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, 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 assemblies 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 following design criteria apply to the naval corrosion-resistant can:

- 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.

- The containment boundary of the corrosion-resistant can shall retain all fuel-bearing items larger than 0.06 in. in diameter after a 1.05 m/s axial impact of a loaded naval SNF canister with an unyielding surface. This criterion ensures that the naval corrosion-resistant can provides confinement of radionuclides for mechanical loads at least as severe as those at which the cladding of commercial SNF in a commercial SNF waste package would fail (SNL 2007b, Table 6.5-10).

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 external surface temperatures calculated as boundary conditions. The IHF will be designed such that, once the naval M-290 transportation cask is open, the naval SNF canister external surface temperature (1) shall not exceed 400°F during a loss of ventilation lasting no more than 30 days, and (2) shall be kept below 320°F for all normal operations (DOE 2008d, Section 10.3.2.2).

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 naval SNF canister surface temperature shall not exceed 400°F during a 30-day loss of ventilation and 320°F during normal operating conditions in the IHF (DOE 2008d, Section 10.3.2.2). 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 four 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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