4 STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA EVALUATION

4.1 Conduct of Review

Sections 3.1, "Purpose of Installation;" 4.2, "Storage Structures;" and 7.2, "Radiation Sources;" and Appendix A, "Safety Evaluation of DOE-ID Provided Transfer Cask," of the Idaho Spent Fuel (ISF) Facility Safety Analysis Report (SAR) (Foster Wheeler Environmental Corporation, 2003a) identify the material to be stored at the ISF Facility. This material is spent nuclear fuel (SNF) consisting of fuel elements from the Peach Bottom Unit 1 reactor and from Training, Research, and Isotope reactors built by General Atomics (TRIGA), as well as reflector modules and rods from the Shippingport reactor.

Section 3.4, "Classification of Structures, Systems, and Components," of the SAR identifies all the structures, systems, and components classified as important to safety. The structures, systems, and components important to safety are designed to maintain conditions required to store the SNF, to prevent damage to the SNF, and to provide reasonable assurance that the SNF can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Chapter 3, "Principal Design Criteria," of the SAR identifies the design criteria for the ISF Facility. These design criteria are derived from the requirements of 10 CFR Part 72 and applicable industry codes and standards. Section 3.6, "Summary of Design Criteria," of the SAR provides a summary of the key design criteria for the ISF Facility. Chapter 3 of Appendix A of the SAR identifies the principal design criteria for the U.S. Department of Energy-Idaho Operations Office transfer cask system. These design criteria are derived from the requirements of 10 CFR Part 72 and applicable industry codes and standards.

The staff's evaluation of the ISF Facility is based on NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000) and the transfer cask evaluation is based on NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997). The design criteria are compared to the actual design in subsequent chapters of this Safety Evaluation Report (SER).

The staff reviewed the structures, systems, and components and design criteria with respect to the following regulatory requirements:

- 10 CFR §72.2(a)(1) specifies that a license issued for an independent spent fuel storage installation (ISFSI) under Part 72 applies to power reactor spent fuel to be stored in a complex that is designed and constructed specifically for storage of power reactor spent fuel aged for at least one year, and to other radioactive materials associated with spent fuel storage.

- 10 CFR §72.24(c) requires that the design of the ISFSI be described in sufficient detail to support the findings in §72.40, including (1) the design criteria for the ISFSI pursuant to Subpart F of this part, with identification and justification for any additions to or departures from the general design criteria; (2) the design bases and the relation of the design bases to the design criteria; (3) information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all

structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin for safety; and (4) applicable codes and standards.

- 10 CFR §72.24(n) requires that a description of the quality assurance program that satisfies the requirements of Subpart G be applied to the design, fabrication, construction, testing, operation, modification, and decommissioning of the structures, systems, and components of the ISFSI important to safety. The description must identify the structures, systems, and components important to safety. The program must also apply to managerial and administrative controls used to ensure safe operation of the ISFSI.

- 10 CFR §72.102(b) requires that for sites west of the Rocky Mountain Front (west of approximately 104E west longitude) and in other areas of known potential seismic activity, seismicity will be evaluated by the techniques of Appendix A of Part 100 of this chapter. Sites that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided.

- 10 CFR §72.102(c) requires that sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion.

- 10 CFR §72.102(d) requires that site-specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.

- 10 CFR §72.102(f) requires that the design earthquake for use in the design of structures must be determined as follows: (1) for sites that have been evaluated under the criteria of Appendix A of 10 CFR Part 100, the design earthquake must be equivalent to the safe shutdown earthquake for a nuclear power plant, and (2) regardless of the results of the investigations anywhere in the continental U.S., the design earthquake must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

- 10 CFR § 72.104(a) requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv [25 mrem] to the whole body, 0.75 mSv [75 mrem] to the thyroid, and 0.25 mSv [25 mrem] to any other critical organ as a result of exposure to (1) planned discharges of radioactive materials, radon and its decay products excepted, to the general environment; (2) direct radiation from ISFSI or MRS operations; and (3) any other radiation from uranium fuel cycle operations within the region.

- 10 CFR §72.106(a) requires that for each ISFSI site, a controlled area must be established.

- 10 CFR §72.106(b) requires that any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv [5 rem], or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent may not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin or any extremity

may not exceed 0.5 Sv [50 rem]. The minimum distance from the spent fuel, high-level radioactive waste, or reactor-related greater than class C waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 m [328 ft].

- 10 CFR §72.120(a) requires that, pursuant to the provisions of 10 CFR §72.24, an application to store spent fuel in an ISFSI include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety as defined in 10 CFR §72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR §72.122(b)(1) requires that structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents.

- 10 CFR §72.122(b)(2) requires that structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect (i) structures, systems, and components important to safety that are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (ii) The ISFSI also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or structures, systems, and components important to safety.

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

- 10 CFR §72.122(e) requires that an ISFSI or MRS located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public.

- 10 CFR § 72.122(h)(5) requires that the high-level radioactive waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.

- 10 CFR §72.122(l) requires that storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal.

- 10 CFR §72.124(a) requires that spent fuel handling, packaging, transfer, and storage systems be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment under accident conditions.

- 10 CFR §72.124(b) requires that when practicable the design of an ISFSI be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design shall provide for positive means to verify their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.

- 10 CFR § 72.124(c) requires that a criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored that will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under this subpart is not required.

- 10 CFR §72.126(a) requires that radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to (1) prevent the accumulation of radioactive material in those systems requiring access; (2) decontaminate those systems to which access is required; (3) control access to areas of potential contamination or high radiation within the ISFSI; (4) measure and control contamination of areas requiring access; (5) minimize the time required to perform work in the vicinity of radioactive components, for example, by providing

sufficient space for ease of operation and designing equipment for ease of repair and replacement; and (6) shield personnel from radiation exposure.

- 10 CFR § 72.126(b) requires that radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibration and testing their operability.

- 10 CFR §72.126(d) requires that the ISFSI or MRS must be designed to provide means to limit levels as low as is reasonably achievable in release of radioactive materials in effluent during normal operation to; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in §72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in §72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

- 10 CFR §72.128(a) requires that spent fuel storage and other systems that might contain or handle radioactive materials associated with spent fuel or high-level radioactive waste be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (1) a capability to test and monitor components important to safety, (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems, (4) a heat-removal capability having the stability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.

- 10 CFR § 72.144(a) requires that the licensee, applicant for a license, certificate holder, and applicant for a CoC shall establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program that complies with the requirements of this subpart. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with these procedures throughout the period during which the ISFSI or MRS is licensed or the spent fuel storage cask is certified. The licensee, applicant for a license, certificate holder, and applicant for a CoC shall identify the structures, systems, and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

- 10 CFR § 72.144(c) requires that the licensee, applicant for a license, certificate holder, and applicant for a CoC shall base the requirements and procedures of their quality assurance program(s) on the following considerations concerning the complexity and proposed use of the structures, systems, or components: (1) the impact of malfunction or failure of the item on safety, (2) the design and fabrication complexity or uniqueness of the item, (3) the need for special controls and surveillance over processes and equipment, (4) the degree to which functional compliance can be demonstrated by inspection or test, and (5) the quality history and degree of standardization of the item.

4.1.1 Materials to be Stored

As identified in Section 3.1, “Purpose of Installation,” of the SAR (Foster Wheeler Environmental Corporation, 2003a), the materials to be stored at the ISF Facility are Peach Bottom fuel elements, TRIGA fuel elements, and Shippingport reflector modules and rods. Additional information on the characteristics of the fuel elements was provided in the Foster Wheeler Environmental Corporation responses to the staff’s request for additional information (Foster Wheeler Environmental Corporation, 2003b, responses 4-1 through 4-7). The assemblies to be transported in the DOE transfer casks have been specifically identified in Chapter 1 of Appendix A of the SAR. The SNF containers are further described in Chapter 4, Appendix A, of the SAR.

The staff reviewed Section 3.1.1.1, “Peach Bottom Fuel Elements,” of the SAR and supporting documentation, which provides information about the Peach Bottom fuel elements including (i) the reactor type; (i) fuel manufacturer, model, and number; (iii) fuel physical characteristics; and (iv) fuel cladding material (U.S. Department of Energy–Idaho Operations Office, 2002a,b). Figures 3.1-1 through 3.1-5 provide the geometry for the various configurations of the stored fuel elements. The history and census for each of the two cores are given in Section 3.1.1.1 of the SAR. Peach Bottom fuel element characteristics are given in Table 3.1-4, with the initial heavy metal loading given in Tables 3.1-2 and 3.1-3 of the SAR. Decay heat for the elements is given in Figures 3.1-9 and 3.1-10 of the SAR.

The staff reviewed Section 3.1.1.2, “TRIGA Fuel Elements,” of the SAR and supporting documentation, which provides information of the TRIGA fuel elements including (i) the reactor type; (ii) fuel manufacturer, model designation, and number; (iii) fuel physical characteristics; and (iv) fuel cladding material (U.S. Department of Energy–Idaho Operations Office, 2002c). Figure 3.1-6 of the SAR provides the general arrangement of a TRIGA fuel element. Decay heat for the elements is given in Figure 3.1-13 of the SAR.

The staff reviewed Section 3.1.1.3, “Shippingport Fuel Modules,” of the SAR and supporting documentation, which provides information of the Shippingport fuel modules including (i) reactor type; (ii) fuel manufacturer, model designation, and number; (iii) fuel physical characteristics; and (iv) fuel cladding material (U.S. Department of Energy–Idaho Operations Office, 2002d). Figures 3.1-7 and 3.1-8 of the SAR provide the general arrangement of the Shippingport fuel modules. The history and census are given in Section 3.1.1.3 of the SAR. Decay heat for the elements is given in Figures 3.1-11 and 3.1-12 of the SAR.

The staff reviewed Section 7.2.1, “Characterization of Sources,” which provides the source terms in three categories: (i) radionuclide composition, (ii) photon production rate, and (iii) neutron production rate. These data are summarized in Tables 7.2-1 through 7.2-7 of the SAR.

The staff finds that the applicant has provided sufficient information in describing the materials to be stored to satisfy the requirements of 10 CFR §72.2(a)(1).

4.1.2 Classification of Structures, Systems, and Components

The staff reviewed Section 3.4, "Classification of Structures, Systems, and Components," of the SAR, which identifies safety protection systems and provides a brief description of the important characteristics of each system. Structures, systems, and components important to safety are defined in 10 CFR §72.3 as items whose functions are to:

- Maintain the conditions required to store spent nuclear fuel safely;
- Prevent damage to the SNF container during handling and storage; and
- Provide reasonable assurance that SNF can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Section 3.4, "Classification of Structures, Components, and Systems," of Appendix A of the SAR refers to Chapter 3 for identification of structures, systems, and components critical to the safety of the transfer cask.

4.1.2.1 Classification of Structures, Systems, and Components—Items Important to Safety

Table 3.4-1 of the SAR summarizes those structures, systems, and components considered important to safety. The staff reviewed this table and the referenced SAR sections that provide the basis for the classification of structures, systems, and components important to safety. The breakdown of the structures, systems, and components important to safety includes the structures, systems, and components located in the following areas: general area, cask receipt area (CRA), transfer area, and storage area. Those structures, systems, and components related to the transfer cask and considered important to safety are identified in Section 3.4 of the SAR and are provided with corresponding categories.

General Area

The structures, systems, and components important to safety located throughout the ISF Facility are given in Table 4-1 of this SER. Each item is designed to protect the SNF during specific phases of handling and storage. Descriptions of major component interlocks, interlock logic, and functions are identified in Table 5.4-1 of the SAR. Sufficient descriptions of the structures, systems, and components located throughout the ISF facility are provided in the SAR and the supporting documentation. The components, listed in Table 4-1 of this SER, have been properly classified as important to safety items.

Table 4-1. General structures, systems, and components important to safety (ITS) (based on Table 3.4-1 of the SAR)

Structures, Systems, and Components	Reason for classifying as ITS
Seismic switch (including seismic sensor, load interrupters, and connections to power feeds)	Operation during a seismic event ensures that equipment items designed to fail safe but needed to passively shut down do, in fact, shut down. Failure of the seismic switch could result in damage that could potentially impact health and safety of the public.
Electrical interlocks	Prevent damage to the SNF and structures, systems, and components important to safety.

Cask Receipt Area

The structures, systems and components important to safety located within the CRA are given in Table 4-2 of this SER. In some cases, the items are classified as important to safety because they are considered part of the single-failure-proof load path of systems used to handle and transport the SNF. Each item is designed to protect the SNF during specific phases of handling and storage. Sufficient descriptions of the structures, systems, and components are provided in the SAR and the supporting documentation. The components listed in Table 4-2 of this SER have been properly classified as important to safety items.

Table 4-2. CRA structures, systems, and components important to safety (based on Tables 3.4-1 and 3.6-1 of the SAR)

Structures, Systems, and Components	Reason for classifying as ITS
Transfer cask and trunnions	Designed to support the canisters during transfer lift operations and provide radiation shielding while in the receipt area and while in the cask trolley. The transfer cask load-bearing components prevent damage to the SNF during transport and transfer operations for all normal, off-normal, and accident conditions. Trunnions are designed for single-failure-proof 29,484-kg [65,000-lb] lift load.
Cask receipt crane and associated lifting fixtures	Part of the load path for handling the transfer cask of SNF. Damage to the SNF could result from failure of cask receipt crane or associated lifting fixtures and subsequent cask drop.
CRA (structural load path for cask receipt crane)	The cask receipt crane supports and load-bearing structure/footings are part of the load path for handling the transfer cask loaded with SNF. Damage to the SNF could result from failure of load-bearing structures/footings and subsequent cask drop.

Transfer Area

The structures, systems, and components important to safety located within the transfer area are given in Table 4-3 of this SER. They either form part of the confinement boundary or are designed to protect the SNF during specific phases of handling and storage. Sufficient descriptions of the structures, systems, and components are provided in the SAR and the supporting documentation. The components listed in Table 4-3 of this SER have been properly classified as important to safety items.

Storage Area

The structures, systems, and components important to safety located within the storage area are given in Table 4-4 of this SER. Each item is designed to protect the SNF during specific phases of handling and storage. Sufficient description of the structures, systems and components is provided in the SAR and the supporting documentation. The components listed in Table 4-4 of this SER have been properly classified as important to safety items.

Conclusion

The staff concludes the structures, systems, and components important to safety identified and listed in the SAR are acceptable and in compliance with 10 CFR §72.24(n) and §72.144(a).

Table 4-3. Transfer area structures, systems, and components important to safety (based on Table 3.4-1 of the SAR)

Structures, Systems, and Components	Reason for classifying as ITS
Fuel packaging area (FPA) (building structures)	Reinforced concrete structure confines radioactive materials and provides the structure that supports the SNF lifting equipment. The structure and the FPA monorail are part of the load path and have functions required to prevent damage to the SNF during handling and storage The confinement boundary structures, shield windows, encasts (through confinement walls), and shield plugs for the FPA confine radioactive materials to ensure radiation dose rates remain within acceptable and analyzed limits, and that there are no uncontrolled releases, so there is no undue risk to health and safety of the public. The FPA bench containment vessels have structural integrity and geometry control functions necessary to provide reasonable assurance the SNF can be processed safely.

Table 4-3. Transfer area structures, systems, and components important to safety (based on Table 3.4-1 of the SAR) (continued)

Structures, Systems, and Components	Reason for classifying as ITS
Transfer Tunnel (building structure)	The reinforced concrete structure is part of the load path and functions to prevent damage to the SNF during handling and storage, and to provide shielding to ensure radiation dose rates remain within acceptable and analyzed limits. The trolley rails and encasts are part of the support hardware to ensure the various transfer casks and ISF canisters are in stable configurations during transfer, and that tipover or other postulated accidents do not cause damage to the transfer cask, ISF canisters, and SNF during transit. The Transfer Tunnel walls, ceiling, and outer door protect the cask and canister trolleys from tornado wind and missile effects.
Canister closure area (CCA) (building structure)	The south wall of the CCA is shared with the FPA and forms part of the reinforced concrete structure that confines radioactive materials and provides the structure that supports the SNF lifting equipment (FHM). The remainder of the CCA structure prevents damage to the SNF during a design earthquake and provides tornado missile protection. That portion of the CCA (i.e., south wall) that forms part of the confinement boundary confines radioactive materials to ensure that radiation dose rates remain within acceptable and analyzed limits, and that there are no uncontrolled releases, so there is no undue risk to the health and safety of the public.
Confinement boundary through wall (service) penetrations	Through wall penetrations form a part of the confinement boundary.
Cask trolley	Provides the mechanism and support hardware to ensure the various transfer casks are in stable configurations during transfer under all normal, off-normal, and accident conditions. The cask trolley will be designed to preclude tipover for site-specific seismic loads. Components of the cask trolley are part of the single-failure-proof load path. Seismic pins are used to restrain the cask trolley from motion during design bases seismic events.
Cask adapter and inflatable seal	Forms a part of the confinement boundary.
Canister port inflatable seal	Forms a part of the confinement boundary.

Table 4-3. Transfer area structures, systems, and components important to safety (based on Table 3.4-1 of the SAR) (continued)

Structures, Systems, and Components	Reason for classifying as ITS
Canister trolley (including jacking system)	Provides the mechanism and support hardware to ensure the canisters are in stable configurations during transfer, lifting, and positioning for normal, off-normal, and accident conditions. The canister trolley will be designed to preclude tipover for site-specific seismic loads. Components of the canister trolley are part of the single-failure-proof load path. Seismic pins are used to restrain the canister trolley from motion during design bases seismic events.
Check valves, relief valve, and connecting tubing for cask port and canister port seals	Used to establish the confinement boundary, in combination with the cask and canister port seals, when transferring SNF from the trolleys to the FPA.
Fuel handling machine (FHM) (operational and load carrying components)	Manipulates the SNF within the FPA. A lifting component failure and subsequent drop could result in SNF damage. The FHM rails and supports are part of the load path, and damage to the SNF could result from failure and subsequent drop.
FHM lifting devices	Part of the load path for transporting and handling SNF within the FPA. Damage to the SNF could result from lifting device failure and subsequent drop.
Worktable and tipping machine	Part of the load path, and damage to the SNF could result from failure and subsequent drop.
Heating, ventilation, air conditioning (HVAC) [portions of duct work and high-efficiency particulate air (HEPA) filters that form part of the FPA confinement boundary]	Defines the ventilation zones and provides filtration of radioactive materials within the confinement boundary of the FPA. Forms part of the confinement boundary to ensure radiation dose rates remain within acceptable and analyzed limits.
HVAC system breakaway joint	Ensures that portions of the HVAC system not important to safety do not damage important to safety portions during design basis events.
Breathing air system (portion of system that penetrates confinement boundary)	Through-wall piping and penetrations form part of the confinement boundary.
Master/slave manipulator through wall tubes and encasts	Form part of the FPA confinement boundary.
Personnel-shielded access door	Forms part of the FPA confinement boundary.

Table 4-4. Storage area classification of structures, systems, and components important to safety (based on Table 3.4-1 of the SAR)

Structures, Systems, and Components	Reason for classifying as ITS
Storage vaults	Provide structures that dictate, ensure, and maintain the geometry and condition of the fuel storage array. Storage vaults protect the ISF canisters and storage tube assemblies during design basis events.
Storage vault inlet vents and exhaust louvers	Maintain the storage area temperature within design limits to preclude damage to the storage structure or ISF canisters because the ISF storage system is a passive system.
Charge face cover plate	Protects the storage tubes and ISF canisters from tornado wind and missile effects.
Storage tube support stool	Ensures seismic and differential thermal movements do not introduce any axial loads in the storage tube that could damage the tube or the ISF canister. Provides support that aids in maintaining the tube and its enclosed SNF in a critically safe array.
Storage tube assembly	Provides the secondary confinement boundary and ensures an inert atmosphere to minimize corrosion.
ISF canister	Provides the primary confinement barrier; its failure could lead to the release of radioactive material.
ISF basket	Provides orientation and structural support for SNF within the ISF canister. Damage to the basket could result in fuel damage and the failure to maintain a subcritical geometry.
ISF canister impact plates	Located inside the ISF canister upper and lower formed heads. These are flat steel plates that create perpendicular end faces inside the canister for the shield plug and canister basket to react against to prevent damage to the SNF.
ISF canister shield plug	Steel disc placed above the basket within the ISF canister. The shield plug is used to reduce the dose from the canister during canister closing operations.
Canister handling machine (CHM)	Part of the load path for transporting and handling SNF within the storage area. The CHM rails and conductors are part of the load path for transporting and handling SNF within the storage area. The CHM grapple is part of the load path for transporting and handling SNF within the storage area. Damage to the SNF could result from a CHM failure and subsequent drop.

4.1.2.2 Classification of Structures, Systems, and Components—Items Not Important to Safety

Structures, systems, and components not important to safety do not involve a safety-related function and are not subject to special utility requirements or U.S. Nuclear Regulatory Commission-imposed regulatory requirements. Table 4-9-1 was provided as a part of the response to the staff's request for additional information (Foster Wheeler Environmental Corporation, 2003b). This table provides a complete list of the structures, systems, and components and their functions within the ISF Facility by area. The potential impact of their failure is described in the SAR. The staff reviewed Table 4-9-1 and the supporting documentation and concurs with the classification identified. The classification is based on the function of the equipment and its potential to ensure that radiation dose rates remain within acceptable and analyzed limits and that there are no uncontrolled releases, so there is no undue risk to health and safety of the public.

Other than the transfer casks, the staff identified the transfer cask trailers as the only systems relevant to the transfer activity of the SNF from other locations at the Idaho National Engineering and Environmental Laboratory (INEEL) site to the ISF Facility. The transport trailers, Trailer Nos. 71801 and 71808, will carry the transfer casks (Peach Bottom–1 or Peach Bottom–2) during transfer operations. Potential failure mechanisms of the transporter involve accidents where the trailer would either tip over or fail to support the transfer cask and result in the transfer cask falling from the trailer. Potential failures that could drop the cask would have a possibility of damaging the cask or its internal components. Failure scenarios of the transfer trailers are bounded by the drop analyses performed by the U.S. Department of Energy–Idaho Operations Office as indicated in Chapter 12 of Appendix A of the SAR. Therefore, the transfer cask trailers can be classified as an item not important to safety.

The staff concludes that the applicant has appropriately identified the components listed in Table 4-9-1 of the response to the staff's request for additional information (Foster Wheeler Environmental Corporation, 2003b) as not important to safety.

4.1.2.3 Classification of Structures, Systems, and Components—Conclusion

The staff evaluated the classification of structures, systems, and components important to safety by reviewing Section 3.4, "Classification of Structures, Systems, and Components," of the SAR; the documents cited in the SAR; and other relevant literature. The staff also evaluated the classification of structures, systems, and components important to safety with respect to Appendix A of the SAR. The staff determined that the classification of the structures, systems, and components important to safety and their associated categories are consistent with the regulatory requirements of 10 CFR §72.144(a), and that the associated technical information is in compliance with 10 CFR §72.24(n). The staff's detailed evaluation of the Quality Assurance Program is contained in Chapter 12 of this SER.

4.1.3 Design Criteria for Structures, Systems, and Components Important to Safety

The principal design criteria identified for structures, systems, and components important to safety at the ISF Facility are described in Chapters 3, "Principal Design Criteria;" and 4,

“Installation Design;” and Appendix A, “Safety Evaluation of DOE-Provided Transfer Cask,” of the SAR.

Section 3.2 of Appendix A of the SAR states that discussions of structural and mechanical safety criteria are provided in other appropriate sections of the appendix. This approach is acceptable in accordance with NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997). The staff’s evaluation of these criteria is also summarized in appropriate sections of this SER.

This section contains a review of Sections 3.2, “Structural and Mechanical Safety Criteria;” 3.3, “Safety Protection System;” 3.6, “Summary of Design Criteria;” 4.2, “Storage Structures;” 4.7, “Spent Fuel Handling Operations Systems;” and Appendix A, Section 3.0, “Principal Design Criteria,” of the SAR. More detailed discussions of the design criteria are presented in Sections 4.1.3.1 through 4.1.3.7 of this SER.

4.1.3.1 General

The staff reviewed the discussion of the general design criteria for structures, systems, and components presented in the SAR. The type and amount of SNF to be stored in the ISF Facility includes (i) 1,601.5 fuel elements from Peach Bottom Unit 1; (ii) 2,971 Shippingport reflector rods; and (iii) 1,600 TRIGA fuel elements, as described in Section 3.1.1, “Purpose of Installation,” of the SAR. The SNF will be placed in up to 246 ISF canisters, which are, in turn, stored in 246 metal storage tubes (the design storage capacity of the ISF Facility).

Definitions of the normal, off-normal, and accident loads are given in Section 3.2.5.2, “Load Combinations,” of the SAR. The quality standards for the design bases of structures, systems, and components important to safety are provided in Chapters 3, “Principal Design Criteria,” and 11, “Quality Assurance,” of the SAR.

The design life of the structures, systems, and components important to safety is based on their ability to withstand the applied loads. The applied loads are defined using an annual probability of exceeding the design load. Analysis procedures are used to demonstrate the ability of the structures, systems, and components to withstand the applied loads with additional factors applied to the loads and material allowables as identified by the referenced codes and standards.

The staff finds that the design criteria discussed in the SAR satisfy the requirements of 10 CFR §72.24(c), §72.120(a), and §72.122(h)(5), because design criteria are identified properly, and that the structures, systems, and components important to safety will be designed to quality standards commensurate with the important to safety functions to be performed to satisfy the requirements of 10 CFR §72.144(c).

The ISF Facility is adjacent to the Idaho Nuclear Technology and Engineering Center (INTEC) which contains several nuclear facilities, as indicated in Section 3.3.5.4, “Proximity to Other Nuclear Facilities.” Therefore, the ISF Facility must be designed and operated to ensure the cumulative effects of the combined operations within INTEC will not constitute an unreasonable risk to the health and safety of the public as required by 10 CFR §72.122(e). Onsite hazards and cumulative effects have been identified and assessed in Section 2.2, “Nearby Industrial, Transportation, and Military Facilities,” of the SAR. Accident analyses have been performed

and documented in Section 8.2.5.6, “Accidents at Nearby Sites,” of the SAR. The distance between the ISF Facility and other INEEL facilities makes airborne contamination the primary consequence of an emergency condition at one of the nearby nuclear facilities. Review of accidents at nearby nuclear facilities and their consequences is provided in Sections 15.1.2.20, 15.1.2.21, and 15.1.2.22 of this SER.

Structural design criteria and radiological protection and confinement criteria are identified in the SAR. Review of the structural criteria is presented in this chapter of the SER. Review of the radiological protection and confinement criteria is presented in Chapters 7, “Shielding Evaluation;” 8, “Criticality Evaluation;” 9, “Confinement Evaluation;” and 11, “Radiation Protection Evaluation,” of this SER.

4.1.3.2 Structural

The staff reviewed the discussion on structural design criteria of structures, systems, and components presented in Sections 3.2, “Structural and Mechanical Safety Criteria;” 3.3, “Safety Protection System;” 3.6, “Summary of Design Criteria;” 4.2 “Storage Structures;” 4.7, “Spent Fuel Handling Operations Systems;” and Appendix A, Section 3.0, “Principal Design Criteria,” of the SAR.

For concrete components of the ISF Facility, design criteria are based on the American Concrete Institute’s (ACI) ACI 349-97 (1998). ACI 349-97 specifies acceptable design and construction of concrete structures that form part of a nuclear power plant and that have nuclear safety-related functions. Structures included in the ACI code are concrete structures inside and outside the containment system.

The design criteria for the storage canister conform to standard engineering practice, as identified in the ASME Boiler and Pressure Vessel Code (ASME International, 1998). The ASME Boiler and Pressure Vessel Code establishes rules governing design, fabrication, and inspection during construction of boilers and pressure vessels. This code contains mandatory requirements, specific prohibitions, and nonmandatory guidance for selection of materials, design, fabrication, examination, inspection, testing, certification, and pressure relief. In addition, NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) is identified for the design criteria of the SNF handling systems and the support structures, systems, and components for compliance with a single-failure-proof lift system. NUREG–0612 identifies controls for handling heavy loads at nuclear power plants.

Appendix A, Section 3.2, “Structural and Mechanical Safety Criteria,” of the SAR addresses the structural and mechanical design criteria for the transfer casks. The transfer casks to be used for transporting SNF to the ISF Facility are the DOE-supplied Peach Bottom–1 and Peach Bottom–2 casks. In general, design specifications for the structures, systems, and components important to safety relevant to the transfer casks are addressed in the analysis sections of Appendix A. The design criteria for the transfer casks are based on 10 CFR 71 requirements and standard engineering practice, as described in the Battle Memorial Institute SAR (1970). The design criteria for the welding procedures used for the casks are based on ASME Boiler and Pressure Vessel Code Sections VIII and IX (ASME International, 1962).

Additional details about the transfer casks are included by reference to the Battelle Memorial Institute SAR (1970) and the Westinghouse Safety Analysis (1986) for shipping SNF generated from a light-water breeder reactor. Exceptions to code requirements for the Peach Bottom transfer casks are identified in Appendix A (Foster Wheeler Environmental Corporation, 2003a)

Drop analyses performed by DOE-ID are referenced in Section 12 of Appendix A of the SAR. These analyses indicate the transfer casks will maintain closure for nonmechanistic drop scenarios, which bound all potential drop heights encountered during transfer to the ISF Facility and operations in the ISF Facility.

The design criteria for the transfer cask trunnions are in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). The applicant considered the guidance in Section 5.1.6(3) of NUREG-0612, which requires a factor of safety of 10 with respect to ultimate strength for the trunnions as single-failure-proof interfacing lifting points.

As identified, the structures, systems, and components important to safety are designed to withstand the effects of environmental conditions and natural phenomena for normal, off-normal, and accident conditions. The structural design loads for structures, systems, and components important to safety are provided in Section 3.2.5.1, “Design Loads,” of the SAR. Information on the derivation of site-specific design criteria for the meteorology, hydrology, and seismology is contained in SAR Chapter 2, “Site Characteristics.”

<u>Wind</u>

Figure 6-1 in American Society of Civil Engineers (ASCE) ASCE 7-98 (2000) identifies a design basis 3-second gust wind speed of 40 m/s [90 mph] for the region. Information provided in Section 3.2.1.1.1, “Design Basis Wind,” and in Table 2.3-14 of the SAR identifies a maximum recorded wind gust speed at the INEEL site of 35.6 m/s [84 mph]. The wind speed of 40 m/s [90 mph] is used as the design basis wind for the ISF Facility.

The staff reviewed the design basis wind for the ISF Facility and finds it is consistent with that identified in ASCE 7-98 for this location. The staff also finds that the requirements of 10 CFR §72.120(a) and §72.122(b)(1) and (2) are satisfied in that the effects of wind are considered in the design of the ISF Facility.

<u>Tornado</u>

The design basis tornado wind loads for the INEEL site are provided in Section 3.2.1.1.2, “Design Basis Tornado,” and in Table 2.3-15 of the SAR. The design basis tornado includes:

- 89.4 m/s [200 mph], maximum speed;
- 71.5 m/s [160 mph], rotational speed;
- 17.9 m/s [40 mph], translational speed;
- 10.3 kPa [1.5 psi], pressure drop; and
- 4.14 kPa/s [0.6 psi/s], rate of pressure drop.

The parameters for the tornado identified have been reviewed and are consistent with those given in Regulatory Guide 1.76 (U.S. Atomic Energy Commission, 1974). The staff finds that the requirements of 10 CFR §72.120(a) and §72.122(b)(1) and (2) are satisfied in that the

effects of site conditions and environmental conditions are considered in the design of the ISF Facility.

Tornado Missiles

The postulated tornado missiles for the ISF Facility are identified in Table 3.2-1 of the SAR, including a compilation of those specified as Spectrum I missiles as shown in Section 3.5.1.4 of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981), as follows:

- 52.2-kg [115-lb] 9.19-cm [3.62-in] × 28.91-cm [11.38-in] × 3.66-m [12-ft] wood plank at 57.9 m/s [190 ft/s];
- 130.2-kg [287-lb] 15.24-cm [6-in] diameter schedule 40 pipe at 10.1 m/s [33 ft/s]; and
- 4.08-kg [9-lb] 2.54-cm [1-in] diameter steel rod at 7.9 m/s [26 ft/s].

These missiles are considered to be representative of the potential missiles present at the site. Use of either Spectrum I or II missiles is considered acceptable by the U.S. Nuclear Regulatory Commission.

The staff reviewed the design basis tornado conditions for the ISF Facility and finds that they are consistent with design criteria specified by Section 3.5.1.4 of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981), to withstand tornadoes, in accordance with the requirements of 10 CFR §72.120(a) and §72.122(b)(1) and (2).

Flood

Based on the location of the ISF Facility and the site surface hydrology, the applicant concluded in SAR Section 2.4, “Surface Hydrology,” that there is potential for flooding in the vicinity of the ISF Facility. As identified in Section 2.4.3.5, “Water Level Determinations,” of the SAR, the flood waters within the ISF Facility could reach up to 1,499.83 m [4,921 ft] above mean sea level. The final graded ground surface elevation at the ISF Facility site will be 1,498.7 m [4,917 ft]. Therefore, the effects of flood waters on important to safety equipment and the actions to mitigate the effects of flooding are reflected in Chapter 8, “Accident Analysis,” of the SAR. The flood design criteria are defined in Section 3.2.2, “Water Level (Flood) Design,” of the SAR.

The staff reviewed the flood design criteria and concludes that the ISF Facility design is consistent with design criteria of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1996) and ASCE 7-98 (American Society of Civil Engineers, 2000) to withstand floods as required by 10 CFR §72.120(a) and §72.122(b)(1) and (2).

Seismicity

The staff reviewed the data presented in the SAR associated with seismic design criteria at the ISF Facility. Section 3.2.3, “Seismic Design,” of the SAR gives the seismic design criteria based on probabilistic site-specific seismology studies summarized in Section 2.6, “Geology and Seismology,” of the SAR. The response spectra for these design basis ground motions are given in Figures 3.2-1 through 3.2-3 of the SAR. As shown from these SAR figures, the horizontal peak ground acceleration from these response spectra is 0.19 g. The associated vertical spectra are derived from the horizontal spectra based on the vertical-to-horizontal

spectra ratio in Figure 2.6-56 of the SAR. Based on these ground motions, elevated response spectra are calculated for use in the structural analysis of the structures, systems, and components at the ISF Facility. The staff assessment of the adequacy of the site-specific seismic design criteria is contained in Chapter 2 of this SER. The applicant's analysis of the ISF storage system during the site-specific design basis seismic ground motion is evaluated in Chapters 5 and 15 of this SER.

The staff reviewed the seismic design criteria for the ISF Facility and finds they are properly identified as required by 10 CFR §72.120(a) and §72.122(b)(1) and (2). The staff's review of the applicant's request for an exemption from the regulatory requirements in 10 CFR §72.102(f) is provided in Chapter 2 of this SER.

Snow and Ice

The design basis snow and ice load for the ISF Facility is based on site-specific case studies (Sack and Sheikh-Taheri, 1986). Section 3.2.4, "Snow and Ice Loadings," of the SAR states that the design basis ground snow load is 1.68 kPa [35 psf], and that the minimum roof snow load is 1.44 kPa [30 psf]. Figure 7-1 of ASCE 7-98 (American Society of Civil Engineers, 2000) suggests the snow load for the region just south of the INEEL site is approximately 0.96 kPa [20 psf].

The staff reviewed the snow and ice loading criteria and finds that they are appropriate and in accordance with the requirements of 10 CFR §72.120(a) and §72.122(b)(1) and (2).

Temperature Loads

The normal thermal loads for the ISF Facility include loads associated with normal condition temperatures, temperature distributions, thermal gradients within the structure, and the effects of expansion and contraction of structural elements. As identified in Section 2.3, "Meteorology," of the SAR, the climatology of the INEEL site is based on National Oceanic and Atmospheric Administration meteorological observations since 1949 (Clawson, et al., 1989). The normal site ambient maximum and minimum temperatures are presented in Section 2.3.1.2.2, "Regional Temperature," as:

- Minimum normal site temperature: !32.2 °C [!26 °F]
- Maximum normal site temperature: 36.7 °C [98 °F]

Indoor normal air temperatures were calculated using the normal outdoor temperatures for steady-state conditions. Off-normal thermal loads are those produced directly by or as a result of off-normal or design basis accidents, fires, or natural phenomena. The off-normal and accident site maximum and minimum temperatures are identified in Section 3.2.5.1.10, "Off-Normal and Accident Thermal Loads (T_a)," as:

- Minimum off-normal and accident site temperature: !40.0 °C [!40 °F]
- Maximum off-normal and accident site temperature: 38.3 °C [101 °F]

The reference temperature for the concrete is 15.6 °C (60 °F) while the reference temperature for the structural steel is 21.1 °C (70 °F). The reference temperature is the temperature at which the material is considered to be stress-free from thermal effects.

The staff reviewed the thermal loading criteria and finds that they are appropriate and in accordance with the requirements of 10 CFR §72.120(a) and §72.122(b)(1) and (2).

Fire

The staff reviewed Section 8.2.4.4, "Fire and Explosion," of the SAR. For fire hazard evaluation purposes, the ISF Facility is divided into three areas. Each fire area has unique fire loading characteristics and fire protection capabilities to address the postulated fire hazards. In accordance with NUREG–0800 (U.S. Nuclear Regulatory Commission, 1996) and National Fire Protection Association (NFPA) 801 (1998), a fire hazards analysis was prepared. This analysis forms the basis for the overall fire protection design.

The staff reviewed the fire considerations in the SAR and finds that they are consistent with equipment used at the facility and operational restraints as required by 10 CFR §72.122(c). Appropriate design criteria are specified to ensure structures, systems, and components important to safety will be designed and located to perform their safety functions effectively during credible fire exposure conditions.

Explosion

Explosions internal and external to the ISF Facility are not considered credible as discussed in Sections 3.3.6, "Fire and Explosion Protection," and 8.2.4.4, "Fire and Explosion," of the SAR. This determination is based on a hazards assessment for INTEC (Idaho National Engineering and Environmental Laboratory, 2001).

The staff reviewed the explosion considerations in the SAR and finds that they are in accordance with Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) and in compliance with 10 CFR §72.122(c).

Lightning

During thunderstorms, a lightning strike is possible. A lightning risk assessment was conducted for the ISF Facility in accordance with NFPA 780 (1997) which identified a moderate to severe risk factor. A lightning protection system is provided to reduce the risk, as discussed in Section 4.3.8.1.4, "Design Code Compliance," of the SAR.

The staff reviewed the lightning design criterion as identified in the SAR with reference to the ISF Facility and determined that it is acceptable for the design of structures, systems, and components important to safety as required by 10 CFR §72.122(b)(1) and (2).

Load Combinations

The load combinations presented in Section 3.2.5, "Combined Load Criteria," of the SAR are used in the analyses of structures, systems, and components important to safety. The loads considered in the load combinations include

- Dead Loads (D)
- Live Loads (L)
- Soil Pressure (H)

- Soil Reaction Loads (G)
- Wind Load (W)
- Temperature Loads (T)
- Earthquake Loads (E)
- Flood Loads (F)
- Tornado Loads (W_t)
- Off-Normal and Accident Thermal Loads (T_a)
- Accident Load (A)

The specific load combinations used are identified in Sections 3.2.5.2.1, "Reinforced-Concrete Structures;" 3.2.5.2.2, "Reinforced-Concrete Footing/Foundations;" 3.2.5.2.3, "Steel Structures;" and 3.2.5.2.4, "Overturning and Sliding," of the SAR. These load combinations are consistent with those suggested in Section 4.5.3.2 of NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000) and in Table 3-1 of NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997). Load combinations from industry codes and standards are used to analyze specific systems and components within these structures.

The staff reviewed the ISF Facility documentation and determined that the load combination design criteria are appropriately considered for the design of structures, systems, and components important to safety as required by 10 CFR §72.122(b)(1) and (2). Appropriate combinations of the effects of normal and accident conditions and the effects on natural phenomena are considered.

Structural Design Criteria Conclusion

The structural design criteria discussed in the previous sections represent the structural loads that may be present at the site. The ISF Facility structures, systems, and components important to safety must be designed to withstand these structural loads, as applicable. The ability of the structures, systems, and components to perform the intended safety functions under the applicable structural design loads is evaluated in Chapters 5 and 15 of this SER.

4.1.3.3 Thermal

The staff reviewed the discussion on thermal design criteria for structures, systems, and components for normal site ambient maximum and minimum temperatures and normal indoor conditions. These thermal design criteria are presented in Section 3.2.5.1.6, "Temperature Loads (T)," of the SAR. Ambient condition design criteria are based on site-specific meteorological conditions. Design temperatures are based on the data from the region, which are consistent with the values measured to date according to the onsite meteorological measurement program.

The staff reviewed the ambient condition loading design criteria and determined that they are acceptable because they are based on site-specific information, and the values are consistent with the data from the National Oceanic and Atmospheric Administration for the region. Consequently, the ambient condition loading design criteria satisfy the requirements of 10 CFR §72.122(b)(1) and (2) and are detailed in Chapter 6 of this SER.

The allowable temperature limits and their bases for SNF, ISF baskets, ISF canisters, ISF storage tubes, and concrete are listed in Table 4.2-53 of the SAR. Design temperatures for various materials are in compliance with acceptable codes and are presented in the SAR. ACI 349-97 (American Concrete Institute, 1998) specifies the maximum concrete temperature for normal operation and accident conditions. The Boiler and Pressure Vessel Code, ASME Section II, Part D, lists stress allowable values for a range of temperatures (ASME International, 1999). The performance requirements of 10 CFR §72.120(a) have been met for all materials, as identified by the acceptable temperatures identified in conformance with the accepted standards.

Off-normal and accident thermal loads are given in Section 3.2.5.1.10, “Off-Normal and Accident Thermal Loads (T_a),” of the SAR. These loads include the design basis and design criteria with respect to fire protection given in Section 3.3.6, “Fire and Explosion Protection,” of the SAR. Consequently, the off-normal and accident thermal load condition loading design criteria satisfy the requirements of 10 CFR §72.122(c) and are detailed in Chapter 6 of this SER.

Thermal design criteria are based on environmental conditions and heat generated by the materials stored. The storage systems are passive and incorporate passive heat removal. The staff reviewed the thermal design criteria for storage and handling of the SNF and determined that they are appropriately identified as required by 10 CFR §72.120(a), §72.122(b)(1) and (2), and §72.122(c). The staff’s review of the thermal evaluation of the ISF Facility is provided in Chapter 6 of this SER.

4.1.3.4 Shielding and Confinement

The staff reviewed the discussion on shielding and confinement design criteria for the structures, systems, and components of the ISF Facility in Section 3.3.5, “Radiological Protection,” of the SAR. These criteria are summarized in Table 4-5 of this SER. A controlled area is identified to satisfy the requirements of 10 CFR §72.106(a). A radiological alarm system is provided to satisfy the requirement of 10 CFR §72.126(b). The design, administrative controls, and personnel training for the ISF Facility provide the necessary radiological protection to maintain public and occupational doses as low as reasonably achievable (ALARA). Criteria used in design of the ISF canister radiological protection features and confinement design of the ISF canister systems are provided in the SAR. The basic concept for the ISF Facility shielding and confinement system is protection by multiple barriers and systems as required by 10 CFR §72.126(a). The use of the ISF storage system, which is a sealed canister-based system, satisfies the requirements of 10 CFR §72.122(h)(5). The design bases for confinement barriers and systems, ventilation of off-gas systems, radioactive waste treatment, and waste storage facilities are provided in the SAR. Operating procedures, shielding design, and access controls provide the necessary radiological protection to ensure radiological exposures to facility personnel and the public are ALARA as required by 10 CFR §72.126(d). Chapter 7 of the SAR provides further details and procedural considerations for radiation protection for public and occupational doses from the ISF Facility operations. Bounding dose rate design criteria are consistent with the requirements of 10 CFR §72.104(a) and §72.106(b).

The staff reviewed the design criteria for SNF storage and handling and determined that they are appropriately identified as required by 10 CFR §72.128(a). The shielding evaluation is

performed in Chapter 7 of this SER, the confinement evaluation is performed in Chapter 9 of this SER, and the radiation protection evaluation is performed in Chapter 11 of this SER.

Table 4-5. Shielding and confinement design criteria (based on Table 3.3-6 of SAR)

Location	ISF Facility Design Value	Applicable Requirements
Restricted area	ALARA	10 CFR §72.126(d)
	Total effective dose equivalent (TEDE) 0.05 Sv/yr [5 rem/yr]	10 CFR §20.1201
	TEDE 0.01 Sv/yr [1 rem/yr]	ISF facility administrative control limits
Controlled area	Normal and off-normal TEDE 1.0 mSv/yr [100 mrem/yr]	10 CFR §20.1301
	Accident TEDE 0.05 Sv/yr [5 rem/yr]	10 CFR §72.106(b)
Outside controlled area	Normal and off-normal TEDE 0.25 mSv/yr [25 mrem/yr]	10 CFR §72.104(a)
	Accident TEDE 0.05 Sv/yr [5 rem/yr]	10 CFR §72.106(b)

4.1.3.5 Criticality

The staff reviewed the discussion on criticality design criteria of the structures, systems, and components for the ISF Facility in Section 3.3.4, “Nuclear Criticality Safety,” of the SAR. The ISF Facility uses standard criticality control methods in the design of the ISF Facility, incorporating additional analyses and evaluations as appropriate to deal with the unique nature of the SNF to be handled and stored. Criticality safety analyses have demonstrated that there are adequate safety margins for handling and storage operations involving the specific fuel types at the ISF Facility, as discussed in the staff’s evaluation in Chapter 8 of this SER. Criticality design bases for the SNF are given in Sections 4.7.3.4, “Criticality Evaluation of Spent Fuel Handling Operations” and 4.2.3.3.7, “Criticality Evaluation,” and in Chapter 8, “Accident Analysis.” These design bases address the condition of the SNF on site until it is sealed in the ISF canister, after it is sealed in the ISF canister, and at the occurrence of associated potential criticality accidents.

The staff finds that the design criteria for criticality are identified appropriately in the SAR as required by 10 CFR §72.124(a)–(c). The staff criticality evaluation is discussed in Chapter 8 of this SER.

4.1.3.6 Decommissioning

The staff review of Section 3.5, “Decommissioning Considerations,” of the SAR is presented in Chapter 13 of this SER.

4.1.3.7 Retrieval

The staff reviewed the discussion on retrieval design criteria of the structures, systems, and components for the ISF Facility in Sections 3.3.7.1, “Spent Fuel or High-Level Radioactive Waste Handling and Storage;” 5.2.2.1, “Functional Description;” and 8.2, “Accidents,” of the SAR. The SNF will be stored in and handled with the ISF canister. The design criteria for SNF retrievability are given together with the processes to retrieve the SNF, including handling damaged or failed fuel. Retrievability also is discussed for accident events. As discussed in the SAR, the ISF storage system is designed to ensure adequate safety and to protect fuel integrity and retrievability under design basis loads.

Based on the foregoing discussion, the staff finds that the ISF storage system will provide adequate safety and maintain fuel retrievability during the ISF Facility site-specific conditions. Therefore, the staff finds that the consideration of the retrievability of the SNF in the ISF Facility design meets the requirements of 10 CFR §72.122(l) and §72.128(a).

4.1.4 Design Criteria for Other Structures, Systems, and Components

No specific requirements are identified in 10 CFR Part 72 for other structures, systems, and components not important to safety. Therefore, the staff discusses the information provided in the SAR in this section, but no evaluation findings are made. The design criteria for structures, systems, and components classified as not important to safety, but which have security or operational importance, are addressed in Chapter 4 of the SAR. As part of the description, codes and standards also are given in this chapter. The SAR specifies that these structures, systems, and components will be designed to comply with the applicable codes and standards to maintain the capability to mitigate the effects of off-normal or accident events.

Structures, systems, and components not important to safety, whose failure could potentially compromise the integrity of important to safety SSCs, are designed to load combinations necessary to prevent their damaging any important to safety SSCs. For example, the stack is designed to withstand the design basis seismic event so that it will not topple and damage structures, systems, and components important to safety. The steel support structure of the storage area is designed to withstand the design basis earthquake and tornado winds to ensure it does not damage the canister handling machine.

The staff finds that the applicant has identified the proper design criteria for the structures, systems, and components not important to safety, and that the appropriate codes and standards are identified.

4.2 Evaluation Findings

Based on its review of the information presented in the SAR, the staff makes the following evaluation findings regarding the proposed ISF Facility:

- The staff finds that the materials to be transported in the transfer casks and subsequently stored at the ISF Facility are appropriately identified and in compliance with 10 CFR §72.2(a)(1).

- The staff finds that the structures, systems, and components important to safety have been properly classified and that the associated categories are consistent with the regulatory requirements of 10 CFR §72.144(a), and that the supporting technical information presented in the SAR is in accordance with 10 CFR §72.24(n). This list of structures, systems, and components is based on the definition in 10 CFR §72.3 of structures, systems, and components important to safety. The SAR appropriately specifies the design criteria for the structures, systems, and components important to safety in accordance with 10 CFR §72.120(a). The design criteria are to be included in the quality assurance procedures in accordance with 10 CFR §72.144(c).

- The staff finds that the design criteria are described in sufficient detail to satisfy the requirements of 10 CFR §72.24(c).

- The staff finds that the structural design criteria given in Section 3.2, "Structural and Mechanical Safety Criteria," of the SAR for the structures, systems, and components important to safety are developed from site characteristics and are used in the determination of structural loads and load combination analyses. The values for these parameters form the basis for the structural design, mechanical design, shielding, confinement, and criticality assessments of the ISF Facility. These design criteria satisfy the requirements of 10 CFR §72.120(a) and §72.122(h)(5). Additionally, the structures, systems, and components important to safety will be designed to quality standards commensurate with important to safety functions performed to satisfy the requirements of 10 CFR §72.144(c).

- The staff finds that the seismic design criteria are appropriately identified in accordance with 10 CFR §72.102(f)(2), §72.120(a), and §72.122(b)(1) and (2). The seismic design criteria are in accordance with the site-specific seismic hazards analysis given in Chapter 2, "Site Characteristics," of the SAR.

- The staff finds that the explosion considerations in the SAR are consistent with standard design criteria required by 10 CFR §72.122(c). Design peak incident pressures have been appropriately defined.

- The staff finds that the load combinations design criteria are adequately considered for the design of structures, systems, and components as required by 10 CFR §72.122(b)(1) and (2). Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena have been considered.

- The staff finds that the bounding dose rate design criteria given in the SAR are consistent with the requirements of 10 CFR §72.104(a) as identified at the site boundary in accordance with the requirements of 10 CFR §72.106(a). The design criteria for the storage and handling of SNF have been properly specified as required by 10 CFR §72.128. The staff's review of the shielding evaluation is provided in Chapter 7 of this SER. The confinement evaluation is provided in Chapter 9 of this SER. The radiation protection evaluation is provided in Chapter 11 of this SER.

- The staff finds that the design criteria for criticality are identified in the SAR as required by 10 CFR §72.124(a)–(c). The criticality evaluation is provided in Chapter 8 of this SER.

- The decommissioning findings are discussed in Chapter 13 of this SER.

- The staff finds that the ISF Facility design, which includes use of the ISF storage system, allows for retrieval of SNF in accordance with 10 CFR §72.122(l). Storage systems are designed to ensure adequate safety during normal and accident conditions in accordance with 10 CFR §72.128(a).

- The staff finds that the applicant has sufficiently defined the transfer cask trunnion design criteria for single-failure-proof systems in accordance with NUREG–0612. This satisfies the requirements of 10 CFR §72.120(a) and 10 CFR §72.122(h)(5).

4.3 References

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