42	1.50×10^{2}
^{243}Pu	—	1.48×10^{2}	—	2.22×10^{-14}	1.48×10^{2}
^{223}Ra	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{224}Ra	4.95×10^{1}	7.26	7.53	6.11×10^{-8}	6.43×10^{1}
^{225}Ra	2.31	9.37×10^{-1}	2.32×10^{-1}	1.07×10^{-12}	3.48
^{226}Ra	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	9.69×10^{-3}	1.00×10^{-1}

Table 1.5.1-20. Total Radionuclide Inventory for each HLW Glass Type at 2017 (Continued)

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{228}Ra	1.36×10^{1}	6.69	1.64	1.24×10^{-11}	2.19×10^{1}
^{102}Rh	—	—	—	1.99×10^{-5}	1.99×10^{-5}
^{106}Rh	1.70×10^{-2}	2.98×10^{1}	1.41×10^{-7}	—	2.98×10^{1}
^{219}Rn	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{220}Rn	4.95×10^{1}	7.26	7.53	6.11×10^{-8}	6.43×10^{1}
^{222}Rn	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{106}Ru	1.70×10^{-2}	2.98×10^{1}	1.41×10^{-7}	—	2.98×10^{1}
^{125}Sb	4.18×10^{2}	6.22×10^{4}	7.83	1.03	6.26×10^{4}
^{126}Sb	8.11×10^{1}	7.42×10^{2}	1.46×10^{1}	2.62×10^{-1}	8.38×10^{2}
^{126m}Sb	5.79×10^{2}	5.30×10^{3}	1.04×10^{2}	8.91×10^{1}	6.07×10^{3}
^{79}Se	1.22×10^{2}	3.60×10^{3}	6.02×10^{1}	—	3.78×10^{3}
^{146}Sm	—	—	8.63×10^{-8}	—	8.63×10^{-8}
^{147}Sm	—	3.46×10^{-4}	4.44×10^{-7}	1.81×10^{-13}	3.46×10^{-4}
^{151}Sm	3.10×10^{6}	1.01×10^{6}	6.85×10^{4}	—	4.18×10^{6}
^{121}Sn	—	8.95×10^{3}	9.59	—	8.96×10^{3}
^{121m}Sn	—	1.15×10^{4}	1.24×10^{1}	—	1.15×10^{4}
^{126}Sn	5.79×10^{2}	5.30×10^{3}	1.04×10^{2}	8.91×10^{1}	6.07×10^{3}
^{90}Sr	3.43×10^{7}	1.80×10^{8}	3.46×10^{6}	7.04×10^{6}	2.25×10^{8}
^{99}Tc	2.97×10^{4}	6.19×10^{4}	1.70×10^{3}	3.41×10^{3}	9.67×10^{4}
^{125m}Te	1.02×10^{2}	1.52×10^{4}	1.91	1.07×10^{-3}	1.53×10^{4}
^{227}Th	1.31×10^{2}	1.39×10^{-4}	1.21×10^{1}	5.81×10^{-17}	1.43×10^{2}
^{228}Th	4.93×10^{1}	7.23	7.51	1.74×10^{-6}	6.40×10^{1}
^{229}Th	2.31	9.37×10^{-1}	2.32×10^{-1}	1.21×10^{-10}	3.48
^{230}Th	1.42×10^{-2}	9.12×10^{-2}	5.96×10^{-2}	9.50×10^{-7}	1.65×10^{-1}
^{231}Th	9.00	4.49	1.01×10^{-1}	1.29×10^{-1}	1.37×10^{1}
^{232}Th	8.00	9.49	1.64	9.89×10^{-8}	1.91×10^{1}
^{234}Th	1.99×10^{2}	3.22×10^{2}	8.54×10^{-1}	3.19×10^{-4}	5.22×10^{2}
^{206}Tl	2.32×10^{-8}	5.36×10^{-11}	1.88×10^{-10}	2.10×10^{-16}	2.34×10^{-8}

Table 1.5.1-20. Total Radionuclide Inventory for each HLW Glass Type at 2017 (Continued)

Nuclide	Radioactivity (Ci)				
	HS	**SRS**	**WVDP**	**INL[a]**	**Total[a]**
^{207}Tl	1.33×10^{2}	1.41×10^{-4}	1.22×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{208}Tl	1.78×10^{1}	2.61	2.71	2.49×10^{-9}	2.31×10^{1}
^{209}Tl	4.86×10^{-2}	1.97×10^{-2}	4.88×10^{-3}	1.96×10^{-16}	7.32×10^{-2}
^{232}U	3.73×10^{1}	1.82	5.74	4.63×10^{-3}	4.49×10^{1}
^{233}U	5.10×10^{2}	3.79×10^{2}	9.53	1.33×10^{-3}	8.99×10^{2}
^{234}U	2.20×10^{2}	4.89×10^{2}	5.05	9.95×10^{1}	8.14×10^{2}
^{235}U	9.00	4.49	1.01×10^{-1}	5.90×10^{-1}	1.42×10^{1}
^{236}U	6.00	2.48×10^{1}	2.97×10^{-1}	1.54	3.26×10^{1}
^{237}U	1.38	8.38	5.32×10^{-1}	1.76×10^{-2}	1.03×10^{1}
^{238}U	1.99×10^{2}	3.22×10^{2}	8.54×10^{-1}	2.94×10^{-2}	5.22×10^{2}
^{90}Y	3.43×10^{7}	1.80×10^{8}	3.46×10^{6}	7.04×10^{6}	2.25×10^{8}
^{93}Zr	4.81×10^{3}	2.61×10^{3}	2.72×10^{2}	—	7.69×10^{3}
Total	**1.34×10^{8}**	**9.54×10^{8}**	**1.46×10^{7}**	**2.58×10^{7}**	**1.13×10^{9}**

NOTE: [a]Radionuclide inventory for Idaho National Laboratory HLW canister is provided for year 2035.
HS = Hanford Site; INL = Idaho National Laboratory; SRS = Savannah River Site; WVDP = West Valley Demonstration Project.

Table 1.5.1-21. Maximum Radionuclide Inventory per HLW Glass Canister at 2017

Nuclide	Radioactivity (Ci/Canister)			
	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory[a]
^{225}Ac	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	4.26×10^{-17}
^{227}Ac	1.72×10^{-4}	2.09×10^{-8}	1.16×10^{-1}	1.85×10^{-17}
^{228}Ac	9.38×10^{-5}	9.87×10^{-4}	1.47×10^{-2}	2.33×10^{-14}
^{241}Am	4.61×10^{2}	3.38×10^{2}	4.97×10^{2}	1.41×10^{1}
^{242}Am	—	7.36×10^{-2}	2.46	6.32×10^{-5}
^{242m}Am	—	7.39×10^{-2}	2.47	—
^{243}Am	9.98×10^{-2}	1.37	3.27	1.05×10^{-4}
^{217}At	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	4.26×10^{-17}
^{137m}Ba	5.62×10^{4}	4.15×10^{4}	1.84×10^{4}	1.14×10^{4}
^{210}Bi	2.51×10^{-6}	5.99×10^{-9}	5.17×10^{-7}	1.17×10^{-12}
^{211}Bi	1.72×10^{-4}	2.09×10^{-8}	1.16×10^{-1}	1.69×10^{-21}
^{212}Bi	4.85×10^{-4}	1.07×10^{-3}	4.74×10^{-2}	9.20×10^{-12}
^{213}Bi	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	4.26×10^{-17}
^{214}Bi	1.29×10^{-5}	4.60×10^{-8}	1.95×10^{-6}	4.76×10^{-6}
^{14}C	1.06×10^{-7}	—	1.30	8.26×10^{-5}
^{113}Cd	1.47×10^{-9}	2.62×10^{-11}	—	1.48×10^{-9}
^{113m}Cd	1.91×10^{1}	—	2.07	—
^{142}Ce	1.17×10^{-5}	—	1.43×10^{-5}	—
^{144}Ce	—	4.74×10^{-4}	8.90×10^{-14}	—
^{249}Cf	—	2.29×10^{-2}	—	—
^{251}Cf	—	1.84×10^{-2}	—	—
^{242}Cm	6.54×10^{-6}	6.10×10^{-2}	2.04	7.71×10^{-5}
^{243}Cm	3.73×10^{-2}	3.31×10^{-1}	2.53×10^{-1}	3.36×10^{-6}
^{244}Cm	3.27×10^{-1}	2.97×10^{2}	2.57×10^{1}	7.71×10^{-5}
^{245}Cm	—	2.42×10^{-2}	3.19×10^{-3}	2.81×10^{-8}
^{246}Cm	—	2.90×10^{-2}	3.66×10^{-4}	6.61×10^{-10}

Table 1.5.1-21. Maximum Radionuclide Inventory per HLW Glass Canister at 2017 (Continued)

Nuclide	Radioactivity (Ci/Canister)			
	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory[a]
^{247}Cm	—	2.20×10^{-2}	—	2.37×10^{-16}
^{248}Cm	—	—	—	7.16×10^{-17}
^{60}Co	4.14×10^{-1}	4.91×10^{1}	6.63×10^{-1}	3.57×10^{-2}
^{134}Cs	2.12×10^{1}	6.48	4.09×10^{-3}	3.64×10^{-5}
^{135}Cs	—	2.16×10^{-1}	1.09	2.53×10^{-1}
^{137}Cs	5.95×10^{4}	4.39×10^{4}	1.95×10^{4}	1.21×10^{4}
^{152}Eu	3.45	—	3.28×10^{-1}	—
^{154}Eu	4.50	1.85×10^{2}	4.72×10^{1}	6.65
^{155}Eu	1.16×10^{2}	1.52×10^{-1}	1.67	3.75×10^{-2}
^{55}Fe	—	—	2.49×10^{-3}	—
^{221}Fr	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	4.26×10^{-17}
^{223}Fr	2.37×10^{-6}	2.88×10^{-10}	1.60×10^{-3}	2.55×10^{-19}
^{152}Gd	5.22×10^{-14}	—	2.24×10^{-14}	—
^{3}H	—	—	6.54×10^{-2}	4.30
^{129}I	—	3.22×10^{-4}	7.64×10^{-4}	1.65×10^{-2}
^{40}K	1.07×10^{-5}	2.77×10^{-5}	7.17×10^{-5}	—
^{138}La	3.98×10^{-7}	3.04×10^{-8}	—	—
^{93m}Nb	3.30	2.33×10^{-1}	2.33	1.43
^{94}Nb	—	—	—	1.60×10^{-5}
^{144}Nd	4.12×10^{-9}	1.43×10^{-15}	6.77×10^{-10}	—
^{59}Ni	4.96×10^{-1}	8.44×10^{-1}	1.00	—
^{63}Ni	4.89×10^{1}	7.47×10^{1}	6.69×10^{1}	—
^{236}Np	—	—	8.97×10^{-2}	—
^{237}Np	2.51×10^{-1}	2.99×10^{-2}	1.53×10^{-1}	2.75×10^{-2}
^{238}Np	—	3.33×10^{-4}	1.11×10^{-2}	—
^{239}Np	9.98×10^{-2}	1.37	3.27	1.11×10^{-5}
^{231}Pa	4.24×10^{-4}	1.43×10^{-7}	1.44×10^{-1}	1.65×10^{-12}

Table 1.5.1-21. Maximum Radionuclide Inventory per HLW Glass Canister at 2017 (Continued)

Nuclide	Radioactivity (Ci/Canister)			
	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory[a]
^{233}Pa	2.51×10^{-1}	2.99×10^{-2}	1.53×10^{-1}	2.66×10^{-4}
^{234}Pa	1.31×10^{-5}	6.16×10^{-5}	4.33×10^{-6}	1.62×10^{-10}
^{234m}Pa	1.01×10^{-2}	4.74×10^{-2}	3.33×10^{-3}	3.55×10^{-7}
^{209}Pb	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	1.27×10^{-17}
^{210}Pb	2.51×10^{-6}	5.99×10^{-9}	5.16×10^{-7}	6.77×10^{-11}
^{211}Pb	1.72×10^{-4}	2.09×10^{-8}	1.16×10^{-1}	1.69×10^{-21}
^{212}Pb	4.85×10^{-4}	1.07×10^{-3}	4.74×10^{-2}	1.40×10^{-11}
^{214}Pb	1.29×10^{-5}	4.60×10^{-8}	1.95×10^{-6}	4.76×10^{-6}
^{107}Pd	—	1.31×10^{-3}	1.04×10^{-1}	—
^{146}Pm	—	—	1.34×10^{-3}	—
^{147}Pm	—	1.53×10^{2}	2.55×10^{-1}	2.97×10^{-2}
^{210}Po	2.31×10^{-6}	5.29×10^{-9}	4.87×10^{-7}	5.61×10^{-16}
^{211}Po	4.74×10^{-7}	5.74×10^{-11}	3.20×10^{-4}	4.65×10^{-24}
^{212}Po	3.11×10^{-4}	6.87×10^{-4}	3.04×10^{-2}	5.90×10^{-12}
^{213}Po	1.37×10^{-6}	1.36×10^{-4}	9.27×10^{-4}	4.17×10^{-17}
^{214}Po	1.29×10^{-5}	4.60×10^{-8}	1.95×10^{-6}	4.75×10^{-6}
^{215}Po	1.72×10^{-4}	2.09×10^{-8}	1.16×10^{-1}	1.69×10^{-21}
^{216}Po	4.85×10^{-4}	1.07×10^{-3}	4.74×10^{-2}	8.11×10^{-11}
^{218}Po	1.29×10^{-5}	4.60×10^{-8}	1.95×10^{-6}	4.76×10^{-6}
^{144}Pr	—	4.74×10^{-4}	8.90×10^{-14}	—
^{144m}Pr	—	6.63×10^{-6}	1.25×10^{-15}	—
^{236}Pu	—	—	9.98×10^{-3}	—
^{238}Pu	2.17	9.10×10^{2}	3.36×10^{1}	9.99×10^{1}
^{239}Pu	2.13×10^{1}	1.74×10^{1}	8.75	2.01
^{240}Pu	6.42	8.78	6.35	1.75
^{241}Pu	8.70×10^{1}	5.17×10^{2}	1.13×10^{2}	2.15×10^{1}
^{242}Pu	9.91×10^{-4}	2.14×10^{-2}	8.17×10^{-3}	3.80×10^{-3}

Table 1.5.1-21. Maximum Radionuclide Inventory per HLW Glass Canister at 2017 (Continued)

Nuclide	Radioactivity (Ci/Canister)			
	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory[a]
^{243}Pu	—	2.20×10^{-2}	—	1.71×10^{-16}
^{223}Ra	1.72×10^{-4}	2.09×10^{-8}	1.16×10^{-1}	1.69×10^{-21}
^{224}Ra	4.85×10^{-4}	1.07×10^{-3}	4.74×10^{-2}	8.11×10^{-11}
^{225}Ra	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	4.89×10^{-15}
^{226}Ra	1.29×10^{-5}	4.60×10^{-8}	1.95×10^{-6}	7.16×10^{-5}
^{228}Ra	9.38×10^{-5}	9.87×10^{-4}	1.47×10^{-2}	6.21×10^{-14}
^{87}Rb	7.42×10^{-6}	—	—	—
^{102}Rh	—	—	—	2.21×10^{-8}
^{106}Rh	1.37×10^{-4}	4.40×10^{-3}	5.14×10^{-10}	—
^{219}Rn	1.72×10^{-4}	2.09×10^{-8}	1.16×10^{-1}	1.69×10^{-21}
^{220}Rn	4.85×10^{-4}	1.07×10^{-3}	4.74×10^{-2}	8.11×10^{-11}
^{222}Rn	1.29×10^{-5}	4.60×10^{-8}	1.95×10^{-6}	4.76×10^{-6}
^{106}Ru	1.37×10^{-4}	4.40×10^{-3}	5.14×10^{-10}	—
^{125}Sb	3.16	9.17	2.85×10^{-2}	1.14×10^{-3}
^{126}Sb	8.04×10^{-2}	1.10×10^{-1}	1.38×10^{-1}	7.61×10^{-4}
^{126m}Sb	5.74×10^{-1}	7.83×10^{-1}	9.85×10^{-1}	2.59×10^{-1}
^{79}Se	9.15×10^{-2}	5.34×10^{-1}	5.70×10^{-1}	—
^{146}Sm	—	—	3.14×10^{-10}	—
^{147}Sm	—	5.12×10^{-8}	1.61×10^{-9}	2.02×10^{-16}
^{151}Sm	3.43×10^{3}	1.49×10^{2}	6.49×10^{2}	—
^{121}Sn	—	1.33	3.48×10^{-2}	—
^{121m}Sn	—	1.71	4.49×10^{-2}	—
^{126}Sn	5.74×10^{-1}	7.83×10^{-1}	9.85×10^{-1}	2.59×10^{-1}
^{90}Sr	6.21×10^{4}	2.67×10^{4}	1.67×10^{4}	1.16×10^{4}
^{99}Tc	2.31×10^{1}	9.16	8.72	9.92
^{123}Te	1.07×10^{-9}	—	—	—
^{125m}Te	7.72×10^{-1}	2.24	6.95×10^{-3}	1.19×10^{-6}

Table 1.5.1-21. Maximum Radionuclide Inventory per HLW Glass Canister at 2017 (Continued)

Nuclide	Radioactivity (Ci/Canister)			
	Hanford	Savannah River Site	West Valley Demonstration Project	Idaho National Laboratory[a]
^{227}Th	1.70×10^{-4}	2.06×10^{-8}	1.15×10^{-1}	6.47×10^{-20}
^{228}Th	4.84×10^{-4}	1.07×10^{-3}	4.72×10^{-2}	2.31×10^{-9}
^{229}Th	1.40×10^{-6}	1.39×10^{-4}	9.47×10^{-4}	5.53×10^{-13}
^{230}Th	9.41×10^{-7}	1.35×10^{-5}	2.18×10^{-4}	1.06×10^{-9}
^{231}Th	5.56×10^{-4}	6.64×10^{-4}	3.72×10^{-4}	1.44×10^{-4}
^{232}Th	1.50×10^{-4}	1.40×10^{-3}	1.55×10^{-2}	4.96×10^{-10}
^{234}Th	1.01×10^{-2}	4.74×10^{-2}	3.33×10^{-3}	3.55×10^{-7}
^{206}Tl	3.32×10^{-12}	—	6.82×10^{-13}	1.55×10^{-18}
^{207}Tl	1.72×10^{-4}	2.08×10^{-8}	1.16×10^{-1}	1.69×10^{-21}
^{208}Tl	1.74×10^{-4}	3.85×10^{-4}	1.70×10^{-2}	3.31×10^{-12}
^{209}Tl	2.93×10^{-8}	2.91×10^{-6}	1.99×10^{-5}	8.94×10^{-19}
^{232}U	4.40×10^{-4}	2.69×10^{-4}	3.24×10^{-2}	6.15×10^{-6}
^{233}U	2.10×10^{-3}	5.59×10^{-2}	9.03×10^{-2}	6.06×10^{-6}
^{234}U	1.46×10^{-2}	7.23×10^{-2}	2.62×10^{-2}	1.11×10^{-1}
^{235}U	5.56×10^{-4}	6.64×10^{-4}	3.72×10^{-4}	6.57×10^{-4}
^{236}U	1.18×10^{-3}	3.67×10^{-3}	1.08×10^{-3}	1.71×10^{-3}
^{237}U	2.08×10^{-3}	1.24×10^{-2}	2.70×10^{-3}	1.96×10^{-5}
^{238}U	1.01×10^{-2}	4.74×10^{-2}	3.33×10^{-3}	3.27×10^{-5}
^{50}V	2.35×10^{-14}	—	—	—
^{90}Y	6.21×10^{4}	2.67×10^{4}	1.67×10^{4}	1.16×10^{4}
^{93}Zr	5.76	3.86×10^{-1}	2.58	—
Total	2.44×10^{5}	1.42×10^{5}	7.28×10^{4}	4.69×10^{4}

NOTE: [a]Radionuclide inventory for the Idaho National Laboratory HLW canister is provided for year 2035.

Table 1.5.1-22. HLW Analysis Basis

Fuel Form	Preclosure Releases	Criticality	TSPA	Postclosure Criticality
HLW	Analyses of the releases for HLW canisters are provided in Section 1.8.3.2.2.	HLW canister criticality controls for normal repository operations and waste emplacement are unnecessary because of the low concentrations of fissile radionuclides in each HLW canister as discussed in greater detail in Section 1.14.2.3.2.4.	The TSPA model uses an average inventory developed from the inventories from these multiple HLW sources and uses a thermal source based on the most representative HLW (Hanford and Savannah River Site) as discussed in Section 2.3.7.4.2.3.	The postclosure criticality evaluation of HLW can be found in Section 2.2.1.

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
01. U Metal, Zirc Clad, LEU	EBWR ENRICHED HEAVY [64]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required. Analyses involving determination of potential event sequences involving a breach of an MCO (for fuel groups 1, 2, and 7) with DOE SNF has not been completed.	Fuel Groups 1 and 2 are analyzed for preclosure criticality safety as part of Criticality Group 1, Section 1.14. Prior to receipt and acceptance of MCOs, criticality safety analyses of MCOs containing SNF will be performed to demonstrate compliance with the criticality safety requirements in Section 1.14.2.1.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 1 and 2 are evaluated for criticality potential as part of Criticality Group 1. Section 2.2.1.4.1.3.
	EBWR ENRICHED THIN [887]				
	EBWR ET-11 [888]				
	EBWR NORMAL HEAVY [889]				
	EBWR NORMAL THIN [890]				
	HWCTR ETWO [867]				
	HWCTR RMT and SMT [790]				
	HWCTR TWNT [791]				
	N REACTOR [991]				
02. U Metal, Non-Zirc Clad, LEU	EBR-II, TREAT, MTR EXPER. and IPNS TARGET [1088]				
	HFEF FISSION CHAMBERS [894]				
	HWCTR IMT [113]				
	MISCELLANEOUS RSWF FUEL [366]				
	SINGLE PASS REACTOR FUEL [197]				
	SINGLE PASS REACTOR FUEL [198]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
03. U-Zirc	CP-5 CONVERTER CYLINDERS [36]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 3 and 4 are analyzed for preclosure criticality safety as part of Criticality Group 3, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 3 and 4 are evaluated for criticality potential as part of Criticality Group 3. Section 2.2.1.4.1.3.
	HWCTR DRIVER [117]				
	HWCTR IS [977]				
	HWCTR SPR [783]				
	HWCTR TFEN [880]				
	SHIPPINGPORT PWR-C1-S4 [194]				
04. U-Mo	FERMI CORE I and 2 (CORE FOIL) [457]				
	FERMI CORE I and 2 (CORE SHIM) [69]				
	FERMI CORE I and 2 (DECLAD) [453]				
	FERMI CORE I and 2 (SECTIONED) [454]				
	FERMI CORE I and 2 (SODIUM WORTH) [455]				
	FERMI CORE I and 2 (STD FUEL SUBASSEMBLY) [456]				
	HWCTR 3EMT-2 [118]				
	SPEC (ORME) [208]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
05. U Oxide, Zirc Clad, Intact, HEU	BR-3 [927]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 5 and 6 are analyzed for preclosure criticality safety as part of Criticality Group 4, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 5 and 6 are evaluated for criticality potential as part of Criticality Group 4. Section 2.2.1.4.1.3.
	EBWR (SPIKES) [891]				
	SHIPPINGPORT PWR-C2-S1 [195]				
	SHIPPINGPORT PWR-C2-S2 [196]				
	TREAT DRIVER [232]				
	VBWR [855]				
06. U Oxide, Zirc Clad, Intact, MEU	BR-3 FUEL [340]				
	EBWR [65]				
	PULSTAR - BUFFALO [174]				
	SAXTON [788]				
07. U Oxide, Zirc Clad, Intact, LEU	BCD B-17 (TURKEY POINT 3) [19]		Fuel Group 7 is evaluated for criticality potential as part of Criticality Group 9, Section 1.14.		Fuel Group 7 is evaluated for criticality potential as part of Criticality Group 9. Section 2.2.1.4.1.3.
	BRP-B [23]				
	BRP-C [24]				
	BRP-D1 [25]				
	BRP-D2 [26]				
	BRP-E [27]				
	BRP-EG [28]				
	BRP-EG/F [1081]				
	BRP-F [30]				
	BRP-F-PU [1082]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
07. U Oxide, Zirc Clad, Intact, LEU (Continued)	CALVERT CLIFFS 1 [307]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required. Analyses involving determination of potential event sequences involving a drop and breach of an MCO (for fuel groups 1, 2, and 7) with DOE SNF has not been completed.	Fuel Group 7 is analyzed for preclosure criticality safety as a part of Criticality Group 9, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Group 7 is evaluated for criticality potential as part of Criticality Group 9. Section 2.2.1.4.1.3.
	CANDU [979]				
	COOPER NUCLEAR [308]				
	DRCT [701]				
	DRCT [756]				
	DRESDEN I (E00161) [928]				
	DRESDEN I (UN0064) [47]				
	EBWR [60]				
	H. B. ROBINSON (ASSEMBLY) [383]				
	HWCTR IRO [976]				
	HWCTR OT [283]				
	HWCTR SOT [120]				
	HWCTR SPRO [115]				
	HWCTR SPRO [772]				
	LOFT CENTER FUEL MODULE (A1,A2,A3,F1) [127]				
	LOFT CENTER FUEL MODULE (FP-1) [1061]				
	LOFT CORNER FUEL MODULE [128]				
	LOFT SQUARE FUEL MODULE [129]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
07. U Oxide, Zirc Clad, Intact, LEU (Continued)	N.S. SAVANNAH [854]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required. Analyses involving determination of potential event sequences involving a drop and breach of an MCO (for fuel groups 1, 2, and 7) with DOE SNF has not been completed.	Fuel Group 7 is analyzed for preclosure criticality as part of Criticality Group 9, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Group 7 is evaluated for criticality potential as part of Criticality Group 9. Section 2.2.1.4.1.3.
	OCONEE [156]				
	PEACH BOTTOM (ASSEMBLY) [385]				
	POINT BEACH [311]				
	PULSTAR-N.C. STATE UNIV. [175]				
	PULSTAR-SUNY-BUFFALO [176]				
	ROBERT E. GINNA [182]				
	SHIPPINGPORT PWR C1 BLKT (RODS) [189]				
	SHIPPINGPORT PWR C1 BLKT [191]				
	SHIPPINGPORT PWR C2 BLKT [192]				
	SHIPPINGPORT PWR C2 BLKT [193]				
	TURKEY POINT [271]				
	VEPCO (T-11 ASSEMBLY) [993]				
	VEPCO (T-11 RODS) [1049]				
	VEPCO (T-11) [994]				
	VEPCO [286]				
	VEPCO [700]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
08. U Oxide, SST/ Hastelloy Clad, Intact, HEU	APPR (AGE-2) [6]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 8 and 9 are analyzed for preclosure criticality safety as part of Criticality Group 4, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 8 and 9 are evaluated for criticality potential as part of Criticality Group 4. Section 2.2.1.4.1.3.
	BORAX V (SUPERHEATER) [22]				
	GCRE (1B SERIES) [745]				
	GCRE (1Z SERIES) [916]				
	ML-1 (GCRE) [137]				
	PATHFINDER (SUPERHEATER) [166]				
	PATHFINDER (SUPERHEATER) [814]				
09. U Oxide, SST Clad, Intact, MEU	ACRR (PULSED CORE) [757]				
	PBF DRIVER CORE [167]				
	SAXTON [882]				
10. U Oxide, SST Clad, Intact, LEU	CONNECTICUT YANKEE (S004) [34]		Fuel Group 10 is analyzed for preclosure criticality safety as part of Criticality Group 9, Section 1.14.		Fuel Group 10 is evaluated for criticality potential as part of Criticality Group 9. Section 2.2.1.4.1.3.
	CVTR FUEL [37]				
	FFTF-TFA-ABA-1 THRU 6 [318]				
	FFTF-TFA-WBO18 and WBO42 [336]				
	HWCTR SPRO [978]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
11.U Oxide, Non-Alum Clad, Non-Intact or Declad, HEU	ANP [451]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 11 and 12 are analyzed for preclosure criticality safety as part of Criticality Group 4, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 11 and 12 are evaluated for criticality potential as part of Criticality Group 4. Section 2.2.1.4.1.3.
	BMI (CPI-24) [774]				
	BMI (CPI-38) [20]				
	EBWR (FUEL FOLLOWER) [740]				
	FRR TARGET (ARGENTINA) [297]				
	FRR TARGET (CANADA) [671]				
	FRR TARGET (INDONESIA) [672]				
	GCRE CAN (1B-8T 1 and 2) [94]				
	GCRE PELLETS (1B-7T-1) [95]				
	GETR FILTERS [98]				
	HTRE (ANP) [105]				
	ROVER (UBM) [840]				
	SM-1A [201]				
	SPSS (SPERT) [213]				
	TORY-IIA [230]				
	TORY-IIC [231]				
	VBWR (GENEVA) [285]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
12. U Oxide, Non-Alum Clad, Non-Intact or Declad, MEU	DRESII, HBR, BR-3, BRP, TMI [50]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 11 and 12 are analyzed for preclosure criticality safety as part of Criticality Group 4, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 11 and 12 are evaluated for criticality potential as part of Criticality Group 4. Section 2.2.1.4.1.3.
	LOFT CENTER FUEL MODULE FP-2 REMAINS [923]				
	LOFT FUEL RODS [924]				
	MTR CANAL SCRAP [1062]				
	PNL MIXED MATERIAL EXP.DCC-1 [430]				
	PNL MIXED MATERIAL EXP.DCC-2 [431]				
	PNL MIXED MATERIAL EXP.DCC-3 [432]				
	RESIDUE FAILED PBF RODS [381]				
	SP-100 FUEL [777]				
	SPERT-III [209]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
13. U Oxide, Non-Alum Clad, Non-Intact or Declad, LEU	ARKANSAS [7]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Group 13 is analyzed for preclosure criticality safety as part of Criticality Group 9, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Group 13 is evaluated for criticality potential as part of Criticality Group 9. Section 2.2.1.4.1.3.
	COMMERCIAL BWR and PWR SNF [1089]				
	H. B. ROBINSON RODS [864]				
	LOOSE FUEL ROD STORAGE BASKET (LFRSB) [126]				
	LWR COMMERCIAL FUEL [130]				
	LWR SCRAP [309]				
	LWR SNF SCRAP [940]				
	PEACH BOTTOM RODS [386]				
	TMI-2 [228]				
	TMI-2 CORE DEBRIS (D-153 and 388) [229]				
	TMI-2 CORE DEBRIS [914]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
14. U Oxide, Alum Clad, HEU	BSR [31]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 14 and 15 are analyzed for preclosure criticality safety as part of Criticality Group 4, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 14 and 15 are evaluated for criticality potential as part of Criticality Group 4. Section 2.2.1.4.1.3.
	HFBR [102]				
	HFBR [706]				
	HFBR [961]				
	HFIR (INNER) [103]				
	HFIR (INNER) [1083]				
	HFIR (OUTER) [1084]				
	HFIR (OUTER) [707]				
	NIST [154]				
	NIST [752]				
	OMEGA WEST (204) [406]				
	OMEGA WEST (236) [407]				
	OMEGA WEST (250) [408]				
	ORR [461]				
	ORR [753]				
	ORR [903]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
15. U Oxide, Alum Clad, MEU and LEU	ASTRA (AUSTRIA) [1058]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 14 and 15 are analyzed for preclosure criticality safety as part of Criticality Group 4, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 14 and 15 are evaluated for criticality potential as part of Criticality Group 4. Section 2.2.1.4.1.3.
	FRG-1 (GERMANY) [581]				
	FRR ASTRA (AUSTRIA) [556]				
	FRR MTR-C (PERU) [503]				
	FRR MTR-S (INDONESIA) [502]				
	FRR MTR-S (PERU) [504]				
	ORR SPECIAL [163]				
	RSG-GAS (INDONESIA) [288]				
16. U-ALx, HEU	ANLJ [5]		Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8, Section 1.14.		Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	ARMF (PLATES) [8]				
	ARMF/CFRMF MARK I [9]				
	ARMF/CFRMF MARK I LL [10]				
	ARMF/CFRMF MARK II [11]				
	ARMF/CFRMF MARK III [12]				
	ASTRA (AUSTRIA) [566]				
	ASTRA (AUSTRIA) [646]				
	ATR [15]				
	ATR [16]				
	ATR [843]				
	ATSR [17]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
16. U-ALx, HEU (Continued)	BER-II [HMI] (END BOXES) (GERMANY) [892]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	BER-II [HMI] (GERMANY) [758]				
	BNL MEDICAL RX (BMRR) [21]				
	DR-3 (DENMARK) [714]				
	ENEA SALUGGIA (ITALY) [574]				
	ESSOR (ITALY) [762]				
	FMRB (GERMANY) [577]				
	FRG-1 (GERMANY) [742]				
	FRJ (GERMANY) [1000]				
	FRJ (GERMANY) [933]				
	FRM (GERMANY) [805]				
	FRM (GERMANY) [806]				
	FRR ASTRA (AUSTRIA) [654]				
	FRR ASTRA (AUSTRIA) [738]				
	FRR FMRB (GERMANY) [1066]				
	FRR MTR (AUSTRALIA) [649]				
	FRR MTR (CANADA) [294]				
	FRR MTR (JAPAN) [565]				
	FRR MTR (JAPAN) [603]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
16. U-ALx, HEU (Continued)	FRR MTR (JAPAN) [605]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	FRR MTR (NETHERLANDS) [609]				
	FRR MTR (TAIWAN) [628]				
	FRR MTR-C (ARGENTINA) [635]				
	FRR MTR-C (CANADA) [612]				
	FRR MTR-C (GERMANY) [579]				
	FRR MTR-C (JAPAN) [600]				
	FRR MTR-C (PORTUGAL) [631]				
	FRR MTR-C (TURKEY) [643]				
	FRR MTR-C1 (SWITZERLAND) [656]				
	FRR MTR-C2 (SWITZERLAND) [657]				
	FRR MTR-O (TURKEY) [642]				
	FRR MTR-S (CANADA) [720]				
	FRR MTR-S (GERMANY) [1068]				
	FRR MTR-S (GERMANY) [582]				
	FRR MTR-S (GERMANY) [584]				
	FRR MTR-S (GERMANY) [585]				
	FRR MTR-S (GERMANY) [588]				
	FRR MTR-S (JAPAN) [602]				
	FRR MTR-S (NETHERLANDS) [607]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
16. U-ALx, HEU (Continued)	FRR MTR-S (NETHERLANDS) [608]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	FRR MTR-S (PORTUGAL) [632]				
	FRR MTR-S (SWITZERLAND) [658]				
	FRR MTR-S (TURKEY) [644]				
	FRR PIN CLUSTER (CANADA) [661]				
	FRR PIN CLUSTER (CANADA) [662]				
	FRR PIN CLUSTER (CANADA) [663]				
	FRR SLOWPOKE (CANADA) [665]				
	FRR SLOWPOKE (CANADA) [666]				
	FRR SLOWPOKE (CANADA) [668]				
	FRR SLOWPOKE (CANADA) [669]				
	FRR SLOWPOKE (MONTREAL) [667]				
	FRR TUBES (AUSTRALIA) [684]				
	FRR TUBES (AUSTRALIA) [300]				
	FRR TUBES (DENMARK) [676]				
	FRR TUBES (DENMARK) [678]				
	FRR TUBES (GERMANY) [683]				
	FRR TUBES (GERMANY) [685]				
	GENTR [97]				
	GRR (GREECE) [1069]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
16. U-ALx, HEU (Continued)	GRR (GREECE) [440]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	GTRR [87]				
	HIFAR (AUSTRALIA) [680]				
	HOR (NETHERLANDS) [713]				
	IAN-R1 (COLUMBIA) [596]				
	IAN-R1 (COLUMBIA) [803]				
	IEA-R1 (BRAZIL) [954]				
	IOWA ST. UNIV. [792]				
	JEN-1 (SPAIN) [795]				
	JMTR (JAPAN) [123]				
	JMTR (JAPAN) [886]				
	JRR-2 (JAPAN) [606]				
	JRR-2 (JAPAN) [885]				
	JRR-4 (JAPAN) [1070]				
	JRR-4 (JAPAN) [505]				
	KURR (JAPAN) [601]				
	MACMASTER (CANADA) [614]				
	MIT [135]				
	MIT [136]				
	MNR (CANADA) [1064]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
16. U-ALx, HEU (Continued)	MURR (COLUMBIA) [142]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	MURR (COLUMBIA) [143]				
	MURR (COLUMBIA) [144]				
	MURR (COLUMBIA) [962]				
	OHIO STATE [157]				
	PRR-1 (PHILIPPINES) [638]				
	PURDUE UNIVERSITY [177]				
	R-2 SVTR (SWEDEN) [801]				
	RA-3 (ARGENTINA) [634]				
	RA-3 (ARGENTINA) [636]				
	RECH-1 (CHILE) [708]				
	RHF (FRANCE) [179]				
	RINSC [180]				
	SAPHIR (SWITZERLAND) [444]				
	SLOWPOKE (CANADA) [1065]				
	SLOWPOKE (CANADA) [296]				
	THOR (TAIWAN) [629]				
	TRR-1 (THAILAND) [633]				
	UMRR (ROLLA) [881]				
	UNIV OF FLORIDA (ARGONAUT) [272]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
16. U-ALx, HEU (Continued)	UNIV OF MASS-LOWELL [274]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	UNIV OF VIRGINIA [279]				
17. U-ALx, MEU	ENEA SALUGGIA (ITALY) [760]				
	FRR MTR (ARGENTINA) [547]				
	FRR MTR (JAPAN) [551]				
	FRR MTR (TAIWAN) [555]				
	FRR MTR (VENEZUELA) [559]				
	FRR MTR-C (JAPAN) [552]				
	FRR MTR-C (PORTUGAL) [540]				
	FRR MTR-C (SWEDEN) [523]				
	FRR MTR-O (PORTUGAL) [541]				
	FRR MTR-S (JAPAN) [553]				
	FRR MTR-S (PORTUGAL) [542]				
	FRR TUBES (AUSTRALIA) [299]				
	IEA-R1 (BRAZIL) [1076]				
	IEA-R1 (BRAZIL) [545]				
	JEN-1 (SPAIN) [749]				
	JRR-3M (JAPAN) [1056]				
	PRR-1 (PHILLIPPINES) [558]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
17. U-ALx, MEU (Continued)	RPI (PORTUGAL) [943]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	RU-1 (URAGUAY) [1073]				
	RU-1 (URAGUAY) [557]				
	RV-1 (VENEZUELA) [816]				
	UNIV OF FLORIDA (ARGONAUT) [273]				
	UNIV OF MICHIGAN (CONTROL) [1005]				
	UNIV OF MICHIGAN [276]				
	UNIV OF MICHIGAN [277]				
	WORCESTER POLY INSTITUTE [287]				
	ZPRL (TAIWAN) [554]				
18. U3Si2	ASTRA (AUSTRIA) [712]				
	DR-3 (DENMARK) [1059]				
	DR-3 (DENMARK) [759]				
	FRG-1 (GERMANY) [741]				
	FRJ TUBES (GERMANY) [999]				
	FRR ASTRA (AUSTRIA) [515]				
	FRR MTR-C (CANADA) [512]				
	FRR MTR-C (GERMANY) [517]				
	FRR MTR-C (GREECE) [531]				
	FRR MTR-C (JAPAN) [289]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
18. U3Si2 (Continued)	FRR MTR-C (NETHERLANDS) [509]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	FRR MTR-C2 (TURKEY) [527]				
	FRR MTR-S (CANADA) [513]				
	FRR MTR-S (GERMANY) [1067]				
	FRR MTR-S (GERMANY) [519]				
	FRR MTR-S (GREECE) [532]				
	FRR MTR-S (JAPAN) [506]				
	FRR MTR-S (JAPAN) [508]				
	FRR MTR-S (NETHERLANDS) [510]				
	FRR MTR-S (TURKEY) [528]				
	FRR PIN CLUSTER (CANADA) [660]				
	FRR PIN CLUSTER (SO. KOREA) [293]				
	FRR PIN CLUSTER (SO. KOREA) [659]				
	FRR TUBES (DENMARK) [298]				
	FRR TUBES (GERMANY) [673]				
	FRR TUBES (GERMANY) [674]				
	FRR TUBES (GERMANY) [675]				
	IOWA ST. UNIV. [953]				
	JMTR (JAPAN) [507]				
	JRR-4 (JAPAN) [1071]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
18. U3Si2 (Continued)	NEREIDE (FRANCE) [751]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 16, 17, and 18 are analyzed for preclosure criticality safety as part of Criticality Group 8, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 16, 17, and 18 are evaluated for criticality potential as part of Criticality Group 8. Section 2.2.1.4.1.3.
	OHIO STATE [158]				
	ORR [165]				
	ORR [850]				
	ORR [944]				
	ORR EXPERIMENTS [1086]				
	PURDUE UNIVERSITY [178]				
	R-2 SVTR (SWEDEN) [942]				
	RINSC [181]				
	SAPHIR (SWITZERLAND) [443]				
	SAPHIR (SWITZERLAND) [945]				
	UMRR (ROLLA) [146]				
	UNIV OF MASS-LOWELL [275]				
	UNIV OF VIRGINIA [952]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
19. Th/U Carbide, TRISO or BISO coated particles in graphite	FSVR [85]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 19, 20, and 21 are analyzed for preclosure criticality safety as part of Criticality Group 6, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 19, 20, and 21 are evaluated for criticality potential as part of Criticality Group 6. Section 2.2.1.4.1.3.
	FSVR [86]				
	HTGR (PEACH BOTTOM SCRAP) [935]				
	PEACH BOTTOM UNIT I CORE II (INTACT) [206]				
	PEACH BOTTOM UNIT I CORE II [171]				
20. Th/U Carbide, Mono-pyrolytic carbon coated particles in graphite	GA HTGR FUEL [89]				
	PEACH BOTTOM UNIT I CORE I (PTE-1) [1085]				
	PEACH BOTTOM UNIT I CORE I [169]				
	PEACH BOTTOM UNIT I CORE I [170]				
21. Pu/U Carbide, Non Graphite Clad, Not Sodium Bonded	EBR-II, FFTF and MTR EXPERIMENTS [42]				
	FAST REACTOR FUEL [1029]				
	FFTF CARBIDE FUEL EXPER. [347]				
	FFTF-TFA PINS (AC-3) [1046]				
	FFTF-TFA-ACN-1 RODS [865]				
	FFTF-TFA-FC-1 [325]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
22. MOX, Zirc Clad	BRP-EP [29]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 22, 23, and 24 are analyzed for preclosure criticality safety as part of Criticality Group 2, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 22, 23, and 24 are evaluated for criticality potential as part of Criticality Group 2. Section 2.2.1.4.1.3.
	EBWR [63]				
	GE TEST [96]				
	H. B. ROBINSON [99]				
	SAXTON [787]				
23. MOX, SST Clad	BABCOCK and WILCOX SCRAP [18]				
	EBR-II and TREAT EXPERIMENTS [858]				
	EBR-II OXIDE FUEL EXPER [345]				
	EBR-II OXIDE FUEL EXPER [364]				
	EPRI [67]				
	FFTF OXIDE EXPERIMENTS [349]				
	FFTF-DFA/TDFA [71]				
	FFTF-DFA/TDFA PINS [323]				
	FFTF-TFA PINS [320]				
	FFTF-TFA-AB-1 [317]				
	FFTF-TFA-ACN-1 PINS [321]				
	FFTF-TFA-ACO-2, 4 THRU 16 [329]				
	FFTF-TFA-CRBR-3 and CRBR-5 [322]				
	FFTF-TFA-DEA-2 [324]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
23. MOX, SST Clad (Continued)	FFTF-TFA-MFF-1 and 1A (CDE) [330]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 22, 23, and 24 are analyzed for preclosure criticality safety as part of Criticality Group 2, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 22, 23, and 24 are evaluated for criticality potential as part of Criticality Group 2. Section 2.2.1.4.1.3.
	FFTF-TFA-P0-2,4 and 5 [333]				
	FFTF-TFA-SRF-3 and 4 [334]				
	FFTF-TFA-UO-1 [335]				
	LWR SAMPLES (MOX) [134]				
	ORR-BW-1 [160]				
	PNL MOX FUEL (7010) [415]				
	PNL MOX FUEL (7055) [416]				
	PNL MOX FUEL [414]				
	PNL MOX STAR 3 [433]				
	PNL MOX STAR 4 [434]				
	PNL MOX STAR 5 [435]				
	PNL MOX STAR 6 [436]				
	PNL MOX STAR 7 [422]				
	PNL-3 [420]				
	SAXTON [883]				
	SODIUM LOOP SAFETY FAC. [352]				
	SODIUM LOOP SAFETY FAC. [367]				
	US/UK FUEL PINS [356]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
24. MOX, Non-SST/ Non-Zirc Clad	MISCELLANEOUS TREAT FUEL [369]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 22, 23, and 24 are analyzed for preclosure criticality safety as part of Criticality Group 2, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 22, 23, and 24 are evaluated for criticality potential as part of Criticality Group 2. Section 2.2.1.4.1.3.
	MOX SCRAP SNF [368]				
	PNL MOX FUEL (7057) [417]				
	PNL MOX PELLETS (7057) [418]				
	PNL MOX PINS (7057) [419]				
25. Th/U Oxide, Zirc Clad	SHIPPINGPORT (MET MOUNTS) [1087]		Fuel groups 25 and 26 are analyzed for preclosure criticality safety as part of Criticality Group 5, Section 1.14.		Fuel Groups 25 and 26 are evaluated for criticality potential as part of Criticality Group 5. Section 2.2.1.4.1.3.
	SHIPPINGPORT LWBR BLKT I [374]				
	SHIPPINGPORT LWBR BLKT II [375]				
	SHIPPINGPORT LWBR BLKT III [376]				
	SHIPPINGPORT LWBR REFLCT. IV [371]				
	SHIPPINGPORT LWBR REFLCT. V [372]				
	SHIPPINGPORT LWBR SCRAP (LINER 15718) [379]				
	SHIPPINGPORT LWBR SCRAP [377]				
	SHIPPINGPORT LWBR SEED [380]				
26. Th/U Oxide, SST Clad	DRESDEN I [44]				
	ERR [1057]				
	ERR [68]				
	FAST REACTOR FUEL [906]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
27. U-Zirc Hydride, SST/Incoloy Clad, HEU	BER-II TRIGA (GERMANY) [236]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA FFCR (OSU) [1041]				
	TRIGA FLIP (AUSTRIA) [492]				
	TRIGA FLIP (DAMAGED) (SO. KOREA) [819]				
	TRIGA FLIP (GA) [729]				
	TRIGA FLIP (MEXICO) [493]				
	TRIGA FLIP (SLOVENIA) [495]				
	TRIGA FLIP (SO. KOREA) [494]				
	TRIGA FLIP [239]				
	TRIGA FLIP [240]				
	TRIGA FLIP [241]				
	TRIGA FLIP [242]				
	TRIGA FLIP [243]				
	TRIGA FLIP [354]				
	TRIGA FLIP [DAMAGED] (TEXAS A&M) [844]				
	TRIGA FLIP ANL-W (NRAD) [884]				
	TRIGA FLIP FFCR (GA) [996]				
	TRIGA FLIP FFCR (OSU) [702]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
27. U-Zirc Hydride, SST/Incoloy Clad, HEU (Continued)	TRIGA FLIP FFCR (SO. KOREA) [733]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA FLIP UNIV OF WISCONSIN [1035]				
	TRIGA HIGH POWER (GA) [998]				
	TRIGA HIGH POWER (ROMANIA) [302]				
	TRIGA HIGH POWER (ROMANIA) [930]				
28. U-Zirc Hydride, SST/Incoloy Clad, MEU	GA RERTR [90]				
	TRIGA FFCR (AFRRI) [969]				
	TRIGA FFCR (UC-IRVINE) [1050]				
	TRIGA FFCR (UC-IRVINE) [1052]				
	TRIGA STD (HANFORD) [316]				
	TRIGA (DEMOUNTABLE) (U OF AZ) [971]				
	TRIGA 20/30 (GA) [995]				
	TRIGA ACPR (SLOVENIA) [932]				
	TRIGA ACPR (JAPAN) [480]				
	TRIGA ACPR (ROMANIA) [1077]				
	TRIGA ACPR PENN. STATE UNIV. [1002]				
	TRIGA FFCR (DORF) [315]				
	TRIGA FFCR (ENGLAND) [987]				
	TRIGA FFCR (GA) [1003]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
28. U-Zirc Hydride, SST/Incoloy Clad, MEU (Continued)	TRIGA FFCR (HEIDELBERG) [1045]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA FFCR (ITALY) [730]				
	TRIGA FFCR (MNRC) [1055]				
	TRIGA FFCR (MNRC) [703]				
	TRIGA FFCR (MNRC) [737]				
	TRIGA FFCR (OSU) [1039]				
	TRIGA FFCR (PENN. STATE UNIV.) [815]				
	TRIGA FFCR (SLOVENIA) [941]				
	TRIGA FFCR (SO. KOREA) [734]				
	TRIGA FFCR (U OF AZ) [974]				
	TRIGA FFCR (U OF TX AUSTIN) [825]				
	TRIGA FFCR (ZAIRE) [735]				
	TRIGA FFCR [448]				
	TRIGA FLIP (BANGLADESH) [470]				
	TRIGA FLIP (MALAYSIA) [497]				
	TRIGA FLIP (PHILIPPINES) [499]				
	TRIGA FLIP (TAIWAN) [498]				
	TRIGA FLIP (THAILAND) [496]				
	TRIGA STD (ACPR) [895]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
28. U-Zirc Hydride, SST/Incoloy Clad, MEU (Continued)	TRIGA STD (ARRR) [780]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA STD (AUSTRIA) [469]				
	TRIGA STD (BRAZIL) [1063]				
	TRIGA STD (ENGLAND) [485]				
	TRIGA STD (FINLAND) [472]				
	TRIGA STD (GERMANY) [305]				
	TRIGA STD (GERMANY) [474]				
	TRIGA STD (HANNOVER) [473]				
	TRIGA STD (HEIDELBERG) [1044]				
	TRIGA STD (IFE) (ENGLAND) [1043]				
	TRIGA STD (IFE) (ITALY) [929]				
	TRIGA STD (IFE) (OSU) [1040]				
	TRIGA STD (IFE) (U OF AZ) [972]				
	TRIGA STD (IFE) (U OF AZ) [973]				
	TRIGA STD (IFE) (U OF IL) [1048]				
	TRIGA STD (IFE) (UC-IRVINE) [1051]				
	TRIGA STD (IFE) (UC-IRVINE) [824]				
	TRIGA STD (INDONESIA) [475]				
	TRIGA STD (INDONESIA) [476]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
28. U-Zirc Hydride, SST/Incoloy Clad, MEU (Continued)	TRIGA STD (ITALY) [1080]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA STD (ITALY) [477]				
	TRIGA STD (ITALY) [478]				
	TRIGA STD (JAPAN) [479]				
	TRIGA STD (MEXICO) [482]				
	TRIGA STD (MNRC) [1053]				
	TRIGA STD (MNRC) [1054]				
	TRIGA STD (MNRC) [704]				
	TRIGA STD (MSU) [873]				
	TRIGA STD (REED COLLEGE) [775]				
	TRIGA STD (ROMANIA) [1078]				
	TRIGA STD (SLOVENIA) [1079]				
	TRIGA STD (SLOVENIA) [488]				
	TRIGA STD (SO. KOREA) [484]				
	TRIGA STD (SOLVENIA) [731]				
	TRIGA STD (THAILAND) [489]				
	TRIGA STD (TURKEY) [490]				
	TRIGA STD (U OF AZ) [59]				
	TRIGA STD (U OF AZ) [975]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
28. U-Zirc Hydride, SST/Incoloy Clad, MEU (Continued)	TRIGA STD (U OF ILL) [449]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA STD (UC BERKLEY) [874]				
	TRIGA STD (USGS) [964]				
	TRIGA STD (ZAIRE) [486]				
	TRIGA STD [233]				
	TRIGA STD [237]				
	TRIGA STD [244]				
	TRIGA STD [246]				
	TRIGA STD [250]				
	TRIGA STD [251]				
	TRIGA STD [252]				
	TRIGA STD [253]				
	TRIGA STD [254]				
	TRIGA STD [258]				
	TRIGA STD [260]				
	TRIGA STD [261]				
	TRIGA STD [262]				
	TRIGA STD [264]				
	TRIGA STD [265]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
28. U-Zirc Hydride, SST/Incoloy Clad, MEU (Continued)	TRIGA STD [268]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA STD [353]				
	TRIGA STD [370]				
29. U-Zirc Hydride, Alum Clad, MEU	TRIGA STD (AUSTRIA) [462]				
	TRIGA STD (BRAZIL) [471]				
	TRIGA STD (CORNELL) [1047]				
	TRIGA STD (DOW) [970]				
	TRIGA STD (FINLAND) [463]				
	TRIGA STD (GA) [728]				
	TRIGA STD (GA) [870]				
	TRIGA STD (GERMANY) [465]				
	TRIGA STD (HANFORD) [876]				
	TRIGA STD (HANNOVER) [303]				
	TRIGA STD (HEIDELBERG) [464]				
	TRIGA STD (ITALY) [466]				
	TRIGA STD (ITALY) [467]				
	TRIGA STD (JAPAN) [481]				
	TRIGA STD (KSU) [804]				
	TRIGA STD (KSU) [871]				

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
29. U-Zirc Hydride, Alum Clad, MEU (Continued)	TRIGA STD (MSU) [878]	Fuel Groups 1 through 30 are not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Fuel Groups 27, 28, 29, and 30 are analyzed for preclosure criticality safety as part of Criticality Group 7, Section 1.14.	Fuel Groups 1 through 30 are analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Groups 27, 28, 29, and 30 are evaluated for criticality potential as part of Criticality Group 7. Section 2.2.1.4.1.3.
	TRIGA STD (SLOVENIA) [468]				
	TRIGA STD (SO. KOREA) [483]				
	TRIGA STD (U OF UTAH) [699]				
	TRIGA STD (UNIV. OF TEXAS) [877]				
	TRIGA STD (ZAIRE) [487]				
	TRIGA STD [235]				
	TRIGA STD [238]				
	TRIGA STD [256]				
	TRIGA STD [267]				
	TRIGA STD [314]				
	TRIGA STD [447]				
30. U-Zirc Hydride, Declad	SNAP [203]				
31. Metallic Sodium Bonded	Fuel Group 31 is not included in the license application.	Fuel Group 31 is sodium-bonded fuel. Some of this material will be treated into HLW. This material is not included in the license application.	Fuel Group 31 is sodium-bonded fuel. Some of this material will be treated into HLW. This material is not included in the license application.	Fuel Group 31 is sodium-bonded fuel. Some of this material will be treated into HLW. This material is not included in the license application. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Group 31 is sodium-bonded fuel. Some of this material will be treated into HLW. This material is not included in the license application.

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
32. Naval	Fuel Group 32 information is provided by the Naval Nuclear Propulsion Program	Fuel Group 32 information is provided in Section 1.8.1.3.	Fuel Group 32 information is provided in Section 1.14 of the Naval Nuclear Propulsion Program Technical Support Document.	Fuel Group 32 information is provided in Section 2.3.7 of the Naval Nuclear Propulsion Program Technical Support Document.	Fuel Group 32 information is provided in Section 2.2.1.4.1 of the Naval Nuclear Propulsion Program Technical Support Document.
33. Canyon Stabilization	Fuel Group 33 will be processed into HLW.	Fuel Group 33 will be processed into HLW. The HLW radionuclide release from an impact breach of a dropped canister is discussed in Section 1.8.1.3.1.	Fuel Group 33 will be processed into HLW. Section 1.14.	Fuel Group 33 will be processed into HLW. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Fuel Group 33 will be processed into HLW.

Table 1.5.1-23. DOE SNF Fuel Group Disposal Analysis Plan (Continued)

Fuel Groups	Fuel Names[a]	Preclosure Releases	Preclosure Criticality[b]	TSPA[c]	Postclosure Criticality
34. Misc (Not Previously Listed)	AMERICIUM TARGETS [776]	Fuel Group 34 is not analyzed for preclosure releases because there are no normal operations or event sequences that result in a release from DOE SNF canisters. An event sequence involving a drop and breach of a DOE standardized canister with DOE SNF is a beyond Category 2 event sequence, so consequence analyses are not required.	Criticality evaluation for this fuel group has not been completed.	Fuel Group 34 is analyzed for postclosure releases based on use of a single surrogate fuel with instantaneous release and a conservative radionuclide inventory distribution. Sections 2.3.7.4.1.1; 2.3.7.4.2.2; and 2.3.7.8.1.	Criticality evaluation for this Fuel Group has not been completed.
	EBR-II NITRIDE FUEL EXPER [363]				
	HWCTR TMT-1-2 and 1-3 [112]				
	KEMA [861]				
	MISCELLANEOUS TREAT FUEL [905]				
	RERTR MINIPLATES [1090]				
	TRU SCRAP SNF [904]				

NOTE: [a]The bracketed numbers in the Fuel Names column represent the actual record numbers from the SNF database.
[b]For each criticality group, only the representative fuel for that group has been analyzed. The representative fuels for each group are identified in Section 1.14.2.3.2.3.1.
[c] For each postclosure criticality group, only the representative fuel for that group has been analyzed. The representative fuels for each group are identified in Section 2.2.1.4.1.3.

Source: DOE 2007, Table 6.

Table 1.5.1-24. Ranges of Nominal Properties for DOE Spent Nuclear Fuel

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
01. U metal, zirc clad, LEU	2103	1.7 – 0.5	Zirconium	Fair Poor	U metal	None	Plates Tubes	2.1 – 9.9	1.0 – 4.3
02. U metal, nonzirc clad, LEU	8	3.4 – 0.2	SST Aluminum	Poor Good Fair	U metal	None	Cans of scrap Tubes None	0.6 – 0.9	1.4 – 1.9
03. U-zirc	<1	92.9 – 0.5	Zirconium	Fair Good	U metal 2% Zr U-Zr	None	Tube Cylinders Plates	2.0 – 12.5	2.0 – 7.4
04. U-Mo	4	25.8 – 2.4	Zirconium Aluminum None	Good Poor Fair None	U-Mo	None	Rod Tube Plates in can	1.0 – 3.8	0.1 – 2.1
05. U oxide, zirc clad, intact, HEU	<1	92.5 – 23.1	Zirconium	Fair Good	UO_2	ZrO2-CaO Graphite ZrO2	Rod Assembly Plates	3.1 – 9.0	0.3 – 7.4
06. U oxide, zirc clad, intact, MEU	2	6.9 – 5	Zirconium	Fair Good	UO_2	None	Plates Rod Cans of rods Element	2.9 – 5.2	0.3 – 3.8
07. U oxide, zirc clad, intact, LEU	90	4.9 – 0.6	Zirconium	Good Fair	UO_2	None	Tubes Rod Plates Assembly	0.8 – 14.7	0.4 – 8.5
08. U oxide, SST/hastelloy clad, intact, HEU	<1	93.2 – 91.0	SST Hastelloy	Good Fair	U oxide UO_2	SST SST (316L) SST 304B SST 304 None	Tubes Cans of scrap Rod Plates Rod assembly	2.1 – 6.6	0.9 – 3.7

Table 1.5.1-24. Ranges of Nominal Properties for DOE Spent Nuclear Fuel (Continued)

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
09. U oxide, SST clad, intact, MEU	<1	20.0 – 5.5	SST	Good Fair	UO_2-BeO_2 UO_2	ZrO_2-CaO None	Rod Element	2.4 – 4.0	0.3 – 1.5
10. U oxide, SST clad, intact, LEU	<1	1.9 – 0.2	SST	Good Fair	UO_2	None	Tube Rod	1.5 – 12.0	0.4 – 8.5
11. U oxide, nonalum clad, nonintact or declad, HEU	<1	93.3 – 21.0	Nichrome Hastelloy SST Zirconium None	Poor None	UO_2	BEO SST Nichrome None	Cans of scrap	0.2 – 2.8	2.8 – 5.6
12. U oxide, nonalum clad, nonintact or declad, MEU	<1	18.6 – 5.2	None Zirconium SST	Poor	UO_2	Gd_2O_3 None SST	Experiment capsule Scrap Cans of scrap	3.4 – 9.9	0.4 – 9.1
13. U oxide, nonalum clad, nonintact or declad, LEU	83	3.2 – 1.1	Zirconium SST	Poor	UO_2	None	Cans of scrap Scrap Rod	12.4 - 13.5	0.5 – 14.0
14. U oxide, alum clad, HEU	5	89.9 – 58.1	Aluminum	Good Fair	U_3O_8	Alum	Plates	2.0 – 3.6	2.8 – 17.2
15. U oxide, alum clad, MEU and LEU	<1	20.0 – 8.9	Aluminum	Good Fair	U_3O_8	Alum	Plates Assembly	2.2 – 3.3	3.0 – 4.8

Table 1.5.1-24. Ranges of Nominal Properties for DOE Spent Nuclear Fuel (Continued)

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
16. U-ALx, HEU	8	93.3 – 21.9	Aluminum	Good Fair	U-ALX	Alum	Rods Tubes Plates Pin cluster Assemblies Elements	0.4 – 10.1	1.3 – 16.3
17. U-ALx, MEU	3	20.0 – 9.0	Aluminum	Good Fair	U-ALX	Alum	Assembly Element Plates	2.0 – 3.4	2.1 – 4.1
18. U_3Si_2	8	22.0 – 5.2	Aluminum	Good Fair Poor	U_3SI_2	Alum	Tubes Multi-pin cluster Assembly Cans of scrap	2.0 – 3.4	2.6 – 4.1
19. Th/U carbide, TRISO or BISO coated particles in graphite[b]	25	84.4 – 71.4	BISC TRISO	Good	ThC_2-UC_2 ThC-UC	Graphite	Tubes Cans of scrap	2.6 – 10.5	3.5 – 14.2
20. Th/U carbide, mono-pyrolytic carbon coated particles in graphite[b]	2	93.2 – 80.6	Mono-pyrolytic carbon	Poor	ThCO-UCO ThC_2-UC_2	Graphite	Element Carbon coated part Cans of scrap	~12.0	~3.5

Table 1.5.1-24. Ranges of Nominal Properties for DOE Spent Nuclear Fuel (Continued)

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
21. Pu/U carbide, nongraphite clad, not sodium bonded	<1	67.3 – 1	SST	Good Fair Poor	Pu/U carbide	None	Element Cans of scrap Rod	7.7 – 12.0	0.2 – 5.2
22. MOX, zirc clad	2	21.3 – 1.3	Zirconium	Poor Good Fair	PuO_2-UO_2	None	Rod Cans of scrap Plates Element	3.3 – 7.1	0.3 – 6.6
23. MOX, SST clad	11	87.4 – 2.1	SST	Poor Good Fair	PuO_2-UO_2 PuO_2	None	Rod Plates Element Cans of scrap Scrap	1.1 – 12.0	0.2 – 9.1
24. MOX, non-SST/non zirc clad	<1	54.3 – 5.0	Unknown	NA Poor	PuO_2-UO_2	None Unknown	Scrap Cans of scrap	Unknown	Unknown
25. Th/U oxide, zirc clad	43	98.4 – 10.1	Zirconium	Good Poor NA	ThO_2-UO_2 ceramic	None	Rod Assembly Cans of scrap	~11.8	9.0 – 22.3
26. Th/U oxide, SST clad	8	97.8 – 7.6	SST	Fair Good Poor	ThO_2-UO_2	None	Assembly Cans of scrap Rod	5.2 – 11.7	0.4 – 11.9
27. U-zirc hydride, SST/incoloy clad, HEU	<1	93.2 – 42.5	SST Incoloy	Good Fair	U-ZrHX-Er	None	Rod Element	2.4 – 3.8	0.5 – 3.2

Table 1.5.1-24. Ranges of Nominal Properties for DOE Spent Nuclear Fuel (Continued)

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
28. U-zirc hydride, SST/incoloy clad, MEU	2	20.0 – 11.9	SST Incoloy	Good Poor	U-ZrHX U-ZrHX-Er	None	Element Cans of scrap	2.4 – 3.8	~1.5
29. U-zirc hydride, alum clad, MEU	<1	20.0 – 16.8	Aluminum	Good	U-ZrHX	None	Element	~2.4	~1.5
30. U-zirc hydride, declad	<1	~89.7	None	NA	U-ZrHX	None	Declad rod	~1.2	~1.2
31. Metallic sodium bonded	NA	93.2 – <0.1	SST None Unknown	Poor Good NA Fair	PuO_2-UO_2 U-10Zr U-Mo U-10Zr U metal U-Pu-Zr UO_2 U metal Pu/U alloy U-5 fissium Pu/U carbide	None	Fuel in sodium Rod Assembly Cans of Scrap Scrap	1.8 – 12.0	0.2 – 9.1
32. Naval[c]	65	—	—	—	—	—	—	—	—
33. Canyon stabilization	NA	—	—	—	—	—	—	—	—

Table 1.5.1-24. Ranges of Nominal Properties for DOE Spent Nuclear Fuel (Continued)

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
34. Misc (not previously listed)	<1	90.0 – 14.6	None Zirconium Unknown Aluminum SST	Fair Poor NA Good	ThO_2-UO_2 U-Th metal U metal Am oxide Pu/U nitride	None Alum (1100) Unknown	Cans of scrap Tube Rod	0.3 – 9.9	0.5 – 2.6

NOTE: [a]MTHM are rounded to next higher whole number or reported as <1 MTHM, as applicable.
[b]For fuel groups 19 and 20, cladding composition and cladding condition are reporting particle coating composition and condition. Group 31 is sodium-bonded fuel. Some of this material will be treated into HLW. The disposition of this material is not included in the initial license application. Group 32 information is provided in the Naval Nuclear Propulsion Technical Support Document. Group 33 will be processed into glass and is included in Section 1.5.1.2 as HLW.
[c]See Section 1.5.1.4 of the Naval Nuclear Propulsion Program Technical Support Document.
NA = not applicable.

Source: DOE 2007, Table 5.

Table 1.5.1-25. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the DOE SNF Canister

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
DOE and Commercial Waste Package System (DS)	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.	The standardized DOE SNF canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the standardized DOE SNF canister capacity curve.
				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.	The standardized DOE SNF canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the standardized DOE SNF canister capacity curve.
				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.	The standardized DOE SNF canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the standardized DOE SNF canister capacity curve.
				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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The standardized DOE SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The standardized DOE SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-25. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the DOE SNF Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The standardized DOE SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The standardized DOE SNF canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the standardized DOE SNF canister capacity curve.
				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.	The standardized DOE SNF canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the standardized DOE SNF canister capacity curve.
Mechanical Handling System (H)	Waste Transfer/ Canister Transfer	DOE Canister Staging Racks (and Fire and Thermal Barrier) (060-HTC0-RK-00006-10)	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.	The thermal barrier (around the staging racks) is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The standardized DOE SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the reliabilities of each component.)

Table 1.5.1-25. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the DOE SNF Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
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 transportation cask resulting from the spectrum of fires shall be less than or equal to 2×10^{-6} per fire event.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The standardized DOE SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

NOTE: "Protect against" in this table means either "reduce the probability of" or "reduce the frequency of."
For casks, canisters, and associated handling equipment that were previously designed, the component design will be evaluated to confirm that the controlling parameters and values are met.
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.
The numbers appearing in parentheses in the third column are component numbers.
Facility Codes: CR: Canister Receipt and Closure Facility; SB: Balance of Plant.
System Codes: DS: DOE and Commercial Waste Package System; H: Mechanical Handling System.
Subsystem Codes: HTC: Canister Transfer.

Table 1.5.1-26. Deterministic Evaluation of Drop Events

Drop Event	18 in. Standardized Canister	24 in. Standardized Canister	Multicanister Overpack
30 ft vertical	Yes[a]	Yes[b]	No
30-ft center of gravity over the corner	Yes[a]	Yes[b]	No
30 ft horizontal	Yes[a]	Yes[b]	No
30 ft at 45°	Yes[a]	Yes[b]	No
30 ft worst orientation	Yes[a]	Yes[b]	No
40 in. horizontal onto a 6-in. post	Yes[a]	Yes[b]	Yes[c]
23 ft vertical	Yes[a]	Yes[b]	Yes (0°, 1°, 3°)[d]
2 ft worst orientation	Yes (80°)[a]	Yes (65°)[b]	Yes (60°, 90°, 115°)[d]
23-ft edge-to-collar drop	Not applicable[e]	Not applicable[e]	Yes[c]
2-ft drop simulating the drop onto the edge of a waste package with toppling onto the opposite edge	Yes[f]	No	No

NOTE: [a]Blandford 2003, Tables 3 and 8.
[b]Blandford 2003, Tables 3 and 6.
[c]BSC 2004a, Table 2.
[d]Snow 2003, Table 4.
[e]The standardized canisters do not have a collar, so this event is not applicable to them.
[f]Morton et al. 2002, Part II, Section 6.2, for canister 18-15-PW-08.

Table 1.5.1-27. Calculated Peak Equivalent Plastic Strains for Drop Events Evaluated

Drop Event	18 in. Standardized Canister[a]	24 in. Standardized Canister[a]	Multicanister Overpack[a]
30 ft vertical	7% outside[b] 3% midplane[b] 6% inside[b]	6% outside[c] 0.6% midplane[c] 4% inside[c]	Not performed
30-ft center of gravity over the corner; 6° for the 18-in. standardized canister and 7° for the 24-in. standardized canister	9% outside[b] 3% midplane[b] 10% inside[b]	0.7% outside[c] 0.1% midplane[c] 0.6% inside[c]	Not performed
30 ft horizontal	40% outside[b] 15% midplane[b] 26% inside[b]	34% outside[c] 16% midplane[c] 22% inside[c]	Not performed
30 ft at 45°	33% outside[b] 9% midplane[b] 36% inside[b]	48% outside[c] 22% midplane[c] 42% inside[c]	Not performed
30 ft worst orientation	57% outside[b] 19% midplane[b] 42% inside[b]	57% outside[c] 23% midplane[c] 48% inside[c]	Not performed
40 in. horizontal onto a 6-in. post	39% outside[b] 14% midplane[b] 40% inside[b]	16% outside[c] 15% midplane[c] 17% inside[c]	10% to 63% outside[d] 8% to 60% midplane[d] 12% to 56% inside[d]
23 ft vertical	10% outside[b] 3% midplane[b] 6% inside[b]	6% outside[c] 0.6% midplane[c] 4% inside[c]	5% outside at 0°[e] 20% outside at 1°[e] 35% outside at 3°[e]
2 ft worst orientation	24% outside[b] 11% midplane[b] 13% inside[b]	23% outside[c] 15% midplane[c] 16% inside[c]	22% outside[f] 15% inside[f]
23-ft edge-to-collar drop	Not applicable[g]	Not applicable[g]	130% outside[h] 17% midplane[h] 6% inside[h]
2-ft drop simulating the drop onto the edge of a waste package with toppling onto the opposite edge	20% outside[i] 7% midplane[i] 18% inside[i]	Not performed	Not performed

NOTE: [a]The maximum strain generally occurs at the outside surface.
[b]Blandford 2003, Tables 3 and 8.
[c]Blandford 2003, Tables 3 and 6.
[d]Snow 2004, Table 1 and Section 8.1. A range of values is provided because the strains are highly dependent upon the location of the impact. The lower strains are associated with an impact centered on a fuel basket and the higher strains are associated with an impact offset 1 in. from a fuel basket base plate.
[e]Snow 2003, Table 5, for Mark IV baskets.
[f]Snow 2003, Table 6, for Mark IV baskets for the bottom and main shell.
[g]The standardized canisters do not have a collar, so this event is not applicable to them.
[h]Snow 2003, Section 8.3.3.
[i]Morton et al. 2002, Part II, Table 25, for canister 18-15-PW-08.

Table 1.5.1-28. Total Thermal Power of DOE Spent Nuclear Fuel at a Specified Time

2010		2030	
Heat Output (W)		Heat Output (W)	
Nominal	Bounding	Nominal	Bounding
7.18×10^5	1.25×10^6	4.67×10^5	8.54×10^5

Source: DOE 2007, Table 4.

Table 1.5.1-29. Total DOE SNF Radionuclide Inventory

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
^{227}Ac	4.98×10^{1}	1.01×10^{2}
^{110}Ag	2.74	4.86
^{110m}Ag	2.06×10^{2}	3.65×10^{2}
^{111}Ag	—	—
^{241}Am	2.11×10^{6}	3.90×10^{6}
^{242}Am	5.11×10^{3}	9.48×10^{3}
^{242m}Am	5.13×10^{3}	9.53×10^{3}
^{243}Am	4.06×10^{3}	7.59×10^{3}
^{136m}Ba	—	—
^{137m}Ba	3.60×10^{7}	6.64×10^{7}
^{140}Ba	—	—
^{10}Be	6.12×10^{-1}	1.29
^{211}Bi	4.99×10^{1}	1.01×10^{2}
^{212}Bi	2.48×10^{4}	5.06×10^{4}
^{14}C	1.83×10^{4}	2.79×10^{4}
^{113}Cd	—	—
^{113m}Cd	5.35×10^{3}	9.93×10^{3}
^{115m}Cd	7.78×10^{-8}	1.39×10^{-7}
^{141}Ce	9.27×10^{-9}	1.63×10^{-8}
^{142}Ce	1.44×10^{-2}	2.52×10^{-2}
^{144}Ce	2.94×10^{6}	5.30×10^{6}
^{36}Cl	2.98×10^{2}	4.67×10^{2}
^{242}Cm	4.24×10^{3}	7.88×10^{3}
^{243}Cm	1.69×10^{3}	3.23×10^{3}
^{244}Cm	2.32×10^{5}	4.45×10^{5}
^{245}Cm	7.14×10^{1}	1.39×10^{2}
^{246}Cm	1.10×10^{1}	2.16×10^{1}
^{247}Cm	4.37×10^{-5}	8.62×10^{-5}

Table 1.5.1-29. Total DOE SNF Radionuclide Inventory (Continued)

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	**Bounding Fuel Inventories (Ci)**
^{60}Co	7.49×10^{6}	9.79×10^{6}
^{51}Cr	1.64×10^{-13}	2.88×10^{-13}
^{134}Cs	1.85×10^{6}	3.20×10^{6}
^{135}Cs	3.13×10^{2}	5.78×10^{2}
^{136}Cs	—	—
^{137}Cs	3.81×10^{7}	7.02×10^{7}
^{152}Eu	4.67×10^{3}	8.22×10^{3}
^{154}Eu	8.37×10^{5}	1.51×10^{6}
^{155}Eu	2.94×10^{5}	5.27×10^{5}
^{156}Eu	—	—
^{55}Fe	7.67×10^{5}	9.71×10^{5}
^{59}Fe	6.83×10^{-8}	1.23×10^{-7}
^{223}Fr	6.87×10^{-1}	1.40
^{153}Gd	6.66	1.18×10^{1}
^{3}H	2.45×10^{5}	4.21×10^{5}
^{129}I	1.95×10^{1}	3.63×10^{1}
^{131}I	—	—
^{114}In	2.65×10^{-9}	4.38×10^{-9}
^{114m}In	2.77×10^{-9}	4.58×10^{-9}
^{115m}In	5.47×10^{-12}	9.77×10^{-12}
^{85}Kr	1.86×10^{6}	3.42×10^{6}
^{140}La	—	—
^{54}Mn	1.56×10^{3}	2.96×10^{3}
^{93}Mo	1.42×10^{2}	2.21×10^{2}
^{93m}Nb	1.31×10^{3}	2.18×10^{3}
^{94}Nb	2.37×10^{2}	3.49×10^{2}
^{95}Nb	4.04	7.18
^{95m}Nb	1.35×10^{-2}	2.40×10^{-2}

Table 1.5.1-29. Total DOE SNF Radionuclide Inventory (Continued)

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
^{144}Nd	6.78×10^{-7}	1.21×10^{-6}
^{147}Nd	—	—
^{59}Ni	4.56×10^{4}	7.13×10^{4}
^{63}Ni	5.28×10^{6}	8.20×10^{6}
^{237}Np	2.02×10^{2}	3.76×10^{2}
^{231}Pa	7.04×10^{1}	1.43×10^{2}
^{233}Pa	2.02×10^{2}	3.76×10^{2}
^{234}Pa	6.44×10^{-1}	1.22
^{234m}Pa	4.95×10^{2}	9.41×10^{2}
^{210}Pb	1.74×10^{-2}	2.74×10^{-2}
^{211}Pb	4.99×10^{1}	1.01×10^{2}
^{212}Pb	2.48×10^{4}	5.06×10^{4}
^{107}Pd	4.50×10^{1}	8.55×10^{1}
^{145}Pm	5.39×10^{2}	9.09×10^{2}
^{147}Pm	7.51×10^{6}	1.38×10^{7}
^{148}Pm	1.38×10^{-8}	2.49×10^{-8}
^{148m}Pm	2.44×10^{-7}	4.42×10^{-7}
^{212}Po	1.59×10^{4}	3.24×10^{4}
^{215}Po	4.99×10^{1}	1.01×10^{2}
^{216}Po	2.48×10^{4}	5.06×10^{4}
^{143}Pr	—	—
^{144}Pr	2.94×10^{6}	5.30×10^{6}
^{144m}Pr	3.52×10^{4}	6.36×10^{4}
^{236}Pu	2.68	4.85
^{237}Pu	9.77×10^{-11}	1.58×10^{-10}
^{238}Pu	9.72×10^{5}	1.79×10^{6}
^{239}Pu	4.75×10^{5}	7.71×10^{5}
^{240}Pu	3.65×10^{5}	6.21×10^{5}

Table 1.5.1-29. Total DOE SNF Radionuclide Inventory (Continued)

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	**Bounding Fuel Inventories (Ci)**
^{241}Pu	1.54×10^{7}	3.21×10^{7}
^{242}Pu	5.05×10^{2}	8.38×10^{2}
^{244}Pu	8.54×10^{-5}	1.61×10^{-4}
^{223}Ra	4.99×10^{1}	1.01×10^{2}
^{224}Ra	2.48×10^{4}	5.06×10^{4}
^{226}Ra	3.71×10^{-2}	5.39×10^{-2}
^{228}Ra	3.39	6.94
^{87}Rb	1.23×10^{-2}	2.19×10^{-2}
^{103m}Rh	3.67×10^{-6}	6.48×10^{-6}
^{106}Rh	6.26×10^{5}	1.14×10^{6}
^{219}Rn	4.99×10^{1}	1.01×10^{2}
^{220}Rn	2.48×10^{4}	5.06×10^{4}
^{103}Ru	4.07×10^{-6}	7.19×10^{-6}
^{106}Ru	6.26×10^{5}	1.14×10^{6}
^{124}Sb	6.47×10^{-5}	1.11×10^{-4}
^{125}Sb	2.35×10^{5}	4.30×10^{5}
^{126}Sb	3.93×10^{1}	7.21×10^{1}
^{126m}Sb	2.81×10^{2}	5.15×10^{2}
^{79}Se	2.91×10^{2}	5.39×10^{2}
^{145}Sm	9.93	1.97×10^{1}
^{147}Sm	1.29×10^{-2}	2.26×10^{-2}
^{151}Sm	5.95×10^{5}	1.08×10^{6}
^{119m}Sn	5.30×10^{2}	1.01×10^{3}
^{121m}Sn	7.97×10^{2}	1.11×10^{3}
^{123}Sn	2.53×10^{1}	4.54×10^{1}
^{125}Sn	—	—
^{126}Sn	2.81×10^{2}	5.15×10^{2}
^{89}Sr	7.33×10^{-3}	1.30×10^{-2}

Table 1.5.1-29. Total DOE SNF Radionuclide Inventory (Continued)

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	**Bounding Fuel Inventories (Ci)**
^{90}Sr	3.12×10^{7}	5.72×10^{7}
^{160}Tb	4.96×10^{-4}	8.95×10^{-4}
^{99}Tc	8.85×10^{3}	1.63×10^{4}
^{123m}Te	7.72×10^{-3}	1.21×10^{-2}
^{125m}Te	5.73×10^{4}	1.05×10^{5}
^{127}Te	1.81×10^{1}	3.25×10^{1}
^{127m}Te	1.84×10^{1}	3.31×10^{1}
^{129}Te	3.74×10^{-10}	6.61×10^{-10}
^{129m}Te	5.75×10^{-10}	1.01×10^{-9}
^{227}Th	4.92×10^{1}	9.99×10^{1}
^{228}Th	2.48×10^{4}	5.05×10^{4}
^{229}Th	3.35×10^{1}	6.86×10^{1}
^{230}Th	3.34	4.79
^{231}Th	1.62×10^{2}	2.67×10^{2}
^{232}Th	8.01	8.17
^{234}Th	4.95×10^{2}	9.41×10^{2}
^{206}Tl	8.56×10^{-6}	1.77×10^{-5}
^{207}Tl	4.97×10^{1}	1.01×10^{2}
^{208}Tl	8.93×10^{3}	1.82×10^{4}
^{232}U	2.42×10^{4}	4.92×10^{4}
^{233}U	1.82×10^{4}	2.21×10^{4}
^{234}U	7.25×10^{3}	1.02×10^{4}
^{235}U	1.46×10^{2}	2.16×10^{2}
^{236}U	2.83×10^{2}	4.98×10^{2}
^{237}U	4.16	7.59
^{238}U	7.77×10^{2}	7.89×10^{2}
^{131m}Xe	—	—
^{133}Xe	—	—

Table 1.5.1-29. Total DOE SNF Radionuclide Inventory (Continued)

Radionuclide	**2010**	
	Nominal Fuel Inventories (Ci)	**Bounding Fuel Inventories (Ci)**
^{90}Y	3.12×10^{7}	5.72×10^{7}
^{91}Y	2.71×10^{-1}	4.81×10^{-1}
^{65}Zn	5.97×10^{3}	1.07×10^{4}
^{93}Zr	1.68×10^{3}	2.82×10^{3}
^{95}Zr	1.82	3.23
TOTAL	1.91×10^{8}	3.48×10^{8}

NOTE: The DOE SNF inventory was originally estimated at 2010 and has not been decayed to the repository proposed operational date of 2020.

Source: DOE 2007, Table 3.

Table 1.5.1-30. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Naval SNF Canister

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
Naval SNF Waste Package System (DN)	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.	The canister transfer machine is required to be designed in accordance with the requirements of ASME NOG-1-2004 for Type I cranes. The canister transfer machine is required to be designed with the following features: • Two hoist upper travel limit switches • Hoist adjustable speed drive that stops the hoist at setpoints that are independent from the hoist upper travel limit switches • Load cell overload limit that stops hoist • Sensor to stop hoist when load clears canister transfer machine slide gate. The naval SNF canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the naval SNF canister capacity curve. (Note: The PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The naval SNF canister is required to be designed such that the maximum effective plastic strain from a drop meets the required reliability when evaluated against the naval SNF canister capacity curve.
				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.	The naval SNF canister is required to be designed such that the maximum effective plastic strain from a low speed impact or collision meets the required reliability when evaluated against the naval SNF canister capacity curve.

Table 1.5.1-30. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Naval SNF Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The naval SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The shield bell of the canister transfer machine is required to be designed such that the thermal penetration meets the required reliability when evaluated against the spectrum of fires. The naval SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
				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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The naval SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.1-30. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Naval SNF Canister (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		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.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.	The cask is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The naval SNF canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

NOTE: For casks, canisters, and associated handling equipment that were previously designed, the component design will be evaluated to confirm that the controlling parameters and values are met.
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.
Facility Codes: IHF: Initial Handling Facility; SB: Balance of Plant.
System Codes: DN: Naval Spent Nuclear Fuel Waste Package.

Table 1.5.1-31. Naval SNF Postclosure Nuclear Safety Design Basis

Barrier	Feature	SSC	Safety Classification	Barrier Function	Relevant Control Parameter Characteristics
EBS	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 Section 4.4 of Waste Acceptance System Requirements Document (DOE 2008b).
EBS	Waste Form and Waste Package Internals—Naval SNF Canister System Components	Naval SNF Baskets	ITWI	Reduces the probability of criticality.	See Section 4.4 of *Waste Acceptance System Requirements Document* (DOE 2008b).
		Naval SNF Basket Spacers			
		Naval Neutron Poison Assemblies (includes retention hardware)			
		Naval Control Rods (includes retention hardware)			
		Naval Corrosion Resistant Cans			
EBS	Waste Form and Waste Package Internals—Naval SNF	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 Section 4.4 of *Waste Acceptance System Requirements Document* (DOE 2008b) and the *Integrated Interface Control Document* (DOE 2008d, Section 10.3.2.2).

Table 1.5.1-32. Radionuclide Inventory for a Representative Naval SNF Canister 5 Years after Reactor Shutdown

Isotope	Activity (curies)
^{227}Ac	2.12×10^{-4}
^{241}Am	3.56×10^{1}
^{242}Am	3.84×10^{-1}
^{242m}Am	3.86×10^{-1}
^{243}Am	4.66×10^{-1}
^{137m}Ba	2.93×10^{5}
^{14}C	6.40×10^{0}
^{113m}Cd	2.33×10^{1}
^{144}Ce	1.47×10^{4}
^{249}Cf	1.04×10^{-6}
^{251}Cf	7.15×10^{-8}
^{252}Cf	8.08×10^{-6}
^{36}Cl	1.36×10^{-1}
^{242}Cm	9.70×10^{-1}
^{243}Cm	4.68×10^{-1}
^{244}Cm	4.40×10^{1}
^{245}Cm	3.85×10^{-3}
^{246}Cm	1.20×10^{-3}
^{247}Cm	1.54×10^{-8}
^{248}Cm	6.50×10^{-8}
^{60}Co	1.18×10^{3}
^{134}Cs	4.95×10^{4}
^{135}Cs	3.68×10^{0}
^{137}Cs	3.11×10^{5}
^{152}Eu	3.71×10^{1}
^{154}Eu	7.17×10^{3}
^{155}Eu	2.12×10^{3}
^{55}Fe	1.68×10^{3}

Table 1.5.1-32. Radionuclide Inventory for a Representative Naval SNF Canister 5 Years after Reactor Shutdown (Continued)

Isotope	Activity (curies)
^{3}H	1.15×10^{3}
^{129}I	8.03×10^{-2}
^{85}Kr	2.41×10^{4}
^{93m}Nb	2.27×10^{3}
^{94}Nb	2.06×10^{2}
^{59}Ni	1.34×10^{1}
^{63}Ni	1.63×10^{3}
^{236}Np	4.92×10^{-5}
^{237}Np	1.17×10^{0}
^{238}Np	1.74×10^{-3}
^{239}Np	4.66×10^{-1}
^{231}Pa	7.77×10^{-4}
^{210}Pb	2.97×10^{-6}
^{107}Pd	4.42×10^{-2}
^{147}Pm	9.20×10^{4}
^{144}Pr	1.47×10^{4}
^{236}Pu	6.33×10^{-1}
^{237}Pu	1.84×10^{-7}
^{238}Pu	7.80×10^{3}
^{239}Pu	9.87×10^{0}
^{240}Pu	1.04×10^{1}
^{241}Pu	2.56×10^{3}
^{242}Pu	5.65×10^{-2}
^{244}Pu	6.72×10^{-9}
^{226}Ra	1.50×10^{-5}
^{228}Ra	9.03×10^{-10}
^{102}Rh	1.12×10^{-2}
^{106}Rh	3.20×10^{3}
^{106}Ru	3.20×10^{3}

Table 1.5.1-32. Radionuclide Inventory for a Representative Naval SNF Canister 5 Years after Reactor Shutdown (Continued)

Isotope	Activity (curies)
^{125}Sb	4.13×10^{3}
^{126}Sb	1.34×10^{-1}
^{126m}Sb	9.55×10^{-1}
^{79}Se	2.67×10^{-1}
^{147}Sm	2.48×10^{-5}
^{151}Sm	9.78×10^{2}
^{121m}Sn	2.58×10^{1}
^{126}Sn	9.55×10^{-1}
^{90}Sr	3.05×10^{5}
^{99}Tc	5.11×10^{1}
^{125m}Te	1.01×10^{3}
^{229}Th	2.14×10^{-5}
^{230}Th	3.22×10^{-3}
^{232}Th	1.19×10^{-5}
^{208}Tl	8.76×10^{-2}
^{232}U	5.29×10^{-1}
^{233}U	6.52×10^{-2}
^{234}U	1.86×10^{1}
^{235}U	2.65×10^{-1}
^{236}U	1.84×10^{0}
^{237}U	6.13×10^{-2}
^{238}U	9.20×10^{-4}
^{90}Y	3.05×10^{5}
^{93}Zr	8.69×10^{0}

Source: Naval Nuclear Propulsion Program Technical Support Document, Section 2.3.7.

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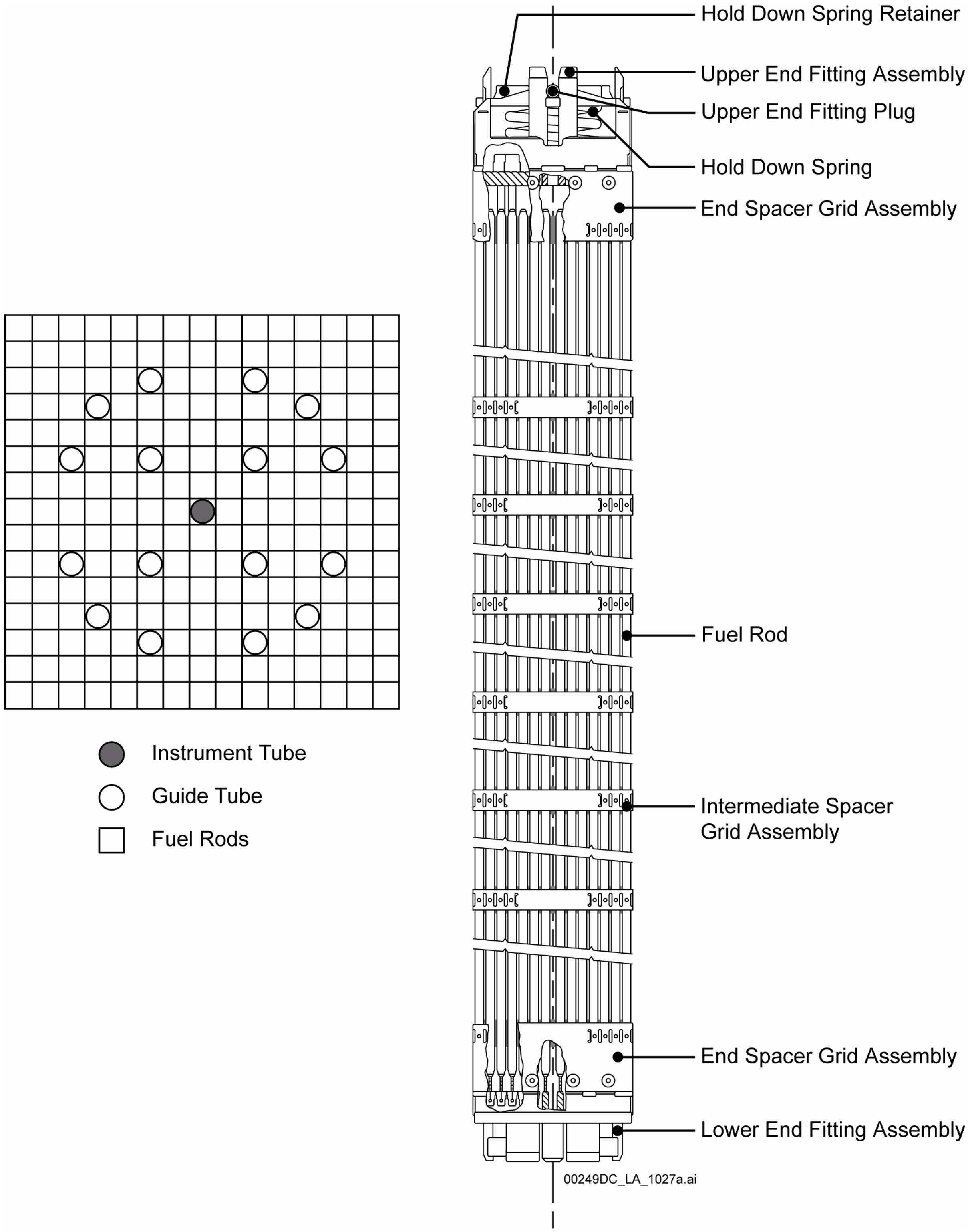

Figure 1.5.1-1. Representative 15 × 15 Pressurized Water Reactor Fuel Assembly

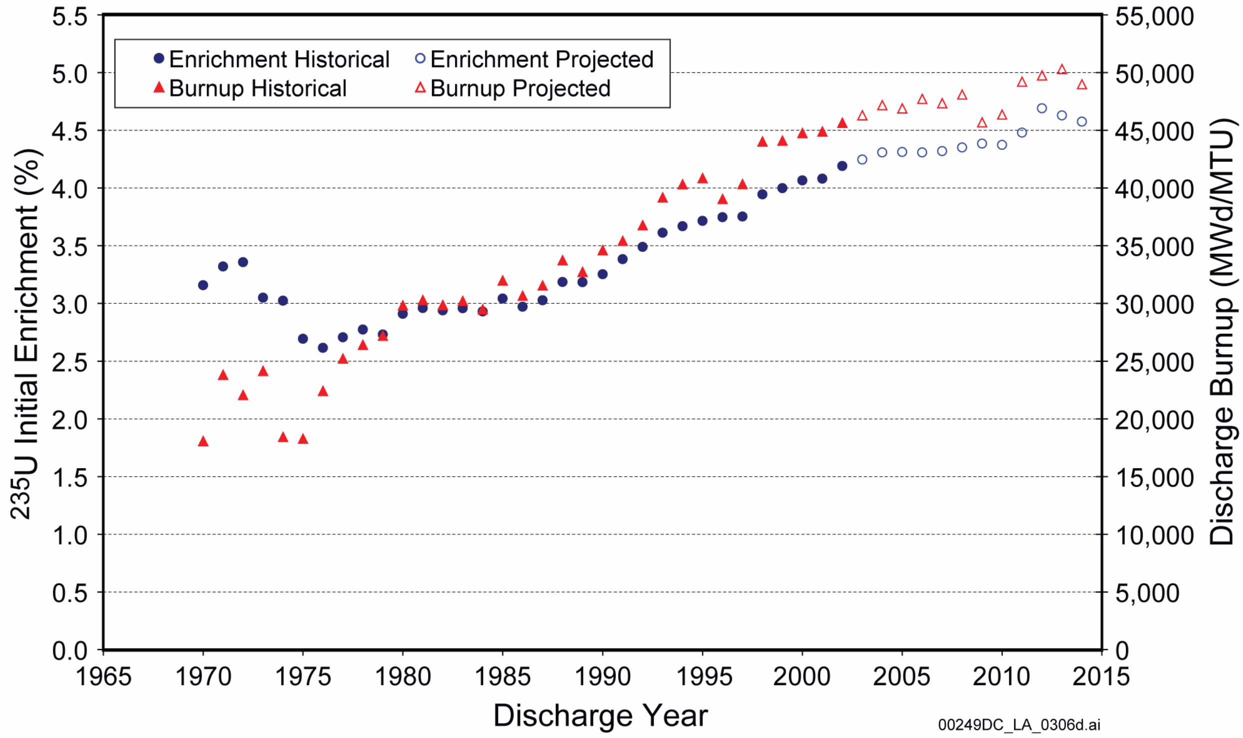

Figure 1.5.1-2. Initial Enrichment and Discharge Burnup Evolution for Pressurized Water Reactors as of December 31, 2002, and Projections for the Next Five Cycles

NOTE: This figure presents the average initial ^{235}U enrichment and average discharge burnup for all assemblies permanently discharged during a calendar year.

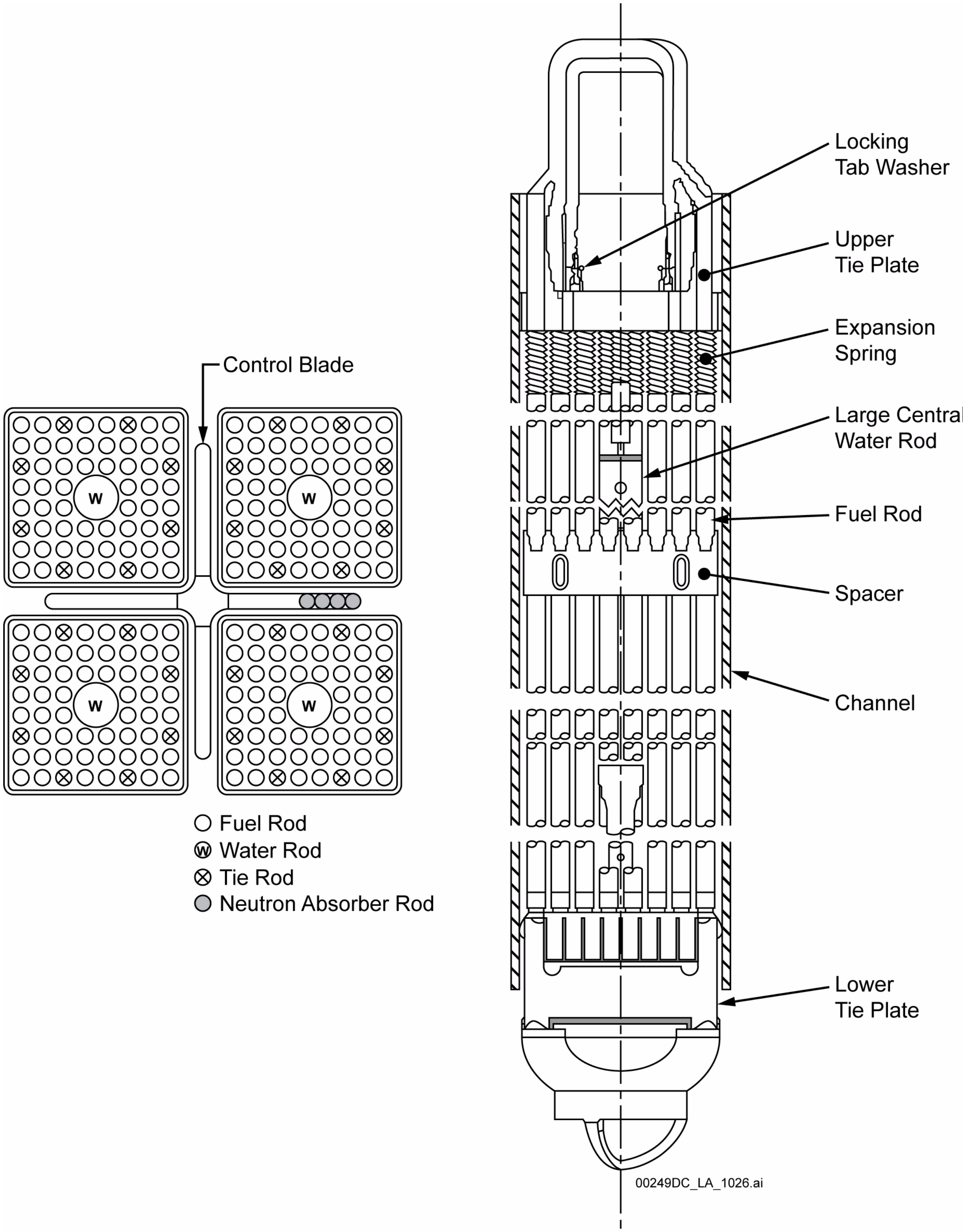

Figure 1.5.1-3. Representative 8 × 8 Boiling Water Reactor Fuel Assembly

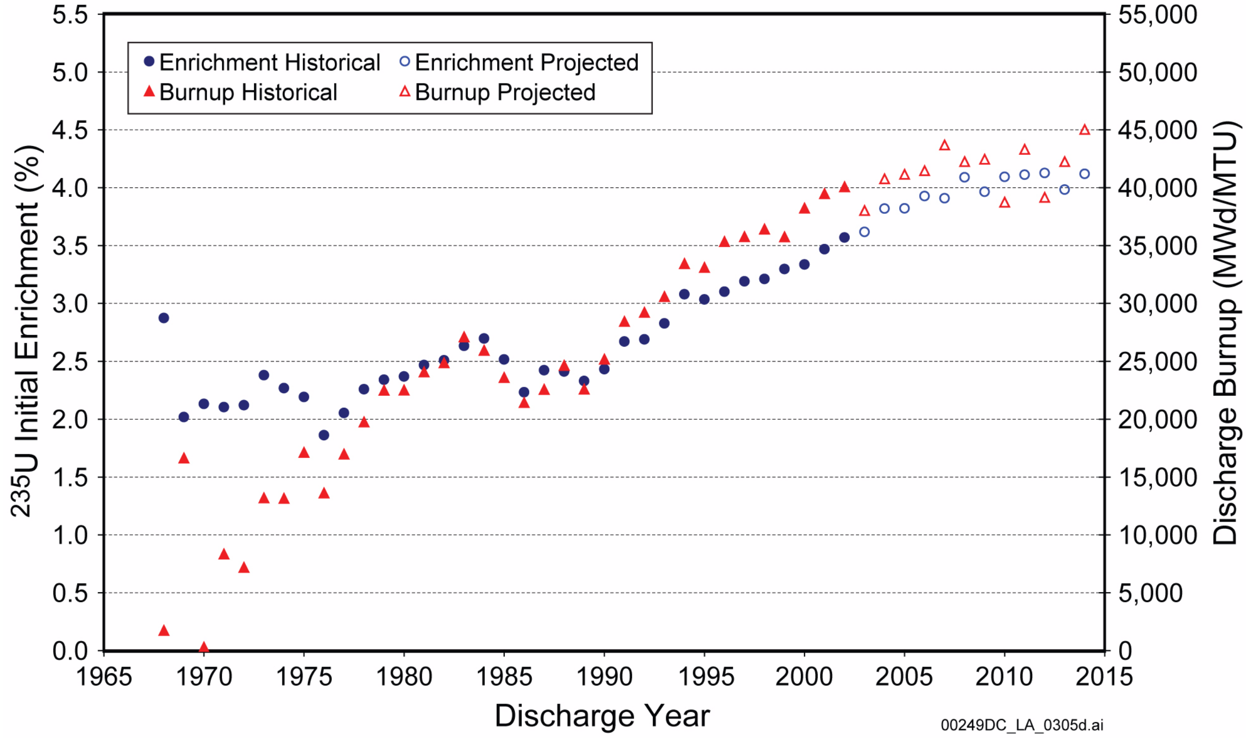

Figure 1.5.1-4. Initial Enrichment and Discharge Burnup Evolution for Boiling Water Reactors as of December 31, 2002, and Projections for the Next Five Cycles

NOTE: This figure presents the average initial ^{235}U enrichment and average discharge burnup for all assemblies permanently discharged during a calendar year.

Figure 1.5.1-5. Transportation, Aging, and Disposal Canister (Conceptual Representation)

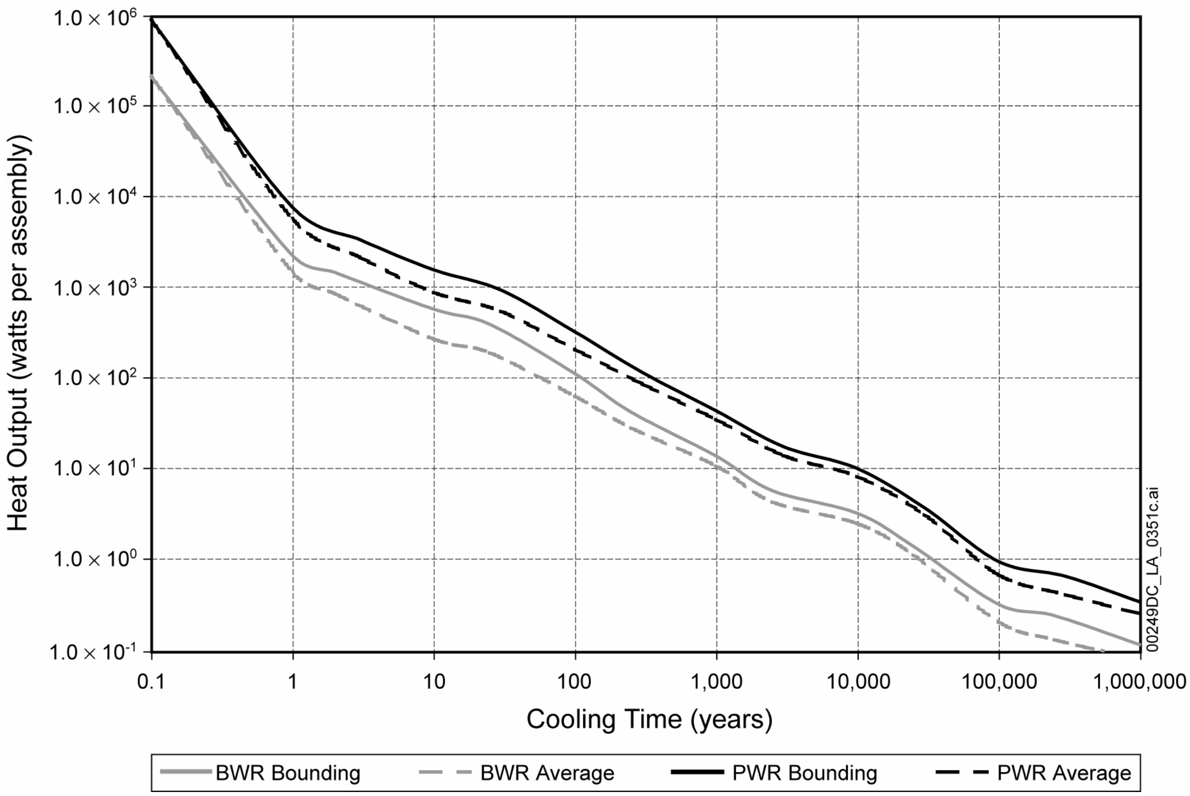

Figure 1.5.1-6. Thermal Power after Discharge: Comparison of Pressurized Water Reactor and Boiling Water Reactor

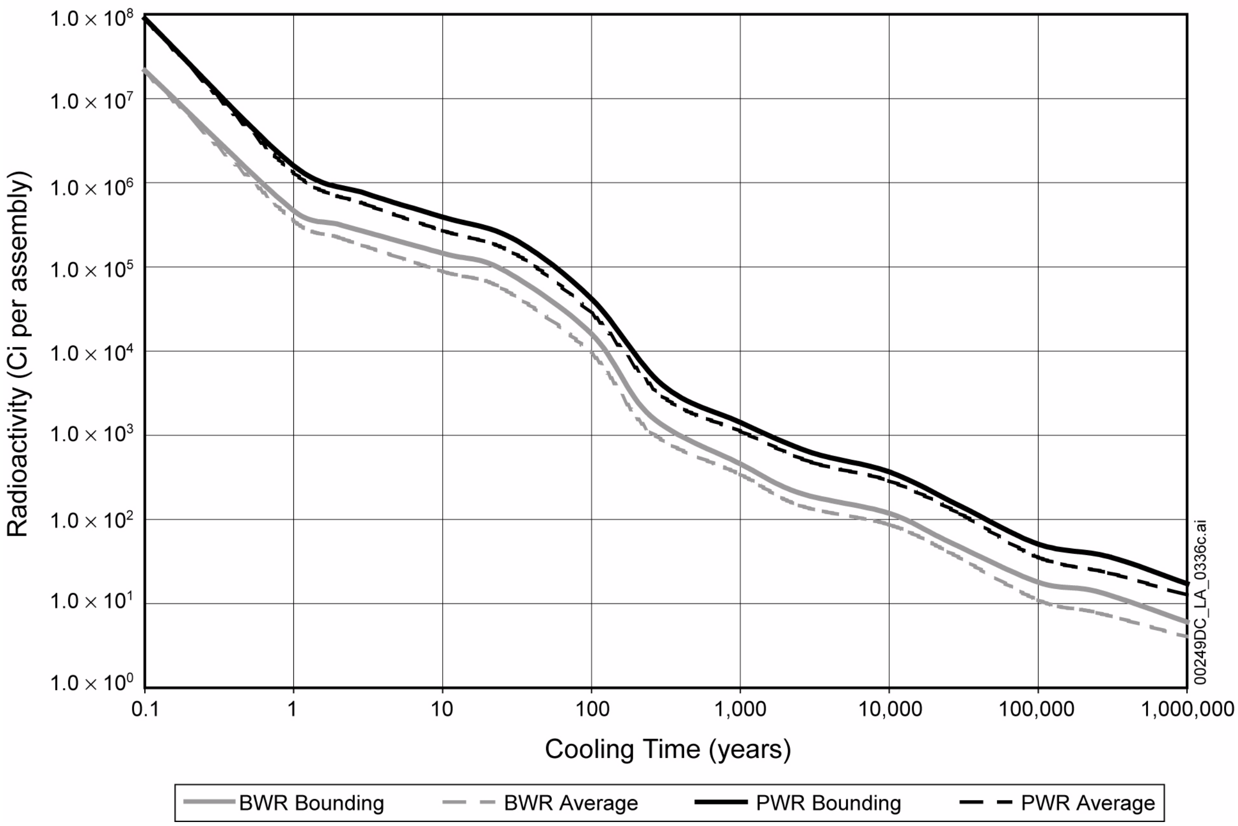

Figure 1.5.1-7. Radioactivity after Discharge: Comparison of Pressurized Water Reactor and Boiling Water Reactor

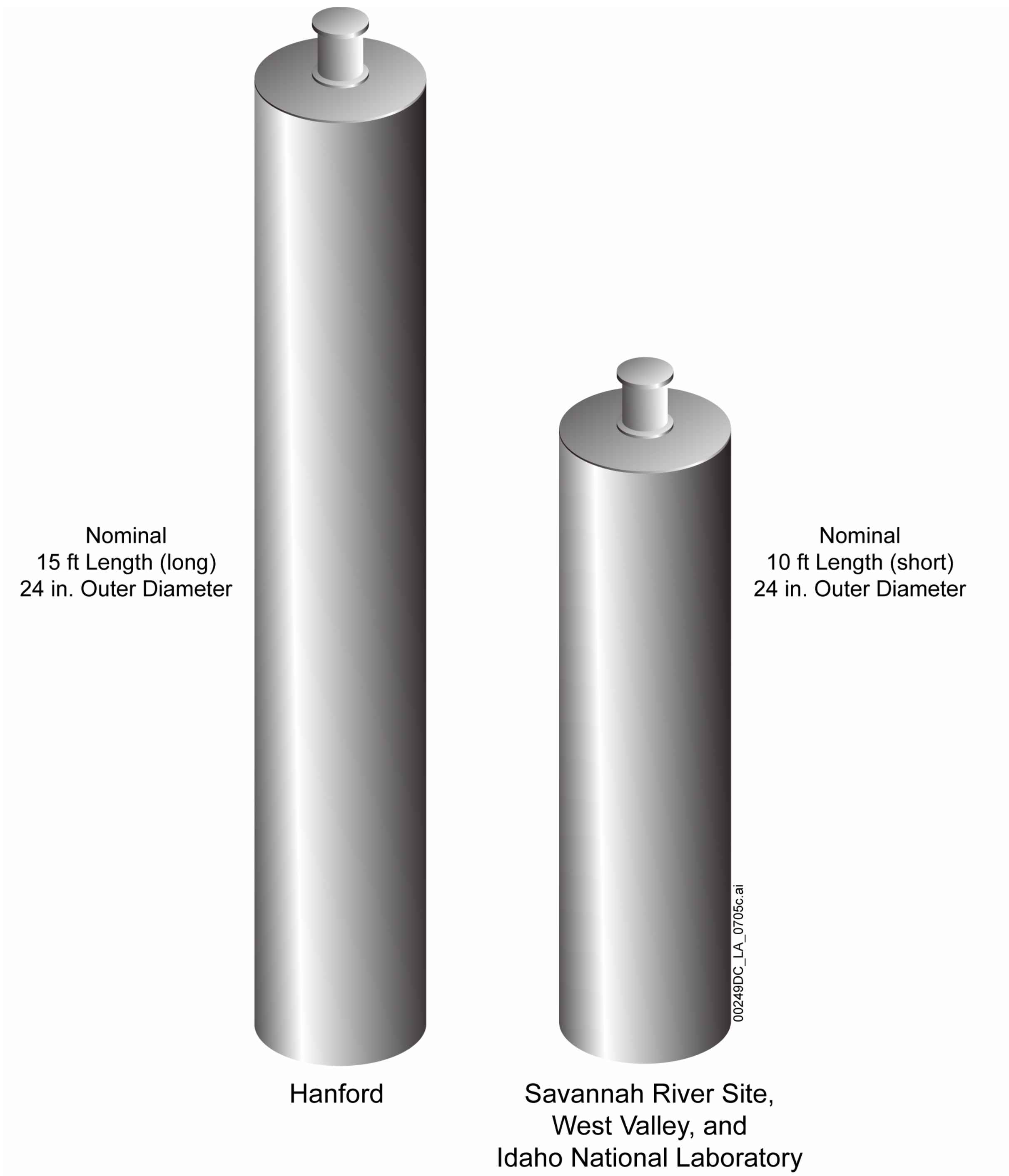

Figure 1.5.1-8. High-Level Radioactive Waste Standardized Canisters

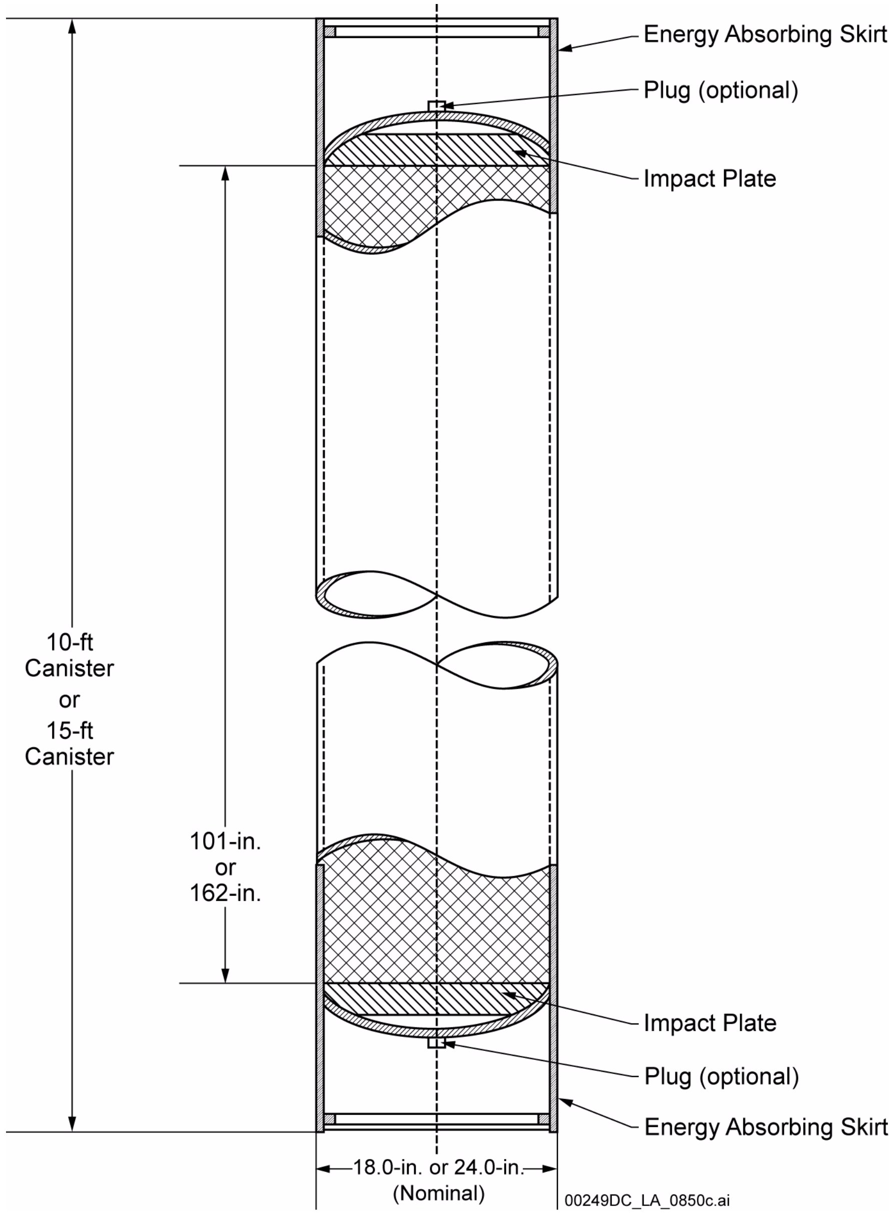

Figure 1.5.1-9. Standardized DOE SNF Canister

Source: DOE 2004b, Figure 3 and Section 2.1.

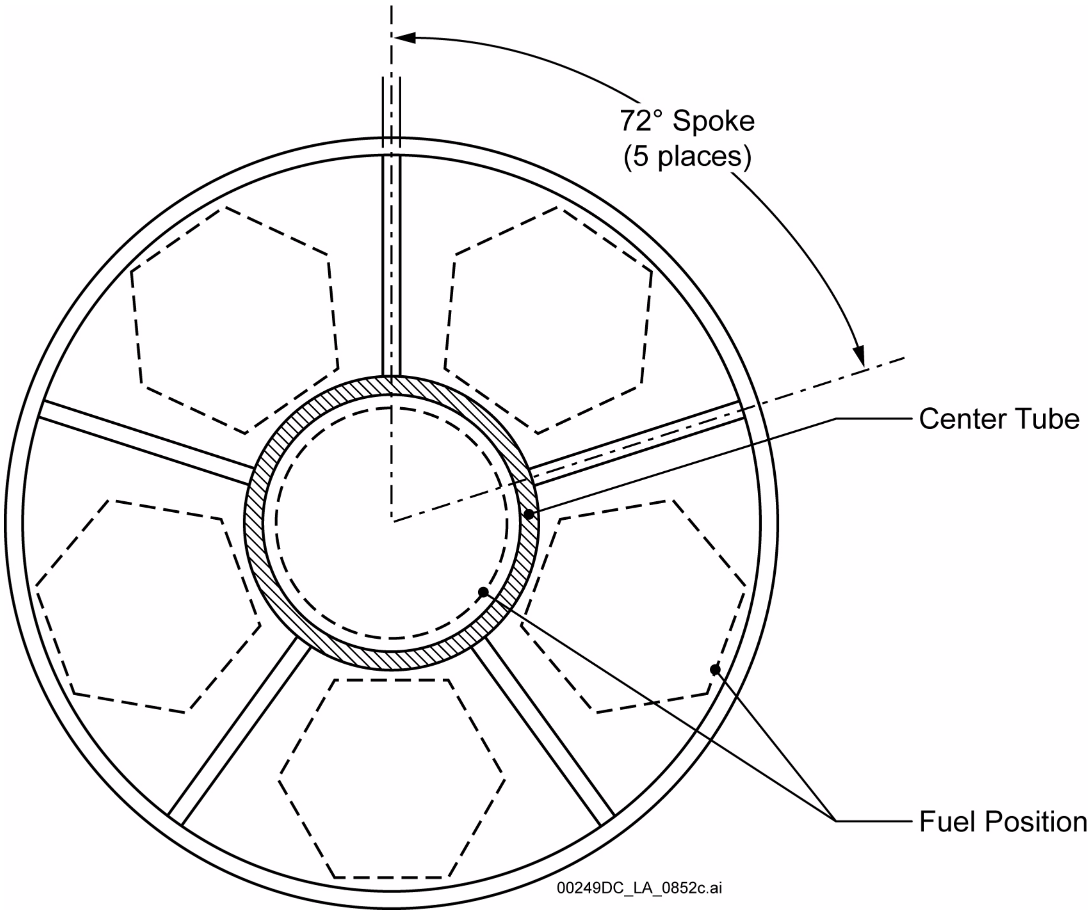

Figure 1.5.1-10. Cross-Sectional Layout for a FFTF-MOX Basket

NOTE: FFTF = Fast Flux Test Facility, MOX = mixed oxide.

Source: DOE 2004b, Figure 8.

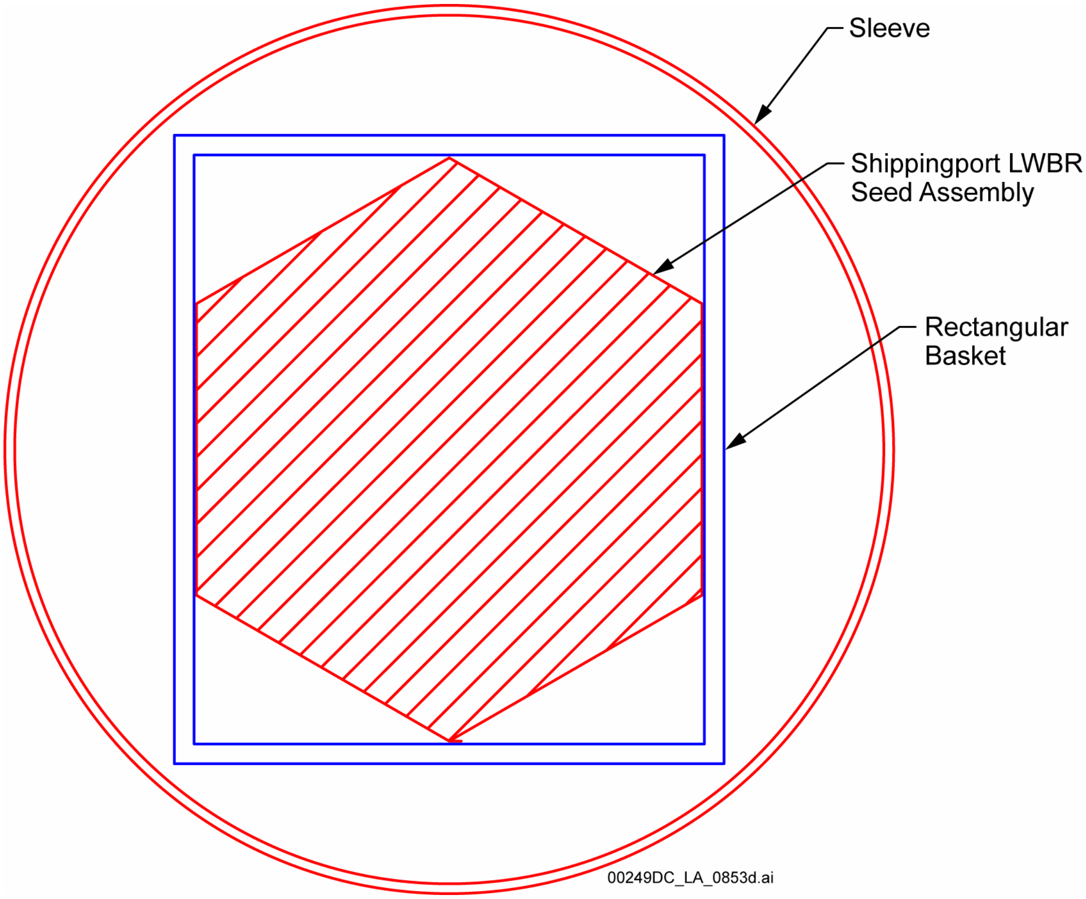

Figure 1.5.1-11. Cross-Sectional Layout for a Shippingport LWBR Fuel Basket

NOTE: LWBR = Light Water Breeder Reactor.

Source: DOE 2006, Figure 4.

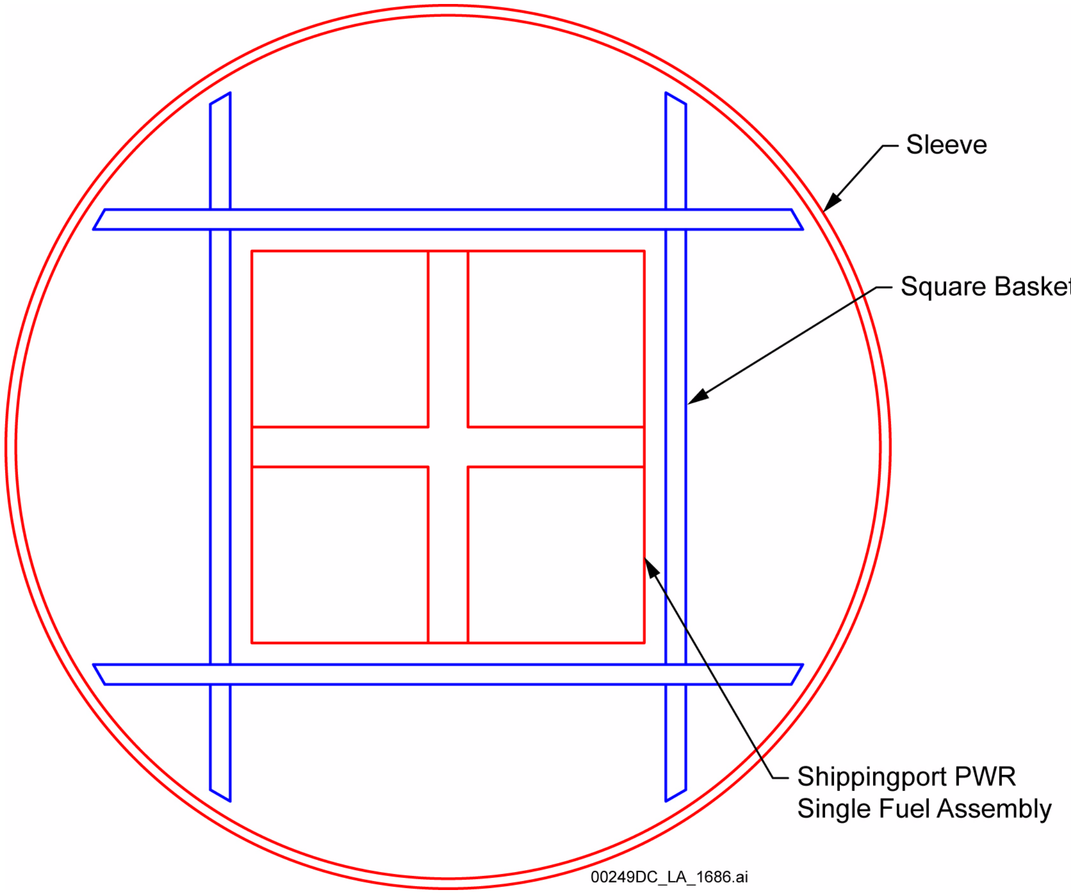

Figure 1.5.1-12. Cross-Sectional Layout for a Shippingport PWR Core 2 Fuel Basket

Source: DOE 2006, Figure 5.

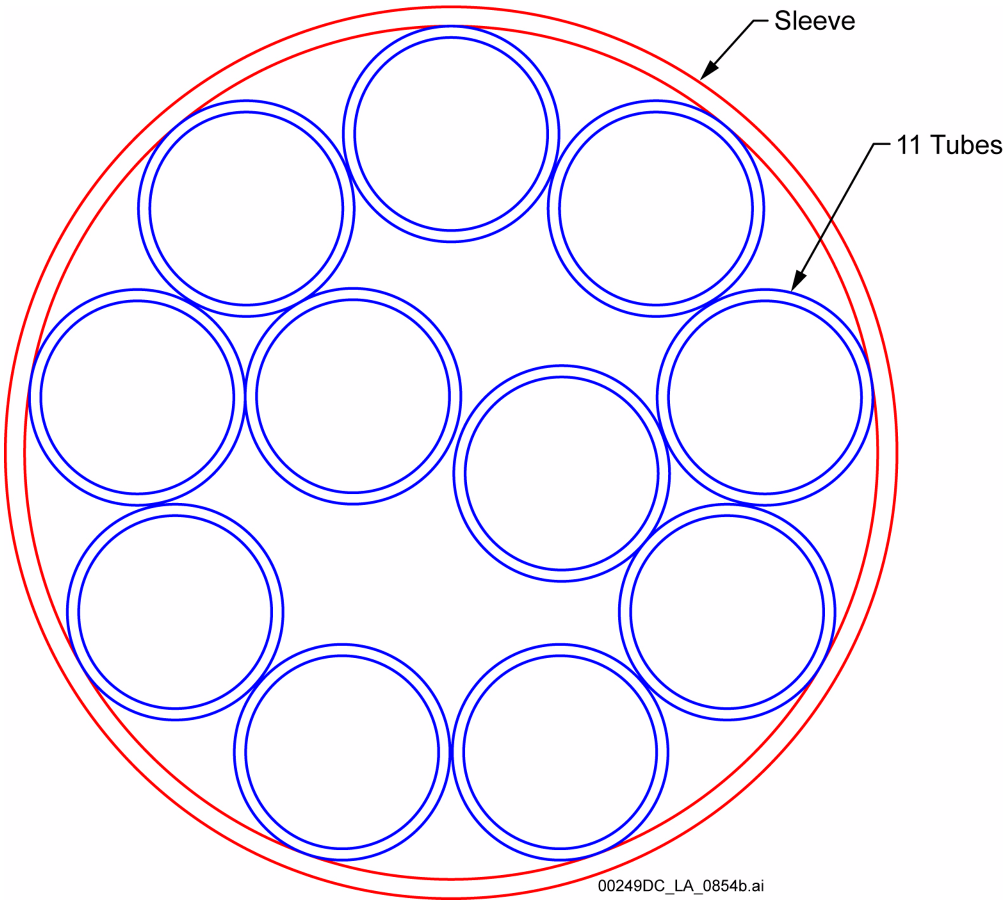

Figure 1.5.1-13. Cross-Sectional Layout for a Fermi Fuel Basket

Source: DOE 2006, Figure 6.

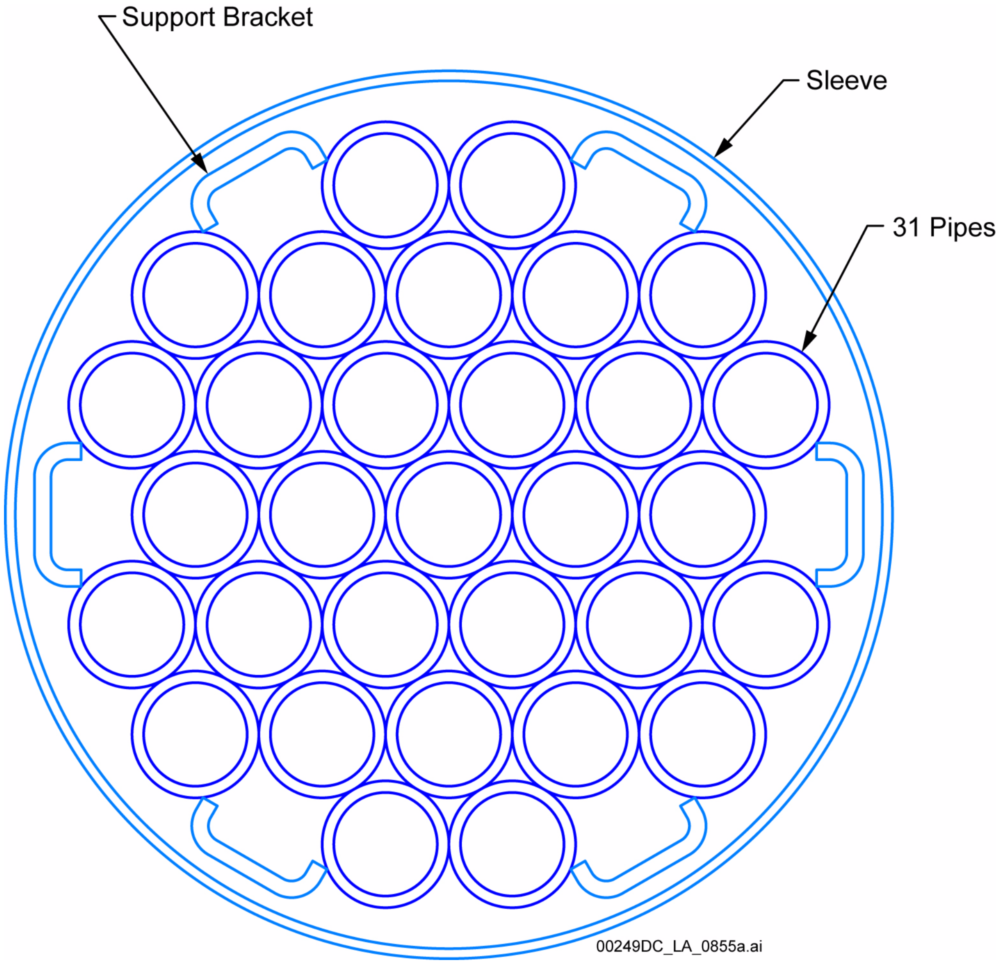

Figure 1.5.1-14. Cross-Sectional Layout for a TRIGA Fuel Basket

Source: DOE 2006, Figure 7.

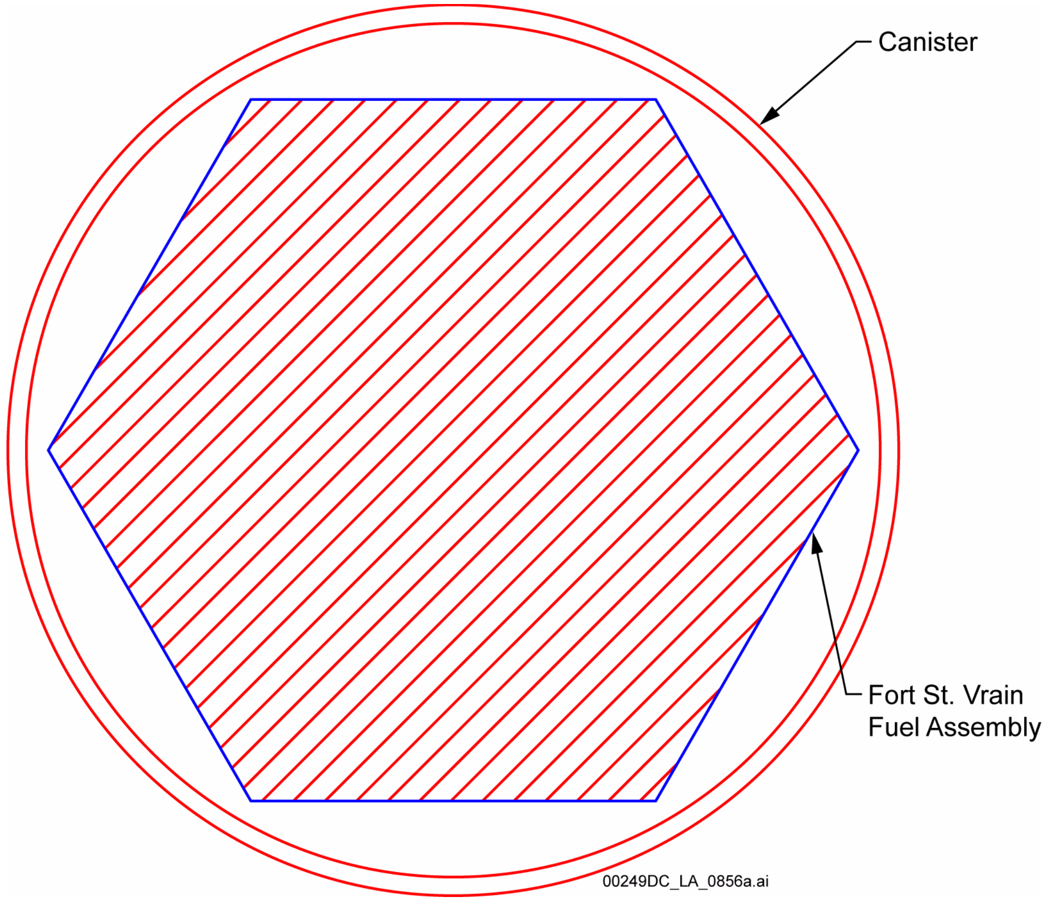

Figure 1.5.1-15. Cross-Sectional Layout for a Fort St. Vrain Fuel Basket

Source: DOE 2006, Figure 8.

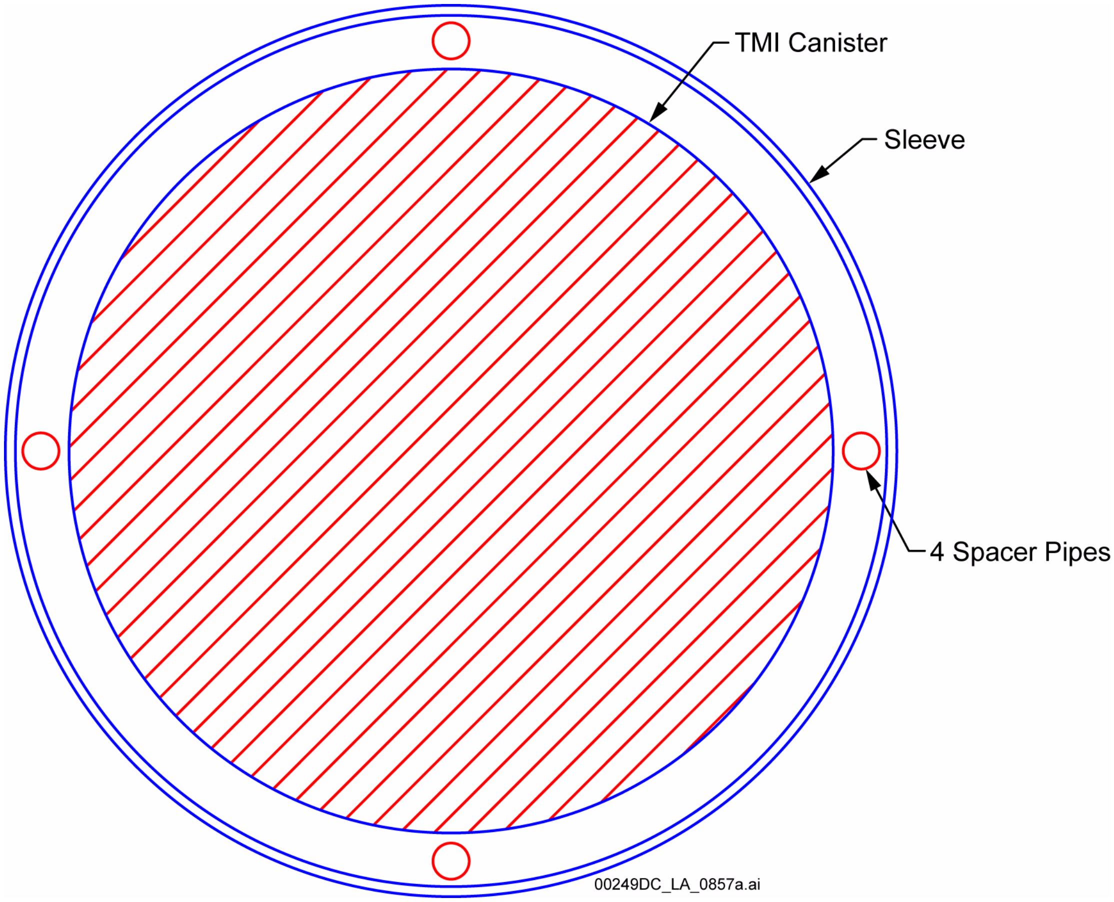

Figure 1.5.1-16. Cross-Sectional Layout for a Three Mile Island Unit 2 Canister Basket

Source: DOE 2006, Figure 9.

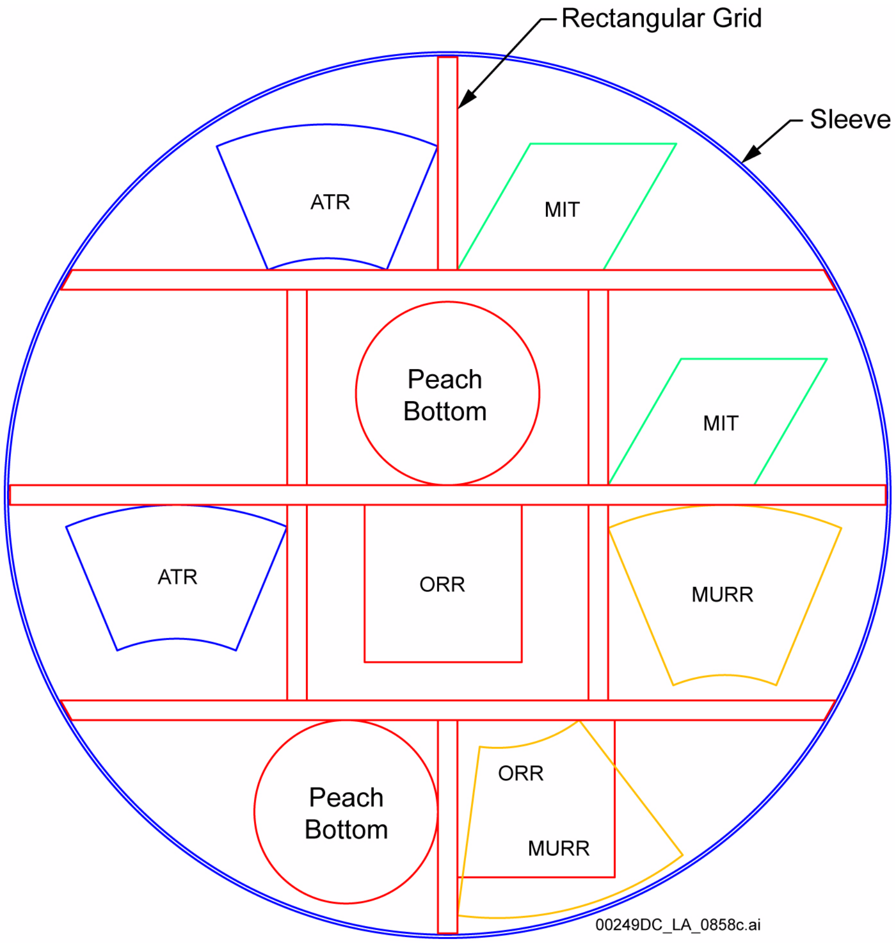

Figure 1.5.1-17. Cross-Sectional Layout for an Aluminum Fuels Basket

NOTE: All aluminum fuels can fit into any part of the basket.
ATR = Advanced Test Reactor; MIT = Massachusetts Institute of Technology; MURR = University of Missouri Research Reactor; ORR = Oak Ridge Research Reactor.

Source: DOE 2006, Figure 10.

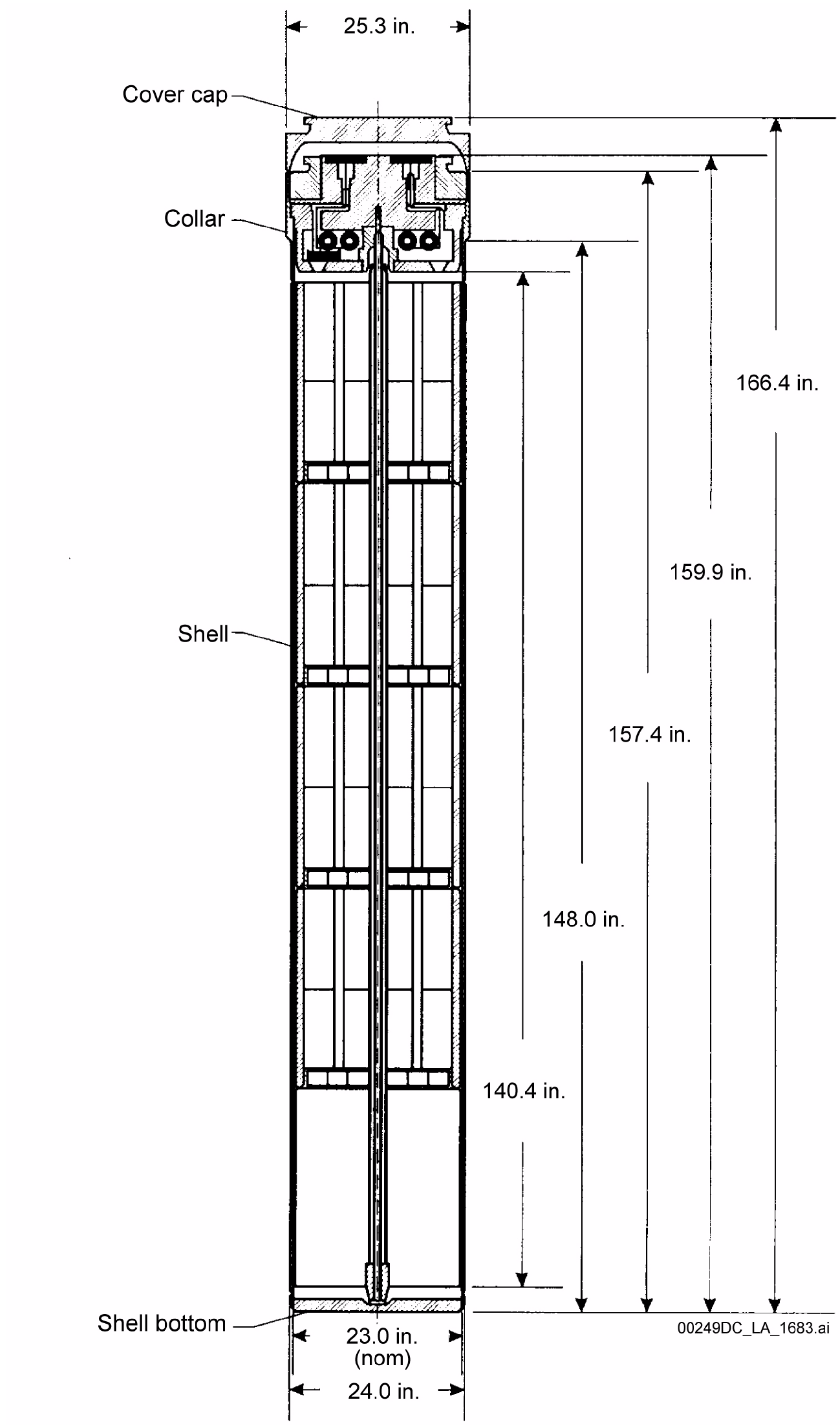

Figure 1.5.1-18. Multicanister Overpack

NOTE: The nominal value for the MCO length may be slightly shorter due to weld shrinkage. Likewise, the MCO head diameter value may be slightly wider due to the buildup of the closure weld. Source: DOE 2008d, Figure C-5, Notes 5 and 6.

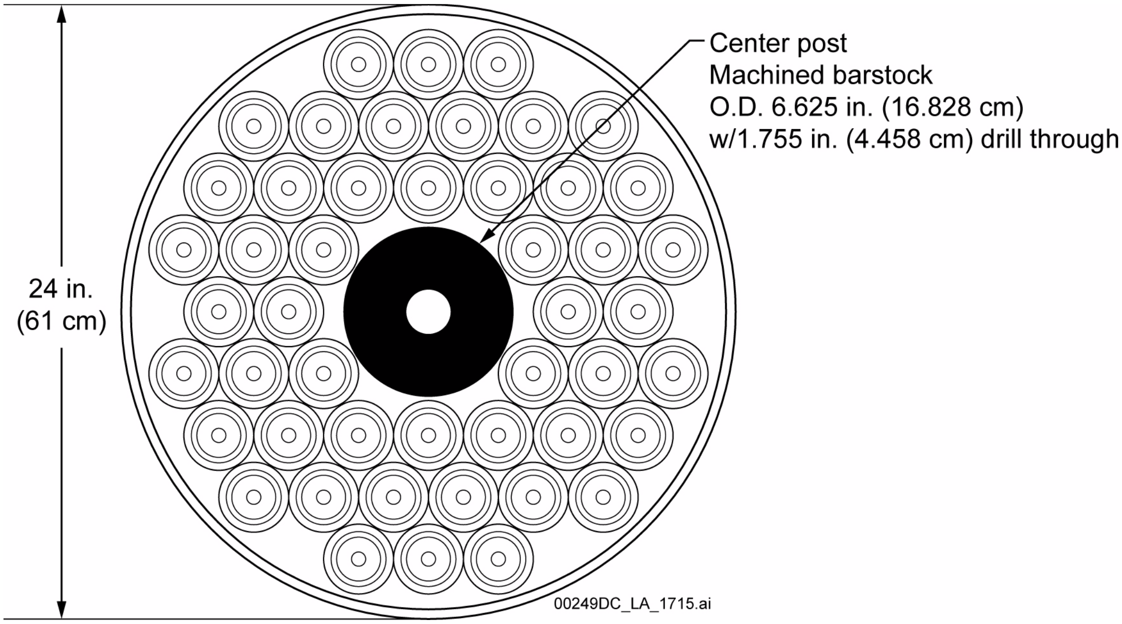

Figure 1.5.1-19. Cross-Sectional Layout for Mark IA Fuel Basket

NOTE: O.D. = outer diameter.

Source: DOE 2004b, Figure 6.

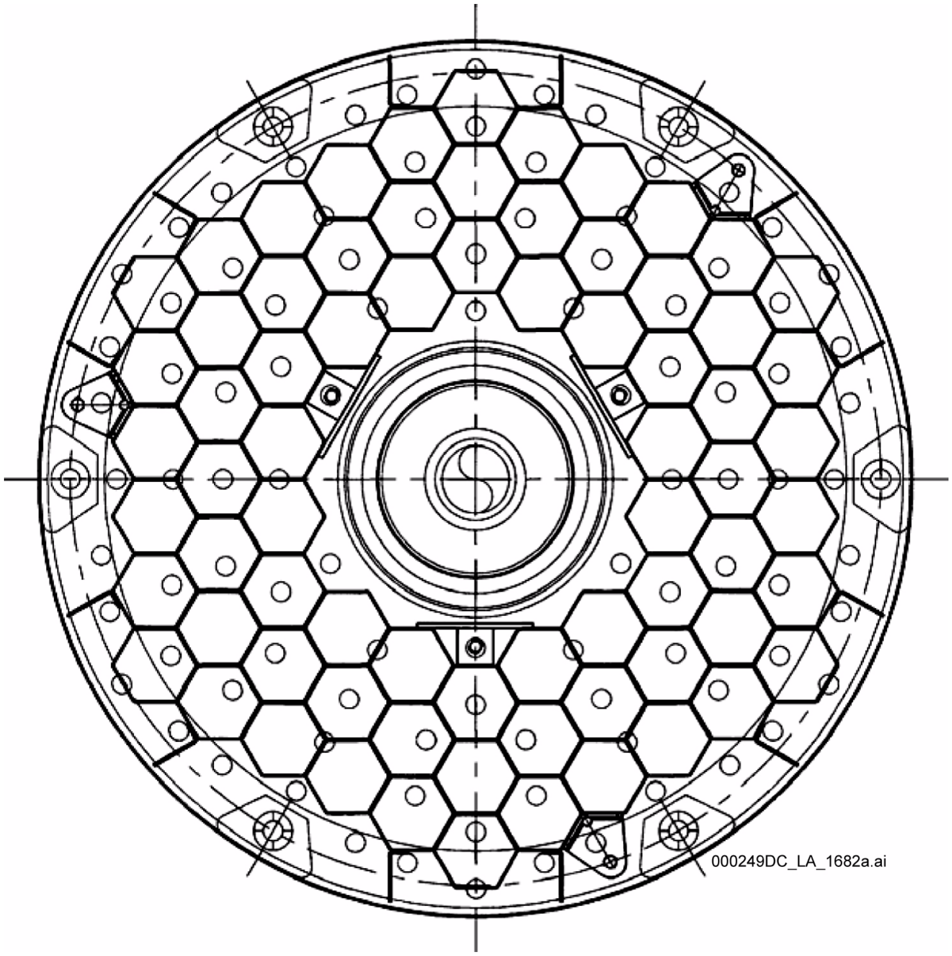

Figure 1.5.1-20. Cross-Sectional Layout for a Single Pass Reactor Basket

Source: DOE 2002.

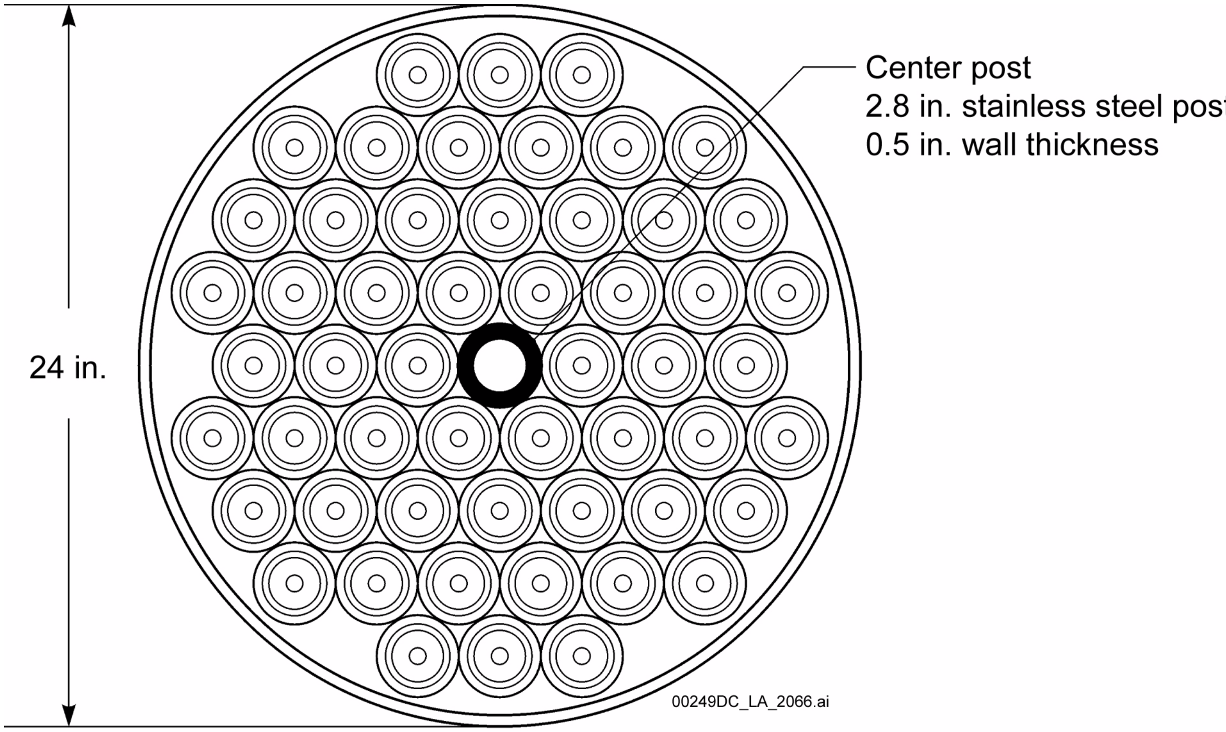

Figure 1.5.1-21. Cross-Sectional Layout for Mark IV Fuel Basket

Source: DOE 2004b, Figure 7.

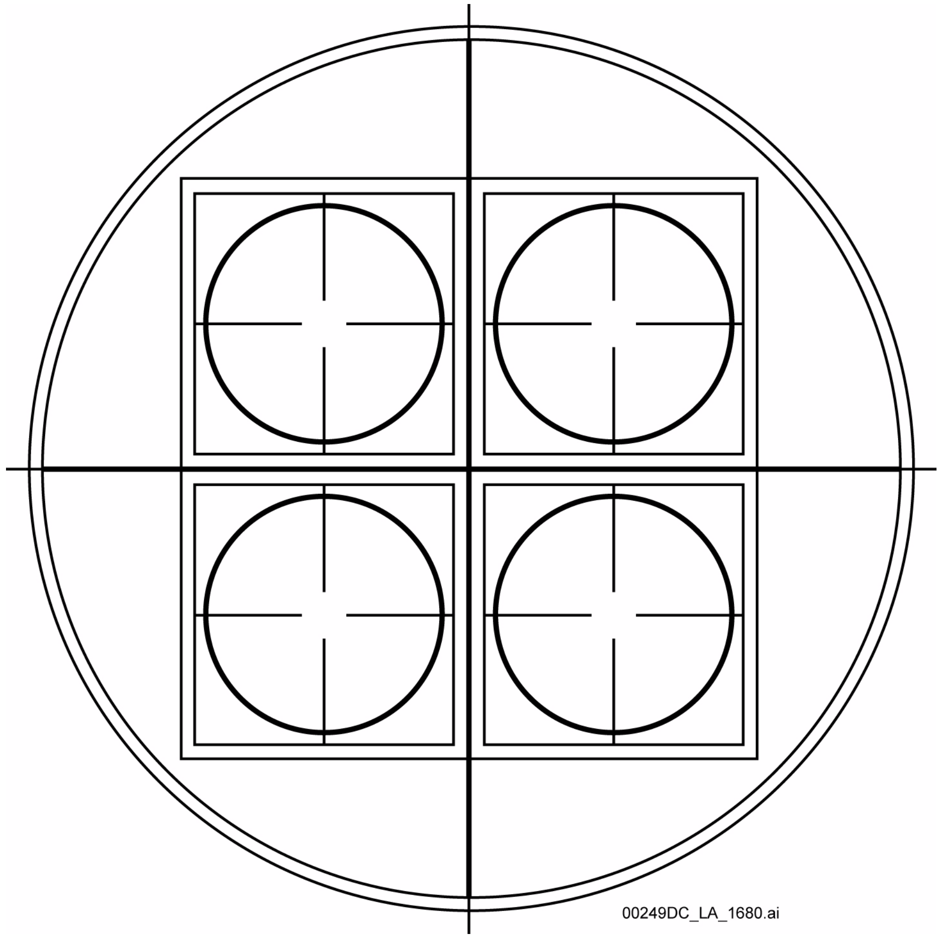

Figure 1.5.1-22. Cross-Sectional Layout for Shippingport PWR Core 2 Multicanister Overpack Blanket Insert

Source: Fluor Hanford 2000, Figure 1.

Figure 1.5.1-23. Deformed Shape of Standardized Canister for 30-ft Center-of-Gravity over Corner Drop (Lower End-Side View)

Source: Morton et al. 2002, Part II, p. 86, Figure 18.

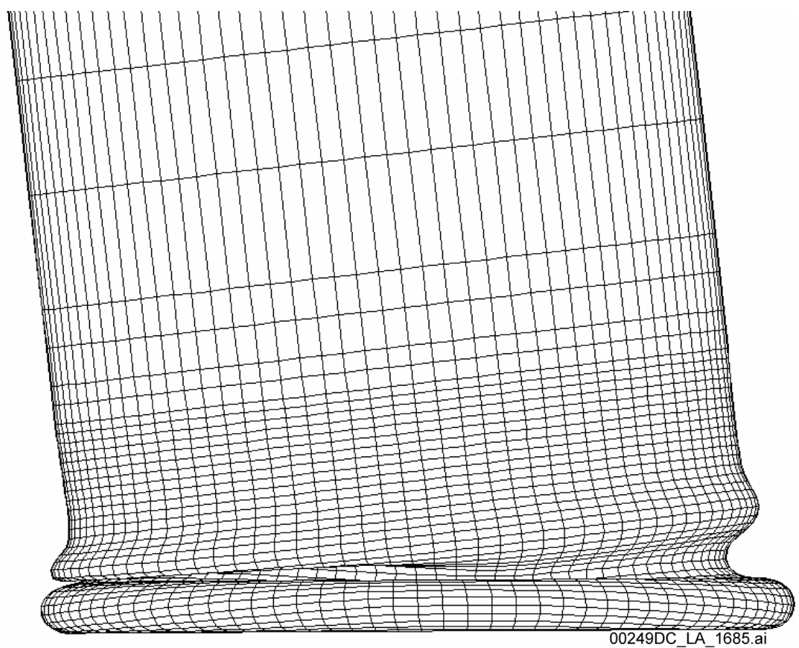

Figure 1.5.1-24. Deformed Shape of Standardized Canister for 30-ft Center-of-Gravity over Corner Drop (Lower End-Side View)

Source: Morton et al. 2002, Part II, p. 86, Figure 19.

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Figure 1.5.1-25. Deformed Shape of Standardized Canister for 45° Drop from 30 ft (Lower End-End View)

Source: Morton et al. 2002, Part II, p. 91, Figure 28.

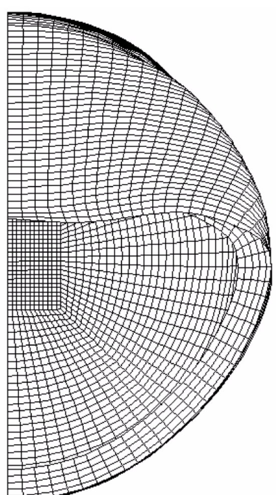

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Figure 1.5.1-26. Deformed Shape of Standardized Canister for 45° Drop from 30 ft (Lower End-End View)

Source: Morton et al. 2002, Part II, p. 91, Figure 29.

00249DC_LA_0861.ai

Figure 1.5.1-27. Deformed Shape of Standardized Canister for 40-in. Drop onto a 6-in. Post (Isometric View)

Source: Morton et al. 2002, Part II, p. 106, Figure 58.

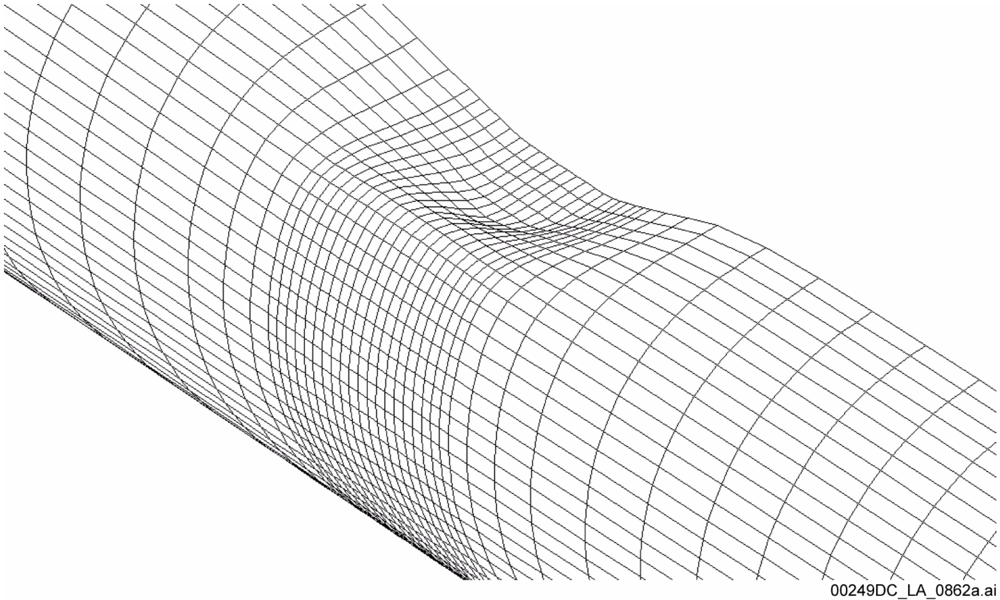

Figure 1.5.1-28. Deformed Shape of Standardized Canister for 40-in. Drop onto a 6-in. Post (Isometric View)

Source: Morton et al. 2002, Part II, p. 106, Figure 59.

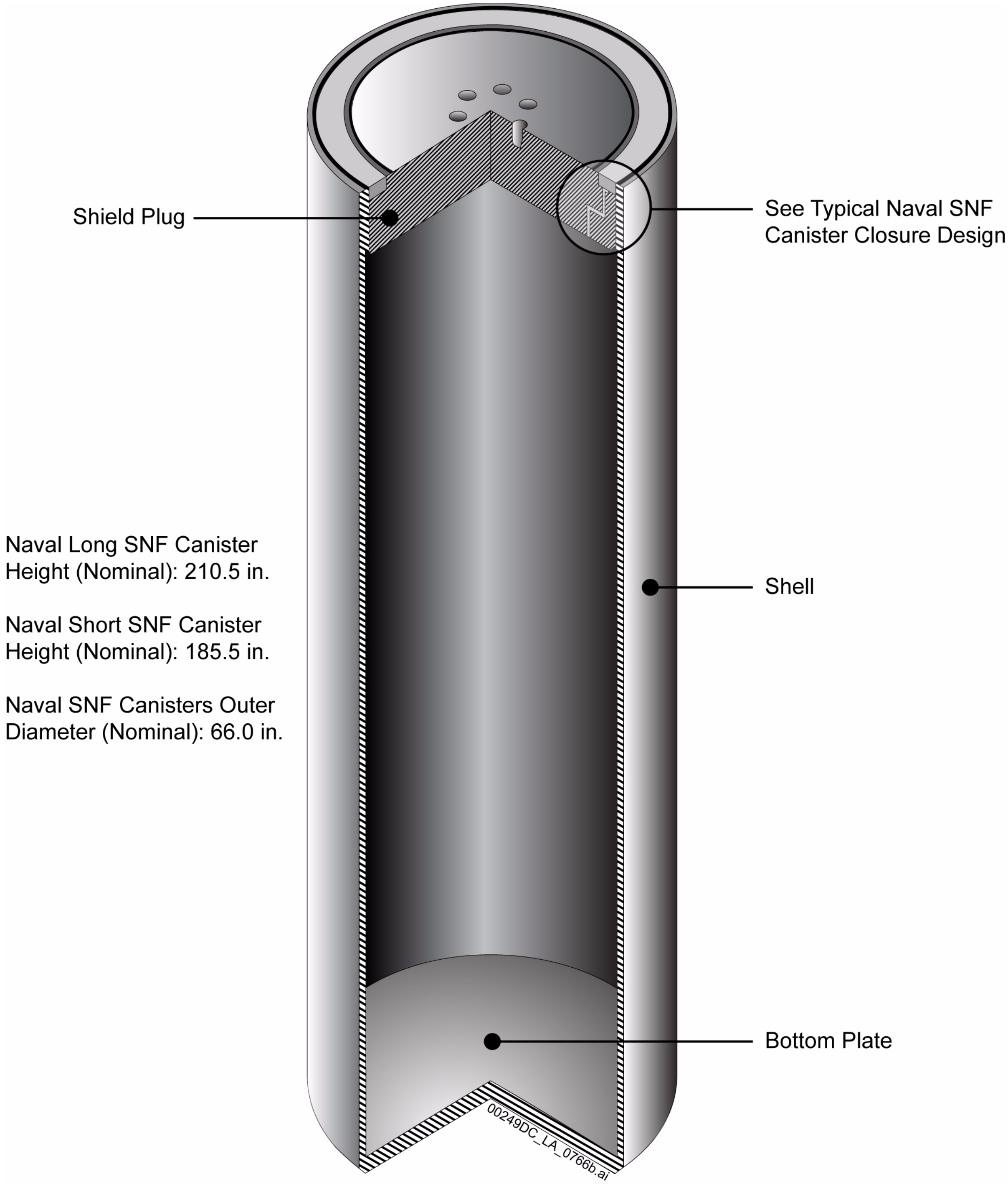

Figure 1.5.1-29. Typical Naval Spent Nuclear Fuel Canister

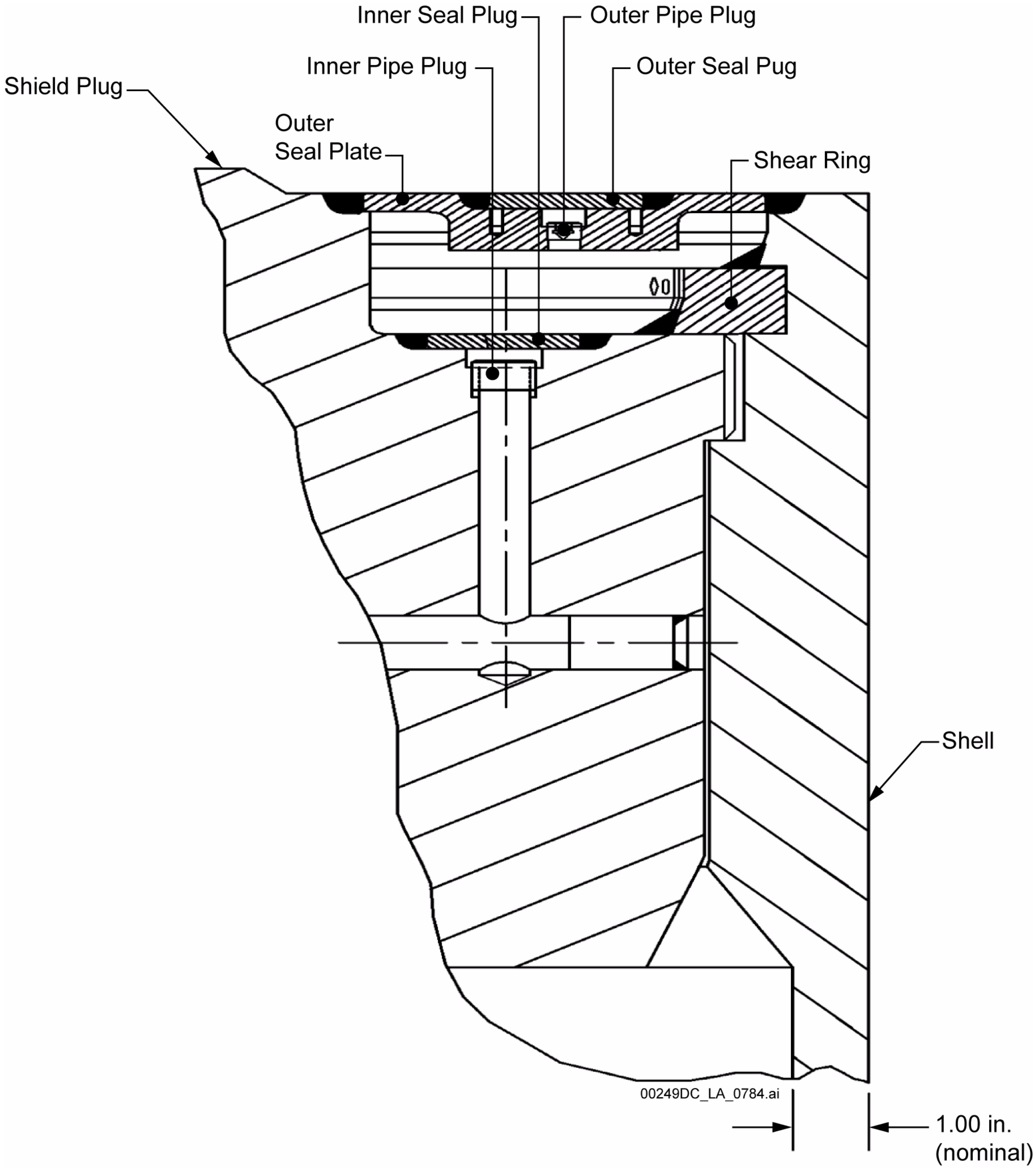

Figure 1.5.1-30. Typical Naval Spent Nuclear Fuel Canister Closure Design

NOTE: Detail from typical naval SNF canister figure.

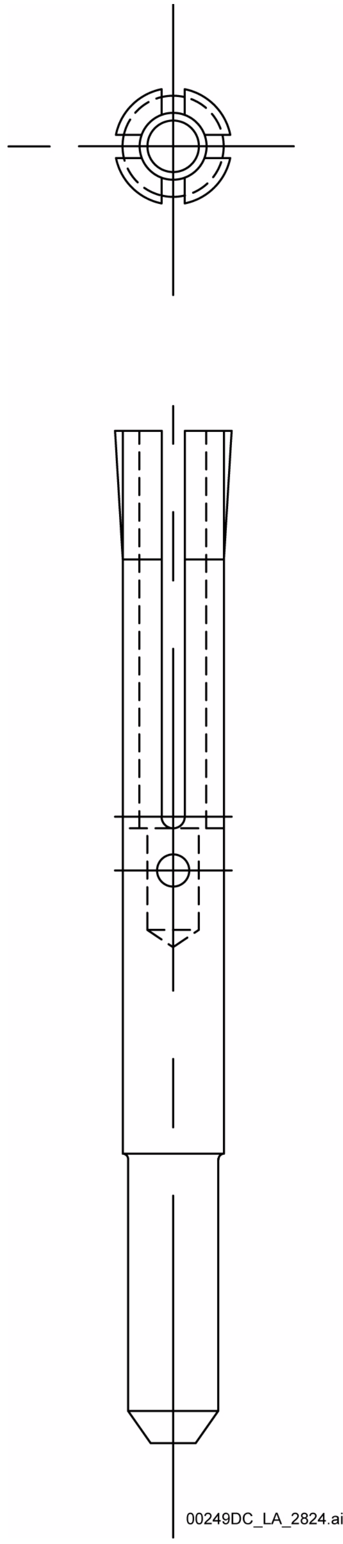

Figure 1.5.1-31. Typical Control Rod Retention Pin

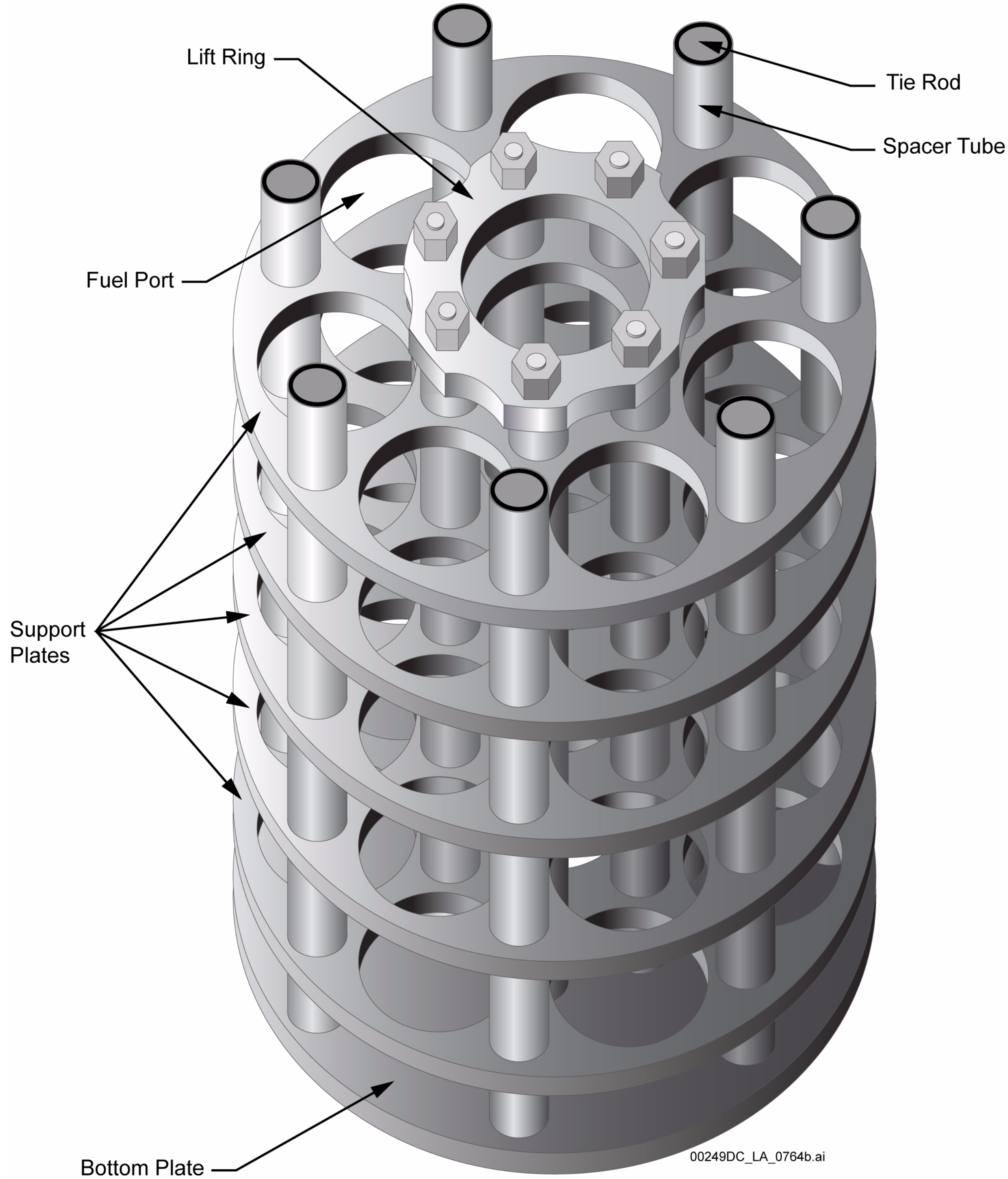

Figure 1.5.1-32. Typical Packaging Method A Naval Spent Nuclear Fuel Basket

NOTE: Size and number of fuel ports vary.

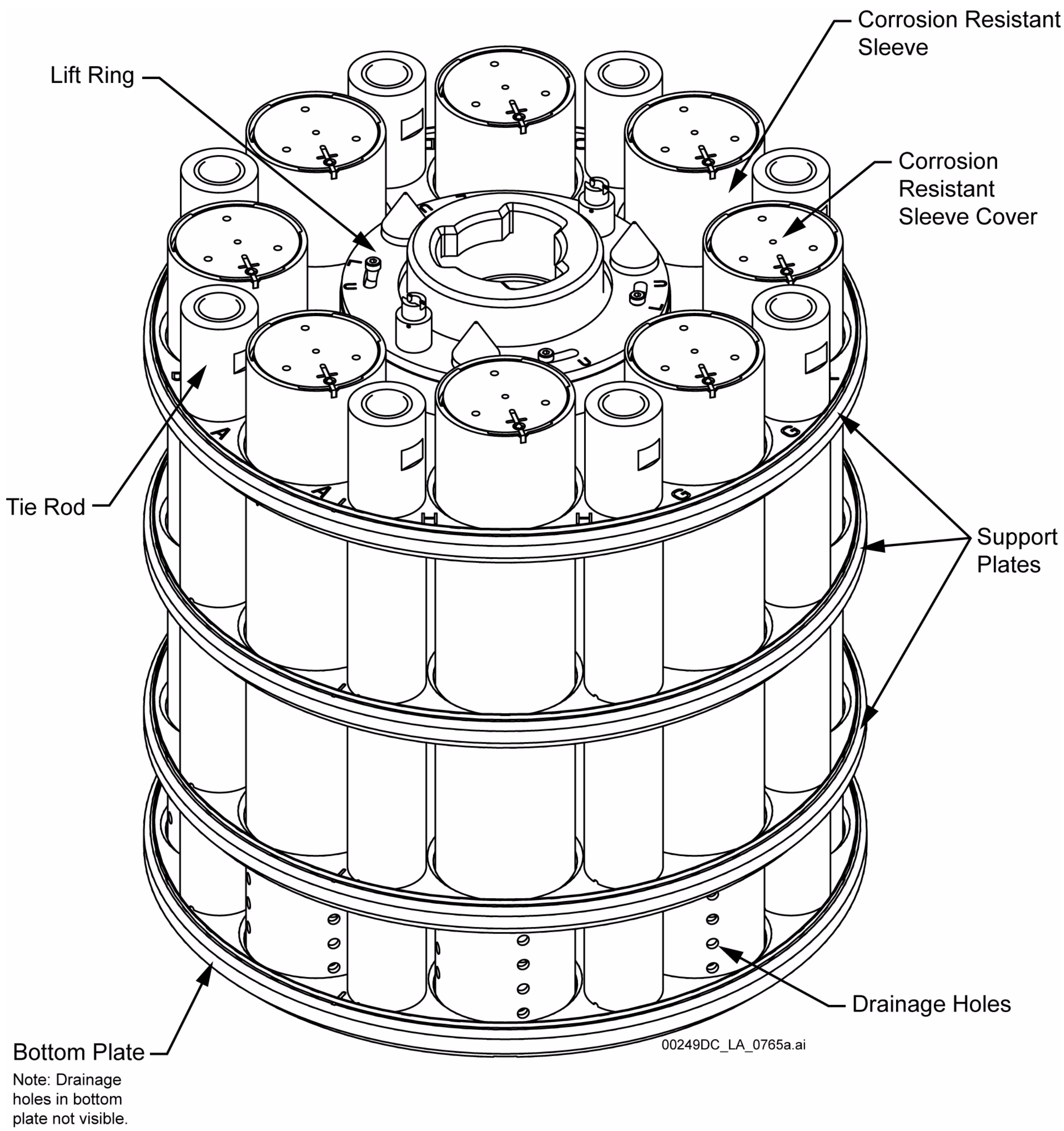

Figure 1.5.1-33. Conceptual Packaging Method B Naval SNF Basket

NOTE: Size and number of corrosion-resistant sleeves vary.

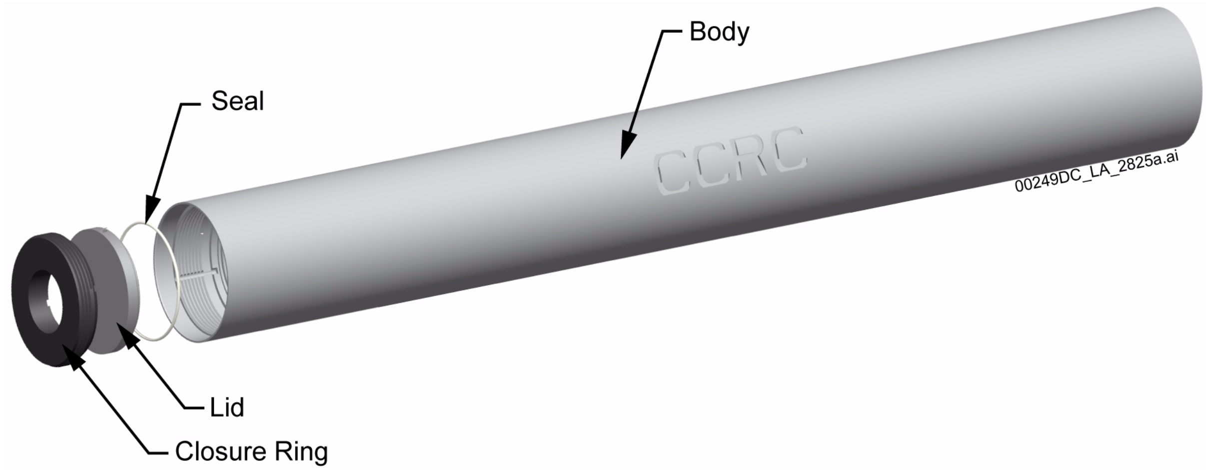

Figure 1.5.1-34. Conceptual Naval Corrosion-Resistant Can

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FIGURES

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1.5.2 Waste Packages and Their Components

[NUREG-1804, Section 2.1.1.2.3: AC 5, AC 6; Section 2.1.1.6.3: AC 1, AC 2; Section 2.1.1.7.3.1: AC 1; Section 2.1.1.7.3.2: AC 1; Section 2.1.1.7.3.3(III): AC 1]

Commercial spent nuclear fuel (SNF) disposed of at the repository is loaded into transportation, aging, and disposal (TAD) canisters either at a utility site or within the Wet Handling Facility (WHF) at the repository. U.S. Department of Energy (DOE) SNF of commercial origin is loaded into TAD canisters within the WHF. Within the Canister Receipt and Closure Facility, the TAD canister is loaded into waste packages. High-level radioactive waste (HLW) canisters and naval SNF canisters are received and loaded into waste packages in the Initial Handling Facility. The Canister Receipt and Closure Facility is capable of loading TAD, DOE SNF, and HLW canisters into waste packages.

Once the waste package closure lids are welded in place, the waste package protects against the release of radioactive gases or particulates during normal operations and Category 1 and Category 2 event sequences. This function classifies the waste package as important to safety (ITS). After repository closure, in conjunction with the other structures, systems, and components (SSCs) of the Engineered Barrier System, the waste package prevents or limits the contact of water with the disposed waste form and prevents or limits the release of radionuclides to the environment. This function makes the waste package important to waste isolation (ITWI).

The following sections describe the principal characteristics of various waste package configurations, including dimensions, weights, materials, design codes, structural analyses, fabrication methods, welding requirements, and nondestructive examination requirements.

1.5.2.1 Waste Package Description

[NUREG-1804, Section 2.1.1.2.3: AC 5(1), (2); Section 2.1.1.7.3.3(III): AC 1(1), (2), (3)]

The waste package consists of a single design with multiple configurations. The different waste package configurations may have waste form specific internal structures and have different external dimensions to allow acceptance of various waste forms. The waste forms received and packaged for disposal are commercial SNF in TAD canisters; canistered DOE SNF, including canistered naval SNF; and canistered HLW from prior commercial and defense fuel-reprocessing operations.

The information in this section applies to all waste package configurations unless otherwise noted. The waste package consists of two concentric cylinders in which the waste forms are placed. The inner vessel includes the inner cylinder, bottom inner lid, and top closure inner lid. The outer corrosion barrier includes the outer cylinder, outer bottom lid, and top closure outer lid. The inner vessel is Stainless Steel Type 316 (UNS S31600), modified with additional constraints on the nitrogen and carbon content. The outer corrosion barrier is restricted Alloy 22 (UNS N06022), a corrosion-resistant, nickel-based alloy. Restrictions on Alloy 22 constituents are imposed to ensure adequate postclosure performance. These restrictions are discussed in Section 2.3.6.

Each waste package has two sleeves on the ends of the outer corrosion barrier. The upper and lower sleeves are also made of restricted Alloy 22. The two sleeves serve as additional structural support for the outer corrosion barrier.

The sleeve on the nonclosure side of the waste package is extended past the outer corrosion barrier to form a skirt that acts as an energy absorber should the waste package be impacted on that surface.

The inner vessel is constructed in accordance with the provisions of *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components. ASME code Section III applies because it (1) is commonly specified by commercial nuclear plants for SNF storage containers, (2) is established and proven in service, and (3) is accepted by industry and regulatory authorities. The inner vessel is N-stamped, signifying compliance with *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001). While the inner vessel is designed for internal pressure and deadweight loads per the ASME code, the loaded waste package is analyzed as an integral unit to demonstrate that structural integrity is maintained during various design basis events.

The outer corrosion barrier is specifically designed as a corrosion barrier, not a pressure vessel. However, it is constructed in accordance with applicable technical requirements of *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components, including material, fabrication, and examination requirements and selected administrative requirements. The outer corrosion barrier is evaluated both against stress limits, consistent with the design limits specified in *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001), and against an energy absorption failure measure for margin analysis. The outer corrosion barrier is not stamped with an N Symbol.

The two closure lids are the top inner lid and the top outer lid, and they provide a leak-tight closure. The top outer lid is made of Alloy 22. The top inner lid is modified Stainless Steel Type 316. The top inner lid has a purge port to allow purging and backfilling. Section 1.2.4.2.3 describes the methods, including applicable codes, for the closure welds.

After the inner lid has been installed and the Stainless Steel Type 316 inner vessel is sealed, the inner vessel is evacuated and helium is added as an inert fill gas through the purge port. Afterward the purge port plug is seal welded. The helium helps transfer heat from the waste form to the wall of the inner vessel and displaces oxidizing gases during the purging process that might react with the inner vessel and the waste form canisters.

The closure methods for the inner and outer lids differ. The inner stainless steel lid is held in place by a spread ring and is seal welded. The outer Alloy 22 lid is narrow-groove welded with low-plasticity burnishing for stress mitigation. The welds are made by a cold-wire-feed gas tungsten arc welding method (Lundin 2002, Executive Summary).

1.5.2.1.1 Waste Package Configurations

The waste package consists of a single design with multiple configurations. The different waste package configurations have multiple internal structures and different external dimensions to allow acceptance of various waste forms.

Table 1.5.2-1 lists the waste package configurations and provides a brief description of the waste form capacity for each. Table 1.5.2-2 provides a breakdown of the estimated percentage of waste package configurations. Figures 1.5.2-1 and 1.5.2-2 illustrate the waste forms and associated waste package configurations.

The waste package design is flexible, allowing various internal configurations and external dimensions for the anticipated waste forms. There are six waste package configurations:

- 21-PWR/44-BWR TAD (holding a TAD canister containing either pressurized water reactor fuel assemblies or boiling water reactor fuel assemblies)
- 5-DHLW/DOE Short Codisposal
- 5-DHLW/DOE Long Codisposal
- 2-MCO/2-DHLW
- Naval Short
- Naval Long.

The fundamental design of the six waste package configurations is similar. Each of the waste package configurations uses the same materials for the inner vessel and the outer corrosion barrier. Each inner vessel is helium filled. Each waste package has two closure lids. If required, internal divider plates are installed in the waste package, depending upon the waste form. The waste package internals, when used, provide structural components for locating the waste forms. There are no waste package internal components used in conjunction with the TAD configuration and the naval SNF configurations.

This section presents structural analyses both to demonstrate the robustness of three waste package configurations given the imposition of mechanical loads associated with potential event sequences and to demonstrate the methodology used to determine the acceptability of all waste package configurations for repository operations. The three configurations considered are:

- 21-PWR/44-BWR TAD (with maximum-sized TAD canister per *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008))
- 5-DHLW/DOE Short Codisposal
- Naval Long.

Section 5.10 discusses the licensing process to be used to incorporate waste package configurations into repository operations.

1.5.2.1.2 Waste Package Configurations for Commercial SNF

Commercial SNF, including DOE SNF of commercial origin, will be loaded into TAD canisters prior to the TAD canister being loaded into a TAD waste package. The waste package configuration for disposal of commercial SNF is the TAD waste package.

The characteristics of commercial SNF are discussed in Section 1.5.1.

1.5.2.1.2.1 Internal Design

There are no internal components, structural guides, or support tubes for the TAD waste package configuration, because it is designed to accept only one TAD canister.

1.5.2.1.2.2 Transportation, Aging, and Disposal Waste Package Specifications

The engineering specifications for the TAD waste package configuration include the waste form characteristics, physical dimensions of the waste package, and material specifications.

Section 1.5.1 discusses typical fuel assembly characteristics, including age, burnup, initial enrichment, and mass of heavy metal. Section 1.5.1 also discusses specifications for TAD canisters. Table 1.5.2-3 shows the physical dimensions and weights of the commercial waste package configuration. Table 1.5.2-4 shows the material specifications of the waste package components, including the TAD waste package used for commercial SNF.

Figure 1.5.2-3 shows component dimensions, weights, and materials for the TAD waste package configuration.

1.5.2.1.3 Waste Package Configurations for HLW and DOE SNF

HLW and DOE SNF is loaded into canisters, which are sealed and shipped to the repository. The canisters are designed for direct loading into the 5-DHLW/DOE Short Codisposal, the 5-DHLW/DOE Long Codisposal, or the 2-MCO/2-DHLW waste package configurations for disposal. The selection of waste package configuration will depend upon the type of waste and the canister into which it is loaded.

The three waste package configurations are described as follows:

- **5-DHLW/DOE Short Codisposal**—As shown in Figure 1.5.2-4, this configuration holds up to five HLW canisters from the Savannah River site or the West Valley site, having a nominal diameter of 24 in. and nominal length of 118 in., along with an 18-in. standardized short DOE SNF canister in the center. Alternatively, it can be loaded with a 24-in. standardized short DOE SNF canister in a peripheral location if the center location is empty. With this loading pattern, the remaining four peripheral locations are loaded with HLW canisters. Or, it can be loaded with up to five HLW canisters in the peripheral locations with the center location empty.

- **5-DHLW/DOE Long Codisposal**—This configuration is identical to the 5-DHLW/DOE Short Codisposal waste package, except for length. As shown in Figure 1.5.2-5, this configuration holds up to five HLW canisters from the Hanford site, having a nominal diameter of 24 in. and nominal length of 180 in., or canisters of the same size from the Idaho National Laboratory, along with an 18-in. standardized long DOE SNF canister in the center. Alternatively, it can be loaded with a 24-in. standardized long DOE SNF canister in a peripheral location if the center location is empty. With this loading pattern, the remaining four peripheral locations are loaded with HLW canisters. Or, it can be loaded with up to five HLW canisters in the peripheral locations with the center location empty.

- **2-MCO/2-DHLW**—As shown in Figure 1.5.2-6, this waste package configuration holds two multicanister overpacks (MCOs) with a maximum diameter of 25.51 in. and a nominal length of approximately 166 in. and two Hanford-type HLW canisters.

1.5.2.1.3.1 Internal Design

Internal Plates—For HLW and DOE SNF waste package configurations, internal plates are arranged to accommodate the particular waste forms intended for disposal in each configuration. The internal structure for the 5-DHLW/DOE Short Codisposal and 5-DHLW/DOE Long Codisposal configurations is created by divider plates that are arranged to form cavities with hexagonal cross sections. The central region consists of a support tube.

The internal structure for the 2-MCO/2-DHLW waste package is divided into four cells by divider plates. Two diagonally opposite cells are designed with a fuel support plate assembly.

For these three DOE waste package configurations, the inner vessel lid acts as an integral shield plug. The inner vessel lid is the same material as the other waste package configurations but is thicker to provide shielding against ionizing radiation for the top of the waste package. For the configurations with commercial SNF and naval SNF, such shielding is provided within the canister loaded into the waste package.

1.5.2.1.3.2 HLW and DOE SNF Waste Package Specifications

The engineering specifications for the HLW and DOE SNF waste package configurations include the waste form characteristics in terms of physical dimensions of the waste package and material specifications. Table 1.5.2-4 shows the material specifications of the waste package components, and Figures 1.5.2-4 through 1.5.2-6 show material specifications, dimensions, and weights of these waste packages. Table 1.5.2-5 shows the physical dimensions and loaded weights for the three HLW and DOE SNF waste package configurations.

1.5.2.1.4 Waste Package Configurations for Naval SNF

Naval SNF arrives in sealed and inerted canisters that are designed for direct loading into the waste package. There is one canister per waste package. Because there are two canister sizes, one short and one long, there are two naval waste package configurations.

- **Naval Short**—This waste package configuration holds one short naval SNF canister with a maximum diameter of 66.5 in. and a maximum length of 187 in.

- **Naval Long**—This waste package configuration holds one long naval SNF canister with a maximum diameter of 66.5 in. and a maximum length of 212 in.

Component dimensions and materials for the two naval waste package configurations are shown in Figures 1.5.2-7 and 1.5.2-8.

1.5.2.1.4.1 Internal Design

There are no internal structures, structural guides, or support tubes for the naval waste package configurations, because they are designed to accept only a single naval SNF canister.

1.5.2.1.4.2 Naval SNF Waste Package Specifications

The engineering specifications for the naval waste package configurations include the waste form characteristics in terms of physical dimensions of the waste package and material specifications. Table 1.5.2-5 shows the physical dimensions and loaded weights for both naval SNF waste package configurations. Table 1.5.2-4 shows the material specifications of the waste package components, including the waste package used for naval SNF.

1.5.2.2 Operational Processes and Procedures

[NUREG-1804, Section 2.1.1.2.3: AC 6(1); Section 2.1.1.7.3.3(III): AC 1(9)]

The fuel assemblies and canistered waste received at the repository are loaded in the waste packages at the handling facilities. Uncanistered, commercial SNF assemblies are packaged within a TAD canister in the WHF prior to placement in a waste package. Section 1.5.1 discusses loaded fuel characteristics, including criticality and thermal controls of the TAD canister.

There are procedures to ensure that thermal requirements are satisfied. Loading plans are generated, checked, and approved prior to operations being performed; records of the loading are made and maintained. Section 1.2.1.4 describes waste package loading. Section 1.3.1 describes the thermal loading of the repository.

Procedures and inspections are implemented to ensure that the waste packages remain suitable for disposal. If a waste package is found to be unsuitable for disposal due to handling or other factors, that waste package will either be remediated, if feasible, or removed from service.

1.5.2.2.1 Waste Package Loading to Satisfy Thermal Requirements

The waste packages are loaded with waste forms in such a manner that, in conjunction with facility and component design, the temperature limits for the waste forms, waste package, and drift walls are not exceeded during operations in the surface facilities, during transport to the subsurface, or during the preclosure period. TAD canister loading, in conjunction with the thermal management process described in Sections 1.3.1 and 5.10, ensures that the repository and waste package temperature limits are not exceeded during the thermal pulse following closure of the repository.

Thermal analyses are discussed within the license application sections appropriate for conditions under consideration. Section 1.2 provides a discussion of thermal analysis in the surface facilities and Section 1.3 provides a discussion of thermal analysis in the subsurface facility. Section 2.3.5 describes postclosure thermal analysis.

1.5.2.2.2 Waste Package Loading to Satisfy Criticality Requirements

As described in Section 1.5.1, criticality controls will be integral to the waste forms loaded into the waste packages. Section 1.14 describes the additional criticality controls relied upon during waste package loading and handling within repository facilities.

1.5.2.2.3 Inerting

After the waste package is loaded, the stainless steel closure lid is installed. Next, the spread ring is installed and seal welded. The stainless steel inner vessel is then evacuated and backfilled with helium. NUREG-1536 (NRC 1997, Section 8.0) provides guidance for this process. Section 1.2.4.2.3 describes the waste package closure process in detail. Helium gas promotes heat transfer from the interior of the waste package to the waste package inner vessel and provides an inert environment. After completion of the inerting process, the purge port is closed and seal welded shut.

1.5.2.3 Considerations Important to Safety and Important to Waste Isolation

1.5.2.3.1 Structures, Systems, and Components Important to Safety

Section 1.9 identifies ITS SSCs and their design bases.

The waste package is ITS. The waste package is designed to sustain loads from normal operations, as well as event sequences. For normal operational loads, the waste package is designed to sustain the loads imposed from horizontal lifting using the emplacement pallet.

1.5.2.3.2 Structures, Systems, and Components Important to Waste Isolation

Section 1.9 identifies ITWI SSCs and their design bases.

The waste package is ITWI. 10 CFR 63.113(b), (c), and (d) require that the entire repository system meets specific dose limits. The waste package is one of many features relied upon to meet these limits. The objective is to design a waste package that works in concert with the Upper Natural Barrier and Lower Natural Barrier and other SSCs of the Engineered Barrier System to meet performance standards. Section 2.3.6 describes the modeling of the long-term corrosion performance of the waste package.

1.5.2.4 Design Bases and Design Criteria

[NUREG-1804, Section 2.1.1.7.3.1: AC 1(1), (3), (4), (5), (6), (8), (9)]

The nuclear safety design bases for ITS and ITWI SSCs and features are derived from the preclosure safety analysis presented in Sections 1.6 through 1.9 and the postclosure performance assessment presented in Sections 2.1 through 2.4. The nuclear safety design bases identify the safety function to be performed and the controlling parameters with values or ranges of values that bound the design.

The quantitative assessment of event sequences, including the evaluation of component reliability and the effects of operator action, is developed in Section 1.7. Any SSC or procedural safety control appearing in an event sequence with a prevention or mitigation safety function is described in the applicable design section of the SAR. Section 1.7.2.3.1 describes the determination of passive SSC reliability and discusses several types of failures for passive SSCs in the PCSA, including the structural challenge causing loss of containment (breach) of a waste form container (e.g., waste package).

Section 1.9 describes the methodology for safety classification of SSCs and features of the repository. The tables in Section 1.9 present the safety classification of the SSCs and features, including those items that are non-ITS or non-ITWI. These tables also list the preclosure and postclosure nuclear safety design bases for each structure, system, or major component.

To demonstrate the relationship between the nuclear safety design bases and the design criteria for the repository SSCs and features, the nuclear safety design bases are repeated in the appropriate SAR sections for each individual SSC or feature that performs a safety function. The design criteria are specific descriptions of the SSCs or features (e.g., configuration, layout, size, efficiency, materials, dimensions, and codes and standards) that are utilized to implement the assigned safety functions. Table 1.5.2-6 presents the nuclear safety design bases and design criteria for the waste package. Table 1.5.2-7 presents the derived requirements and associated design solutions for the ITWI function of the waste package component.

1.5.2.5 Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

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

There are no procedural safety controls to prevent event sequences or mitigate an event sequence.

1.5.2.6 Design Methodologies

[NUREG-1804, Section 2.1.1.7.3.2: AC 1(1); Section 2.1.1.7.3.3(III): AC 1(1), (7), (8), (9)]

1.5.2.6.1 Structural Design

The waste package is designed and analyzed for lifting operations, normal loads, and event sequences. The following terms are used to describe and discuss structural analysis methodology, acceptance criteria, and results:

S_u	=	engineering tensile strength
S_y	=	engineering yield strength
S	=	maximum allowable stress
e_u	=	engineering uniform strain
e_y	=	engineering yield strain
ε_y	=	true yield strain
ε_u	=	true uniform strain
σ_y	=	true yield strength

σ_u = true tensile strength
σ_{int} = stress intensity
τ = shear stress
I_T = toughness index
I_T' = wall-averaged expended toughness
σ_m = mean stress
σ' = effective stress
η = stress triaxiality term
ξ = deviatoric state parameter
P_m = general primary membrane stress intensity
P_L = local primary membrane stress intensity
P_b = primary bending stress intensity.

1.5.2.6.1.1 Structural Design Methodology

Tables 1.5.2-8 and 1.5.2-9 list the various load combinations that are analyzed and applicable acceptance criteria. The waste package is designed so that, with normal loads, stresses remain in the elastic range and below the threshold for stress corrosion cracking. The normal loads are analyzed using finite-element methods. The normal loads include dead weight, internal pressure, and thermal expansion.

To protect against breach, the waste package is also designed and analyzed to demonstrate that, during an event sequence, the Alloy 22 outer corrosion barrier supported by the inner vessel will not exceed stress limits. While the seal welds of the inner vessel are anticipated to be sound welds, at present no credit for resistance against dynamic events is taken. Therefore, for dynamic structural events, where the inner vessel in the vicinity of the seal welds may be reasonably anticipated to experience considerable loads, these welds are not credited to maintain the hermeticity of the inner vessel. The suitability of a waste package for emplacement following an event sequence requires further evaluation of preclosure and postclosure performance objectives and disposition on a case-by-case basis.

The analysis of event sequences allows loads beyond the elastic range; relies on elastic-plastic, finite-element analysis methods; and permits deformation. Acceptance criteria for event sequences invoke the tiered screening criteria for material failure shown in Table 1.5.2-10. The tiered screening criteria method is a deterministic approach based on elastic-plastic analysis methods described in the ASME code (ASME 2001, Section III, Appendix F, F-1341.2). Under the tiered screening criteria, if a case does not satisfy one tier of screening, a more detailed analysis is conducted to determine whether the case is acceptable. This process continues until the load and, hence, the design configuration are determined to be either acceptable or unacceptable. Once a given tier is satisfied, further analysis is not required. Satisfaction of the tiered screening criteria provides confidence that the Alloy 22 outer corrosion barrier is not likely to breach during an event sequence.

To determine the margins inherent in the code compliance case, an energy absorption methodology is developed to define the capacity of the outer corrosion barrier given the occurrence of event sequence loading. The potential for strain energy absorption of the wall-averaged stress fields defines the capacity. The capability is the potential strain energy absorption, or toughness index, for the outer corrosion barrier material.

The toughness index, I_T, is the flow stress multiplied by the uniform strain, ε_u. The flow stress is the average of the yield strength, σ_y, and the ultimate strength, σ_u, and the uniform strain is the strain at the ultimate strength. I_T is defined in terms of true stress and strain values.

$$I_T = \frac{1}{2} \cdot \varepsilon_u(\sigma_y + \sigma_u) \qquad \text{(Eq. 1.5.2-1)}$$

The basis for this definition is illustrated in Figure 1.5.2-9.

The variability of the three-dimensional stress-strain state (the triaxiality) affects the failure behavior in metals. Ductile failure theories have been developed based on the stress triaxiality term, $\eta = \sigma_m/\sigma'$ and a deviatoric state parameter, ξ. The hydrostatic, or mean, stress, σ_m, is defined below in terms of principal stresses, while σ' is the effective stress—usually the Von Mises stress (maximum distortion energy criterion).

$$\sigma_m = \frac{\sigma_1 + \sigma_2 + \sigma_3}{3} \qquad \text{(Eq. 1.5.2-2)}$$

Under the constraint of plane stress that occurs in the cold forming of steel sheet near the tensile instability level, a deviatoric state parameter can be uniquely defined in terms of the triaxiality term, irrespective of the constitutive behavior of the material. This permits a unique mapping of the equivalent strain to rupture versus stress triaxiality onto the failure space of maximum and minimum in-plane membrane strains, ε_1 and ε_2. This is the same failure space used by the sheet metal cold forming community (i.e., tensile forming limit diagrams), and is the triaxiality adjusted failure space used in the energy absorption methodology.

The toughness (strain energy) expended versus toughness available at the governing wall section is used as a measure of damage. The wall-averaged Von Mises effective stress and strain time histories are used to compute a wall-averaged toughness expended, I_T' (approximate area under a constructed stress-strain response curve), from initiation of each event sequence loading to the time of unloading. An expended toughness fraction, defined as I_T'/I_T, is calculated for each event sequence load. The expended toughness fraction is a measure of damage and when the expended toughness fraction equals 1.0, failure is assumed.

The outer corrosion barrier breach level of impact velocity may be iteratively determined using LS-DYNA finite element analysis calculations and is conservatively defined as that level of loading that just leads to the initiation of outer corrosion barrier tensile instability (strain concentration and void initiation). This precedes breach because further loading is needed for the voids to coalescence across a wall section with sufficient porosity to lead to a bifurcation (rupture) of the wall. This microstructural material response is exhibited in a uniaxial tension test as the initiation of necking and reduction in load carrying capacity (i.e., the maximum "ultimate" (engineering) tensile strength) is reached. Necking in room-temperature tension tests begins with internal void formation at second phase particles which breaks down the uniaxial stress state and introduces complex multiaxial stress states adjacent to the voids.

To demonstrate the margin inherent in the satisfaction of the tiered screening process, the expended toughness fraction is calculated for the bounding event sequence load configuration using vendor-cited typical material strength properties and the margin to failure, considering uniaxial weldment coupon test data scatter in the material strength properties, and compared to that obtained from the tiered screening process.

The satisfaction of the tiered screening process by the waste package configuration design protects against waste package breach given the occurrence of an event sequence. However, consistent with the risk-informed preclosure safety compliance requirements of 10 CFR Part 63, DOE has developed reliability estimates for designs that follow current codes and standards. These reliability estimates are needed to calculate the probabilities of radionuclide release from waste packages given the occurrence of event sequences. The development of reliability estimates for passive components such as waste packages and the demonstration of compliance with the preclosure safety requirements of 10 CFR Part 63 are provided in Sections 1.7 to 1.9.

1.5.2.6.1.2 Structural Analysis Software

Structural analysis is performed using ANSYS, LS-DYNA, and Mathcad.

ANSYS is a structural analysis finite-element software package used to solve a variety of engineering problems. Two- or three-dimensional, finite-element representations of waste packages are prepared using ANSYS, depending on the symmetry of the design or the loading. ANSYS is used primarily for static loads on the waste package during normal operations. ANSYS is qualified for use in waste package structural design and is used within its range of qualification.

LS-DYNA is a finite-element program for three-dimensional, nonlinear dynamic analysis of structures. LS-DYNA is used primarily for dynamic impact analyses (Category 1 and Category 2 event sequences) of the waste packages. LS-DYNA is qualified for use in waste package structural design and is used within its range of qualification.

Mathcad is used for solving complex thermal expansion, impact velocities, and internal pressurization calculations, using vectors and matrices. Mathcad is appropriate for these calculations and is used within its range.

1.5.2.6.1.3 Finite-Element Mesh Discretization

For each structural calculation, a sensitivity analysis is performed on the mesh representation of the waste package. The purpose of mesh refinement is to ensure the mesh objectivity of the finite-element analysis so that the results obtained are not mesh sensitive. The mesh-refinement study consists of the development of an optimum mesh that yields mesh-insensitive results.

The mesh is initially sized by selecting an appropriate element type and refining the mesh in the highest stress and strain regions. The initial mesh refinement is across the thickness in the region of high stress and is kept unchanged in the hoop and axial directions. Mesh refinement is repeated in the other directions until the relative difference in results is approximately an order of magnitude smaller than the relative differences in successive mesh sizes.

Figure 1.5.2-10 is a schematic drawing showing the finite-element representation for a typical structural analysis. The sensitivity analysis assigns a higher number of finite elements in regions of high stress and a lower number of finite elements in regions of low stress. As shown in the figure, the point of impact has a higher number of finite elements than expected low-stress areas.

1.5.2.6.1.4 Load Combinations

The normal load combinations include handling, dead weight, internal pressure, and thermal expansion.

The event sequence load combinations are based on ASME code Service Level D (ASME 2001) loads for the faulted plant condition. These sets of limits permit gross general deformation with some consequent loss of dimensional stability for nuclear plant components. Waste package event sequences allow some deformation, provided the Alloy 22 outer corrosion barrier does not exceed stress limits.

1.5.2.6.1.5 Basis for Acceptance Criteria

Anticipated normal loads will remain in the elastic range. Stress within the outer corrosion barrier will remain below limits established by postclosure analysis. Acceptance criteria will be developed for surface marring to permit assessment of suitability for long-term disposal.

The acceptance criteria for event sequences are based on elastic-plastic analysis methods described in the *Waste Package Component Design Methodology Report* (BSC 2007a). Subsection NC-3211.1(c), Appendix XIII, and Table NC-3217-1, Note (4) of *2001 ASME Boiler and Pressure Vessel Code* provide acceptance criteria for Level D loads for vessel designs in accordance with NC-3200, provided a complete stress analysis is performed (ASME 2001). For the Alloy 22 outer corrosion barrier, the general primary membrane stress intensity (P_m) does not exceed the greater of 0.7 S_u or $S_y + {}^1/_3\,(S_u - S_y)$. The maximum primary stress intensity at any location does not exceed 0.9 S_u.

The Pressure Vessel Research Council of the Welding Research Council has provided recommended guidelines (Hechmer and Hollinger 1998) to the American Society of Mechanical Engineers rule committees for assessing stress results from three-dimensional, finite-element analysis in terms of stress limits in the design-by-analysis rules of the ASME code (ASME 2001, Section III, Division 1, Subsection NB and Section VIII, Division 2). These guidelines explain how to implement applicable ASME code requirements and are used to develop the tiered screening criteria.

The structural criteria method developed for the waste package event sequences directly addresses the dominant failure mode of ductile rupture and limits the membrane stresses to acceptable limits.

Specialized reduced modulus studies indicate that the wall bending stresses are secondary and need not be included in the primary stress intensity evaluation. Due to the established characteristics of Alloy 22, it is not necessary to treat the welds and the heat-affected zones of the welds differently from the base metal (Kokajko 2005, Enclosure Section 4.2, p. 4; Allegheny Technologies 2004). Ductile rupture has been recognized as the failure mode for Alloy 22 for dynamic events (Reamer 2004, Enclosure, Section 4.2).

Plastic analysis is conducted using true stress and true strain-based load and deformation relationships. In accordance with the ASME code, Section III, Appendix F, F-1322.3(b) and F-1341.2 (ASME 2001), for Alloy 22:

- The limit on P_m is $0.7\sigma_u$.
- The limit on P_L is $0.9\sigma_u$, where $P_b = 0$.

As stated in the ASME code, Section III, Appendix XIII, XIII-1123(j) (ASME 2001), the local primary membrane stress intensity P_L must be limited to preclude excessive distortion in the transfer of load to either portion of the structure. Interpretation of this guidance with respect to the Appendix F limits results in requiring P_L values exceeding $0.77\sigma_u$ to not extend greater than $\sqrt{R \cdot t}$ in any direction, where R is the median wall radius and t is the thickness of the outer corrosion barrier.

1.5.2.6.1.6 Acceptance Criteria

Tables 1.5.2-8 and 1.5.2-9 collectively summarize the analyses performed to address the normal load combinations and event sequence load combinations. The tiered screening criteria for material failure shown in Table 1.5.2-10 are based on the ASME code (ASME 2001, Section III, Appendix F, F-1341.2), as discussed above. These criteria are converted to specific acceptance criteria by applying specific material design properties. Table 1.5.2-11 shows Alloy 22 design properties for room temperature. The design properties include yield strength, tensile strength, and true tensile strength and factors thereof based on the acceptance criteria and tiered screening criteria shown in Table 1.5.2-10. The design properties of yield strength, tensile strength, and maximum allowable stress are obtained or interpolated from the ASME code, Section II, Part D, “Properties,” Tables Y-1, U, and 1B for Alloy 22 plate (ASME 2001). True tensile strength is calculated from the equation $\sigma_u = S_u\,(1 + e_u)$, where S_u is the tensile strength and e_u is the corresponding engineering strain.

1.5.2.6.1.7 Cases

All waste package configurations will be evaluated against load combinations for various cases. The cases are grouped into normal loads and event sequence loads. The cases analyzed for normal loads include:

- Static loading on waste package emplacement pallet
- Axial and radial thermal expansion
- Tensile stresses from internal pressurization.

The cases analyzed for event sequence loads include:

- Vibratory ground motion damages waste package in the transport and emplacement vehicle (TEV)
- TEV collision with an emplaced waste package
- Oblique waste package drop in the TEV
- Vibratory ground motion damages horizontally oriented waste package during transfer to the TEV
- General drift collapse in the lithophysic portions of the repository
- Rockfall on the waste package in the nonlithophysic portions of the repository
- Horizontal drop on the emplacement pallet and invert.

1.5.2.6.1.8 Structural Analysis Results for Normal Loads

Table 1.5.2-8 summarizes specific cases analyzed, acceptance criteria, and results of structural analysis for normal loads. Satisfaction of the acceptance criteria provide confidence that the waste package is designed to withstand normal loads. The results are reported in terms of maximum stress intensity, which is compared with the acceptance criteria. However, axial and radial gaps are reported as minimum gaps that result in zero contact stress.

Static Loading on Waste Package Emplacement Pallet—During preclosure operations, the tensile stresses imposed on the Alloy 22 of the waste package outer barrier shall be less than 257 MPa (the approximate stress corrosion cracking threshold for Alloy 22). Analysis determines the structural response of the outer corrosion barrier while statically resting on a waste package emplacement pallet. ANSYS is used to solve the finite-element representation of the problem. Table 1.5.2-8 presents the acceptance criteria for this load case.

Axial and Radial Thermal Expansion—As temperature rises, the inner vessel expands faster than the outer corrosion barrier. Thermal expansion has the potential of causing contact stresses in the waste package walls. The design needs to ensure that no contact stresses are introduced. The calculation shows that the minimum required axial gap between the inner vessel and outer corrosion barrier to produce zero contact stress in the axial direction is calculated as 8.1 mm. Table 1.5.2-8 shows the value conservatively rounded up to 10 mm for design purposes. For radial expansion, Table 1.5.2-8 shows that with a radial gap of 1 mm or greater, there is zero contact stress in the radial direction.

Tensile Stresses from Internal Pressurization—As temperature rises, the inner vessel expands faster than the outer corrosion barrier, and the gap volume decreases. This decrease in the gap volume increases pressure. The tensile stresses imposed on the Alloy 22 of the waste package outer barrier shall be less than 257 MPa (the approximate stress corrosion cracking threshold for

Alloy 22). Table 1.5.2-8 presents the results and acceptance criteria for this load case. Based on the results for this load case, stresses in the outer corrosion barrier do not exceed 257 MPa.

1.5.2.6.1.9 Structural Analysis Results for Event Sequences

Table 1.5.2-9 summarizes specific cases, acceptance criteria, and the results of structural analysis of event sequences. The acceptance criteria for the specific cases are based on the tiered screening criteria shown in Table 1.5.2-10. Satisfaction of the tiered acceptance criteria, as demonstrated in this section, provides confidence that the outer corrosion barrier is protected from breach during event sequences. Event sequence cases are bounded by the room temperature results. That is, the margin to stress limits is greater at higher temperatures than at lower temperatures. The reduced material strength at higher temperatures is offset by higher ductility and better impact energy absorption in the plastic range.

The cases chosen for analysis envelope the potential mechanical challenges the waste package may experience during the preclosure period during Category 1 and 2 event sequences.

1.5.2.6.2 Radiation Shielding Methodology

For the 5-DHLW/DOE Short Codisposal, the 5-DHLW/DOE Long Codisposal, and the 2-MCO/2-DHLW waste package configurations, the inner vessel lid provides shielding. Shielding on the top is required during the waste package closure process. For the configurations with commercial SNF and naval SNF, shielding is provided within the canister loaded into the waste package. Radiation shielding methodology is discussed in Section 1.10. Table 1.5.2-12 presents the required integral closure lid shield plug thickness along with the calculated dose for the three HLW/DOE SNF codisposal configurations.

1.5.2.6.3 Fire Analysis Methodology

The waste package provides protection of the waste form given the occurrence of a fire. Short-term temperature limits on the waste form will not be exceeded. Calculations for fire analyses are performed parametrically utilizing a 2-D ANSYS simulation of a waste package, assuming the worst fire conditions will not exceed those defined in the U.S. Nuclear Regulatory Commission regulations for transportation. Duration and temperature of the fire conditions are varied. The waste package inner components are integrally connected, and fuel assemblies are modeled with an effective thermal conductivity. The integral connection minimizes thermal resistance to the fire and is therefore conservative in estimating waste package internal temperatures (from a fire).

The duration and intensity of these fires are expected to be only a few minutes with only a small part of the waste package exposed to flame. Results for a TAD waste package exposed to a fully engulfing, 800°C (1472°F) fire shows the TAD canister surface temperature increases from about 150°C (302°F) to 450°C (852°F) in 30 minutes and that clad remains below 570°C (1058°F). Fires in any room where the waste package transfer trolley is located are expected to be far less severe, both in duration and intensity.

1.5.2.6.4 Waste Package Prototype Program

The waste package prototype procurement strategy is addressed in *Prototype Procurement Strategy for Waste Packages, Pallets, and Drip Shields* (BSC 2006). The waste package prototype procurement strategy is closely related to the waste package prototype testing program. The following subsection summarizes the prototype program.

Procurement of prototype waste packages over the next several years will confirm fabrication processes well before manufacture of the production waste packages. Six waste package prototypes have been planned. This prototype strategy will also ensure that a population of qualified suppliers and fabricators will be identified and updated right up to the time of procurement of the production units, and the variability between vendors will be identified and reduced.

It is anticipated that multiple vendors will be required to fabricate the approximately 11,000 waste packages needed for the Yucca Mountain Project. Prototypes will be used to evaluate manufacturing process variability for individual vendors and between multiple vendors.

The various fabrication processes used by the fabricators, although guided by procurement requirements, industry codes and standards, and technical and quality requirements, will not be completely uniform from one fabricator to the next. A prudent way to identify and evaluate these differences is to evaluate the waste package prototypes from the various fabricators before the actual manufacture of the production waste packages. The variability between fabricators and processes will be identified and addressed as early as possible in order to develop appropriate mitigation measures, if needed. In order to evaluate any differences between fabricators and processes, the project will receive, as part of the prototype procurement process, numerous fabrication and nondestructive examination reports documenting compliance with the license conditions, quality requirements, procurement requirements, and industry codes and standards.

The following are the functions and goals for the development of six waste package prototypes that will be used to support design, testing, startup, and preoperational testing:

- Demonstrate fabricability.

- Develop a cadre of qualified vendors.

- Evaluate manufacturing process variability.

- Verify process operations, including mechanical handling, fuel loading, and closure weld system operations. This program is in progress and is also closely coordinated with the waste package closure demonstration program described in Section 1.2.4.2.

- Perform testing to confirm as-built conditions, including nondestructive testing, destructive testing, mechanical properties testing, and drop testing.

1.5.2.7 Consistency of Materials with Design Methodologies
[NUREG-1804, Section 2.1.1.7.3.3(III): AC 1(3), (4), (5)]

The waste package inner vessel is fabricated and has the N Symbol stamped in accordance with the requirements of the ASME code (ASME 2001, Section III, Division 1, Subsection NC). The outer corrosion barrier is also fabricated to the provisions of ASME code, Section III, Division 1, Subsection NC, but is not code stamped. The outer corrosion barrier is inspected by an authorized nuclear inspector and certified to the specific provisions of the ASME code, as identified in the waste package fabrication specifications. The basket assembly attachment welds to the inner vessel will be performed in accordance with the requirements of the ASME code, Section III, Division I, Subsection NC. These welds will be performed by the N Certificate Holder prior to affixing the N Symbol stamp to the inner vessel.

ASME code, Section III, Division 1, Subsection NC (ASME 2001), is specified for the fabrication of the waste package, based on:

- Section III rules are intended for the design and fabrication of nuclear vessels.
- Section III pertains to unfired pressure vessels. Fired pressure vessels (Section I) are exposed to higher temperatures and have different corrosion considerations compared to unfired pressure vessels.
- Common industry practice is to use Section III (rather than Section VIII) for new, nuclear safety applications.
- Subsection NC is appropriate for the service conditions (e.g., pressure, temperature, external loadings) that the waste package will be subjected to.
- An advantage of using Subsection NC rules is the increased availability of qualified fabricators (it is anticipated that multiple fabricators will be required to produce waste packages in the years required for production).

The waste package is fabricated by rolling and welding plates to form concentric cylinders. The basket assembly is installed, if required, inside the waste package at the fabricator. The bottom lids are welded to the inner vessel and outer corrosion barrier. The cutting, forming, machining, and fitting operations are performed in accordance with the approved shop procedures or drawings and the provisions of the waste package fabrication specification. The cutting, forming, machining, and fitting operations applied to the inner vessel and the outer corrosion barrier meet the requirements of ASME code, Section III, Division 1, Subsection NC-4000 (ASME 2001).

1.5.2.7.1 Fabrication Materials and Process

The inner vessel is fabricated from Stainless Steel Type 316, modified with the following additional restrictions on chemical composition: 0.020% maximum carbon and 0.060% minimum to 0.10% maximum nitrogen. The restrictions provide better corrosion properties, while maintaining the mechanical properties of Stainless Steel Type 316. The outer corrosion barrier is fabricated from restricted Alloy 22, a corrosion-resistant, nickel-based alloy. To ensure the postclosure performance of the Alloy 22, limits are set on the chemical constituents of the alloy. These restrictions and the postclosure basis for the values are discussed in Section 2.3.6.

Materials for the inner vessel and the outer corrosion barrier comply with the requirements of the ASME code, Section III, NC-2000 as augmented by Section II, material specification (ASME 2001). Examination and repair are in accordance with Section III for the inner vessel. The more restrictive requirements of Section III Subsection NB-2530 (Class 1) are specified for examination and repair for the outer corrosion barrier to provide added assurance for material integrity (ASME 2001).

Basket materials comply with the requirements of the ASME code, Section II, “Materials” (ASME 2001).

The welding filler metals for ASME code welds conform to the requirements of the ASME code, Section III, Division 1, NC-2400 (ASME 2001) and applicable project requirements. Welding processes used on the inner vessel, the outer corrosion barrier, and the welds that attach basket materials to the inner vessel comply with ASME code, Section IX and Section III, NC-4000 (ASME 2001). Only the following two welding processes, with the restrictions listed, are permitted for the outer corrosion barrier:

- **Gas Tungsten Arc**—An inert gas backing purge is used for the first 3/16 in. of deposited weld metal thickness for full penetration welds having an open root weld.

- **Gas Metal Arc—**This process, except for the short-circuiting arc mode, is used for Alloy 22. The short-circuiting arc mode is prohibited for fabrication of the outer corrosion barrier.

The heat treatment procedure for the vessel shells meets ASME code, Section III, Subsection NC-4600 (ASME 2001). The heat treatment may be accomplished by any suitable method of heating and cooling, provided the required heating and cooling rates, metal temperature uniformity, and temperature control are maintained. Heat treatment is performed with thermocouples in contact with the material.

The outer corrosion barrier is solution annealed after initial fabrication. Once solution annealing is complete, machining of the outer surface of the outer corrosion barrier is not permitted. However, machining is allowed on the inner diameter and final closure weld groove. The outer corrosion barrier is furnace heated at a soak temperature of 2050°F +50°F/–0°F for 20 minutes minimum. Cooling is achieved by immersion in water or spray quenching. The cooling rate for the entire outer corrosion barrier is greater than 275°F per minute from soak temperature to below 700°F.

After heat treatment, the solution anneal film will be removed from the outer barrier surface of the waste package. Numerous treatment options are available for this process, including electropolishing and grit blasting. A decision on a treatment option will be made after consideration of the technology available and the effectiveness of the available options.

1.5.2.7.2 Lid Closure Processes

1.5.2.7.2.1 Welding Requirements

Gas tungsten arc welding is used for the following reasons (Lundin 2002, Executive Summary):

- It is amenable to weld joints in the waste package closure.
- There are more than 35 years experience of gas tungsten arc welding applications for critical welds in the nuclear industry, as well as high-quality pipe and tube applications.
- It is readily automated for remote operations.
- It is applicable to both the Stainless Steel Type 316 inner vessel and the Alloy 22 outer corrosion barrier materials.
- Nondestructive examination using ultrasonic, eddy current, and visual methods is readily automated for remote operations on gas tungsten arc welds.
- Weld repairs are readily accomplished with gas tungsten arc welding, which would likely be used for weld repairs regardless of the main welding method employed for the closure welds.
- Weld residual stresses in a gas tungsten arc weld can be mitigated using controlled low-plasticity burnishing.

Further, the gas tungsten arc welding process limits the size and nature of weld discontinuities for the materials chosen for the waste packages. The discontinuities generated by gas tungsten arc welding are well defined and include lack of penetration and lack of fusion, porosity, and microfissuring. The extent of any of these discontinuities is related to a single-weld pass and does not have a tendency to propagate between passes during welding. Their size and orientation within the weld are naturally limited. Because the possible generation of discontinuities during welding is limited to the specific types discussed previously, they are readily amenable to detection by remote and automated inspection using eddy current, ultrasonic, and visual methods (Lundin 2002, p. 6). Project research on welding of Alloy 22 has confirmed the engineering process-related acceptability of the gas tungsten arc welding process for closure welds (Smith 2008). Long-term performance of the closure welds is discussed in Section 2.3.6.

1.5.2.7.2.2 Stress Mitigation of Closure Welds

The waste package outer corrosion barrier is relied upon to maintain containment of the waste form during the postclosure period. A challenge to this containment is stress corrosion cracking in the outer closure lid weld region. Stress corrosion cracking is the initiation and propagation of cracks in structural components because of the simultaneous interaction of three factors:

1. Metallurgical susceptibility because of alloy composition and grain structure
2. Critical environment, such as water
3. Static, residual, or sustained tensile stresses.

Section 2.3.6 discusses the long-term implications of potential stress corrosion cracking. Residual stress mitigation techniques are applied to the waste package outer lid closure weld region to induce compressive stresses in the outer layers and delay any potential initiation of crack growth.

Low plasticity burnishing is a process by which a high-strength metallic ball is moved across the surface under load. It plastically deforms the surface layer of the material, resulting in compressive stresses. The purpose of the mitigation treatment is to eliminate surface tensile stresses that otherwise may be detrimental in the initiation and propagation of stress corrosion cracking.

Low plasticity burnishing can induce compressive hoop stresses to a minimum depth of 3 mm (SNL 2007, Section 6.5.5).

1.5.2.7.3 Nondestructive Examination

1.5.2.7.3.1 Fabrication Nondestructive Examination

Figure 1.5.2-11 shows the major fabrication welds of the waste package, and Table 1.5.2-13 shows the nondestructive examination requirements associated with these welds. Nondestructive examinations, except for liquid dye penetrant testing, are performed after final machining, surfacing, or heat treatment are performed. The welds for the inner vessel and outer corrosion barrier are examined in accordance with the requirements of ASME code, Section III, Division 1, Subsection NC-5000 (ASME 2001), by Level II or Level III nondestructive examination personnel.

1.5.2.7.3.2 Closure Weld Nondestructive Examination

The closure welds are inspected using visual, eddy current, and ultrasonic inspection techniques. The inspections are performed in accordance with examination procedures developed using *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001). Visual inspection is performed using the profile measurement device on the welding end effectors. An eddy current/ultrasonic inspection end effector is used for the outer lid weld and an eddy current inspection end effector is used for the repair cavity ground to remove weld defects. These end effectors include the appropriate probes and transducers. The inspection end effectors are in contact with the weld surface and automatically maintain alignment with the weld seam. The inspection end effectors are capable of detecting weld defects of 1/16 in. or greater. This capability is demonstrated as part of the process qualification, and the end effectors are checked with a calibration block. A weld repair end effector is used for minor repairs. The defect is removed, resulting in an excavation of a predetermined contour. The excavated area is welded and inspected in accordance with the welding and inspection procedure. Section 1.2.4.2.3 provides detailed discussion of closure weld operations.

1.5.2.7.4 Fabrication Pressure Testing

The inner vessel is hydrostatically or pneumatically tested in accordance with the ASME code, Section III, Subsection NC-6220 and NC-6324 (ASME 2001). For hydrostatic tests, the inner vessel is tested to 1.25 times the design pressure for at least 10 minutes prior to initiation of the examination (ASME 2001). For pneumatic tests, the inner vessel is pressurized with helium to 1.1 times the design pressure for a minimum total time of 10 minutes (ASME 2001).

1.5.2.7.5 Fabrication Helium Leakage Test for Inner Vessel

The fabricated inner vessel is helium leak tested. The helium leakage test is in accordance with ASME code, Section V, Article 10, Appendix IX (ASME 2001). The helium leakage test is performed using a pressure differential of not less than 1 atm. The maximum acceptable leakage is 10^{-6} standard cm^3/s helium.

1.5.2.8 Design Codes and Standards

[NUREG-1804, Section 2.1.1.7.3.3(III): AC 1(3), (4), (5)]

The materials, design, fabrication, testing, examination, and shipping of the waste package meet the requirements of the following codes and standards. The codes and standards are applicable to the extent referenced in the proper waste package fabrication specification and the associated drawings. The applicable codes and standards are:

- *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001)

  – Section II, “Materials”
  – Section III, “Rules for Construction of Nuclear Power Plant Components, Division 1”
  – Section V, “Nondestructive Examination”
  – Section IX, “Welding and Brazing Qualifications”

1.5.2.9 General References

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Table 1.5.2-1. Waste Package Configurations

Waste Package Configuration	Capacity of Configuration
21-PWR/44-BWR TAD	Capacity: One TAD canister containing pressurized water reactor or boiling water reactor assemblies
5-DHLW/DOE Short Codisposal	Capacity: Five short HLW canisters and one short DOE SNF canister in center location[a]
5-DHLW/DOE Long Codisposal	Capacity: Five long HLW canisters and one long DOE SNF canister in center location[a]
2-MCO/2-DHLW	Capacity: Two DOE MCOs and two long HLW canisters
Naval Short	Capacity: One short naval SNF canister
Naval Long	Capacity: One long naval SNF canister

NOTE: [a]Alternatively, this waste package can be loaded with a 24-in. DOE SNF canister in a peripheral location if the center location is empty. With this loading pattern, the remaining four peripheral locations are loaded with HLW canisters. Or the waste package can be loaded with up to five HLW canisters in the peripheral locations with the center location empty.

Table 1.5.2-2. Breakdown of Waste Package Configurations

Waste Package Configuration	Approximate Percentage by Waste Package Configuration
21-PWR/44-BWR TAD	71%
5-DHLW/DOE Short Codisposal	11%
5-DHLW/DOE Long Codisposal	12%
2-MCO/2-DHLW	2%
Naval Short	1%
Naval Long	3%

Table 1.5.2-3. Physical Dimensions and Weights of the Commercial Waste Package Configuration

Waste Package Configuration	Nominal Length (in.)	Nominal Diameter (in.)	Maximum Loaded Weight (lb)
21-PWR/44-BWR TAD	230.32	77.28	162,000

Table 1.5.2-4. Waste Package Configuration Component Materials

Component	Material
Inner vessel	Modified Stainless Steel Type 316
Outer corrosion barrier	Restricted Alloy 22
Inner vessel fill gas	Helium
Divider plates and support tube for 5-DHLW/DOE Codisposal waste package configurations	Carbon steel (SA 516 (UNS K02700))
Divider plates for 2-MCO/2-DHLW waste package	Carbon steel (SA 516)
Upper/lower sleeve	Restricted Alloy 22

Table 1.5.2-5. Physical Dimensions of Waste Package Configurations for DOE Waste Forms

Waste Package Configuration	Nominal Length (in.)	Nominal Diameter (in.)	Maximum Loaded Weight (lb)
5-DHLW/DOE Short Codisposal	145.57	83.70	90,000
5-DHLW/DOE Long Codisposal	208.82	83.70	127,900
Naval Short	205.32	77.28	157,000
Naval Long	230.32	77.28	162,000
2-MCO/2-DHLW	207.82	72.07	112,500

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package

			Nuclear Safety Design Bases		
System or Facility (System Code)	**Subsystem or Function (as Applicable)**	**Component**	**Safety Function**	**Controlling Parameters and Values**	**Design Criteria**
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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with ASME code.
				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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in the ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
DOE and Commercial Waste Package System (DS) (Continued)	DOE and commercial waste package (Continued)	Entire (Continued)	Provide containment (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
Naval SNF Waste Package System (DN) (Continued)	Naval SNF Waste Package (Continued)	Entire (Continued)	Provide containment (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	**Controlling Parameters and Values**	
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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (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	Design Criteria
DOE and Commercial Waste Package System (DS) (Continued)	DOE and Commercial Waste Package (Continued)	Entire (Continued)	Provide containment (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
DOE and Commercial Waste Package System (DS) (Continued)	DOE and Commercial Waste Package (Continued)	Entire (Continued)	Provide containment (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	**Controlling Parameters and Values**	
DOE and Commercial Waste Package System (DS) (Continued)	DOE and Commercial Waste Package (Continued)	Entire (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
			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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	**Controlling Parameters and Values**	
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 impact.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (Continued)

System or Facility (System Code)	Subsystem or Function (as Applicable)	Component	Nuclear Safety Design Bases		Design Criteria
			Safety Function	Controlling Parameters and Values	
Naval SNF Waste Package System (DN) (Continued)	Naval SNF Waste Package (Continued)	Entire (Continued)	Provide containment (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.
				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.	The waste package is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. The canister is required to be designed such that the fire-induced failure hazard meets the required reliability when evaluated against the spectrum of fires. (Note: PCSA analysis depends on the combination of the reliabilities of each component.)
			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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

Table 1.5.2-6. Preclosure Nuclear Safety Design Bases and their Relationship to Design Criteria for the Waste Package (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	Design Criteria
Naval SNF Waste Package System (DN) (Continued)	Naval SNF Waste Package (Continued)	Entire (Continued)	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.	The structural and metallurgical design of the waste package shall be in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME code) (ASME 2001), including the 2002 addenda. The inner vessel of the waste package is designed as a pressure vessel in accordance with Section III, Division I, Subsection NC, for internal pressure and dead load. Loads on the outer corrosion barrier of the waste package are assessed using the material properties and stress limits found in the ASME code, with extensions to permit assessment of performance with plastic deformation. For risk-informed analysis, outer corrosion barrier performance is assessed against limits based on energy absorption during event sequences. Note that these design criteria assume that the waste package will be fabricated in accordance with the ASME code.

NOTE: "Protect against" in this table means either "reduce the probability of" or "reduce the frequency of."
For casks, canisters, and associated handling equipment that were previously designed, the component design will be evaluated to confirm that the controlling parameters and values are met.
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.
The numbers appearing in parentheses in the third column are component numbers.
Facility Codes: CR: Canister Receipt and Closure Facility; IH: Initial Handling Facility; SS: Subsurface Facility.
System Codes: DN: Naval Spent Nuclear Fuel Waste Package; DS: DOE and Commercial Waste Package System.

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
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	The waste package dimensions and component masses for a commercial SNF waste package are specified in Table 1.5.2-3. The waste package dimensions and components masses for the DOE waste forms (HLW glass codisposed with DOE fuel) are specified in Table 1.5.2-5.	NA
	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	The waste package quantities are specified in Table 1.5.2-1. The inventory included in the license application is indicated in Table 1.5.1-12 for commercial SNF, Table 1.5.1-20 for DOE HLW, Table 1.5.1-29 for DOE SNF and Table 1.5.1-32 for naval SNF. The materials of the waste package are specified in Table 1.5.2-4, and the design properties of the waste package are summarized in Table 1.5.2-11. The dimensions of the waste package are illustrated in Figures 1.5.2-3 through 1.5.2-8 for the different configurations.	NA
	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	The minimum thickness of the waste package outer corrosion barrier will not be less than 25 mm for codisposal, naval, and TAD canister packages, as illustrated in Figures 1.5.2-3 through 1.5.2-8 for the different configurations.	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
Waste Package (Continued)	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	The design of the radial gap for the waste package is presented in Section 1.5.2.6.1.8, which shows that 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. The confirmation that the gap sizes meet this criterion is presented in Table 1.5.2-8. The different waste package design configurations are illustrated in Figures 1.5.2-3 through 1.5.2-8.	NA
	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	The different waste package design configurations are illustrated in Figures 1.5.2-3 through 1.5.2-8 and show that 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.	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
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	The inner vessel of the waste package is a pressure vessel within the definition of the *2001 ASME Boiler and Pressure Vessel Code*. For such a device, design limits must be established to size (or confirm the appropriateness of) the wall thicknesses. For the inner vessel, which includes the inner vessel, inner vessel bottom lid, inner vessel closure lid, spread ring, purge port cap plug, and divider plate assembly attachment weld (for high-level waste-bearing waste packages), the ASME design pressures are: • Design Pressure: 150 psi at 650°F • Service Level A (normal): 62.1 psia (47.4 psi) at 662°F • Service Level B (upset): 69.6 psia (54.9 psi) at 662°F • Service Level C (emergency): not specified • Service Level D (accident): 140 psia (125.3 psi) at 707°F. Service Level A is based on no cladding hull rupture and a temperature of 350°C, the former limiting temperature for the fuel cladding, which bounds the inner vessel maximum temperature. Service Level B has the same temperature but it is assumed that 10% of the cladding hulls have ruptured and released the free gas inventory. The accident condition (i.e., Service Level D) is based on exposure to a hypothetical fire and includes the effect of pressure increase with temperature for the initial fill gas of one atmosphere gauge and the contents of the free volume of the cladding hulls for a representative PWR fuel assembly.	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
Waste Package (Continued)	03-08 Seismic Design of Waste Package	The seismic design spectra, time histories, and ground accelerations for the subsurface facilities shall be controlled through the configuration management system (Section 5).	No	The seismic design spectra, time histories and ground accelerations used in the analysis of the postclosure performance of the subsurface facilities are presented in Section 2.3.4.3.2.	NA
	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	Dose rate on the waste package and representative neutron flux is provided in Section 1.10. See Table 1.5.1-8 for the bounding dose rate basis.	NA
	03-12 Waste Package Fabrication	The waste package outer corrosion barrier cylinder shall be fabricated from no more than three 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	Procurement of the waste package inner vessel will require that it be fabricated in accordance with the requirements of *2001 ASME Boiler and Pressure Vessel Code* as identified in Section 1.5.2.7. The outer corrosion barrier is also fabricated to the provisions of ASME code and that it is inspected by an authorized nuclear inspector and certified to the specific provisions of the ASME code, as identified in the waste package fabrication specifications. Nondestructive examination techniques are provided in Table 1.5.2-13. The fabrication sections of the waste package are illustrated in Figure 1.5.2-11. (See Parameter 03-18 for inspection requirements.)	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	**Postclosure Control Parameter** **Parameter Number and Title**	**Postclosure Control Parameter** **Values, Ranges of Values or Constraints**	**Relevant to ITWI**	**Design Criteria/Configuration**	**Postclosure Procedural Safety Control**
Waste Package (Continued)	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	The procurement specification for the waste package will require that the nondestructive examination techniques, of the subject constraint, are provided to the vendor as fabrication requirements. The requirements are specified in Table 1.5.2-13. As discussed in Section 1.5.2.7.3.2, the closure welds are inspected using visual, eddy current, and ultrasonic inspection techniques. The inspections are performed in accordance with examination procedures developed using *2001 ASME Boiler and Pressure Vessel Code*.	NA
	03-14 Waste Package Welding Materials	The waste package fabrication welds shall be conducted in accordance with standard nuclear industry requirements.	Yes	The procurement specification for the waste package will include welding requirements in accordance with *2001 ASME Boiler and Pressure Vessel Code* as specified in Section 1.5.2.7.	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter		Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
	Parameter Number and Title	Values, Ranges of Values or Constraints			
Waste Package (Continued)	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	The procurement specification for the waste package will require the vendor to use two welding processes, with the restrictions listed for the outer corrosion barrier, as presented in Section 1.5.2.7.1. • Gas Tungsten Arc—An inert gas backing purge is used for the first 3/16 in. of deposited weld metal thickness for full penetration welds having an open root weld. • Gas Metal Arc—This process, except for the short-circuiting arc mode, is used for Alloy 22. The short-circuiting arc mode is prohibited for fabrication of the outer corrosion barrier. As discussed in Section 1.5.2.7.3.2, the closure welds are inspected using visual, eddy current, and ultrasonic inspection techniques. The inspections are performed in accordance with examination procedures developed using *2001 ASME Boiler and Pressure Vessel Code*. The repair process, developed by the fabricator, will be reviewed and approved by the repository design organization prior to its use by the fabricator.	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

	Postclosure Control Parameter				
Structure, System and Component	**Parameter Number and Title**	**Values, Ranges of Values or Constraints**	**Relevant to ITWI**	**Design Criteria/Configuration**	**Postclosure Procedural Safety Control**
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 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	The procurement specification will contain the waste package fabrication materials and processes, including annealing per the subject constraint. These requirements are specified in Section 1.5.2.7.1 (*Waste Package Fabrication* (BSC 2007b)).	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter		Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
	Parameter Number and Title	Values, Ranges of Values or Constraints			
Waste Package (Continued)	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, 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 a compressive 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, and ultrasonic testing.	Yes	NA (Background: The outer lid welding process including welding and stress mitigation is presented in Section 1.5.2.7.2. The closure weld examination process is described in Section 1.5.2.7.3. Section 1.2.4.2.3 describes the equipment and operational processes associated with closure of a waste package.)	Procedures controlling the outer lid welding process including welding and stress mitigation will require that such activities satisfy the listed requirements. The process is presented in Section 1.5.2.7.2. The closure weld examination process is described in Section 1.5.2.7.3 and will also be controlled by procedure. Procedures will be developed to implement the subject constraints. The design of the remote waste package closure cells in the CRCF and IHF can accommodate the implementation of these requirements.

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter		Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
	Parameter Number and Title	Values, Ranges of Values or Constraints			
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 as follows: 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 material tensile stresses greater than 257 MPa.	Yes	NA	Procedures will be developed for controlling waste package inspection per the subject constraint. Procedures for repairs shall control modifications in order not to exceed the 257 MPa limit.

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
Waste Package (Continued)	03-19 Waste Package Outer Barrier Material Specifications	The waste package Alloy 22 material will be manufactured to ASTM B575-99a 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	The procurement specification will control the elemental restrictions on the waste package outer corrosion barrier per the identified constraint. These restrictions are described in Sections 1.5.2.1 and 2.3.6.7.4.	NA
	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	The procurement specification will control the pertinent surface finish requirements per the subject constraint.	NA

Table 1.5.2-7. Summary of Conformance of Waste Package Design to Postclosure Control Parameters (Continued)

Structure, System and Component	Postclosure Control Parameter: Parameter Number and Title	Postclosure Control Parameter: Values, Ranges of Values or Constraints	Relevant to ITWI	Design Criteria/Configuration	Postclosure Procedural Safety Control
Waste Form and TAD Canister	04-07 Waste Package Capacities	Waste package capacities shall be as follows: (a) TAD canister-bearing waste package: one commercial SNF TAD canister. (b) Naval waste packages: one naval SNF canister. (c) 2-MCO/2-HLW waste package: two MCOs 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	The waste package configurations that satisfy the specified waste package capacities are in Table 1.5.2-1 and illustrated in Figures 1.5.2-1 and 1.5.2-2.	Section 1.2.1 discusses procedural safety controls on waste package loading.

NOTE: DHLW = defense high-level radioactive waste; PWR = pressurized water reactor.

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

Table 1.5.2-8. Summary of Structural Analyses for Normal Loads

Case	Description	Acceptance Criteria	Results	Comments
Tensile Stresses from Static Loading on Waste Package Emplacement Pallet	Analysis determines the structural response of the outer corrosion barrier while statically resting on a waste package emplacement pallet	The tensile stresses imposed on the Alloy 22 of the waste package outer barrier shall be less than 257 MPa (the approximate stress corrosion cracking threshold for Alloy 22).	Calculated results indicate tensile stresses less than 257 MPa within the waste package outer barrier while resting on the pallet.	—
Contact Stresses from Axial Thermal Expansion	The inner vessel has a greater thermal expansion coefficient than the outer corrosion barrier. The waste package is designed with an initial axial gap at room temperature to prevent interference between the inner vessel and outer corrosion barrier at elevated temperatures or with high overall heat transfer rates.	To eliminate the possibility of induced stress corrosion cracking, the inner vessel and outer corrosion barrier has an axial gap in between. These distances account for differences in thermal expansion values for Alloy 22 and Stainless Steel Type 316.	There is no interaction between the inner vessel and outer corrosion barrier if the axial gap is at least 10 mm (0.394 in.). The different waste package design configurations are illustrated in Figures 1.5.2-3 through 1.5.2-8. The configurations are designed with axial gaps much greater than 10 mm to accommodate postclosure parameters.	—
Contact Stresses from Radial Thermal Expansion	The waste package is designed with a radial gap to prevent interference between the inner vessel and outer corrosion barrier at elevated temperatures or with high overall heat transfer rates.	To eliminate the possibility of induced stress corrosion cracking, the inner vessel and outer corrosion barrier has a radial gap in between. These distances account for differences in thermal expansion values for Alloy 22 and Stainless Steel Type 316.	There is zero tangential stress from thermal expansion if the radial gap is 1.0 mm or greater for all waste package configurations. The different waste package design configurations are illustrated in Figures 1.5.2-3 through 1.5.2-8. The radial gaps are greater than 1.0 mm.	—

Table 1.5.2-8. Summary of Structural Analyses for Normal Loads (Continued)

Case	Description	Acceptance Criteria	Results	Comments
Tensile Stresses from Internal Pressurization	Internal pressure is a result of increased temperature and decreased volume between the inner vessel and outer corrosion barrier, which is caused by thermal expansion. This internal pressure creates hoop and longitudinal tensile stress in the outer corrosion barrier. The calculation is based on an outer surface temperature of 239°C.	The tensile stresses imposed on the Alloy 22 of the waste package outer barrier shall be less than 257 MPa (the approximate stress corrosion cracking threshold for Alloy 22).	The maximum tensile stresses for 21°C (70°F) and 250°C (482°F) are located in the Outer Corrosion Barrier of the waste package with values of 56.8 MPa, which are significantly less than the 257 MPa limit.	—

NOTE: NA = not applicable.

Table 1.5.2-9. Summary of Structural Analyses of Outer Corrosion Barrier for Event Sequences

Case	Description	Acceptance Criteria	Results	Comment
Vibratory Ground Motion Damages Waste Package in TEV	The TEV will not tip over due to vibratory ground motion; however, it may derail. Given a derailment, the waste package will be subject to dynamic forces as it interacts with the TEV.	Meet tiered screening criteria of Table 1.5.2-10.	The analysis of this event used the drop velocities and resulting wall-averaged stress intensity to determine that a velocity of 5.76 m/s is required before the wall-averaged stress intensity reaches the project-tiered second condition acceptance criteria of 0.77.	The results of this analysis provide input to the probabilistic analysis utilized for the preclosure safety analysis described in Sections 1.6 through 1.9.
Collision with Emplaced Waste Package	If the TEV is overdriven, it could collide with a line of emplaced waste packages causing loading on the packages.	Meet tiered screening criteria of Table 1.5.2-10.	To illustrate compliance with this case, a conservative analysis was performed. The use of a 2-MCO/2-DHLW waste package on the opposite end of the 5-DHLW/DOE Short Codisposal waste package impacted by the overdriven TEV will produce a conservative structural response. The orientation of the neighboring waste package as well as the size of the neighboring waste package will have an effect on the structural response of the 5-DHLW/DOE Short Codisposal waste package. The rationale for this assumption is that the 2-MCO/2-DHLW waste package has the smallest diameter of the configurations of waste packages expected to be emplaced in the repository and because of the smaller diameter, it is expected to create a smaller area of contact and higher concentration of the collision force to produce a higher structural response in the 5-DHLW/DOE Short Codisposal waste package. The total driving force exerted on the impacted waste package will be equal to that generated by the TEV in order to propel itself at a speed of 2.0 mph. The TEV force generated and then transferred to the impacted waste package is found by first calculating the tractive effort required by the fully loaded, 300-ton vehicle. Calculation of maximum stress intensity indicates the maximum calculated stress intensity values in the outer corrosion barrier of the TEV impacted waste package are below 0.7 σ_u and hence meet Tier 1 of the screening criteria of Table 1.5.2-10.	—

Table 1.5.2-9. Summary of Structural Analyses of Outer Corrosion Barrier for Event Sequences (Continued)

Case	Description	Acceptance Criteria	Results	Comment
Oblique Waste Package Drop in TEV	During handling operations, the waste package has been lifted in a horizontal position to a height of 2.49 ft (0.759 m). The waste package is dropped and impacts the TEV rail ledge obliquely.	Meet tiered screening criteria of Table 1.5.2-10.	The Naval Long waste package configuration was dropped obliquely onto the TEV surface from a height of 0.759 m. The Naval Long configuration is the maximum weight and is thus considered bounding for this event; hence, all configurations will meet these criteria. The calculation shows that the project-tiered acceptance criteria of wall-averaged σ_{int} <0.77σ_u is met at the assumed failure or breach location.	—

Table 1.5.2-9. Summary of Structural Analyses of Outer Corrosion Barrier for Event Sequences (Continued)

Case	Description	Acceptance Criteria	Results	Comment
Vibratory Ground Motion Damages Horizontally Oriented Waste Package during Transfer to TEV	During loadout operations within the surface facilities, vibratory ground motion causes damage to the waste package. The waste package shall not breach in an event where the waste package while horizontal inside the waste package transfer trolley on the waste package transfer carriage is subjected to the dynamics imposed by vibratory ground motion. The waste package is ejected from the emplacement pallet and falls into the shielded enclosure of the waste package transfer trolley or TEV.	Meet tiered screening criteria of Table 1.5.2-10.	Using the velocity of 5.76 m/s in which the waste package dropped in this configuration reaches the project-tiered second condition acceptance criteria of effective wall-averaged stress intensity ratio of wall-averaged 0.77 and using Newton's equation of motion, the minimum drop height at which the waste package will reach this velocity can be determined: $V^2 = Vo^2 + 2gh$ where Vo = initial velocity V = final velocity g = acceleration due to gravity h = vertical drop height For this calculation: V = 5.76 m/s (waste package final velocity) Vo = 0.0 m/s (waste package initially at rest) g = 9.81 m/s^2 (acceleration due to gravity) Solving for h: $h = (V^2 - Vo^2) / 2g = 1.691$ m A drop height of 1.691 m is more than twice any possible drop that the waste package might experience during loadout operations.	—
General Drift Collapse in the Lithophysic Portion of the Subsurface	Prior to drip shield emplacement, the waste package shall not breach in an event of general drift collapse in the lithophysic portions of the repository caused by vibratory ground motion.	Meet tiered screening criteria of Table 1.5.2-10.	The maximum stress intensities observed in the outer corrosion barrier were less than $\sigma_{int} < 0.7\sigma_u$.	—

Table 1.5.2-9. Summary of Structural Analyses of Outer Corrosion Barrier for Event Sequences (Continued)

Case	Description	Acceptance Criteria	Results	Comment
Rockfall on Waste Package in the Nonlithophysal Portions of the Subsurface	For the preclosure period, the drip shields have not yet been emplaced, so rocks in the nonlithophysal portions of the repository may fall onto the emplaced waste packages. Three waste package configurations for the license application are investigated to determine their structural response to rock fall sequences. A finite element analysis is performed by using the commercially available LS-DYNA finite element code.	Meet tiered screening criteria of Table 1.5.2-10.	The largest credible rockfall in the nonlithophysal portions of the repository is a 20-MT block. Bounding rockfall calculations were performed on the three flagship waste package configurations. The maximum stress intensities observed in the outer corrosion barrier were less than $\sigma_{int} < 0.7\sigma_u$. Consequently, the three representative waste packages meet Tier 1 of the screening criteria of Table 1.5.2-10.	A companion waste package capability analysis was performed for the rock fall analysis to determine the margin in the code compliance case. The capability of the waste package in response to the maximum credible preclosure emplacement drift rock fall in terms of the OCB's expended toughness fraction is 0.0993. That is, the outer corrosion barrier has received only 10% of the energy required for breach. Figure 1.5.2-12 shows the capacity of the waste package against failure given a rock fall event. This result is compared with the deterministic analysis of the OCB structural response to the same maximum credible preclosure rock fall which showed that the peak effective wall-averaged stress intensity at the governing location is 91% of the first tier level ductile rupture criterion (using the $0.7s_u$ Stress Intensity threshold of Table 1.5.2-10).

Table 1.5.2-9. Summary of Structural Analyses of Outer Corrosion Barrier for Event Sequences (Continued)

Case	Description	Acceptance Criteria	Results	Comment
Horizontal Drop on Emplacement Pallet	The purpose of this calculation is to determine the structural response of the Naval Long waste package with the emplacement pallet when they are subjected to event sequence impacts onto horizontal surfaces including the drift invert steel structure. This calculation includes considerations of noncentered and angled orientations of the waste package, emplacement pallet, and invert steel. This calculation can address either a drop by the TEV or a vertical impact within the emplacement drift due to a seismic event. This calculation addresses both emplacement pallet and invert steel contacts with the outer corrosion barrier between its end sleeves.	Meet tiered screening criteria of Table 1.5.2-10.	Considering realistic orientations between the waste package and emplacement pallet that can occur during a drop of the waste package on the emplacement pallet from the TEV or in the drifts during vertical motion dominated seismic events, the worst case orientation determined was for a 20 in., room temperature drop (3.15 m/s impact) of the waste package onto the worst case location of the emplacement pallet on a flat surface. The results indicate an effective wall averaged stress intensity value of 587 MPa. The ratio of this value to room temperature true tensile strength (σ_u = 971 MPa) is 0.60. This ratio satisfies the acceptance criteria in Table 1.5.2-10.	A companion expended toughness fraction analysis was performed for the 20 in. drop analysis to determine the margin in the code compliance case. The expended toughness fraction for the 20 in. drop is 0.25. That is, the outer corrosion barrier has received only 25% of the energy required for breach while the governing effective wall averaged stress intensity for the deterministic 20 in. drop is $0.6\sigma_u$ which has a minimum "damage fraction" of 0.67 (using the highest $0.9\sigma_u$ limit of Table 1.5.2-10). Figure 1.5.2-13 shows the capacity of the waste package given a horizontal drop on emplacement pallet.

NOTE: OCB = outer corrosion barrier.

Source: BSC 2007c, Section 7; BSC 2007d, Section 7; BSC 2007e, Section 7.

Table 1.5.2-10. Tiered Screening Criteria for Material Failure for Mechanical Loading

Tier	Criteria	Condition of Acceptance
1	Maximum σ_{int} <0.7σ_u? Answer No: Go to next Tier.	Answer Yes: Meets P_m and P_L limits without the need for wall averaging.
2	Maximum σ_{int} <0.77σ_u? No.	Yes: Meets P_L limit without the need for wall averaging, but the stress field must not be uniform around the entire circumference (only a concern for vertical drop events).
3	Maximum wall-averaged σ_{int} <0.7σ_u? No.	Yes: Meets P_m and P_L limits.
4	Maximum wall-averaged σ_{int} <0.77σ_u? No.	Yes: Meets P_L limit if the stress fields are not uniform around the entire circumference (only a concern for vertical drop events).
5	Maximum wall-averaged σ_{int} <0.9σ_u and wall-averaged σ_{int} <0.77σ_u at $\sqrt{R \bullet t}$ surrounding maximum location? No.	Yes: Meets P_L and average primary shear limit.
6	Perform more rigorous evaluation using all six stress components. Replace maximum wall-averaged σ_{int} in above simplified tiered criteria with component-based membrane stress intensity. OR Use multiple stress classification lines and extrapolation to avoid nonmembrane stress contributions in governing location. Meets criteria and condition of acceptance? No. Fails screening criteria, consider performing rigorous ASME code evaluation (ASME 2001) using quantitative instead of bounding stress classifications.	Yes: Meets screening criteria.

NOTE: P_L is local primary membrane stress intensity; P_m is general primary membrane stress intensity; R is median wall radius; t is wall thickness; τ is shear stress; σ_{int} is stress intensity (equal to twice maximum shear stress τ); σ_u is true tensile strength.

Source: BSC 2007a, Section 7.1.4.

Table 1.5.2-11. Alloy 22 Design Properties

Design Properties	Room Temperature Strength (MPa)
S_y	310
S_u	689
σ_u	971
0.7 σ_u	680
0.77 σ_u	748
0.84 σ_u	816
0.9 σ_u	874

NOTE: Design properties are obtained or interpolated from the ASME code, Section II, Part D, "Properties," Tables U, Y-1, and 1B (ASME 2001) for Alloy 22 plate. The allowable design limit stress for Alloy 22 is 300 MPa at 250°C for load-bearing components such as emplacement pallets (ASME 2001, Code Cases, Nuclear N-621, Table 2). However, a more conservative design limit of 257 MPa is used for the waste package outer barrier based on corrosion considerations for postclosure analysis. For event sequences, room temperature values are used. Lower temperatures are slightly more challenging to the structural performance of the waste packages than higher temperatures. This is due to the increased elongation and disproportionate reduction in yield versus ultimate strength at elevated temperatures.
Design properties are shown in megapascals (MPa). $\sigma_u = S_u (1 + e_u)$, where S_u is the engineering tensile strength and e_u is the corresponding engineering strain.
S_y = engineering yield strength which is approximately equal to true yield strength (σ_y); σ_u = true tensile strength.

Table 1.5.2-12. Radiation Shielding of Inner Lid

Waste Package Configuration	Inner Lid Thickness (in.)	Dose Rate on Contact (mrem/hr)
5-DHLW/DOE Short Codisposal	9	~60
5-DHLW/DOE Long Codisposal	9	~60
2-MCO/2-DHLW	8	~50

Table 1.5.2-13. Nondestructive Examination of Major Fabrication Welds of the Waste Package

Name of Weld	Acceptance Standards[a]
Inner Vessel Longitudinal Weld	Radiographic Examination (NC-5320) Liquid Dye Penetrant Testing (NC-5350)
Inner Vessel Circumferential Weld	Radiographic Examination (NC-5320) Liquid Dye Penetrant Testing (NC-5350)
Inner Vessel Bottom Lid Weld	Radiographic Examination (NC-5320) Liquid Dye Penetrant Testing (NC-5350)
Divider Plate Assembly Weld to Inner Vessel	Liquid Dye Penetrant Testing (NC-5350)
Outer Corrosion Barrier Longitudinal Weld	Radiographic Examination (NC-5320, with maximum project acceptable indication length 1/16 in.) Liquid Dye Penetrant Testing (NC-5350) Ultrasonic Examination (NC-5330, with maximum project acceptable indication length 1/16 in.)
Outer Corrosion Barrier Circumferential Weld	Radiographic Examination (NC-5320, with maximum project acceptable indication length 1/16 in.) Liquid Dye Penetrant Testing (NC-5350) Ultrasonic Examination (NC-5330, with maximum project acceptable indication length 1/16 in.)
Outer Corrosion Barrier Bottom Lid Weld	Radiographic Examination (NC-5320, with maximum project acceptable indication length 1/16 in.) Liquid Dye Penetrant Testing (NC-5350) Ultrasonic Examination (NC-5330, with maximum project acceptable indication length 1/16 in.)
Upper Sleeve to Outer Corrosion Barrier	Radiographic Examination (NC-5320, with maximum project acceptable indication length 0.04 in.) Liquid Dye Penetrant Testing (NC-5350) Ultrasonic Examination (NC-5330, with maximum project acceptable indication length 0.04 in.)
Lower Sleeve to Outer Corrosion Barrier	Liquid Dye Penetrant Testing (NC-5350)
Inner Vessel Support Ring to Outer Corrosion Barrier	Liquid Dye Penetrant Testing (NC-5350)
Inner Vessel Lid Lifting Feature Weld	Liquid Dye Penetrant Testing (NC-5350)
Outer Lid Lifting Feature Weld	Liquid Dye Penetrant Testing (NC-5350)

NOTE: [a]ASME code, Section III, Division 1, Subsection NC (ASME 2001).

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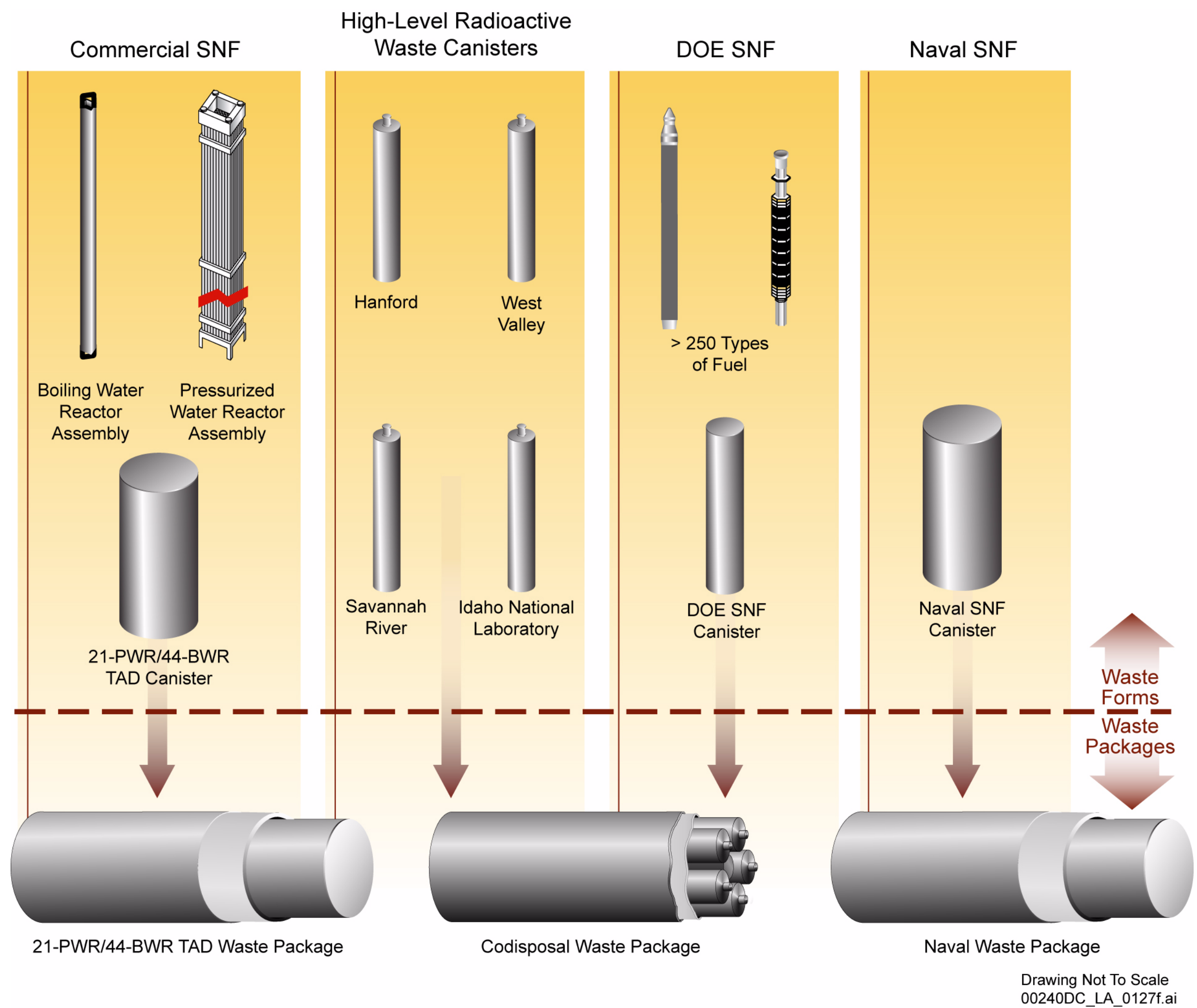


Figure 1.5.2-1. Waste Form and Waste Package Configurations

NOTE: The figure depicts the types of waste forms to be disposed of in the repository and their representative waste package configurations. Section 1.5.1.4 provides information on naval fuel.

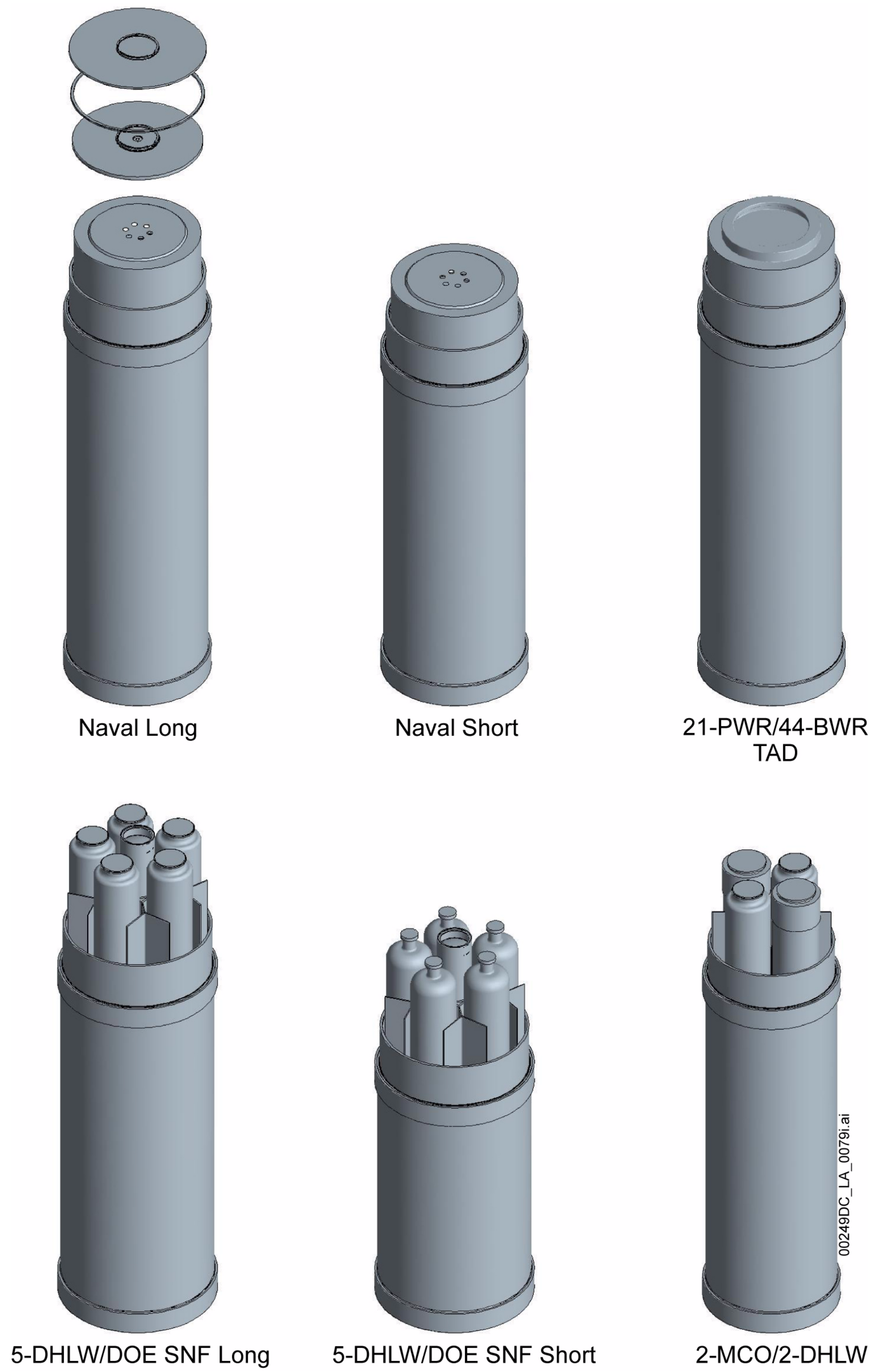
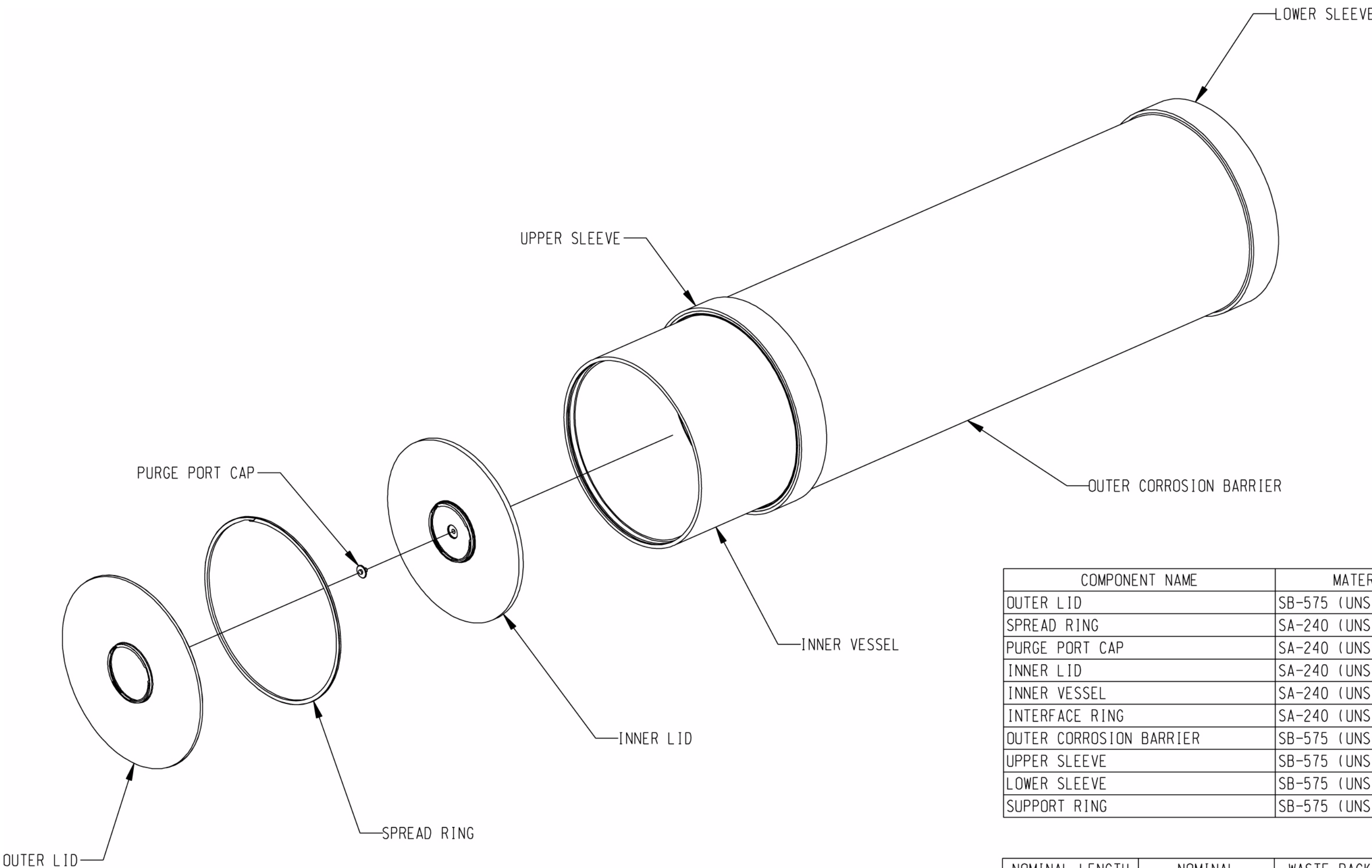


Figure 1.5.2-2. Waste Package Configurations

NOTE: The waste package configurations are depicted close to scale, along with the waste-form configuration contained in each design.



COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
INNER LID	SA-240 (UNS S31600)	1
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOTES:

1. ALL DIMENSIONS ARE REFERENCE ONLY.

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
230.32 IN	77.28 IN	53,500 LBM	162,000 LBM
5850.1 MM	1962.9 MM	24,300 KG	73,500 KG

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Figure 1.5.2-3. Transportation, Aging, and Disposal Waste Package Configuration (Sheet 1 of 3)

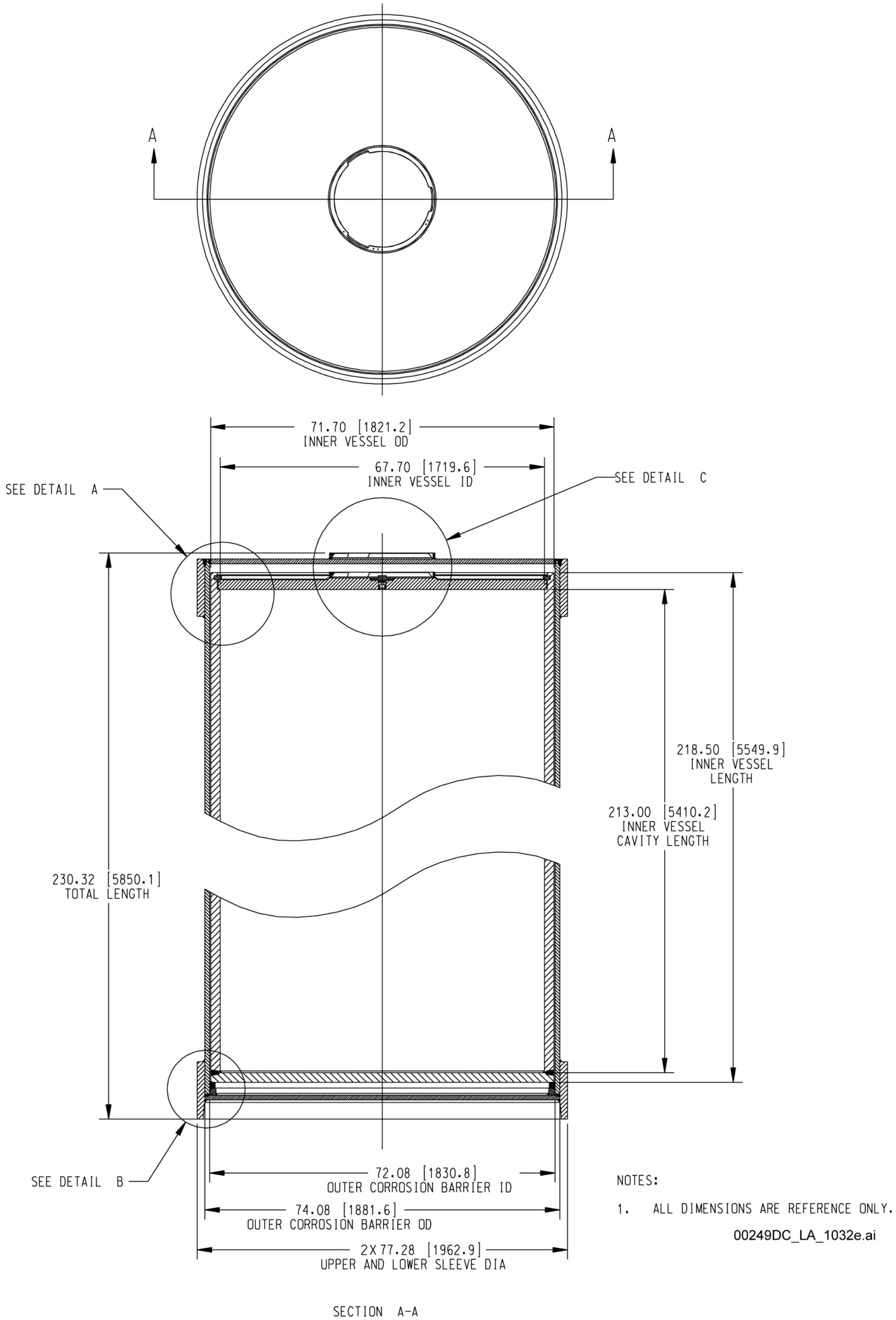


Figure 1.5.2-3. Transportation, Aging, and Disposal Waste Package Configuration (Sheet 2 of 3)

NOTE: Details A, B, and C are shown on Figure 1.5.2-3 on Sheet 3. All dimensions are in inches, and millimeters are in brackets. ID = inner diameter; OD = outer diameter.

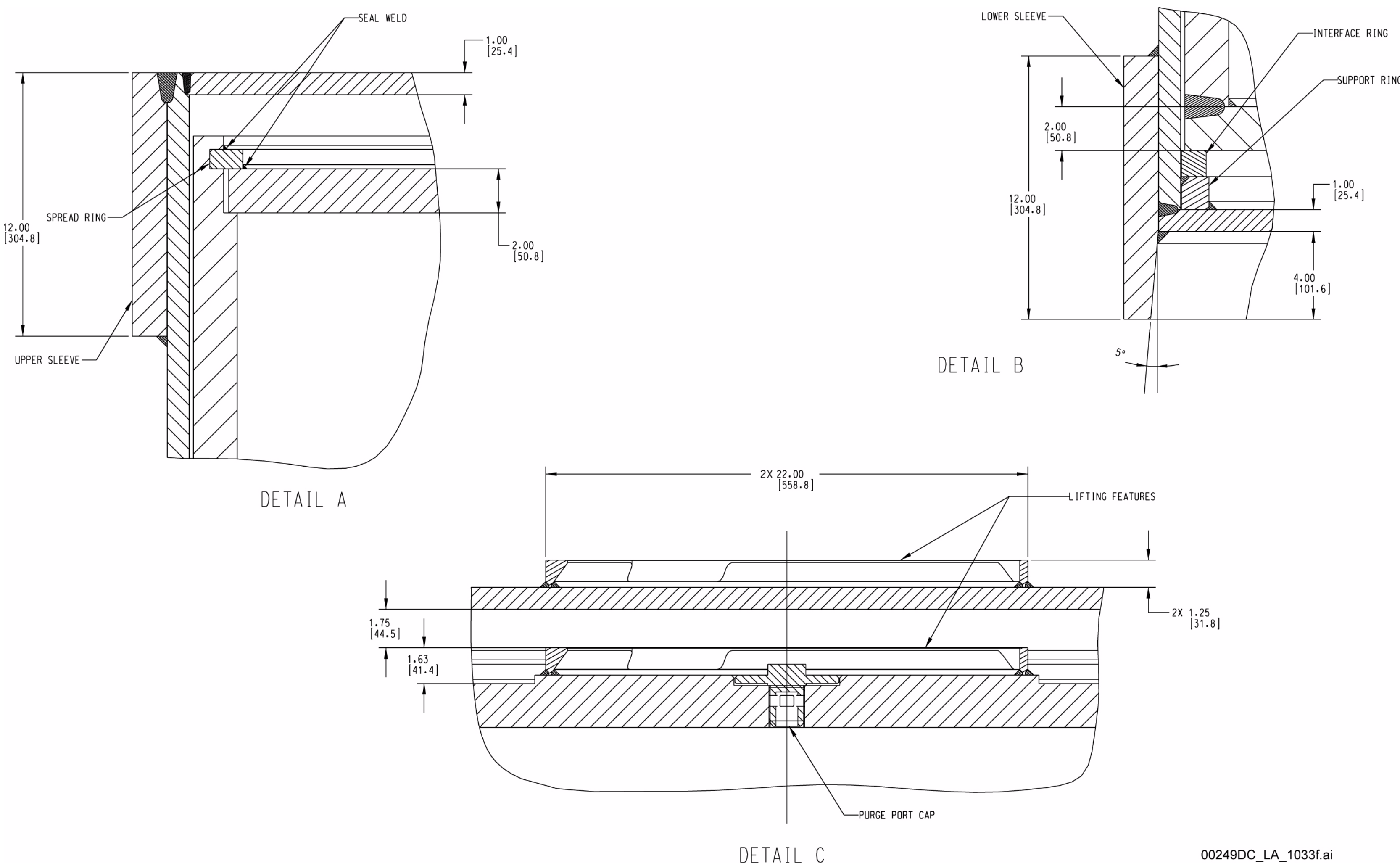
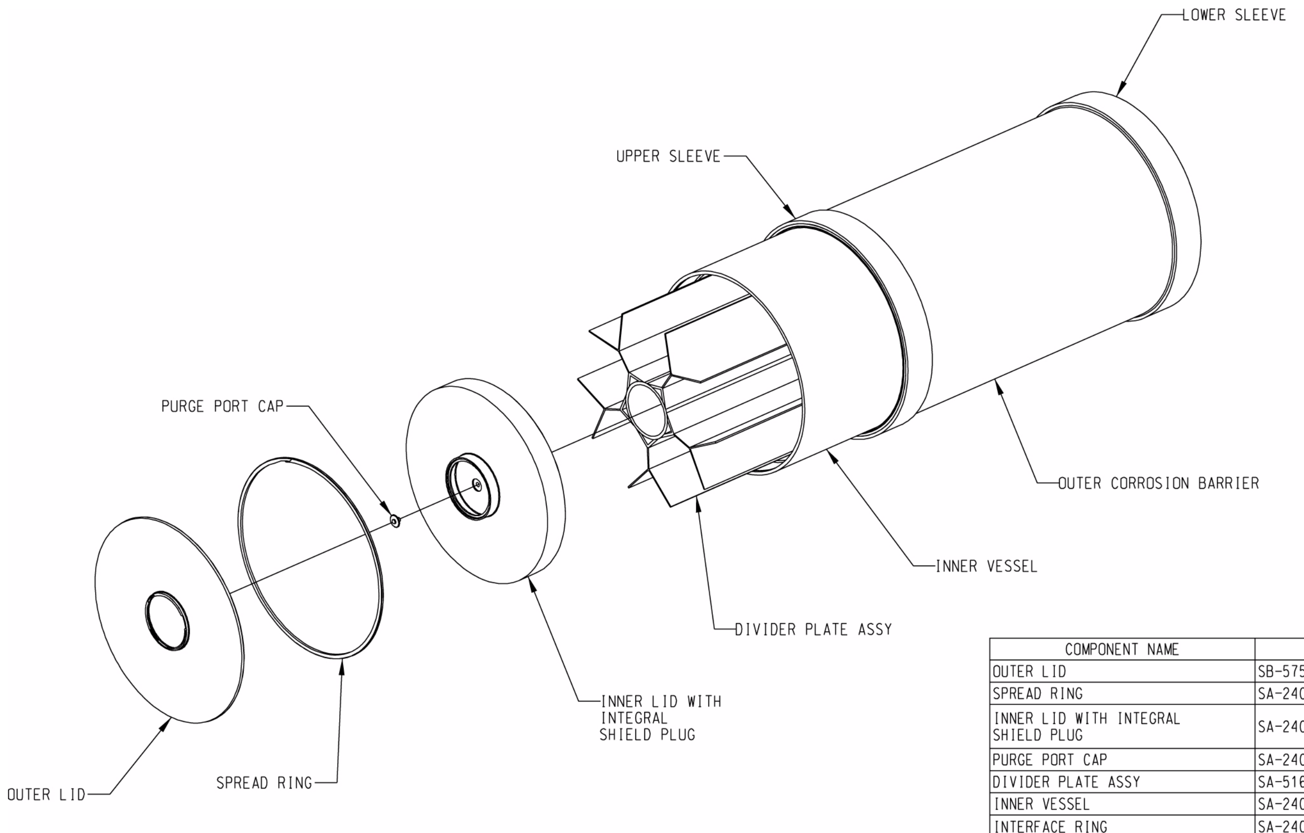


Figure 1.5.2-3. Transportation, Aging, and Disposal Waste Package Configuration (Sheet 3 of 3)

NOTE: All dimensions are in inches, and millimeters are in brackets.



COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
INNER LID WITH INTEGRAL SHIELD PLUG	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
DIVIDER PLATE ASSY	SA-516 (UNS K02700)	1
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOTES:

1. ALL DIMENSIONS ARE REFERENCE ONLY.
2. THE WASTE PACKAGE CAN BE LOADED WITH ONE 24-IN DOE SNF CANISTER IN A PERIPHERAL LOCATION IF THE CENTER LOCATION IS EMPTY. THE REMAINING FOUR PERIPHERAL LOCATIONS ARE THEN LOADED WITH HLW CANISTERS.

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
145.57 IN	83.70 IN	57,400 LBM	90,000 LBM
3697.4 MM	2126.0 MM	26,100 KG	40,800 KG

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Figure 1.5.2-4. 5-DHLW/DOE Short Codisposal (Sheet 1 of 3)

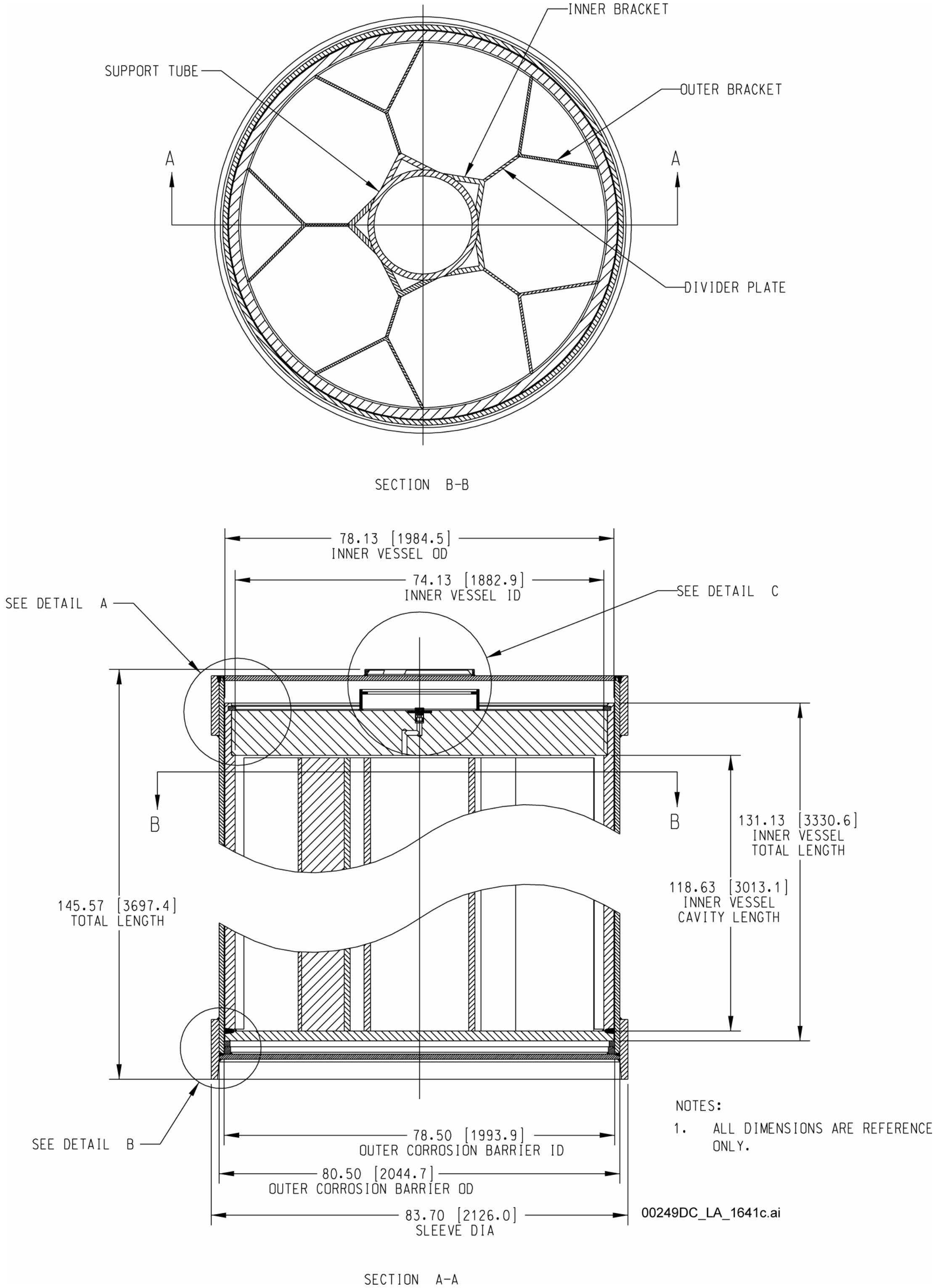


Figure 1.5.2-4. 5-DHLW/DOE Short Codisposal (Sheet 2 of 3)

NOTE: Details A, B, and C are shown on Figure 1.5.2-4 on Sheet 3. All dimensions are in inches, and millimeters are in brackets. ID = inner diameter; OD = outer diameter.

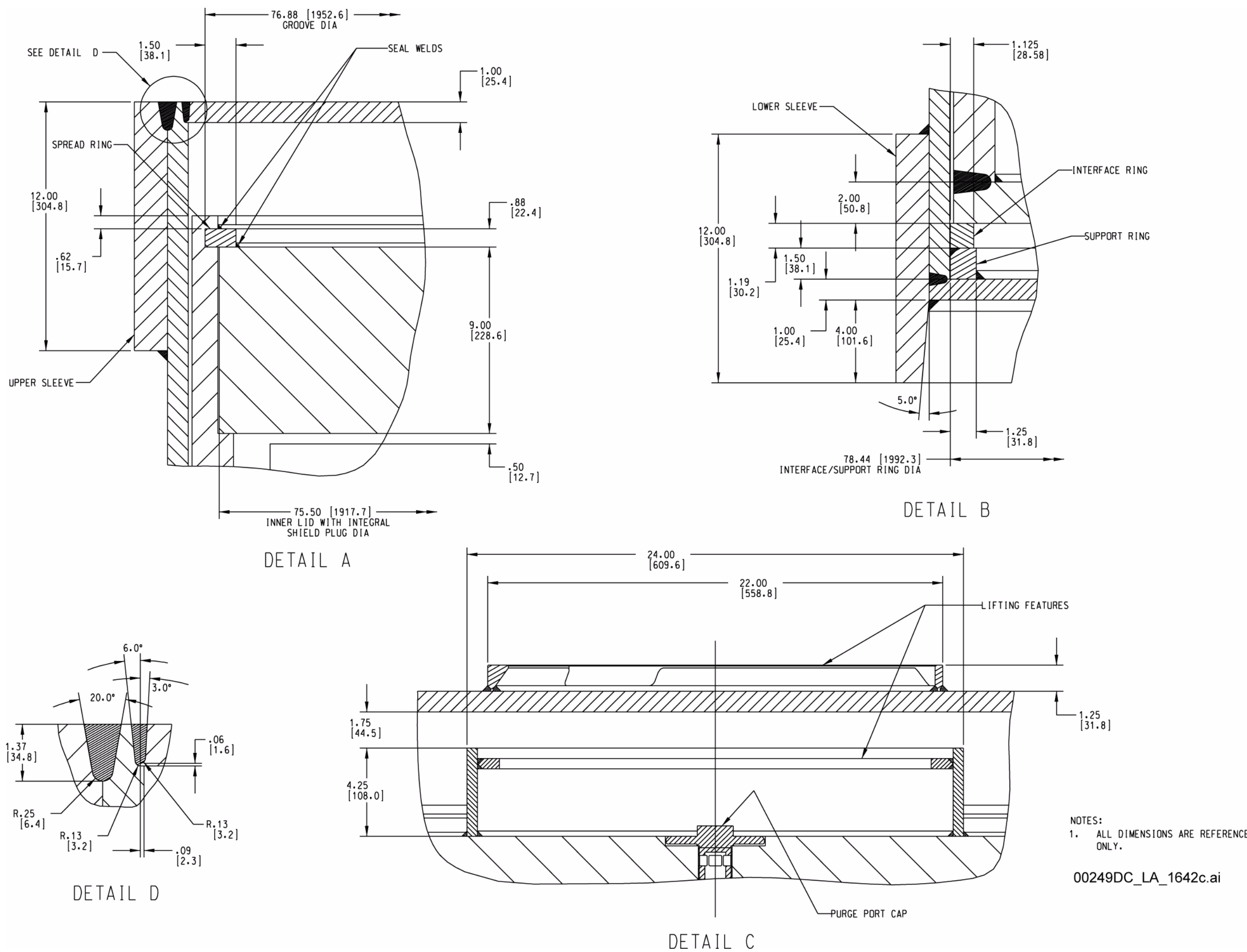
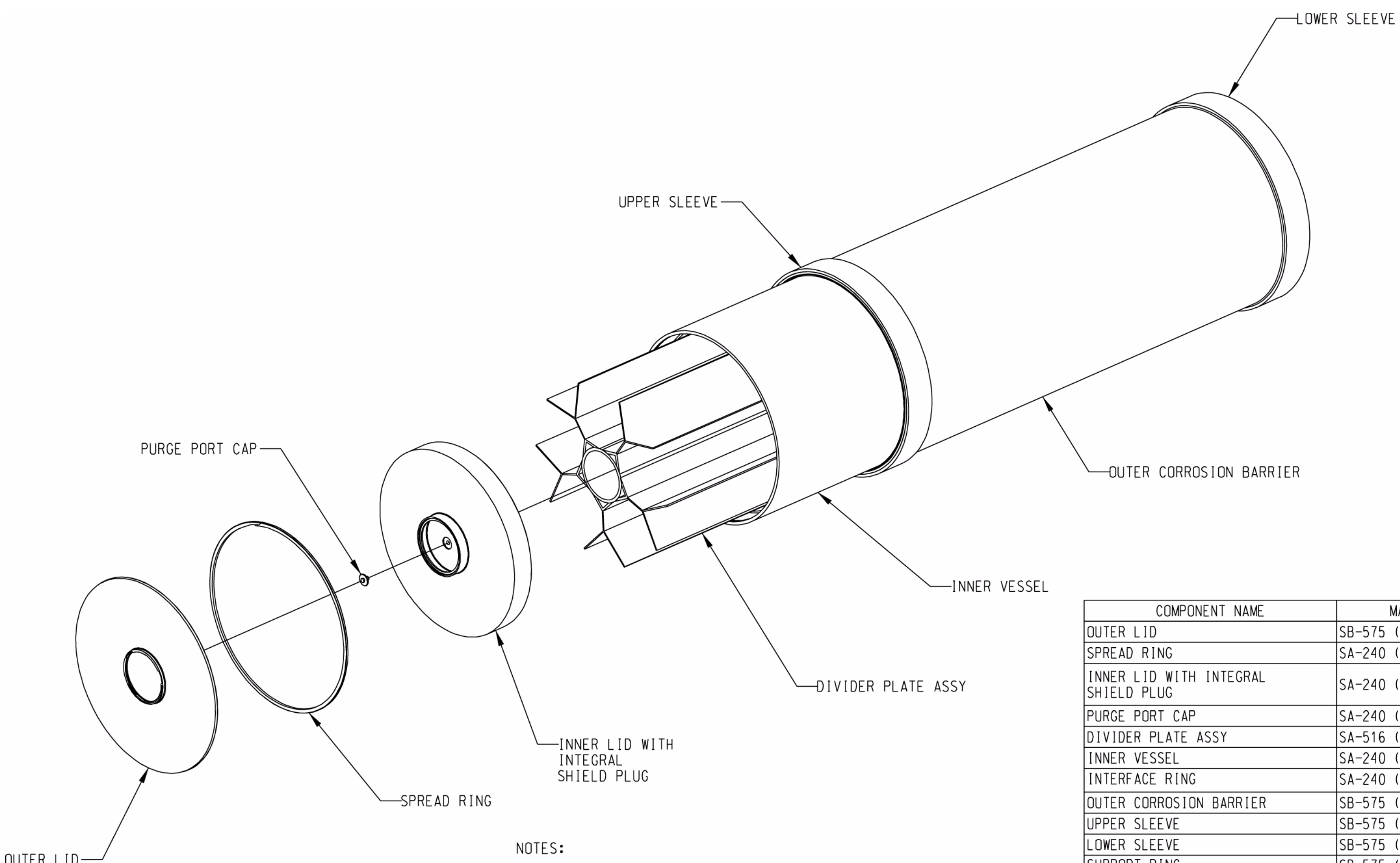


Figure 1.5.2-4. 5-DHLW/DOE Short Codisposal (Sheet 3 of 3)

NOTE: All dimensions are in inches, and millimeters are in brackets.



NOTES:

1. ALL DIMENSIONS ARE REFERENCE ONLY.
2. THE WASTE PACKAGE CAN BE LOADED WITH ONE 24-IN DOE SNF CANISTER IN A PERIPHERAL LOCATION IF THE CENTER LOCATION IS EMPTY. THE REMAINING FOUR PERIPHERAL LOCATIONS ARE THEN LOADED WITH HLW CANISTERS.
THE LOADED MASS WAS CALCULATED USING THE STANDARD FIVE HLW AND ONE DOE SNF CONFIGURATION.

COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
INNER LID WITH INTEGRAL SHIELD PLUG	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
DIVIDER PLATE ASSY	SA-516 (UNS K02700)	1
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
208.82 IN	83.70 IN	75,600 LBM	127,900 LBM
5303.9 MM	2126.0 MM	34,300 KG	58,000 KG

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Figure 1.5.2-5. 5-DHLW/DOE Long Codisposal (Sheet 1 of 3)

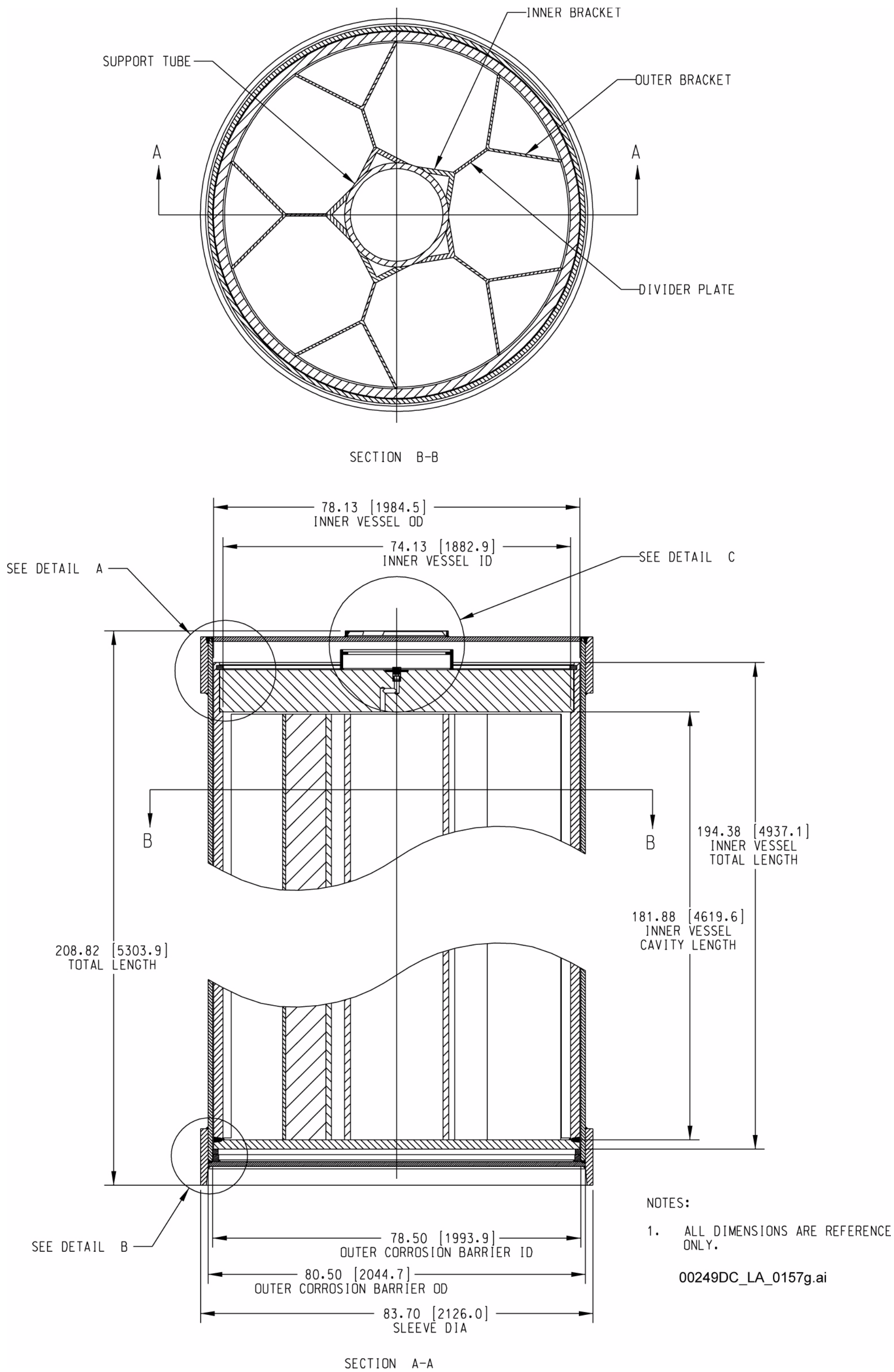


Figure 1.5.2-5. 5-DHLW/DOE Long Codisposal (Sheet 2 of 3)

NOTE: Details A, B, and C are shown on Figure 1.5.2-5 on Sheet 3. All dimensions are in inches, and millimeters are in brackets. ID = inner diameter; OD = outer diameter.

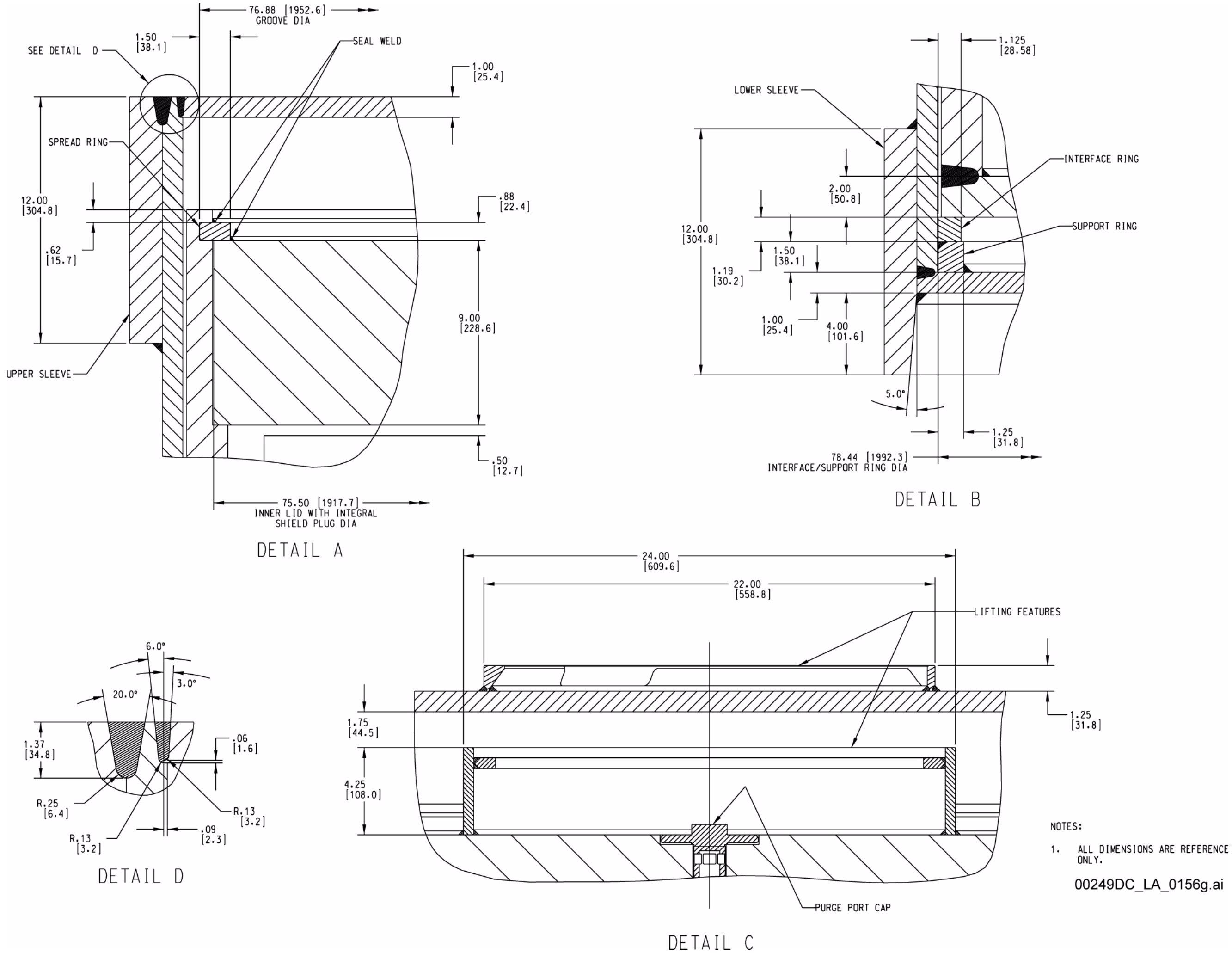
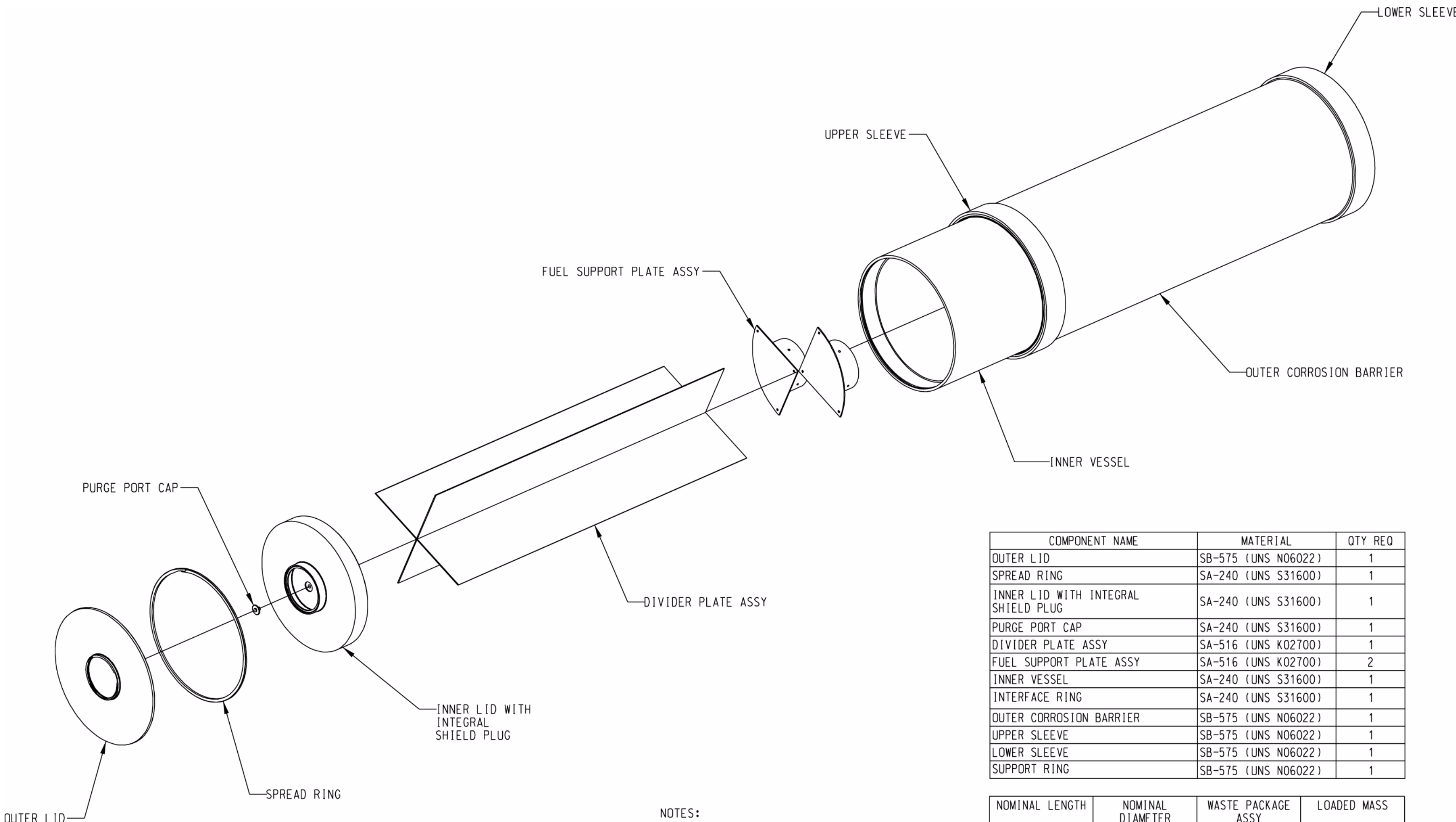


Figure 1.5.2-5. 5-DHLW/DOE Long Codisposal (Sheet 3 of 3)

NOTE: All dimensions are in inches, and millimeters are in brackets. R = radius.



COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
INNER LID WITH INTEGRAL SHIELD PLUG	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
DIVIDER PLATE ASSY	SA-516 (UNS K02700)	1
FUEL SUPPORT PLATE ASSY	SA-516 (UNS K02700)	2
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOTES:

1. ALL DIMENSIONS ARE REFERENCE ONLY.

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
207.82 IN	72.07 IN	53,600 LBM	112,500 LBM
5278.6 MM	1830.6 MM	24,300 KG	51,100 KG

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Figure 1.5.2-6. 2-MCO/2-DHLW (Sheet 1 of 3)

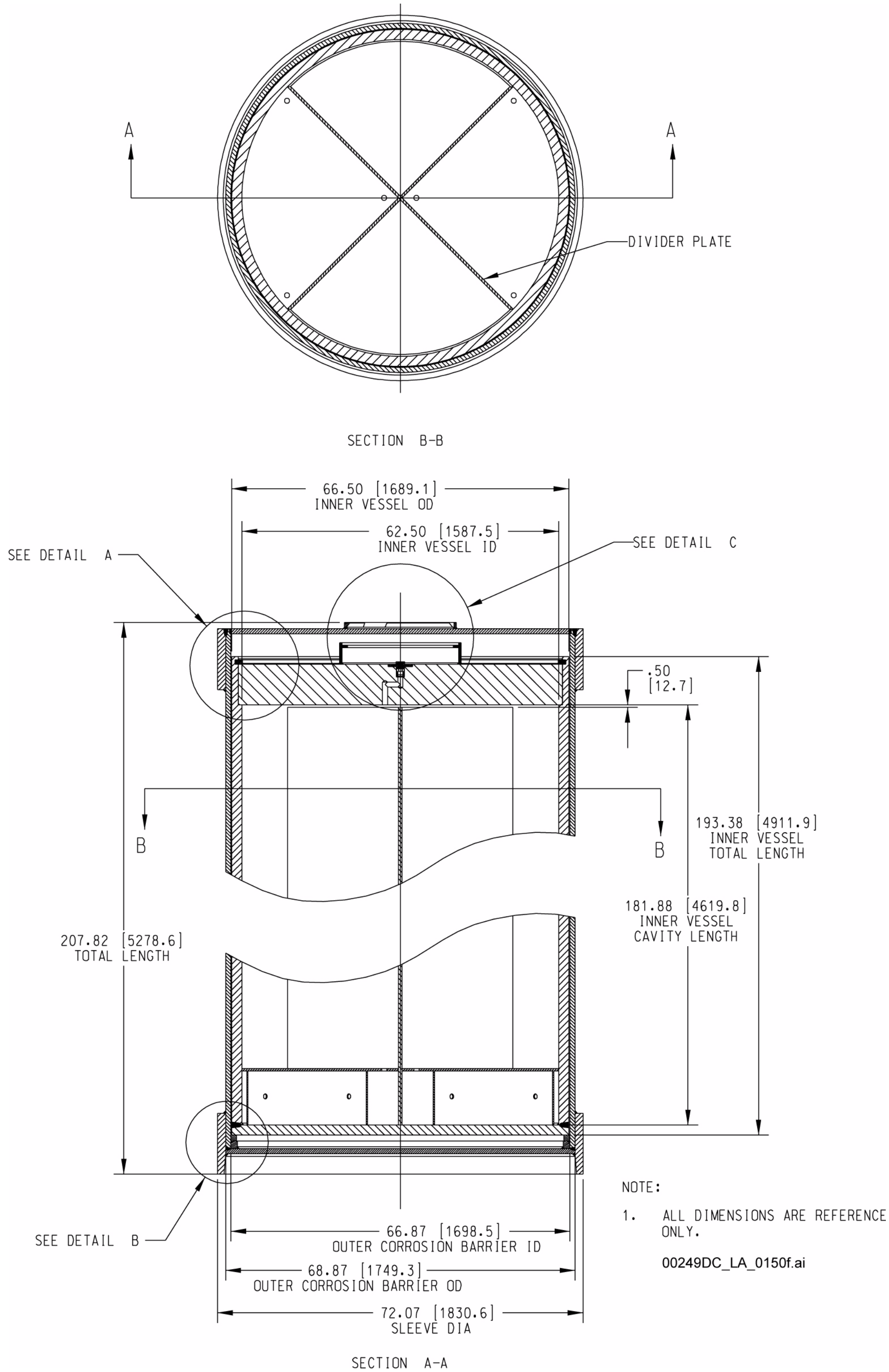


Figure 1.5.2-6. 2-MCO/2-DHLW (Sheet 2 of 3)

NOTE: Details A, B, and C are shown on Figure 1.5.2-6 on Sheet 3. All dimensions are in inches, and millimeters are in brackets. ID = inner diameter; OD = outer diameter.

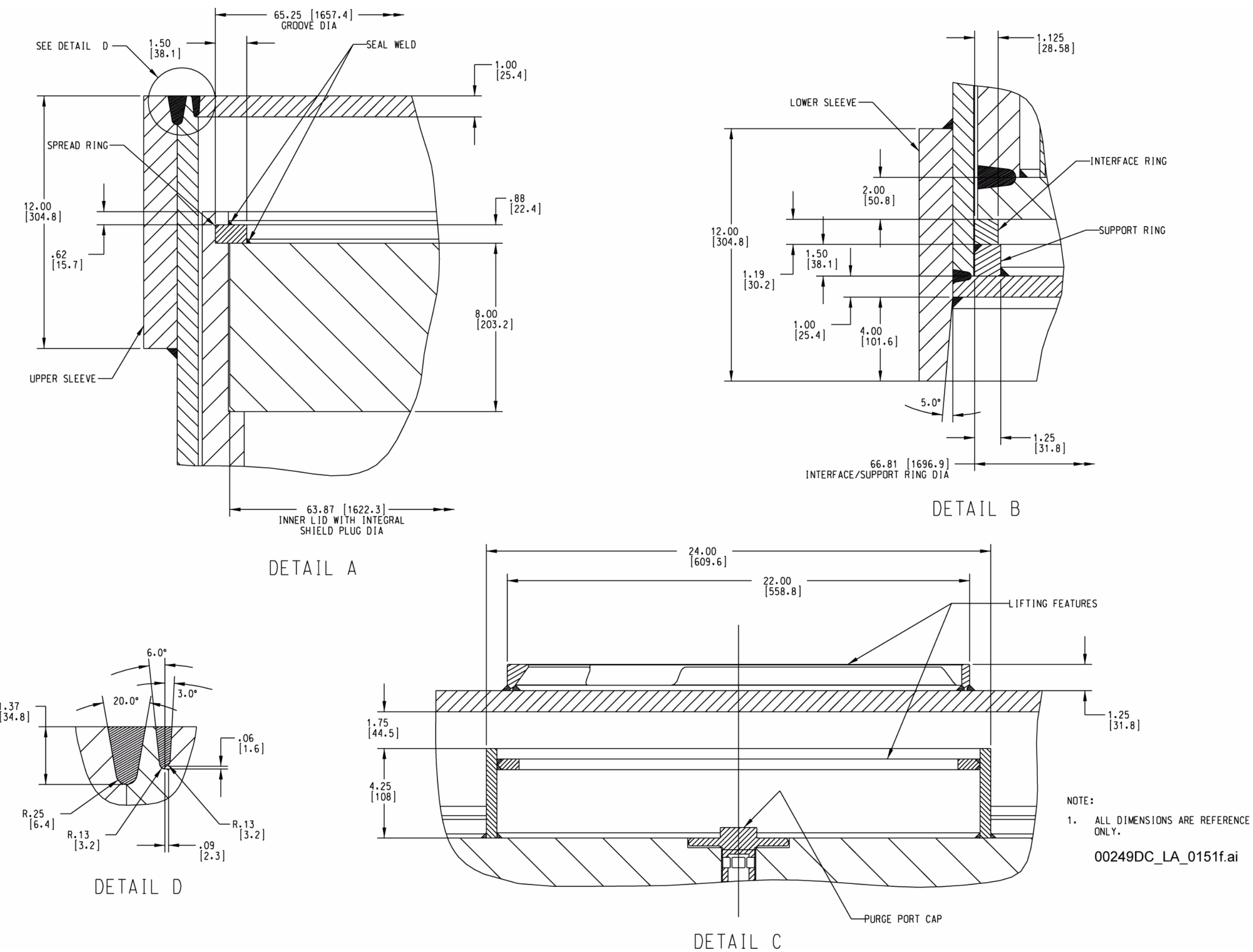
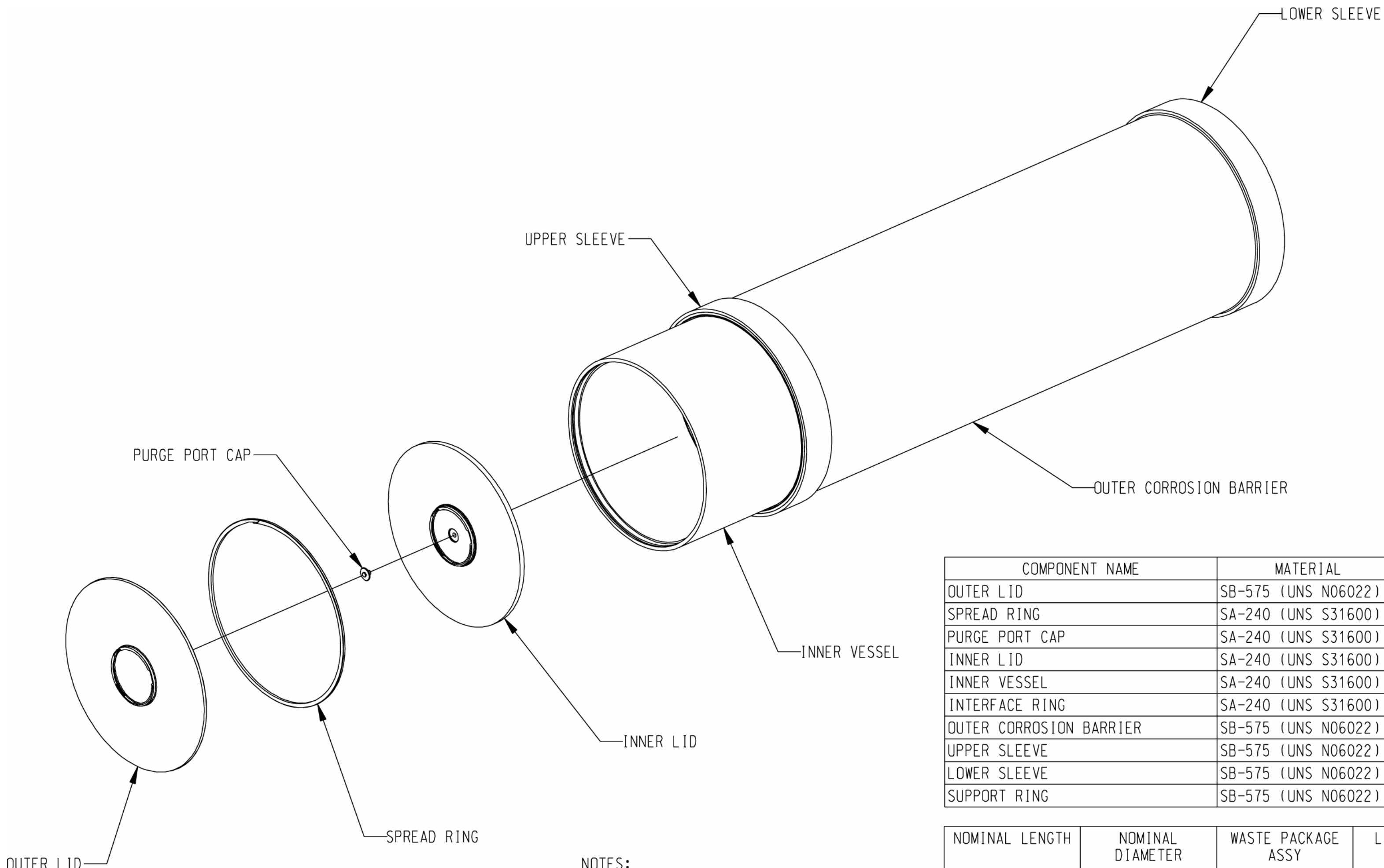


Figure 1.5.2-6. 2-MCO/2-DHLW (Sheet 3 of 3)

NOTE: All dimensions are in inches, and millimeters are in brackets.



COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
INNER LID	SA-240 (UNS S31600)	1
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
230.32 IN	77.28 IN	53,500 LBM	162,000 LBM
5850.1 MM	1962.9 MM	24,300 KG	73,500 KG

NOTES:

1. ALL DIMENSIONS ARE REFERENCE ONLY.

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Figure 1.5.2-7. Naval Long Waste Package Configuration (Sheet 1 of 3)

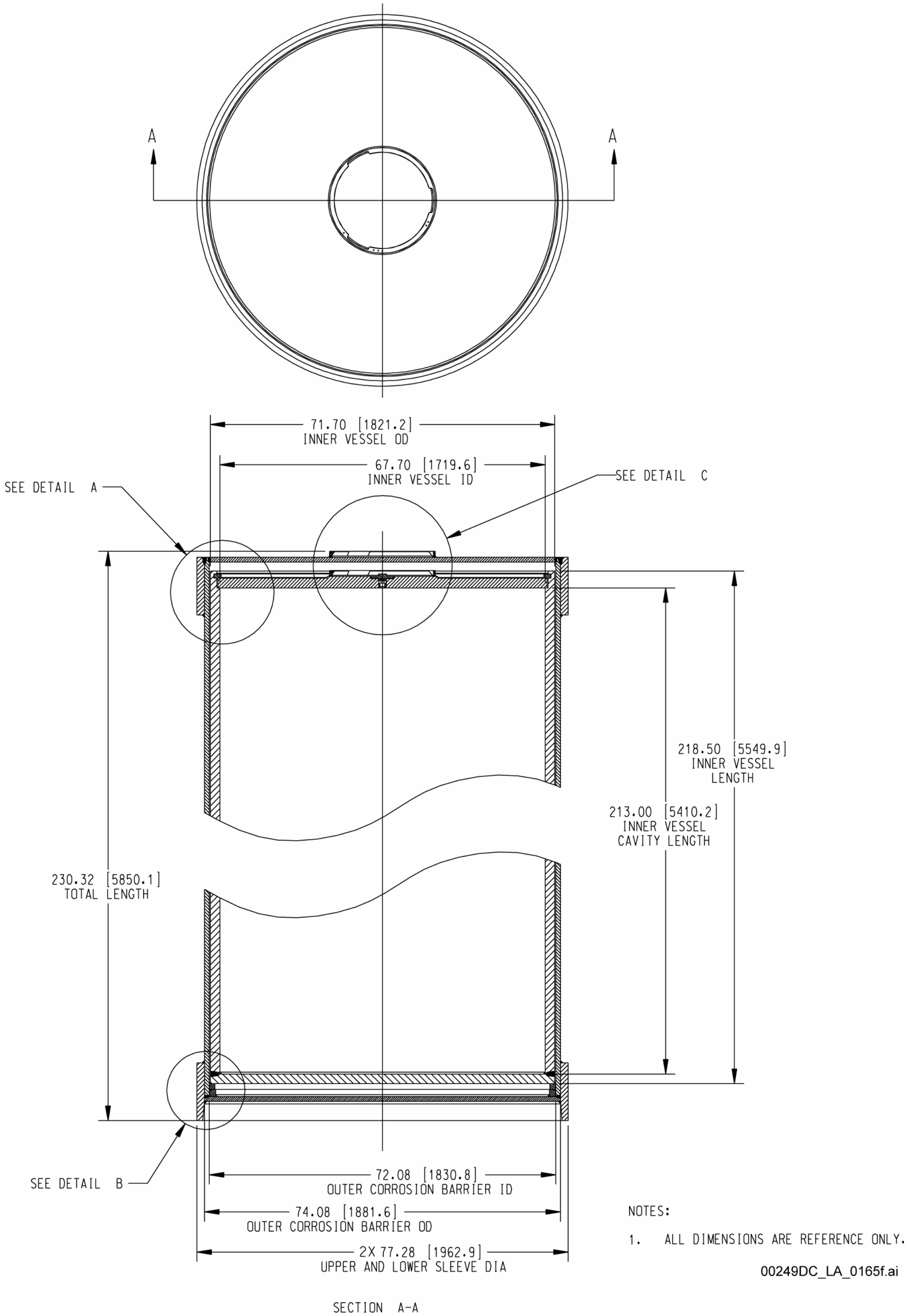


Figure 1.5.2-7. Naval Long Waste Package Configuration (Sheet 2 of 3)

NOTE: Details A, B, and C are shown on Figure 1.5.2-7 on Sheet 3. All dimensions are in inches, and millimeters are in brackets. ID = inner diameter; OD = outer diameter.

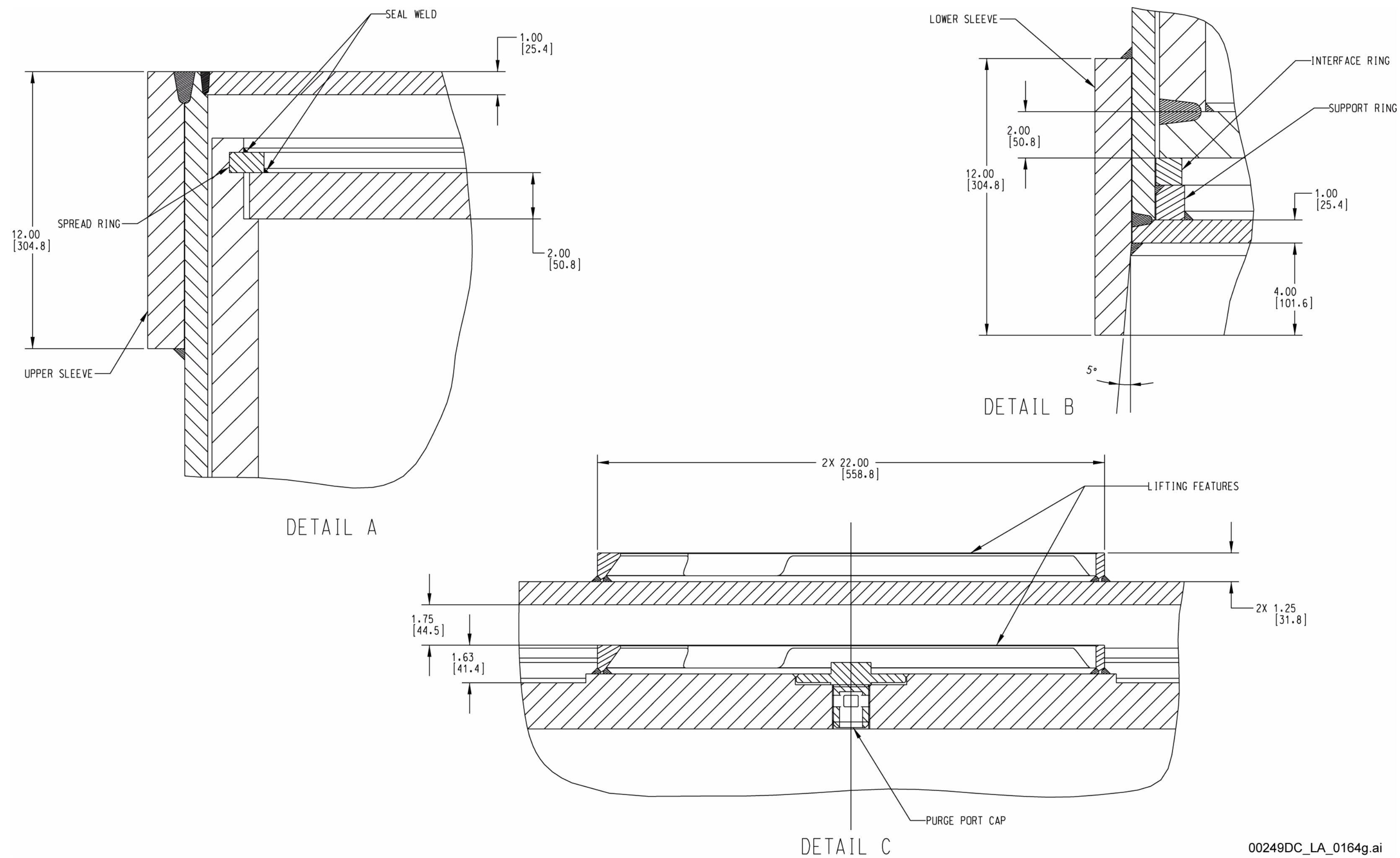
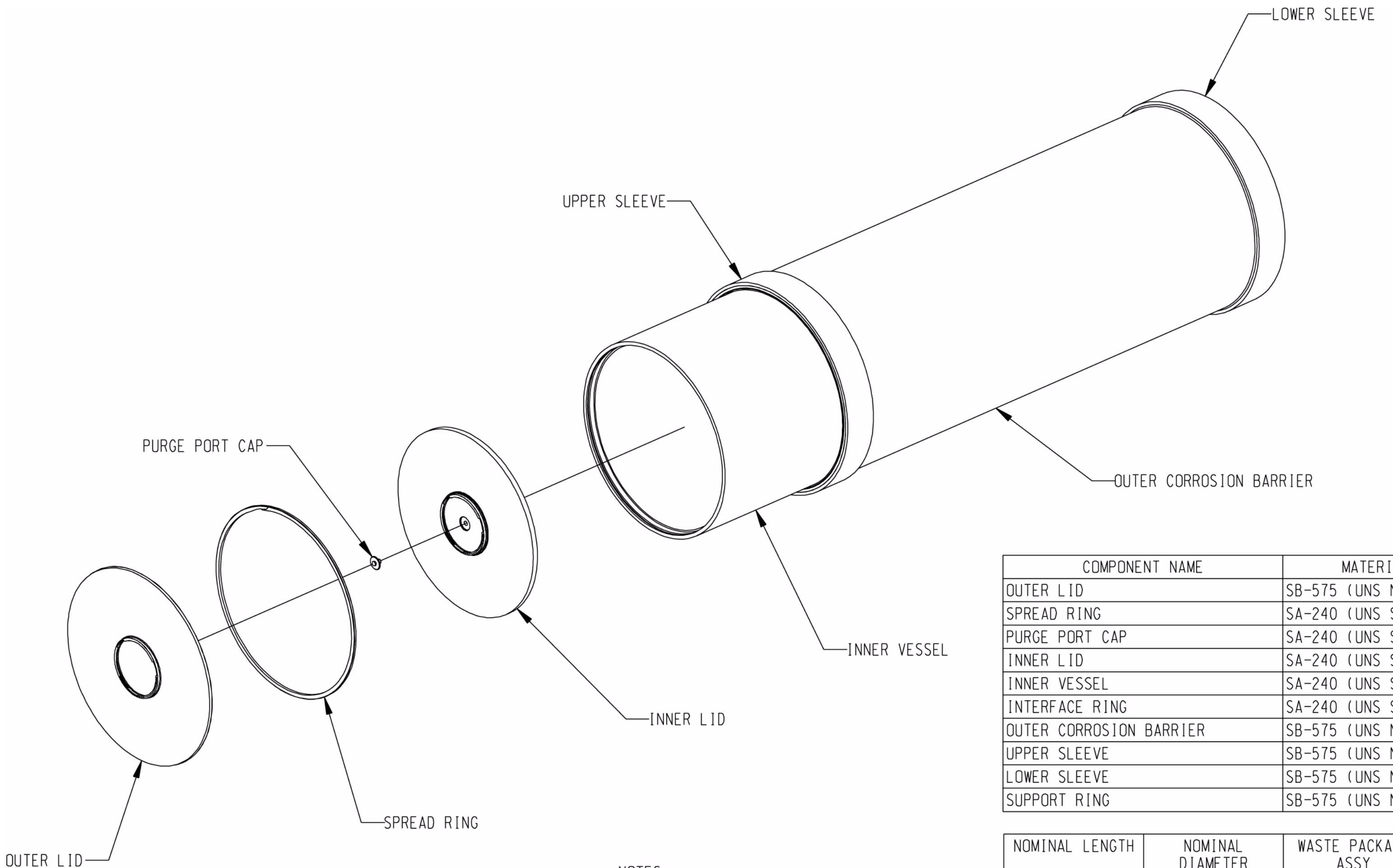


Figure 1.5.2-7. Naval Long Waste Package Configuration (Sheet 3 of 3)

NOTE: All dimensions are in inches, and millimeters are in brackets.



COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
INNER LID	SA-240 (UNS S31600)	1
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOTES:

1. ALL DIMENSIONS ARE REFERENCE ONLY.

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
205.32 IN	77.28 IN	48,500 LBM	157,000 LBM
5215.1 MM	1962.9 MM	22,000 KG	71,200 KG

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Figure 1.5.2-8. Naval Short Waste Package Configuration (Sheet 1 of 3)

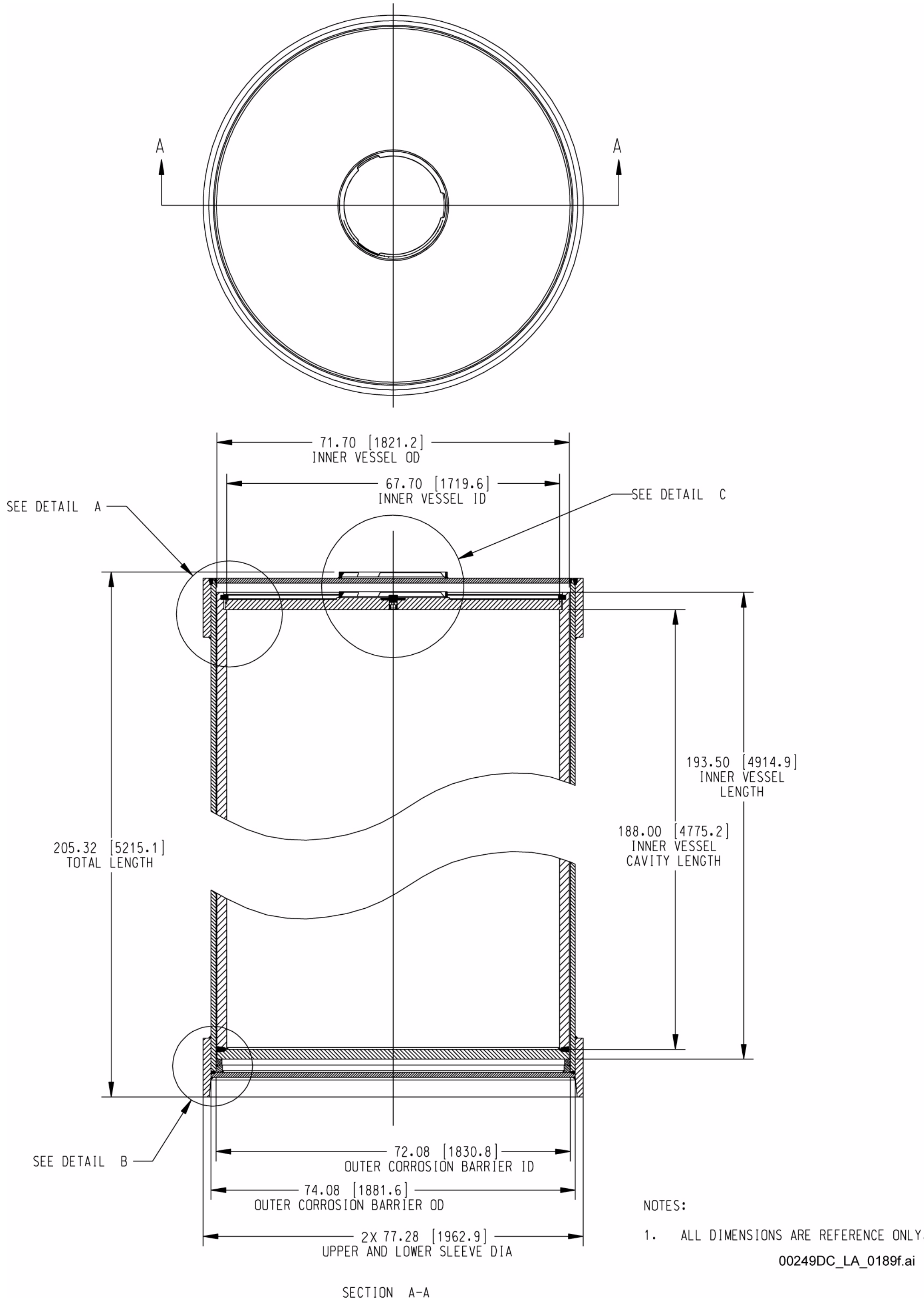

Figure 1.5.2-8. Naval Short Waste Package Configuration (Sheet 2 of 3)

NOTE: Details A, B, and C are shown on Figure 1.5.2-8 on Sheet 3. All dimensions are in inches, and millimeters are in brackets. ID = inner diameter; OD = outer diameter.

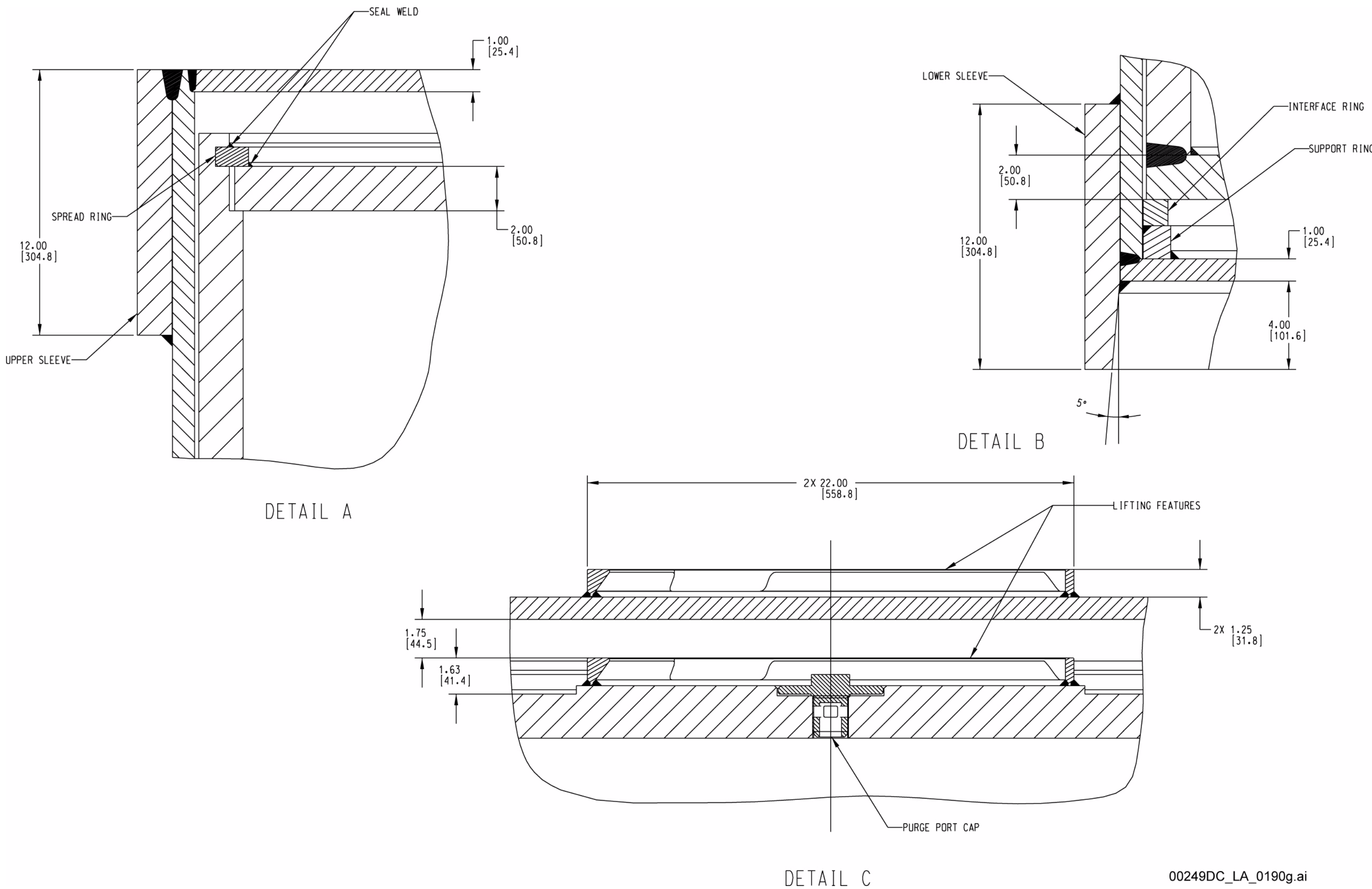

Figure 1.5.2-8. Naval Short Waste Package Configuration (Sheet 3 of 3)

NOTE: All dimensions are in inches, and millimeters are in brackets.

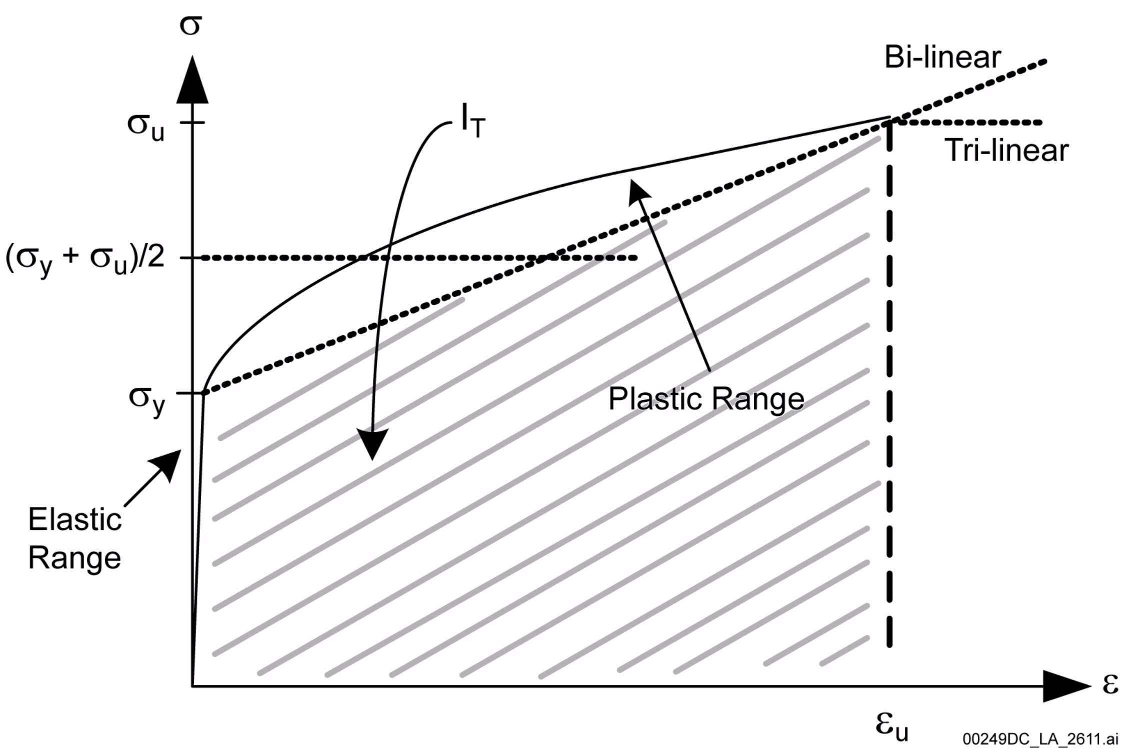

Figure 1.5.2-9. Toughness Index Representation

NOTE: I_T = toughness index.

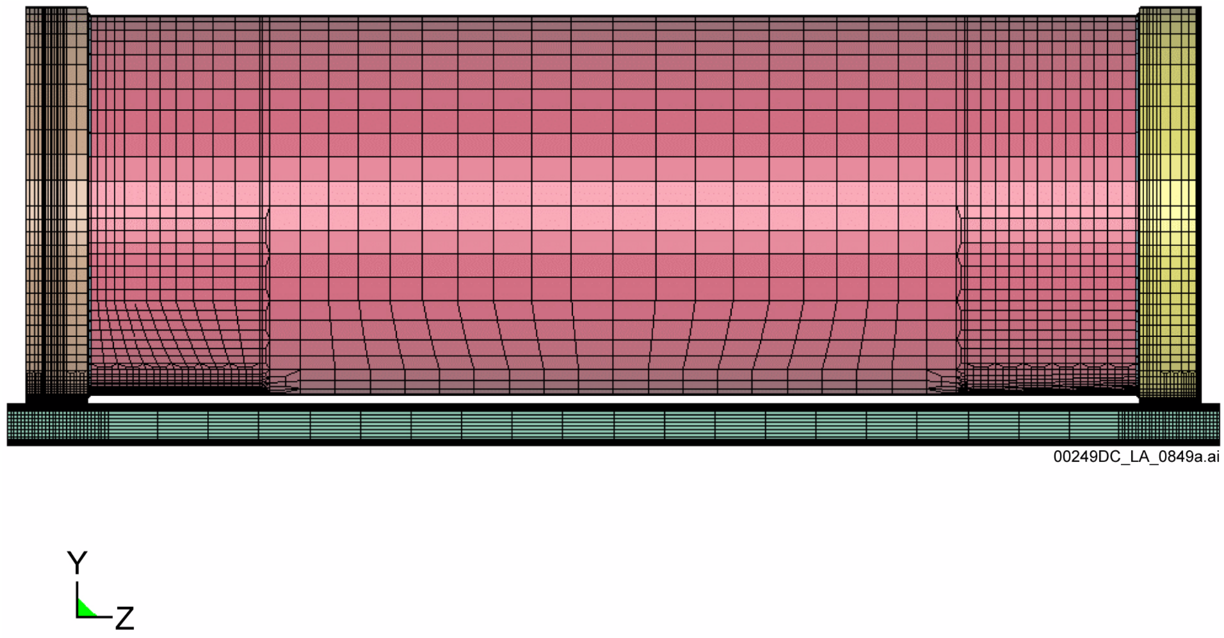

Figure 1.5.2-10. Finite-Element Representation for Typical Structural Analysis

Source: BSC 2007f, Figure 6-5.

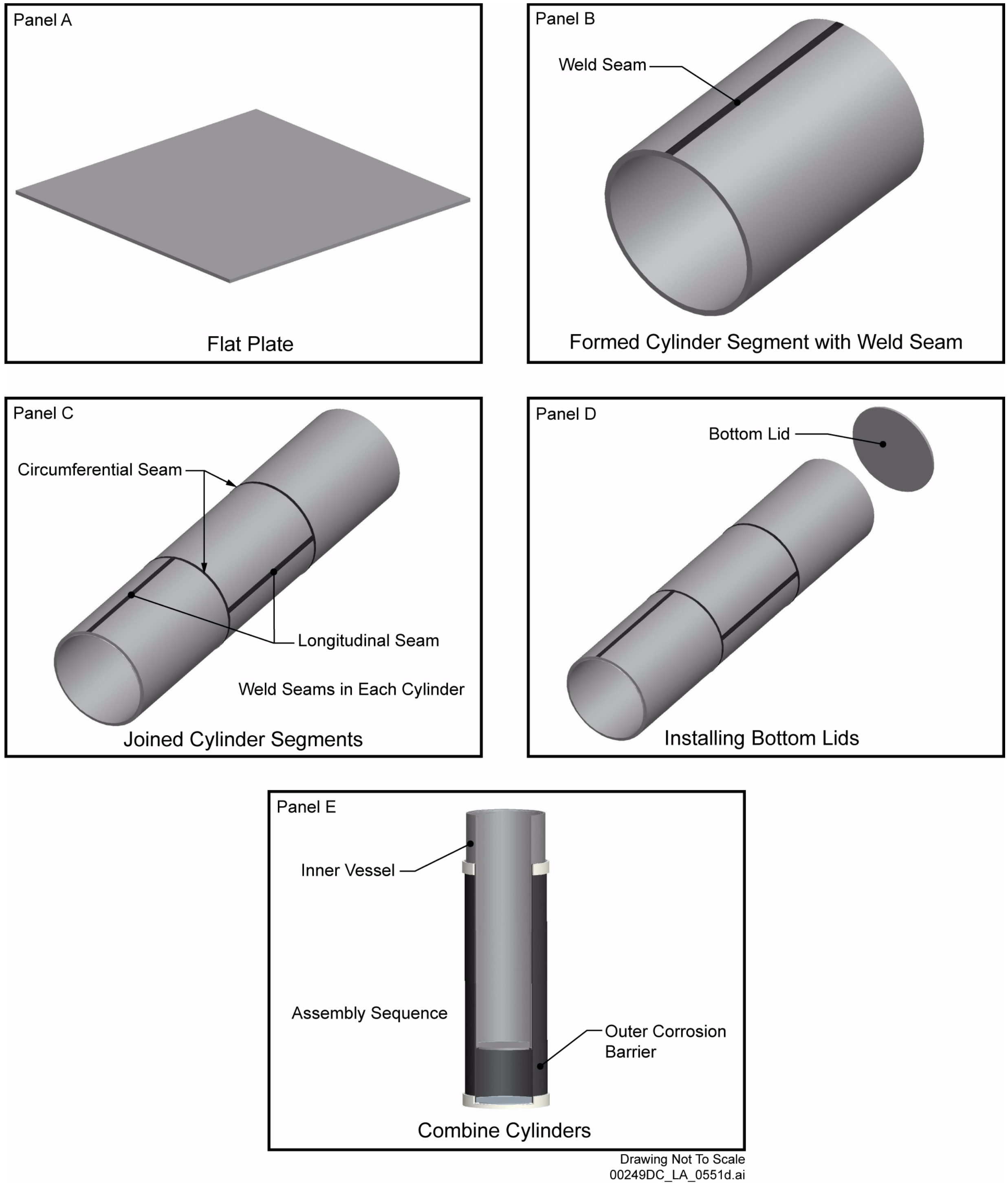

Figure 1.5.2-11. Fabrication of the Waste Package

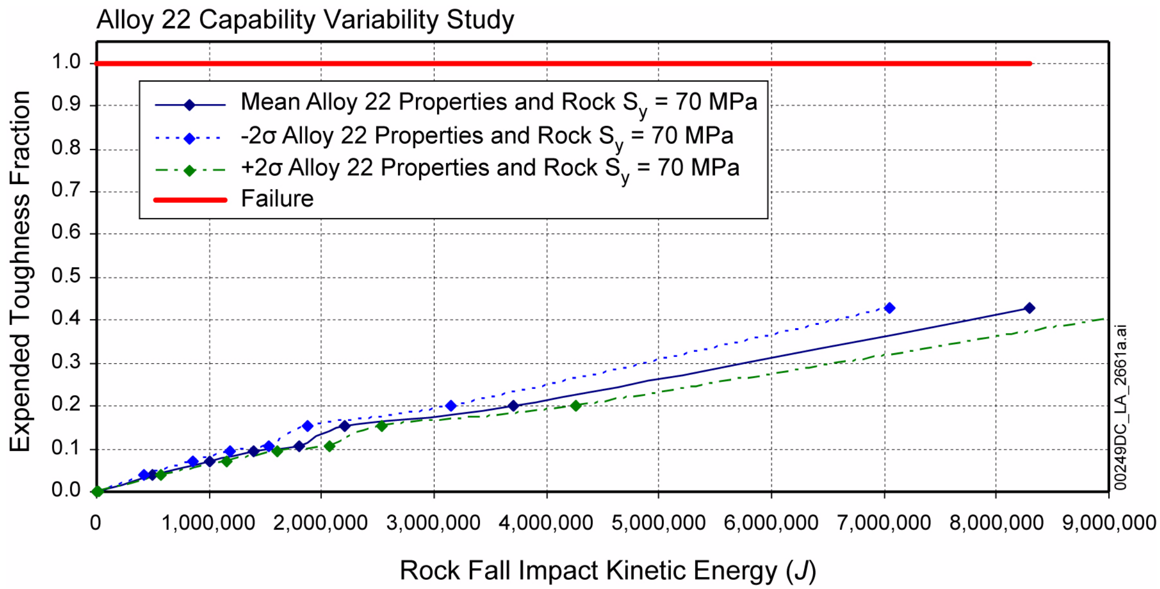

Figure 1.5.2-12. Mean and 2σ Outer Corrosion Barrier Capability for Lower Waste Package Sleeve Rock Impacts on TAD Canister–Bearing Waste Package

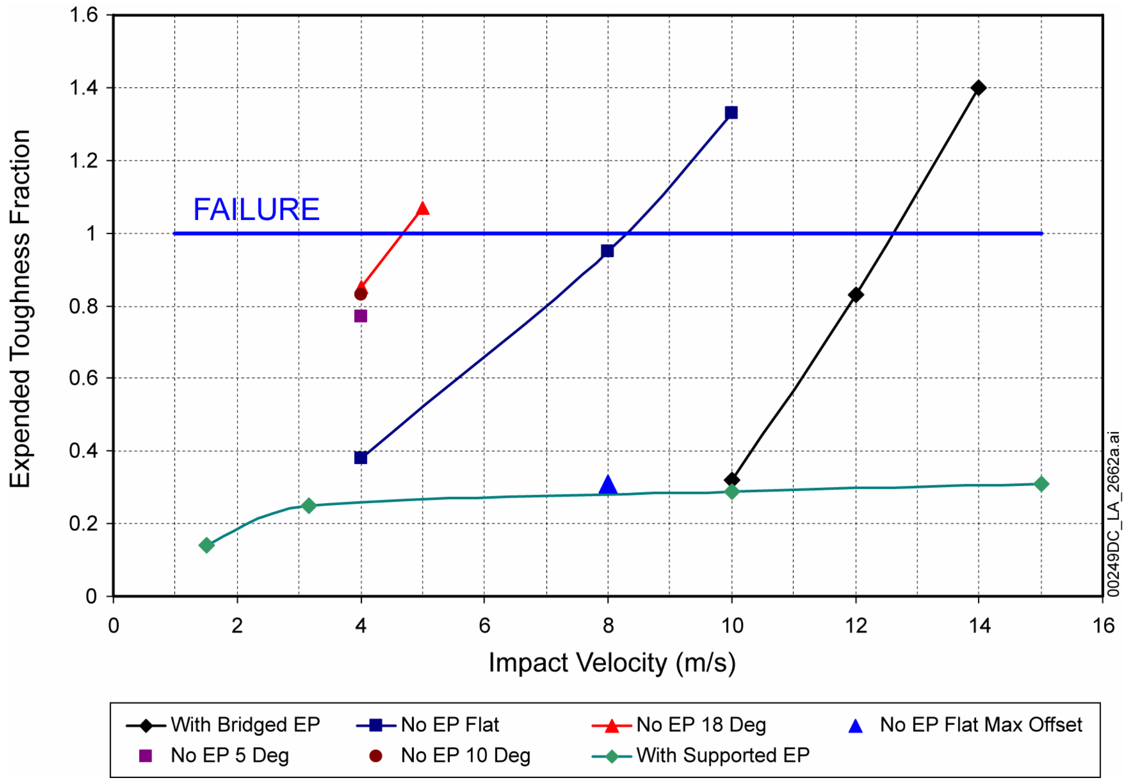

Figure 1.5.2-13. Mean Waste Package Capability for Worst-Case Impacts

NOTE: EP = emplacement pallet.

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