# 4 STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA EVALUATION

## 4.1 Conduct of Review

Chapter 3 of the SAR identifies the principal design criteria for the Facility. These design criteria are derived from the requirements of 10 CFR Part 72 and applicable industry codes and standards. The SAR identifies PWR and BWR spent fuel as the material to be stored. It also identifies the general design criteria for structures, systems, and components classified as important to safety with respect to withstanding the effects of environmental conditions and natural phenomena. The worst case loads for normal, off-normal, and accident conditions are identified. Structures, systems, and components important to safety are designed for safe confinement and storage of the spent nuclear fuel without the release of radioactive material. This chapter also categorizes all structures, systems, and components as either important to safety or not important to safety. Table 3.6-1 of the SAR provides a summary of the key design criteria for the Facility. The design criteria are compared to the actual design in subsequent chapters.

The storage cask to be used at the Facility is the HI-STORM 100 Cask System as described in the HI-STORM 100 FSAR (Holtec International, 2000). The HI-STORM 100 Cask System has been approved by NRC for general use under Certificate of Compliance No. 1014 (Nuclear Regulatory Commission, 2000a). Where applicable, the staff relied on the review carried out during the certification process of the cask system, as documented in the NRC's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b).

### 4.1.1 Materials to be Stored

The materials to be stored at the Facility are PWR and BWR spent fuel assemblies that are approved for storage in the HI-STORM 100 Cask System. The approved contents are specified in Appendix B of Certificate of Compliance No. 1014. The physical, thermal, and radiological characteristics of the spent nuclear fuel are described in detail and evaluated in the HI-STORM 100 FSAR. Section 3.1.1 of the PFS Facility SAR provides a brief discussion of the materials to be stored at the Facility. This discussion is consistent with the information in the HI-STORM 100 FSAR.

**Table 4-1. Summary of Private Fuel Storage Facility materials to be stored and spent fuel specifications**

| Design Parameters | Design Conditions | Applicable Reference |
|---|---|---|
| Type of Fuel | See Appendix B of HI-STORM 100 Certificate of Compliance | HI-STORM 100 FSAR |
| Fuel Characteristics | See Appendix B of HI-STORM 100 Certificate of Compliance | HI-STORM 100 FSAR |

### 4.1.2 Classification of Structures, Systems, and Components

This section contains a review of SAR Section 3.4, Classification of Structures, Systems, and Components. The staff reviewed the discussion on classifications of structures, systems, and components with respect to the following regulatory requirements:

- 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 72.144(a) requires that the licensee establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this subpart. The licensee 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 is licensed. The licensee 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.

In 10 CFR 72.3, structures, systems, and components are identified as items whose functions are to: (1) maintain the conditions required to store spent nuclear fuel safely; (2) prevent damage to the spent fuel container during handling and storage; and (3) provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. The SAR list the structures, systems, and components based on this definition as required by 10 CFR 72.120(a).

SAR Section 3.3, Safety Protection Systems identifies safety protection systems and provides a brief description of the important characteristics of each system. The classification consists of two levels: important to safety and not important to safety. The important to safety classification contains three categories based on the potential impact to safe operation:

**Classification Category A—Critical to Safe Operation**, whose failure or malfunction could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.

**Classification Category B—Major Impact on Safety**, whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with failure of an additional item, could result in an unsafe condition.

**Classification Category C—Minor Impact on Safety**, whose failure or malfunction would not be likely to create a situation adversely affecting public health and safety.

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

Those structures, systems, and components considered important to safety are identified in Sections 3.3 and 3.4 of the SAR and are provided, with corresponding categories, in Table 4-2 in this SER. Details associated with the first four items (spent nuclear fuel canister, storage cask, transfer cask, and associated lifting devices) in Table 4-2 are cask-specific and presented in the HI-STORM 100 FSAR.

**Table 4-2. Quality assurance classification of structures, systems, and components important to safety (Based on SAR Table 3.4-1)**

| Category | Structures, Systems, and Components | Logic |
|---|---|---|
| A | Spent Nuclear Fuel Canister | Serves as the primary confinement structure for the Spent Nuclear Fuel assemblies and is designed to remain intact under all accident conditions analyzed. It provides confinement, criticality control, heat transfer capability, and radiation shielding. |
| B | Storage Cask | Serves as the primary component for protecting the canister during storage from environmental conditions and provides radiation shielding and canister heat rejection. |
| B | Transfer Cask | Designed to support the canister during transfer lift operations and provide radiation shielding and canister heat rejection. |
| B | Associated Lifting Devices | Designed to preclude the accidental drop of a canister. |
| B | Canister Transfer Building | Designed to protect the canister from adverse natural phenomena during shipping cask load/unload operations and canister transfer operations, provide radiological shielding to workers during transfer operations, and support for the canister transfer crane. The Canister Transfer Building also houses the fire-suppression system. |
| B | Canister Transfer Overhead Bridge Crane | Designed as a single failure proof system to preclude the accidental drop of a shipping cask during load/unload operations or a canister during the canister transfer operations. |
| B | Canister Transfer Semi-Gantry Crane | Designed as a single failure proof system to preclude the accidental drop of a shipping cask during load/unload operations or a canister during the canister transfer operations. |
| B | Seismic Support Struts | Designed to ensure that the transfer, storage, and shipping casks will remain stable and not topple in the event of an earthquake. |
| C | Cask Storage Pads | Designed to ensure a stable and level support surface for the storage cask under normal, off-normal, and accident conditions. It provides a yielding surface for the drop/tip-over of the storage cask. |

The  spent nuclear fuel canister has been properly classified as a Category A important to safety item because it serves as the primary confinement structure.  Its failure could lead to the release of radioactive material.  Sufficient description of the spent nuclear fuel canister is provided in SAR Sections 3.4.1.1, Canister, and 4.2.1.4, Components.

The following components have been properly classified as Category B important to safety items:  the storage cask, the transfer cask, the canister transfer building, the canister transfer overhead bridge crane, the canister transfer semi-gantry cranes, associated lifting devices, and the seismic support struts.  Each of these items is designed to protect the spent nuclear fuel canister during specific phases of handling and storage of the spent fuel.  Failure of one or more of these Category B items, combined with the subsequent failure of the canister, is necessary to lead to a condition adversely affecting public health and safety.

The HI-STORM 100 storage cask protects the spent nuclear fuel canister during storage. Sufficient description of the storage cask is provided in SAR Sections 3.4.1.2, Concrete Storage Cask, and 4.2.1.4, Components.  The transfer cask protects the spent nuclear fuel canister during transfer lift operations in the Canister Transfer Building.  Sufficient description of the transfer cask is provided in SAR Sections 3.4.1.3, Transfer Cask, and 4.7.3, HI-STORM Transfer Equipment.  The Canister Transfer Building is designed to protect the canisters from adverse natural phenomena during cask loading/unloading and canister transfer operations. The building also provides radiation protection to workers during canister transfer operations and support of the canister transfer cranes. Sufficient description of the Canister Transfer Building is provided in SAR Section 4.7.1, Canister Transfer Building.  The canister transfer cranes and associated lifting devices are important to safety because they will be used to support the spent nuclear fuel canister and transfer cask during the transfer process. Sufficient description of the canister transfer cranes is provided in SAR Sections 3.4.4, Canister Transfer Cranes, and 4.7.2, Canister Transfer Cranes.  Sufficient description of the associated lifting devices is provided in SAR Sections 3.4.1.4, Lifting Devices, and 4.7.3, HI-STORM Transfer Equipment.  Seismic support struts are used to support the storage, transfer, and shipping casks during canister transfer operations in case of an earthquake.  These support struts are to be attached to the walls of the Canister Transfer Building and provide support during a seismic event.  Adequate description of the seismic struts is provided in SAR Sections 3.4.5, Seismic Support Struts, and 4.7.1.4.1, Seismic Support Struts.  Based on the above discussion, the staff concludes that these Category B important to safety items are correctly classified.

Failure of the cask storage pads would not create a situation adversely affecting public health and safety. Sufficient description of the cask storage pad is provided in SAR Sections 3.4.2, Cask Storage Pads, and 4.2.3, Cask Storage Pads. Consequently, the staff concluded that the cask storage pads have been correctly identified as Category C important to safety items.

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

Based on  SAR Table 3.4-1, the classification of structures, systems, and components not important to safety includes items or services that do not involve a safety related function and that are not subject to special utility requirements or NRC-imposed regulatory requirements.

Structures, systems, and components not important to safety include: the PFS Facility infrastructure, Security and Heath Physics Building, Administration Building, Operations and Maintenance Building, fire detection and suppression systems, security systems, electrical systems, radiation monitors, temperature monitoring system, flood control berm, cask transporter, and offsite transportation components. The storage facility infrastructure, buildings, and facilities are necessary to support operation of the Facility. However, they are not necessary to ensure safe storage of the spent fuel because the storage cask system is passive. Therefore, they are classified as not important to safety.

The fire detection and suppression systems are contained within the Canister Transfer Building. The construction materials of the Canister Transfer Building do not support combustion, and the fire-prone materials are limited to diesel fuel and tires of the heavy haul trucks. Fires are analyzed in the accident analysis section of the SAR. The area surrounding the storage pads and Canister Transfer Building includes a gravel-covered fire break with vegetation control to limit potential fuel for fires. The nonflammable nature of the materials of construction, other passive design features, and the limited fuel sources at the Facility lead to the conclusion that the fire detection and suppression systems are correctly classified as not important to safety.

There are a number of systems that are security related: intrusion detection system, closed circuit television system, restricted area lighting, and security alarm stations. Each system is used to support the activities of the security personnel who monitor the controlled area of the facility. If systems fail, the security personnel can still perform their required functions. Therefore, the security systems are correctly classified as not important to safety.

Because the HI-STORM 100 storage cask system is a passive system, the uninterrupted power supply, backup diesel generator, and normal electrical power can also be classified as not important to safety. No electrical power is required for the storage system to perform its design functions.

The passive design of the cask also affects classification of the radiation monitors and temperature monitoring system. The radiation monitors are established to protect the health and safety of the workers. It has been demonstrated by analysis that the radiation levels at the site boundary will be below those identified in the applicable radiation protection regulations. The public is restricted from access into the controlled area. Therefore, the radiation monitors are correctly classified as not important to public safety.

The thermal monitors track the temperature of the air in the cooling passages of the storage cask. Upon loss of thermal monitoring, an alarm will sound and repair of the monitoring system will begin. The thermal monitoring system is intended to identify blockage of the cask cooling air passages and resulting rise of the cask temperature. The cask, by design, is not adversely affected by complete blockage of the air passages for 72 h. It is not necessary to continuously monitor the temperature since the canister since at least 72 h. must pass before the canister and fuel cladding temperature reach the allowable limits. Therefore, the thermal monitoring system is appropriately classified as not important to safety.

The flood control berm and drainage ditch are to prevent sheet flow over the site, to facilitate maintenance at the site and to maintain access to the casks on the storage pads in case of flooding. The flood control berm is not important to safety because the Facility elevation is above the PMF level. Further, the HI-STORM 100 storage cask is designed to resist the effects of full immersion in flood waters.

The cask transporter is also classified as not important to safety. Potential failure mechanisms of the transporter involve the drive-train, brakes, electrical system, or lift beam hydraulic ram. None of these potential failures would cause the transporter or the cask to tipover. Of these potential failures, only those that could drop the cask would have a possibility of damaging the cask or its internal components. The HI-STORM 100 FSAR (Holtec International, 2000) has demonstrated that the storage cask can be dropped a height of 11 in. without impairing confinement system integrity or fuel retrievability. However, the 11 in. drop height is based on a softer pad than is proposed at the PFS Facility. The cask storage pads at the proposed PFS Facility will be stiffer due to increased stiffness of the soil-cement layer overlaying the existing soil. Therefore, the applicant has stated that the transporter will be designed to limit the lift height of the cask to 9 in. This height is based on site-specific analyses of drop events on the PFS Facility storage pads to estimate the limiting deceleration level on the fuel rods (Holtec International, 2001). As calculated by PFS, a vertical drop of the PFS cask upon the cask storage pad, up to 9 in. will produce decelerations bounded by the 45g design basis. The cask transporter will also be designed to preclude tipover under site-specific seismic, tornado winds, and tornado missile loads. Therefore, the cask transporter can be classified as an item not important to safety.

Another group of structures, systems, and components that are classified as not important to safety are the road transport and railroad line alternatives. These are classified as such because the shipping casks that will be used to transport the spent fuel are designed and approved under 10 CFR Part 71. Transportation equipment is outside the scope of this review.

#### 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 SAR Chapter 3, Principal Design Criteria; documents cited in the SAR; and other relevant literature. The staff found that the SAR appropriately classifies the structures, systems, and components important to safety. The design criteria for the structures, systems, and components important to safety are adequately identified as required by 10 CFR 72.120(a). Details of the quality assurance program evaluation are contained in Chapter 12 of this SER. 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 associated technical information content of the application, as specified in 10 CFR 72.24(n).

### 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 Facility are described in SAR Chapter 3, Principal Design Criteria. This section contains a review of Section 3.2, Structural and Mechanical Safety Criteria; Section 3.3, Safety Protection Systems; and SAR Section 3.6, Summary of Design Criteria. Details of the design criteria evaluation are provided in Sections 4.1.3.1 to 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 with respect to the following regulatory requirements:

- 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(h) specifies the criteria for confinement barriers and systems, including: 72.122(h)(1), which requires that the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage; 72.122(h)(3), which requires that ventilation systems and off-gas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions; 72.122(h)(4), which requires that storage confinement systems have the capability for monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions; and 72.122(h)(5), which requires that the high-level radioactive waste be packaged to allow handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits (also, the package must be designed to confine the high-level radioactive waste for the duration of the license).

- 10 CFR 72.144(c) requires that the licensee base the requirements and procedures of its quality assurance program 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.

A summary of the PFS Facility general design criteria is provided in Table 4-3 of this SER.

**Table 4-3. Summary of Private Fuel Storage Facility Design Criteria—General (Based on SAR Table 3.6-1)**

| Design Parameters | Design Conditions | Reference |
|---|---|---|
| Design Life | 40 yr | PFS Facility SAR |
| Storage Capacity | 40,000 MTU of commercial spent nuclear fuel | PFS Facility SAR |
| Number of Casks | Approximately 4,000 casks | PFS Facility SAR |

The design life of structures, systems, and components important to safety is based on their ability to withstand the applied loads. The applied loads are defined in terms of 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 by the referenced codes and standards. The majority of design loads for the PFS Facility, identified in Table 3.6-1 of the SAR, are based on a 50 yr mean recurrence interval ($P_a = 0.02$) or longer. The loads specified in the SAR are consistent with standard engineering practice, as identified in American Society of Civil Engineers (ASCE) ASCE 7-95 (American Society of Civil Engineers, 1996) that identifies the minimum design loads for buildings and other structures. The storage capacity and number of casks to be stored at the Facility have been identified in the SAR. Based on a given annual probability ($P_a$) of exceeding the design load and service life (n), the probability ($P_n$) that the design load will be equaled or exceeded at least once during the service life is given by (Commentary Section of ASCE 7–95, 1996):

$$P_n = 1 - (1 - P_a)^n$$

The principal criteria used in the design are given in Section 3.2, Structural and Mechanical Safety Criteria, and summarized in SAR Section 3.6, Summary of Design Criteria. The storage system characteristics are given in Table 4-4 of this SER. These characteristics are based on information provided in the HI-STORM 100 FSAR.

**Table 4-4. Summary of Private Fuel Storage Facility Design Criteria—Storage System Characteristics (Based on SAR Table 3.6-1)**

| Design Parameters | Design Conditions | Applicable Criteria and Codes |
|---|---|---|
| Canister Capacity | Maximum 24 PWR assemblies/canister<br>Maximum 68 BWR assemblies/canister | HI-STORM FSAR, Section 1.1 |
| Weights (maximum) | Storage Cask 268,334 lb<br>Loaded Canister 87,241 lb<br>Transfer Cask 152,636 lb<br>Shipping Cask 153,080 lb | HI-STORM FSAR, Table 3.2.1<br>HI-STORM FSAR, Table 3.2.1<br>HI-STORM FSAR, Table 3.2.2<br>Shipping Cask SAR |

The staff reviewed the general design criteria for the storage system characteristics identified in Tables 4-3 and 4-4 of this SER. The staff found that they are consistent with the HI-STORM 100 FSAR. Definitions of the normal, off-normal, and accident loads are given in SAR Section 3.2, Structural and Mechanical Safety Criteria. The quality standards for design basis of structures, systems, and components important to safety are provided in SAR Chapters 3, Principal Design Criteria, and 11, Quality Assurance. These design criteria, in part satisfy the requirements of 10 CFR 72.120(a) and 72.122(h) in that design criteria are identified, and 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).

#### 4.1.3.2 Structural

The staff reviewed the discussion on structural design criteria of structures, systems, and components in the SAR with respect to the following regulatory requirements:

- 10 CFR 72.102(f) requires that the design earthquake for use in the design of structures 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 72.120(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 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.

- 72.122(b)(2) requires structures, systems, and components important to safety to 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) 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 (ii) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also 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 high-level radioactive waste or on structures, systems, and components important to safety.

- 72.122(b)(4) specifies that if the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

- 10 CFR 72.122(c) requires that structures, systems, and components important to safety 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.

SAR Section 3.2, Structural and Mechanical Safety Criteria, addresses the structural and mechanical design criteria. The design criteria for the site include: dead loads, live loads, wind, tornado, tornado missiles, flood, seismicity, snow and ice, soil pressure, explosion overpressure, fire, ambient temperature and humidity, solar radiation, and lightning. Information on the derivation of site-specific design criteria for the meteorology, hydrology, and seismology are contained in SAR Chapter 2, Site Characteristics.

The structural design criteria for major components are provided in Table 4-5 of this SER. The design of the proposed Facility is based on the use the HI-STORM 100 Cask System, which has been approved by the NRC for use under the general license provisions of 10 CFR Part 72.

The design criteria for the pressure vessel portions of the cask system conform to standard engineering practice, as identified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998). The ASME Boiler and Pressure Vessel Code establishes rules of safety governing the 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.

For concrete components, as identified in Table 4-5, the design criteria are based on the American Concrete Institute's ACI 349-90. ACI 349-90 specifies the proper design and construction of concrete structures that form part of a nuclear power plant and that have nuclear safety related functions, but does not cover concrete reactor vessels and concrete containment structures.  The structures covered by the ACI code include concrete structures inside and outside the containment system.

The design criteria for structural steel, as identified in Table 4-5, are based on the American Institute of Steel Construction's ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction, 1994).  ANSI/AISC N690 is the specification and commentary for the design, fabrication, and erection of structural steel for safety-related structures for nuclear facilities.

The design criteria for the cranes are in accordance with ASME NOG–1.  The ASME NOG–1 covers electric overhead and gantry multiple girder cranes with top running bridge and trolley and components of cranes used at nuclear facilities.  In addition, NUREG–0554 (Nuclear Regulatory Commission, 1979), and NUREG–0612 (Nuclear Regulatory Commission, 1980) are identified for the design criteria of the cranes. NUREG–0554 identifies design criteria for single-failure-proof cranes for nuclear power plants. NUREG–0612 identifies controls for handling heavy loads at nuclear power plants.

**Table 4-5. Summary of Private Fuel Storage Facility Design Criteria— Component Structural Design Criteria (Based on SAR Table 3.6-1)**

| Design Parameters | Design Conditions | Applicable Criteria and Codes |
|---|---|---|
| HI-STORM 100 Cask System Load Criteria | Canister:<br>Internals:<br>Storage Cask:<br>Transfer Cask:<br>} See HI-STORM 100 FSAR Table 2.2.6 | ASME III, NB<br>ASME III, NG<br>ASME III, NF, ACI–349<br>ASME III, NF, ANSI N14.6 |
| Canister Transfer Crane Designs | Type I, single-failure-proof<br>200-ton overhead bridge crane<br>150-ton semi-gantry crane | ASME NOG–1,<br>NUREG–0554, and<br>NUREG–0612 |
| Cask Storage Pad Designs | Normal, off-normal, and accident loading | ACI 349-90 |
| Canister Transfer Building Reinforced Concrete Designs | Normal, off-normal, and accident loading | ACI 349-90 |
| Canister Transfer Building Structural Steel Designs | Normal, off-normal, and accident loading | ANSI/AISC N690 |

The structural design loads for structures, systems, and components important to safety are provided in Table 4-6 of this SER.  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.  Important to safety design criteria for the HI-STORM 100 storage system are described in its FSAR. Table 4-6 identifies the HI-STORM 100 design criteria in relationship to the PFS Facility design criteria. Review in this SER is limited to identification of enveloping design criteria. Adequacy of the HI-STORM 100 Cask System design criteria are discussed in the NRC's HI-STORM SER.

Site-specific design criteria not enveloped by the HI-STORM FSAR criteria are identified in Section 3.2 of the SAR. These criteria include the specific site criterion, storage system affected, and the corresponding section in the SAR where it is addressed. Structural design criteria and radiological protection and confinement criterion are identified. The structural criteria are discussed in this chapter of the SER. Consideration of the radiological protection and confinement criterion are contained in other chapters of the SER.

**Table 4-6. Summary of Private Fuel Storage Facility Design Criteria—Structural Design Loads (Based on SAR Table 3.6-1)**

| Design Parameters | PFS Facility Design Criteria | Applicable Criteria and Codes | HI-STORM 100 MPC Design Criteria (HI-STORM 100 FSAR, Table 2.0.1) | HI-STORM 100 Overpack Design Criteria (HI-STORM FSAR, Table 2.0.2) | HI-TRAC Transfer Cask Design Criteria (HI-STORM 100 FSAR, Table 2.0.3) |
|---|---|---|---|---|---|
| Wind | 90 mph, normal speed | ASCE–7 (0.02 annual frequency) | Protected by overpack | Enveloped by Tornado Wind | Protected in transfer facility |
| Tornado | 240 mph, maximum speed<br>190 mph, rotational speed<br>50 mph, translational speed<br>150 ft, radius of maximum speed<br>1.5 psi, pressure drop<br>0.6 psi/sec rate of drop | Regulatory Guide 1.76 | Protected by overpack | 360 mph, maximum speed<br>290 mph, rotational speed<br>70 mph, translational speed<br>3.0 psi, pressure drop | Protected in transfer facility |
| Tornado Missiles | 3990 lb automobile, 134 ft/sec<br>750 lb 12 in. schedule 40 pipe, 23 ft/sec<br>1124 lb wooden utility pole, 85 ft/sec<br>9 lb 1 in. diameter steel rod, 26 ft/sec<br>287 lb 6 in. schedule 40 pipe, 33 ft/sec<br>115 lb wood plank, 190 ft/sec | NUREG–0800, Section 3.5.1.4 | Protected by overpack | 3990 lb automobile, 185 ft/sec<br>275 lb 8 in. rigid solid steel cylinder, 185 ft/sec<br>1 in. diameter steel sphere, 185 ft/sec | 3990 lb automobile, 185 ft/sec<br>275 lb 8 in. rigid solid steel cylinder, 185 ft/sec<br>1 in. diameter steel sphere, 185 ft/sec |
| Flood | PFS Facility is not in a flood plain and is above the PMF elevation. Details contained in Section 2.3.2.3 of the PFS Facility SAR. | NUREG–0800, Section 3.4.1 | 125 ft. water depth | 125 ft. flood height<br>15 ft/sec flood velocity | Protected in transfer facility |
| Seismic | PGA of 0.711g, horizontal (both directions) and 0.695g vertical. Probabilistic design basis ground acceleration identified in Section 2.6 of the PFS Facility SAR. | 10 CFR 72.102 | $G_H + 0.53\ G_V \leq 0.53$ | $G_H + 0.53\ G_V \leq 0.53$ | NA |

**Table 4-6. Summary of Private Fuel Storage Facility Design Criteria—Structural Design Loads (Based on SAR Table 3.6-1)**

| Design Parameters | PFS Facility Design Criteria | Applicable Criteria and Codes | HI-STORM 100 MPC Design Criteria (HI-STORM 100 FSAR, Table 2.0.1) | HI-STORM 100 Overpack Design Criteria (HI-STORM FSAR, Table 2.0.2) | HI-TRAC Transfer Cask Design Criteria (HI-STORM 100 FSAR, Table 2.0.3) |
|---|---|---|---|---|---|
| Snow and Ice | P(g) = 45 psf | ASCE–7, Tooele County Building Department | Protected by Overpack | 100 psf | Protected in transfer facility |
| Allowable Soil Pressure | Static = 4 ksf max<br>Dynamic = Varies by footing type/size. Details contained in Section 2.6.1.12 of the SAR. | NUREG–0800, Section 2.5.4 | NA | NA | NA |
| Explosion Overpressure | The PFS Facility design and layout shall assure that the peak positive incident overpressure at important to safety structures, systems, and components does not exceed 1.0 psi from credible and offsite explosions. | Reg. Guide 1.91 | 60 psig (external) | 10 psid for 1 seconds<br>5 psid steady state | NA |
| Ambient Conditions | Low Temperature = −30 °F<br>Max. Annual Average Temp. = 51 °F<br>Average Daily Max. Temp. = 95 °F<br>Humidity = 0–100 percent | National Oceanic and Atmospheric Administration Data–Salt Lake City, Utah, Climate Data | See Tables 2.0.2 and 2.0.3 | Min. Ambient Temp. = -40 °F<br>Max. Ambient Temp. = 100 °F<br>Max. Yearly Average Temp. = 80 °F<br>Extreme Environmental Temperature = 125 °F | Min. Ambient Temp. = 0 °F<br>Max. Ambient Temp. = 100 °F<br>Max. Yearly Average Temp. = 100 °F |

$G_H$ = peak seismic horizontal ground acceleration
$G_V$ = peak seismic vertical ground acceleration
NA = not applicable

## Wind

Figure 6-1 in ASCE 7-95 (American Society of Civil Engineers, 1996) identifies a design basis wind speed of 90 mph for the region. Information provided in SAR Section 2.3.1.3.2, Extreme Winds, for Salt Lake City region identifies the wind speed with a 50-yr return period as 70.4 mph, which, taking into account a gust response factor, results in a 88.7 mph design wind speed. As identified in Table 2.3.5 of the SAR, the mean wind speed and direction at the PFS Facility site and Salt Lake City are consistent. Therefore, data obtained at Salt Lake City can be used to represent the site. The staff reviewed the design basis wind (90 mph) for the Facility and found that it is consistent with that identified in ASCE 7-95 (American Society of Civil Engineers, 1996) for this location. The requirements of 10 CFR 72.120(a) and 72.122(b) are satisfied in that the effects of site conditions and environmental conditions are considered in the Facility design.

## Tornado

The design basis tornado wind loads are based on information provided in Regulatory Guide 1.76 (U.S. Atomic Energy Commission, 1974). Tooele County is located in Tornado Intensity Region III, as defined by Regulatory Guide 1.76. The parameters for the tornado identified in the SAR are those given in Regulatory Guide 1.76. Based on data provided in SAR Section 2.3.1.3.3, Tornadoes, the most severe tornado observed in the region was classified as F1 with a corresponding wind speed of 73 to 112 mph. The specified design criteria specify greater wind speeds than those observed. Specifically, the PFS Facility design criterion for tornado specifies a maximum speed of 240 mph with an associated pressure drop of 1.5 psi. The probability of a tornado striking the PFS Facility site is given as $1.37 \times 10^{-6}$ per year in the PFS Facility SAR Section 2.3.1.3.3.

## Tornado Missiles

The tornado missiles, identified in the SAR, are those specified as Spectrum II missiles for Region III in NUREG–0800, Section 3.5.1.4 (Nuclear Regulatory Commission, 1981). These are considered to be representative of potential missiles present at the site. As identified in the HI-STORM 100 FSAR, the cask-specific tornado missiles correspond to Spectrum I missiles in NUREG–0800. Use of either Spectrum I or II missile is considered acceptable by the NRC. The staff reviewed the design basis tornado conditions for the Facility and found that they are consistent with design criteria, as specified by NUREG–0800, Section 3.5.1.4, to withstand tornadoes, in accordance with the requirements of 10 CFR 72.120(a) and 72.122(b).

## Flood

The maximum probable flood for the site is at elevation 4,468.8 ft above mean sea level in the southeast corner and 4,456.8 ft above mean sea level in the northeast corner. The corresponding site elevations are 4,475 and 4,463 ft above mean sea level, respectively. Analysis of the maximum probable flood level is based on the maximum probable precipitation and the surface hydrology of the region given in SAR Section 2.4, Surface Hydrology. In addition, an earthen berm and drainage ditch system are to be constructed at the site. The berm is designed to ensure that sheet flow will not approach the storage casks on the pads or the Canister Transfer Building. Therefore, the forces due to flood waters and flood protection measures do not need to be considered in design of structures, systems, and components important to safety. The staff therefore concludes that the Facility design is consistent with design criteria of NUREG–0800 and ASCE 7–95 to withstand floods as required by 10 CFR 72.120(a) and 72.122(b).

**Seismicity**

The staff reviewed the data presented in the SAR associated with seismic design criteria at the Facility. SAR Section 3.2.10, Seismic Design, gives the seismic design criteria, based on probabilistic site-specific seismology studies summarized in SAR Section 2.6, Geology and Seismology. PFS has requested an exemption from the seismic requirement of 10 CFR 72.102(f). Discussions of the implications of this request for exemption are contained in Section 2.1.6 of this SER. The resulting site-specific design response spectra are anchored at a peak ground acceleration (PGA) of 0.711g in both horizontal directions and 0.695g in vertical direction. The horizontal and vertical design response spectra curves have been identified in the Geomatrix Consultants, Inc. report (Geomatrix Consultants, Inc., 2001). The site-specific seismic design criteria of the Facility are not bounded by the HI-STORM 100 seismic design criteria. The seismic design criteria are based on the site-specific probabilistic seismic hazards analysis given in SAR Chapter 2, Site Characteristics, which has been evaluated in Chapter 2 of this SER. The applicant's analysis of the HI-STORM 100 storage cask under the site-specific design basis seismic event is evaluated in Chapters 5 and 15 of this SER. The staff reviewed the seismic design criteria for the Facility and found that they are properly identified as required by 10 CFR 72.120(a) and 72.122(b).

**Snow and Ice**

Figure 7-1 of ASCE 7-95 (American Society of Civil Engineers, 1996) identifies that the snow load at the proposed site should be estimated from results of a site-specific case study. Based on the elevation of the Goshute Reservation (elevation 4,600–4,700 ft above mean sea level), the Tooele County Building Department stated that a ground snow design load of 43 $lb/ft^2$ would be required to comply with the Uniform Building Code (UBC) (International Conference of Building Officials, 1997). SAR Section 3.2.3, Snow and Ice Load, states that the PFS Facility has a design ground snow load of 45 $lb/ft^2$ that bounds the UBC requirements. As identified in the HI-STORM 100 FSAR (Holtec International, 2000), the overpack is designed for 100 $lb/ft^2$ snow and ice load. This bounds the PFS Facility site design criteria. The staff reviewed the snow and ice loading criteria and determined that they are appropriate and in accordance with the requirements of 10 CFR 72.120(a) and 72.122(b).

**Allowable Soil Pressure**

The allowable soil pressure design criteria are based on site-specific investigations summarized in SAR Section 2.6.4, Stability of Subsurface Materials. Review of the soil classification and soil properties identified at the site are presented in Section 2.1.6.4, Stability of Subsurface Materials, of this SER. The allowable soil pressure design criteria are applicable to the storage pad and Canister Transfer Building designs. The applicant has presented acceptable analysis to determine values of allowable bearing pressure consistent with the site-specific soil properties and accepted design earthquake loading, as identified in Section 2.1.6.4 of this SER. The staff reviewed the SAR and determined that the allowable soil pressure design criteria are appropriately specified in accordance with the requirements of 10 CFR 72.120(a) and 10 CFR 72.122(b).

**Explosive Overpressure**

The explosive overpressure design criterion for the Facility is based on the assumption that all credible events that produce a peak positive incident overpressure will result in a pressure at the structures, systems, and components important to safety of less than 1 psi.

The 1 psi limit is based on Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1978) that states:

> A method for establishing the distances referred to above can be based on a level of peak positive incident overpressure (designated as $P_{so}$ in Ref.1) below which no significant damage would be expected. It is the judgement of the Staff that, for the structures, systems, and components, this level can be chosen as 1 psi (approximately 7 kPa).

During a blast event, the normal reflected pressure is twice the peak incident pressure $P_{so}$ of 1 psi. Therefore, a normal reflected pressure of 2 psi should be considered in the analysis of the structure when considering the blast pressure loading. The following identifies the analyses used by PFS to demonstrate that the site is configured so that no credible explosion will produce a peak positive incident overpressure greater than 1 psi.

The location of the Facility is beyond the range at which cargo explosions, such as a fuel tank truck or a truck loaded with explosives traveling on Skull Valley Road, could produce peak positive incident overpressures greater than 1 psi, as per Table 4-7 of this SER. The amount of hazardous cargo and distance are those identified in Regulatory Guide 1.91.

Explosive overpressure from other accident conditions is discussed in SAR Section 8.2.4, Explosion. Potential explosions at nearby industrial, transportation, and military facilities are considered but the distance from the Facility is such that the resulting pressure will be less than 1 psi. Potential explosions of the onsite diesel fuel oil storage tanks are not considered credible because of the low volatility and high flash point of the fuel. Diesel fuel is not a flammable liquid but is classified as a Class II combustible liquid with a flash point between 100 °F and 140 °F.

The propane tanks located on the Facility are also considered in Section 8.2.4 of the SAR. PFS has proposed four 5,000 gallon propane storage tanks to be located at least 1,800 ft from the closest structures, systems, and components important to safety. The postulated explosion includes atmospheric dispersion modeling to determine the maximum downwind distance from the tank that the concentration of propane in the plume could be above the lower explosive limit and to determine the overpressure created by delayed ignition of the resulting cloud.

As identified in the SAR, the explosive overpressure design criterion is enveloped by the wind pressure loading from a tornado, 1.5 psi differential plus the dynamic wind pressure load. The dynamic wind pressure is given by $q_z = 0.00256V^2$, where $q_z$ is in $lb/ft^2$ and V is in mph, as identified in ASCE 7-95. Therefore, a 240-mph wind speed relates to a dynamic wind pressure equal to 1.02 psi. To obtain the load on the structure, this value is multiplied by the wall pressure coefficient that varies from 0.5–0.8 which gives dynamic pressure load from 0.51 to 0.82 psi. Adding the tornado differential pressure drop to the dynamic wind pressure results in a total pressure load of 2.01 to 2.32 psi. This load exceeds both the explosion peak incident and normal reflected pressure. Therefore the staff concludes that the tornado total pressure load is a bounding load with respect to the explosive overpressure load.

The staff reviewed the explosion considerations in the SAR and found that they are consistent with standard design criteria, NFPA 30 (National Fire Protection Association, 1996) and Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1981) as required by 10 CFR 72.122(c).

**Table 4-7. Summary of Private Fuel Storage Facility Blast Overpressures (Based on Regulatory Guide 1.91)**

| Condition | Trinitrotoluene (TNT) Equivalent (lb) | Distance for Pressure ≤1.0 psi (ft) | Comment |
|---|---|---|---|
| Offsite based on maximum cargo for a Highway Truck | 50,000 | 1,658 | 1.9 mi to Facility (10,032 ft) |
| Offsite based on maximum cargo for a Railroad Car | 132,000 | 2,291 | 24 mi to Facility (126,720 ft) |
| Offsite based on maximum cargo for a River Vessel | 10,000,000 | 9,696 | No nearby river |
| Offsite Space Shuttle Rocket Testing at Tekoi | 1,200,000 | 4,782 | 2.3 mi to Facility (12,144 ft) |
| Onsite 5,000 gal. Propane Tanks (with atmospheric dispersion) | 7,135 | 866 | 1,800 ft to Canister Transfer Building and nearest storage cask |

**Lightning**

During thunderstorms, a lightning strike is possible. Therefore, structures, systems, and components located outdoors will be designed to withstand the effects of a lightning strike. The Canister Transfer Building is provided with lightning protection in accordance with the requirements of NFPA 780 (National Fire Protection Association, 1997). NFPA 780 provides for the protection of people, buildings, special occupancies, structures containing flammable liquids and gases, and other entities against lightning damage. The HI-STORM 100 storage cask is also designed for lightning protection. Any lightning strike on the cask will discharge through its steel shell to the ground and have no adverse impact on the cask or fuel. The staff reviewed the lightning design criterion 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).

**Load Combinations**

The load combinations identified in Table 3.6-1 of the SAR are used in the analysis of structures, systems, and components important to safety. These load combinations are based on the requirements of ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992), American Concrete Institute ACI 349-90 (American Concrete Institute, 1989), ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction,1994), and ASME NOG–1 (American Society of Mechanical Engineers,1995a). The staff reviewed the PFS Facility documentation and determined that the load combinations design criteria are appropriately considered for the design of structures, systems, and components important to safety as required by 10 CFR 72.122(b). 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 above represent the structural loads that may be present at the site. The PFS Facility structures, systems, and components that are important to safety must be designed to withstand these structural loads, as applicable. The ability of the structures, systems, and components to perform their intended safety functions under the applicable structural design loads is evaluated in Chapters 5 and 15 of this SER.

As shown in Table 4-6 of this SER, the PFS Facility site-specific structural design criteria are bounded by the applicable structural design criteria for the HI-STORM 100 Cask System, except for the seismic design criteria. Thus, except for the seismic analysis, the structural analysis presented in the HI-STORM 100 FSAR and the NRC's structural evaluation as documented in the HI-STORM 100 SER are valid for the PFS Facility. Because the seismic design loads for the PFS Facility are not enveloped by the seismic design loads for the HI-STORM 100 Cask System, the applicant performed an analysis to demonstrate that the HI-STORM 100 storage cask would perform acceptably under the site-specific design basis seismic event. This analysis is also evaluated in Chapters 5 and 15 of this SER.

The staff reviewed the PFS Facility documentation and determined that the principal design criteria, given in SAR Section 3.2, Structural and Mechanical Safety Criteria, considered for the design of structures, systems, and components are developed from appropriate site characteristics and are used in the determination of appropriate structural loads and load combination analyses. The values for these parameters form the basis for the structural design, mechanical design, and criticality assessment of the Facility.

#### 4.1.3.3 Thermal

The staff reviewed the discussion on thermal design criteria of structures, systems, and components with respect to the following regulatory requirement:

- 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 must 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.

Thermal design criteria are based on both environmental conditions and heat generated by the materials stored.

Ambient condition design criteria are based on site-specific meteorological conditions. Additionally, an onsite meteorological measurement program has been in place. The minimum and maximum annual average and average daily maximum design temperatures identified for the site are −30, 51, and 95 °F, respectively. The design temperatures are based on data from the region, which are consistent with the values measured to date under 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 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).

The site-specific maximum total insolation for a 12-hour period was 684.6 W/m$^{2}$, (706.5 g cal/cm$^{2}$). Using the 30-yr database for Salt Lake City, the maximum total insolation for a 12-hr period was 730.4 W/m$^{2}$ (753.8 g cal/cm$^{2}$). The HI-STORM 100 cask has been evaluated for the solar insolation values specified in 10 CFR Part 71.71(c)(1) which are 774 W/m$^{2}$ (800 g cal/cm$^{2}$) for flat surfaces and 387 W/m$^{2}$ (400 g cal/cm$^{2}$) for curved surfaces. The Standard Review Plan for dry cask storage systems, NUREG–1536 (Nuclear Regulatory Commission, 1997) states, "The NRC staff accepts insolation presented in 10 CFR Part 71 for 10 CFR Part 72 applications. Because of the large thermal inertia of a storage cask, the values listed in 10 CFR 71.71 may be treated as the average insolence, calculated by averaging over a 24-hour day the reported 10 CFR Part 71 values for insolence over a 12-hour solar day, in a steady-state calculation." The staff concluded that both site-specific measurement and regional data are bounded by the design of the HI-STORM 100 cask.

No specific design criteria are identified in Table 3.6-1 for fire. As identified in SAR Section 8.2.5, a fire is classified as a human-induced Design Event IV as defined in ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992). The design of the Facility is such that all structures, systems, and components are located within a region covered with crushed rock with at least 100 ft to the boundary of the restricted area fence. Additionally, PFS will plant a 300 ft crested wheatgrass barrier around the restricted area. Therefore, there is no credible wildfire load on structures, systems, and components important to safety. A range of onsite fire scenarios has been evaluated. Bounding fire events are based on 50 gal. of diesel fuel in the transfer cells or cask storage pads from the cask transporter tank and 300 gal. of diesel fuel from the heavy haul vehicle tanks and vehicle tires in the cask load/unload bay. Operational restrictions are in place to ensure that these levels are not exceeded. The crushed rock fire break and operation restrictions are sufficient to limit the fire load to the identified structures, systems, and components. The staff reviewed the fire considerations in the SAR and found 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 that structures, systems, and components important to safety will be designed and located so that they can perform their safety functions effectively under credible fire exposure conditions. Fire design criteria for the HI-STORM 100 cask system are based on 50 gal. of combustible transporter fuel. They are identified in the FSAR as a 1,475 °F fire with duration from 217 to 288 seconds.

The storage systems are passive and incorporate passive heat removal. Evaluation of the thermal design criteria for the cask system was carried out during licensing of the HI-STORM 100 Cask System and is documented in the NRC's HI-STORM 100 SER.

Design temperatures for various materials are identified in the SAR and are in compliance with acceptable codes. ACI 349-90 (American Concrete Institute, 1989) specifies the maximum temperature for normal operation and accident conditions. Allowable temperature for the fuel cladding is based on NUREG–1536. The Boiler and Pressure Vessel Code, ASME Section II, Part D, Table 1A specifies a design temperature for steel casks under all load conditions (American Society of Mechanical Engineers, 1999). The performance requirements for all materials, as identified by the acceptable temperature, required for compliance with 10 CFR 72.120(a) are in conformance with accepted standards. Cask-specific material properties given in the SAR are derived from the HI-STORM FSAR Table 2.2.3. The PFS Facility design temperatures are identical to those given in the HI-STORM 100 FSAR (Holtec International, 2000).

#### 4.1.3.4 Shielding and Confinement

The staff reviewed the discussion on shielding and confinement design criteria of structures, systems, and components with respect to the following regulatory requirements:

- 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 be limited to 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.

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

- 10 CFR 72.122(h)(1) requires the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage.

- 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 be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given set point 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(c) requires that effluent and direct radiation monitoring meet the following criteria: (1) as appropriate for the handling and storage system, effluent systems must be provided; and (2) areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.

- 10 CFR 72.126(d) requires that the ISFSI be designed to provide means to limit to ALARA levels the release of radioactive materials in effluents during normal operations; 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 10 CFR 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 10 CFR 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, high-level radioactive waste 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 72.128(b) requires that radioactive waste treatment facilities be provided. Provisions must be made for the packing of site-generated, low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

Criteria used in the design of cask radiological protection features and confinement design of the cask systems are provided in the SAR and the HI-STORM 100 FSAR and are summarized in Tables 4-9 and 4-10 of this SER. The basic concept for the PFS Facility shielding and confinement system is protection by multiple barriers and systems, as required by 10 CFR 72.126(a)–(c). The use of the HI-STORM 100 Cask System, which is a sealed canister-based system, satisfies the requirements of 10 CFR 72.122(h)(1). 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).

**Table 4-9. Summary of Private Fuel Storage Facility Design Criteria—Radiation Protection/Shielding Design (Based on SAR Table 3.6-1)**

| Design Parameters | Design Conditions | | Reference |
|---|---|---|---|
| Storage Systems Design Dose Rate Limits | Cask Side Surface<br>Cask Inlet/Exit Vent Area<br>Cask Top Surface | 40 mrem/hr<br>60 mrem/hr<br>10 mrem/hr | HI-STORM 100 FSAR, Section 2.3.5.2 |
| Individual Workers Dose Rate | Total effective dose equivalent<br>Dose to eye lens<br>Dose to skin and extremities | 5 rem/yr<br>15 rem/yr<br>50 rem/yr | 10 CFR 20.1201 |
| Restricted Area Boundary Dose Rate | 2 mrem/hr, maximum | | 10 CFR 20.1301 |
| Owner-Controlled Area Boundary Dose Rate | 25 mrem/yr whole body and<br>75 mrem/yr thyroid, maximum<br>25 mrem/yr to any other critical organ<br>5 rem accident dose or 50 rem total organ dose equivalent (one time) | | 10 CFR 72.104<br>10 CFR 72.106 |

The bounding dose rate design criteria are consistent with the requirements of 10 CFR 72.104(a) and 72.106(a).

**Table 4-10. Summary of Private Fuel Storage Facility Design Criteria—Confinement Design (Based on SAR Table 3.6-1)**

| Design Parameters | Design Conditions | Reference |
|---|---|---|
| Confinement Method | Welded closed steel canister | HI-STORM 100 FSAR, 2.3.2.1 |
| Confinement Barrier Design | HI-STORM canister: ASME III, NB | HI-STORM 100 FSAR, 2.3.2.1 |

The staff reviewed the design criteria for spent nuclear fuel storage and handling, and determined that they are appropriately identified as required by 10 CFR 72.128(a) and (b). A shielding evaluation has been performed in Chapter 7 of this SER, a confinement evaluation has been performed in Chapter 9 of this SER, and a radiation protection evaluation has been performed in Chapter 11 of this SER. Evaluation findings given in this chapter are drawn from Chapters 7, 9, and 11 of this SER.

#### 4.1.3.5 Criticality

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

- 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.

- 10 CFR 72.124(c) requires that a criticality monitoring system be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under Part 72 is not required.

Criteria used in criticality design of the cask systems are provided in the PFS Facility SAR and the HI-STORM 100 FSAR and are summarized in Table 4-11 of this SER. The staff's criticality evaluation is discussed in Chapter 8 of this SER. The design criteria for criticality are identified in the SAR as required by 10 CFR 72.124(a) through (c).

**Table 4-11. Summary of Private Fuel Storage Facility Design Criteria—Criticality Design**

| Design Parameters | Design Conditions | Applicable Criteria and Codes |
|---|---|---|
| Control Method | Geometry of fuel assemblies assuming no moderator | HI-STORM 100 FSAR, Section 2.3.4.1 |
| $k_{eff}$ | $\leq 0.95$ | NUREG–1567 |

#### 4.1.3.6 Decommissioning

The staff's decommissioning evaluation is presented in Chapter 13 of this SER.

#### 4.1.3.7 Retrieval

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

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

- 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.

The spent fuel will be stored in and handled with the HI-STORM 100 Cask System which has been approved for use under the general license provisions of 10 CFR Part 72. As discussed in the HI-STORM 100 FSAR and the staff's related SER, the HI-STORM 100 Cask System is designed to ensure adequate safety and to protect fuel integrity and retrievability under the design basis loads specified in the HI-STORM 100 FSAR. The design basis loads considered in the HI-STORM 100 FSAR bound the structural and thermal loads found at the Facility except for the seismic load (see SER Sections 4.1.3.2 and 4.1.3.3). For the seismic event, the applicant provided an analysis which demonstrated that the HI-STORM 100 storage cask would neither tipover nor slide during a site-specific seismic event. Further, the loads on the canister would remain bounded by the canister loads considered in the HI-STORM 100 FSAR. (The seismic analysis is evaluated in Chapters 5 and 15 of this SER.)

Based on the foregoing discussion, there is reasonable assurance that the HI-STORM 100 Cask System will provide adequate safety and maintain fuel retrievability under the PFS Facility site-specific conditions. Therefore, the staff finds that the requirements of 10 CFR 72.122(l) and 72.128(a) are satisfied.

### 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, no evaluation findings are made in this section; only discussion of the information provided in the SAR is given. The design criteria for structures, systems, and components classified as not important to safety, but which have security or operational importance, are addressed in SAR Sections 4.3, 4.5.4, 4.5.5, and 4.7.5. The SAR specifies that these structures, systems, and components will be designed to comply with their applicable codes and standards to maintain the capability to mitigate the effects of off-normal or accident events.

## 4.2 Evaluation Findings

Based on the review of the information presented in the SAR, the following evaluation findings are made regarding the proposed PFS Facility ISFSI:

- The staff finds that the materials to be stored at the Facility are appropriately identified as those that are approved for storage in the HI-STORM 100 Cask System.

- The staff finds that the structures, systems, and components important to safety have been properly classified and their associated categories are consistent with the regulatory requirements of 10 CFR 72.144(a) and associated technical information content of the application, in accordance with 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, as required in 10 CFR 72.144(a).

- The staff finds that the structural design criteria, given in SAR Section 3.2, Structural and Mechanical Safety Criteria, considered 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, and criticality assessment of the Facility. These design criteria satisfy the requirements of 10 CFR 72.120(a), and 72.122(h). 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 72.120(a) and 72.122(b). The seismic design criteria are in accordance with the site-specific seismic hazards analysis given in SAR Chapter 2, Site Characteristics. The applicant has demonstrated that an exemption to the requirements of 10 CFR 72.120(f) is acceptable.

- The staff finds that the explosion considerations in the SAR are consistent with standard design criteria specified by Regulatory Guide 1.91,as required by 10 CFR 72.122(c). No credible onsite or offsite explosions will result in a peak positive incident overpressure of greater than 1.0 psi (the threshold air overpressure specified in Regulatory Guide 1.91) for structures, systems, and components important to safety.

- 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). 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). The design criteria for spent nuclear fuel storage and handling have been properly specified, as required by 10 CFR 72.128. A shielding evaluation has been performed in Chapter 7 of this SER. A confinement evaluation has been performed in Chapter 9 of this SER. A radiation protection evaluation has been performed in Chapter 11 of this SER.

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

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

- The staff finds that the Facility design, which includes use of the HI-STORM 100 Cask System, allows for retrieval of the spent nuclear fuel 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).

## 4.3 References

American Concrete Institute. 1989. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-90. Detroit, MI: American Concrete Institute.

American National Standards Institute/American Institute of Steel Construction. 1994. *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities.* ANSI/AISC N690. Chicago, IL : American Institute of Steel Construction.

American National Standards Institute/American Nuclear Society. 1992. *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)*. ANSI/ANS 57.9. La Grange Park, IL: American Nuclear Society.

American Society of Civil Engineers. 1996. *Minimum Design Loads for Buildings and Other Structures*. ASCE 7–95. New York: American Society of Civil Engineers.

American Society of Mechanical Engineers. 1995a. *Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).* ASME NOG–1. New York: American Society of Mechanical Engineers.

American Society of Mechanical Engineers. 1998. *ASME Boiler and Pressure Vessel Code, Section III, Division 1.* New York: American Society of Mechanical Engineers.

American Society of Mechanical Engineers. 1999. *ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D–Properties. 1998 Edition with 1999 Addenda.* New York: American Society of Mechanical Engineers.

Geomatrix Consultants, Inc. 2001. *Development of Design Ground Motions for the Private Fuel Storage Facility, Skull Valley, Utah*. Revision 1. Project No. 4790. Oakland, CA: Geomatrix Consultants, Inc.

Holtec International. 2000. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* Volumes I and II. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International.

Holtec International. 2001. *PFSF Site-Specific HI-STORM Drop/Tipover Analyses.* HI–2012653. October 31. Marlton, NJ: Holtec International.

International Conference of Building Officials. 1997. *Uniform Building Code*. Whittier, CA: International Conference of Building Officials.

National Fire Protection Association.1996. *Flammable and Combustible Liquids Code.* NFPA 30. Quincy, MA: National Fire Protection Association.

National Fire Protection Association. 1997. *Standard for the Installation of Lightning Protection Systems*. NFPA 780. Quincy, MA: National Fire Protection Association.

Nuclear Regulatory Commission. 1978. *Evaluation of Explosives Postulated to Occur on Transportation Routes Near Nuclear Power Plants*. Regulatory Guide 1.91. Revision 1. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. 1979. *Single-Failure-Proof Cranes for Nuclear Power Plants*. NUREG–0554. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. 1980. *Control of Heavy Loads at Nuclear Power Plants*. NUREG–0612. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. Missiles generated by natural phenomena. Revision 2, July 1981. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* LWR Edition. NUREG–0800 Section 3.5.1.4: (formerly issued as NUREG–76/087). Washington, DC: Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

Nuclear Regulatory Commission. 1997. *Standard Review Plan for Dry Cask Storage Systems*. NUREG–1536. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. 2000a. 10 CFR Part 72 *Certificate of Compliance No. 1014, Amendment 0, for the HI-STORM 100 Cask System.* Docket No. 72-1014. May 31.

Nuclear Regulatory Commission. 2000b. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report*. Docket No. 72-1014. May.

Private Fuel Storage Limited Liability Company. 2001. *Safety Analysis Report for Private Fuel Storage Facility*. Revision 22. Docket 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

U.S. Atomic Energy Commission. 1974. *Design Basis Tornado for Nuclear Power Plants*. Regulatory Guide 1.76. Washington, DC: U.S. Atomic Energy Commission.