4 STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA EVALUATION

4.1 Conduct of Review

The objective of the review of the structures, systems, and components (SSCs) and design criteria is to ensure that the applicant acceptably defines: (i) the limiting characteristics of the spent fuel or other high-level waste to be stored, (2) the classification of SSCs according to their importance to safety, and (3) the design criteria and design bases, including the external conditions during normal and off normal operations, accident conditions, and natural phenomena events. The Diablo Canyon ISFSI Safety Analysis Report (SAR), Section 3.1.1, identifies spent fuel, consisting of both Westinghouse LOPAR and VANTAGE 5 assemblies, as the material to be stored. SAR Section 4.5 categorizes all SSCs as either important to safety or not important to safety. Those SSCs important to safety are designed for safe confinement and storage of the spent fuel without the release of radioactive material. SAR Chapter 3 also identifies the principal design criteria for the Diablo Canyon ISFSI. These design criteria are derived from the requirements of 10 CFR Part 72 and applicable industry codes and standards. 10 CFR Part 72 also identifies the general design criteria for SSCs 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. SAR Tables 3.4-1 through 3.4-5 summarize the key design criteria for the Diablo Canyon ISFSI. The design criteria are compared to the actual design in subsequent chapters of this safety evaluation report.

The storage system to be used at the Diablo Canyon ISFSI is the HI-STORM 100 System, as described in the HI-STORM 100 System Final Safety Analysis Report (FSAR) (Holtec International, 2002) and the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2003). The HI-STORM 100 System has been approved by the U.S. Nuclear Regulatory Commission (NRC) for general use under Certificate of Compliance (CoC) No. 1014, through Amendment No. 1 (U.S. Nuclear Regulatory Commission, 2002a). Where applicable, the staff relied on the review carried out during the certification and amendment of the cask system, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

4.1.1 Materials to be Stored

As identified in SAR Section 3.1.1, “Materials to be Stored,” the materials to be stored at the Diablo Canyon ISFSI are Westinghouse LOPAR and VANTAGE 5 spent fuel assemblies, damaged fuel, and debris that are approved for storage in the HI-STORM 100 System. PG&E has elected to further limit the fuel to be stored at the Diablo Canyon ISFSI to low burnup fuel (burnup $\leq$ 45 GWd/MTU)(Pacific Gas and Electric Company, 2004), so the analyses in the ISFSI SAR are bounding for these more restrictive fuel limits. These materials fall within the broad range of approved contents as specified in Appendix B of CoC No. 1014, as modified through Amendment 1. SAR Section 3.1.1 provides a brief discussion of the specific materials to be stored at the Diablo Canyon ISFSI. SAR Tables 3.1-1 and 3.1-2 are summaries of the fuel physical, thermal and radiological characteristics. This discussion is consistent with the physical, thermal and radiological characteristics of the spent fuel in the HI-STORM 100 System FSAR.

4.1.2 Classification of Structures, Systems, and Components

This section contains a review of SAR Section 4.5, "Classification of Structures, Systems, and Components." The staff reviewed the discussion on classification of SSCs with respect to the following regulatory requirements:

- 10 CFR §72.120(a) requires that 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.

- 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 that 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, SSCs important to safety are defined as items whose functions are to (1) maintain the conditions required to store spent fuel safely; (2) prevent damage to the spent fuel container during handling and storage; and (3) provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. The SAR lists the SSCs based on this definition as required by 10 CFR §72.24(a) and §72.144(a).

SAR Section 4.5, "Classification of Structures, Systems, and Components," 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:

(1) **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.

(2) **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.

(3) **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 SSCs considered important to safety are identified in SAR Section 4.5, "Classification of Structures, Systems, and Components."

<u>Category A Structures, Systems, and Components</u>

The SSCs classified as Category A are given in SER Table 4-1. The failure of a single Category A item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control. In some cases, the items are classified as Category A because they are considered part of the single-failure-proof load path of systems used to handle and transport the spent fuel. Each of these items is designed to protect the spent fuel during specific phases of handling and storage. The following Category A components have been properly classified.

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the multipurpose canister (MPC) serves as the primary confinement structure. Its failure could lead to the release of radioactive material. Sufficient description of the MPC is provided in SAR Section 4.2.3 "Storage Cask Description."

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the fuel basket and damaged fuel container maintain fuel in a safe geometry. Failure could lead to criticality and reduced ability to retrieve the fuel. Sufficient descriptions of the fuel basket and damaged fuel container are provided in SAR Section 4.2.3, "Storage Cask Description."

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the HI-TRAC 125 Transfer Cask protects the MPC and provides shielding during handling for 10 CFR Part 50 and Part 72 operations and is part of the single-failure-proof load path. Sufficient description of the transfer cask is provided in SAR Section 4.2.3.2.4, "HI-TRAC 125 Transfer Cask."

The transfer cask lift links, MPC downloader slings, MPC lift cleats, HI-STORM 100 System lifting brackets, and HI-STORM 100 System lift links are all part of the single-failure-proof load path. Sufficient descriptions of the associated lifting devices are provided in SAR Sections 4.3.2.5 through 4.3.2.9.

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the HI-STORM mating device bolts and shielding frame provide structural support and shielding at the interface between the top of the open overpack and the bottom of the transfer cask during MPC transfer operations. Sufficient description of these components is provided in SAR Section 3.3.4.2.5, "HI-STORM Mating Device."

Table 4-1. Category A quality assurance classification of SSCs
(Based on SAR Table 4.5-1)

Structures, Systems, and Components	Function
Multi-Purpose Canister (MPC) (HI-STORM)	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.
Fuel Basket (HI-STORM)	Ensures the correct geometry of the stored fuel assemblies and provides the fixed neutron absorber to prevent criticality.
Damaged Fuel Container (HI-STORM)	Maintains damaged fuel or fuel debris in a safe geometry and enables retrieval.
HI-TRAC 125 Transfer Cask (HI-STORM)	Designed to support the canister during transfer lift operations and provide radiation shielding and canister heat rejection.
Transfer Cask Lift Links (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister.
MPC Downloader Slings (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the MPC and preclude the accidental drop of an MPC.
MPC Lift Cleats (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the MPC and preclude the accidental drop of an MPC.
HI-STORM Lifting Brackets (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister.
HI-STORM Lift Links (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister.
HI-STORM Mating Device Bolts and Shielding Frame (HI-STORM)	Provides structural support and shielding at the interface between the top of the open overpack and the bottom of the transfer cask during MPC transfer operations at Cask Transfer Facility (CTF).
Cask Transporter	The load-bearing components prevent damage to the spent nuclear fuel and storage cask system components during transport, lifting, and MPC transfer operations in all normal, off-normal, and accident conditions.
Lateral Restraints (Transporter at CTF)	Used to restrain the transporter from motion during design basis seismic events.

The first 10 items in Table 4-1, those that correspond to the HI-STORM 100 System, are identified in the previous five paragraphs. Details associated with these 10 items are cask-specific, and additional descriptions are presented in the HI-STORM 100 System FSAR.

As identified in SAR Section 4.5.4, “Cask Transport System,” the cask transporter load-bearing components prevent damage to the spent fuel and spent fuel storage cask system components during transport, lifting, and MPC transfer operations in all normal, off-normal, and accident conditions. The cask transporter will also be designed to preclude tip-over under site-specific seismic, tornado winds, and tornado missile loads. Components of the transporter are part of the single-failure-proof load path. Sufficient description of the cask transporter is provided in SAR Section 4.3.2.1, “Cask Transporter.”

The lateral restraints are used to restrain the HI-TRAC 125 Transfer Cask and transporter from motion at the Cask Transfer Facility (CTF) during design-basis seismic events. The lateral restraints are briefly described in SAR Section 4.2.1.2 “CTF Support Structure.” Sufficient description of the lateral restraints is provided in the SAR and in PG&E’s response to additional staff questions (Pacific Gas and Electric Company, 2003a).

Based on the previous discussion, the staff concludes that these Category A important to safety items are correctly classified.

Category B Structures, Systems, and Components

The SSCs classified as Category B are summarized in SER Table 4-2. The failure of a Category B item, in conjunction with failure of an additional item, could result in an unsafe condition. The following Category B components have been properly classified.

As identified in SAR Section 4.5.1.1, “Multi-Purpose Canister and Fuel Basket,” the upper and lower fuel spacer columns and end plates maintain fuel in the correct geometry. Sufficient descriptions of the upper and lower fuel spacer columns and end plates are provided in SAR Section 4.2.3 “Storage Cask Description.”

As identified in SAR Section 4.5.1.3, “Overpack,” the HI-STORM 100SA Overpack protects the MPC during storage. Sufficient description of the storage cask is provided in SAR Section 4.2.3, “Storage Cask Description.”

Details associated with these two items are cask-specific, and additional descriptions are presented in the HI-STORM 100 System FSAR.

As identified in SAR Section 4.5.2, “Cask Storage Pads,” the cask storage pads provide the necessary embedment for the anchorage of the overpack. The overpack anchorage hardware is designed to prevent sliding and tip-over during the design-basis seismic event. Sufficient descriptions of the cask storage pad and overpack anchorage system are provided in SAR Sections 3.3.2, “ISFSI Cask Storage Pads;” and 4.2.1.1, “Cask Storage Pads.”

The CTF is designed to protect the canisters from adverse natural phenomena during cask loading, unloading and canister transfer operations. The CTF jacks are part of the single-failure-proof load path. Sufficient description of the CTF is provided in SAR Sections 4.2.1.2, “CTF Support Structure;” and 4.4.5, “Cask Transfer Facility.”

Table 4-2. Category B quality assurance classification of SSCs (Based on SAR Table 4.5-1)

Structures, Systems, and Components	Function
Upper and Lower Fuel Spacer Columns and End Plates (HI-STORM)	Ensures the correct geometry of the stored fuel assemblies.
HI-STORM 100 SA Overpack (HI-STORM)	Serves as the primary component for protecting the canister during storage from environmental conditions and provides radiation shielding and canister heat rejection.
Independent Spent Fuel Storage Installation (ISFSI) Storage Pads	Designed to ensure a stable and level support surface for the storage cask in all normal, off-normal, and accident conditions. It provides the necessary embedment for the anchorage of the overpack.
Overpack Anchorage Hardware	Designed to prevent sliding and tip-over during a design-basis seismic event.
Cask Transfer Facility (CTF) (except jacks)	Prevents damage to the spent nuclear fuel and spent nuclear fuel storage cask system components during lifting and multipurpose canister (MPC) transfer operations in all normal, off-normal, and accident conditions.
CTF Jacks	Designed to prevent uncontrolled lowering of the load during lifting and MPC transfer operations.
Transporter Connection Pins	Designed to lock the transfer cask in place during transport.
Transfer Cask Horizontal Lift Rig (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask for storage operations and preclude the accidental drop of a canister.
Transfer Cask Lift Slings (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask for storage operations and preclude the accidental drop of a canister.
Helium Fill Gas	Provides an inert environment for storage of the spent nuclear fuel and facilitates canister heat rejection.

As identified in SAR Table 4.3-1, the transporter connector pins are used to connect the transfer cask lift links or the overpack lifting brackets to the cask transporter lift links. The applicable design codes are American National Standards Institute (ANSI) N14.6 (American National Standards Institute/American Nuclear Society, 1993), in accordance with the guidance of NUREG–0612, Section 5.1.6 (U.S. Nuclear Regulatory Commission, 1980). Sufficient description of the transporter connection pins is provided. The transfer cask horizontal lift rig and transfer cask lift slings are designated as special lifting devices in the load path of the transfer cask during lifting and movement between the FHB/AB and the CTF. Sufficient description of the associated lifting devices is provided in SAR Sections 4.3.2.2, “Transfer Cask Horizontal Lift Ring;” and 4.3.2.3, “Transfer Cask Lift Slings.”

The helium fill gas is placed in the MPC during the canister preparation activities inside the DCPP FHB/AB. The characteristics of the helium gas and associated controls are described in SAR Section 10.2.2.4, “ MPC Helium Backfill Characteristics and Purity.”

Failure of one or more of these Category B items, combined with the subsequent failure of the MPC, is necessary to lead to a condition adversely affecting public health and safety. Based on the previous discussion, the staff concludes that these Category B items are correctly classified.

Category C Structures, Systems, and Components

The SSCs summarized in Table 4-3 of this SER have been properly classified as Category C. The HI-STORM 100 System mating devices are conceptually shown in SAR Figure 4.2-11. Except as noted previously, failure of this component would not likely result in an unsafe condition. Based on the previous discussion, the staff concludes that this Category C item is correctly classified.

Table 4-3. Category C quality assurance classification of SSCs (Based on SAR Table 4.5 1)

Structures, Systems, and Components	Function
HI-STORM Cask Mating Device (except bolts and shielding frame)	Used to manipulate the transfer cask bottom lid to facilitate MPC transfer operations at the CTF.

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

In SAR Table 4.5-1, PG&E classified several SSCs as not important to safety because they do not involve a safety-related function and are therefore not subject to NRC-imposed regulatory requirements.

A number of systems are security-related, including the security system, fencing, lighting, and communications systems. Each system is used to support the activities of the security personnel who monitor the controlled area of the Diablo Canyon ISFSI. 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 100SA System is a passive system, 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 automated welding system, MPC helium backfill system, MPC force helium dehydration system, and MPC vacuum drying system are classified as not important to safety, consistent with the classification of these systems in the HI-STORM 100 System FSAR (Holtec International, 2002), as approved by the NRC staff through Amendment 1, effective July 15, 2002.

As identified in SAR Section 4.3.2.4, “Cask Transport Frame,” the cask transport frame is used for rotating the transfer cask between the horizontal and vertical orientations. It is attached to, but does not support, the transfer cask during transport from the FHB/AB to the CTF. Therefore, the cask transport frame is correctly classified as not important to safety.

The cask transporter is designed such that any malfunction in the drive and control systems will cause it to stop in a safe condition. A malfunction of the drive system will initiate the fail-safe braking system. The control system is a dead-man system, in which the operator must actively drive the system. Therefore, it can be considered fail-safe. The CTF drive and control systems are correctly classified as not important to safety.

4.1.2.3 Classification of Structures, Systems, and Components—Conclusion

The staff evaluated the classification of SSCs important to safety by reviewing SAR Chapter 4, “ISFSI Design ;” documents cited in the SAR; and other relevant literature. Details of the quality assurance program evaluation are contained in Chapter 12 of this SER. The staff determined that the classification and descriptions of the SSCs important to safety are consistent with the regulatory requirements of 10 CFR §72.144(a) and 10 CFR §72.24(n).

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

The principal design criteria identified for SSCs important to safety at the Diablo Canyon ISFSI are described in SAR Chapter 3, “Principal Design Criteria.” This section contains a review of Section 3.2, “Design Criteria for Environmental Conditions and Natural Phenomena;” Section 3.3, “Design Criteria for Safety Protection Systems;” and Section 3.4, “Summary of Design Criteria.” Diablo Canyon site-specific design criteria are derived from information in other sections of the Diablo Canyon ISFSI SAR and from the DCPP FSAR update (Pacific Gas and Electric Company, 2001). The design criteria for the Holtec storage system are derived for the HI-STORM 100 System CoC and the HI-STORM 100 System FSAR. Details of the design criteria evaluation are provided in SER Sections 4.1.3.1 to 4.1.3.7.

4.1.3.1 General

The staff reviewed the discussion of the general design criteria for SSCs 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 nuclear 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 SSCs 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 nuclear fuel cladding must 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 must 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 waste 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 §72.144(c) requires that the licensee shall base the requirements and procedures of its quality assurance program(s) on the following considerations concerning the complexity and proposed use of the SSCs: (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) quality history and degree of standardization of the item.

A summary of the Diablo Canyon ISFSI general design criteria is provided in SER Table 4-4. The design of the proposed Diablo Canyon ISFSI is based on the HI-STORM 100 System, which has been approved by the NRC for use according to the general license provisions of 10 CFR Part 72.

Table 4-4. Summary of Diablo Canyon ISFSI design criteria—general and spent nuclear fuel specification (Based on SAR Tables 3.4-2 through 3.4-5)

Design Parameters	Design Conditions	Reference
ISFSI Design Life	20 years	SAR Section 1.1
Storage Pad Design Life	40 years	SAR Table 3.4-3
Cask Transfer Facility Design Life	40 years	SAR Table 3.4-5
HI-STORM 100 System Design Life	40 years	SAR Table 3.4-2
Cask Transporter Design Life	20 years	SAR Table 3.4-4
Storage Capacity	4,400 spent nuclear fuel assemblies	SAR Section 3.1
Number of Casks	140 casks including two spares	SAR Section 3.1
Type of Fuel	Nonconsolidated Pressurized Water Reactor Westinghouse 17 x 17 LOPAR and VANTAGE 5	SAR Sections 3.1.1 and 10.2.1.1 and Tables 10.2-1 through 10.2-5
Fuel Characteristics	Physical, thermal, and radiological characteristics	SAR Sections 3.1.1 and 10.2.1.1 and Tables 3.1-1 and 3.1-2 and 10.2-1 through 10.2-5
Fuel Classification	Intact, damaged, debris	SAR Sections 3.1.1 and 10.2.1.1 and Tables 10.2-1 through 10.2-10 and Diablo Canyon ISFSI Tech Specs
Nonfuel Hardware	Borosilicate absorber rods Wet annular burnable absorber rods Thimble plug devices Rod cluster control assemblies	SAR Section 3.1.1.3 and Tables 3.1-1 and 10.2-10

Table 4-5. Design criteria for Diablo Canyon major ISFSI structures, systems and components (Based on SAR Table 3.4-1)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Seismic	Design Earthquake [5% damping, Zero Period Acceleration (ZPA) = 0.201g horizontal and 0.134g vertical, periods up to 1.0 second] Double Design Earthquake (5% damping, ZPA = 0.402g horizontal and 0.268g vertical, periods up to 1.0 second) Hosgri Earthquake (5% damping, ZPA = 0.75g horizontal and 0.50g vertical, periods up to 0.8 second) LTSP (5% damping, ZPA = 0.83g horizontal and 0.70g vertical, periods up to 2.0 seconds) ILP (5% damping, ZPA = 0.83g horizontal and 0.70g vertical) (envelopes other design spectra and includes extended low frequency components) (SAR Sections 3.2.3, 8.2.1.2)	10 CFR §72.102
Wind	35.77 m/s [80 mph] with gust factor of 1.1 (SAR Section 3.2.1)	ASCE–7
Tornado	89.41 m/s [200 mph], maximum speed 70.19 m/s [157 mph], rotational speed 19.22 m/s [43 mph], translational speed 5.93 kPa [0.86 psi], pressure drop 2.48 kPa [0.36 psi]/sec rate of drop (SAR Section 3.2.1, Table 3.2-1)	Regulatory Guide 1.76
Tornado Missiles	DCPP—Generic 1,814 kg [1.79 ton] automobile, 14.89 m/s [33.3 mph] 34.5 kg [76.1 lb], 25.4-cm [3-in] diameter × 3.04-m [10-ft] schedule 40 pipe, 29.82 m/s [66.7 mph] 49.0 kg [108.03 lb] 10.16 cm [4 in] × 30.48 cm [12 in] × 3.05 m [10 ft] wood plank, 89 m/s [293 ft/sec] Diablo Canyon ISFSI—Site Specific 130 kg [286.60 lb], 15.24-cm [6-in] diameter schedule 40 pipe, 3.13 m/s [7 mph] 510 kg [0.50 ton] wooden utility pole, 15.65 m/s [35 mph] 340 kg [0.33 ton], 30.48-cm [12-in] diameter schedule 40 pipe, 2.24 m/s [5 mph] 3.9 kg [8.60 lb] 5.08 cm [2 in] × 5.08 cm [2 in] × 0.32 cm [1/8 in] × 1.52 m [5 ft] Steel Angle, 70.19 m/s [157 mph] 344.7 kg [0.34 ton] 500-kV insulator string, 70.19 m/s [157 mph] 6.8 kg [14.99 lb] 500-kV insulator segments, 70.19 m/s [157 mph] 4 kg [8.82 lb], 2.54-cm [1-in] diameter steel rod, 2.24 m/s [5 mph]	NUREG–0800, Section 3.5.1.4

Table 4-5. Design criteria for Diablo Canyon major ISFSI Structures, Systems and Components (Based on SAR Table 3.4-1) (continued)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Flood	Design basis flooding events are not considered credible. (SAR Section 3.2.2)	NUREG–0800, Section 3.4.1
Snow and Ice	Design basis snow and ice loadings are not considered credible. (SAR Section 3.2.4)	ASCE–7
Fire	Fuel tank on transporter, load-handling equipment, or other vehicle Local stationary fuel tanks Local combustible materials Nearby grass/brush fire Unit 2 transformers, approximately 49.21 m^3 [13,000 gal] of mineral oil (SAR Sections 2.2.2.2, 3.3.1.6, and 8.2.5)	ANSI/ANS 57.9
Explosion	Fuel tank on transporter, load-handling equipment, or other vehicle 26.5-L [7-gal] propane bottle Standard acetylene bottle 0.95-m^3 [250-gal] propane tank, 7.57-m^3 [2,000-gal] No. 2 diesel fuel oil tank, and 11.36-m^3 [3,000-gal] gasoline tank Standard compressed gas bottle Hydrogen gas facility Acetylene bottle storage (SAR Sections 2.2.2.3, 3.3.1.6, and 8.2.6)	Reg. Guide 1.91
Severe Electrical Lightning	500-kVA Line Drop (SAR Section 3.2-6)	10 CFR §72.122(g)
Ambient Conditions	Annual Average = 12.78 °C [55 °F] Low Temperature = below freezing for a few hours Maximum Recorded = 36.11 °C [97 °F] Extreme Temperature Range = −4.4 °C [24 °F] to 40 °C [104 °F] Insolation = 791 Whr/m^2 [766 g-cal/cm^2] maximum for a 24-hour period (SAR Sections 2.3.2, 3.2.7, 8.2.6, and 8.2.10)	National Oceanic and Atmospheric Administration data

The design life of each SSC important to safety is based on its ability to withstand the applied loads and environmental conditions. 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 SSCs to withstand the applied loads with additional factors applied to the loads and material allowables in accordance with the referenced codes and standards. The storage capacity and number of casks to be stored at the Diablo Canyon ISFSI have been identified in SAR Section 3.1, “Purposes of Installation.”

The staff reviewed the general design criteria identified, and summarized them in SER Table 4-5. The staff found that the criteria are consistent with the HI-STORM 100 System FSAR. Definitions of the normal, off-normal, and accident loads are given in SAR Sections 3.2, “Design Criteria for Environmental Conditions and Natural Phenomena;” and 3.3, “Design Criteria for Safety Protection Systems.” The quality standards for the design bases of SSCs important to safety are provided in SAR Chapters 3, “Principal Design Criteria” and 11, “Quality Assurance.” These design criteria satisfy, in part, the requirements of 10 CFR §72.120(a), §72.122(h), and §72.236(b), in that design criteria are identified and SSCs important to safety will be designed to quality standards commensurate with the important to safety functions to be performed in accordance with the requirements of 10 CFR §72.144(c).

The ISFSI is located near the DCPP and must be designed and operated to ensure the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public as defined in 10 CFR §72.122(e). As identified in the applicant’s response to the staff’s Request for Additional Information (RAI responses 2-8 through 2-11, Pacific Gas and Electric Company, 2002), onsite hazards and cumulative effects have been identified (SAR Section 2.2.2, “Onsite Potential Hazards”) and accident analysis performed (SAR Section 8.2, “Accidents”). Therefore, there will be no undue risk to the public health and safety from combined ISFSI and DCPP operations.

4.1.3.2 Structural

The staff reviewed the discussion on structural design criteria of SSCs in the SAR with respect to the following regulatory requirements:

- 10 CFR §72.102(f) requires that the design earthquake (DE) for use in the design of structures be determined as follows: (1) For sites that have been evaluated in accordance with the criteria of Appendix A of 10 CFR Part 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant. (2) Regardless of the results of investigations anywhere in the continental United States, the DE must have a value for the horizontal ground motion of no less than 0.10g 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 nuclear 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.

- 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.

- 10 CFR §72.122(b)(2)(i) 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 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. (2)(ii) 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 nuclear fuel or high-level waste or on SSC important to safety.

- 10 CFR §72.122(b)(4) specifies that if the ISFSI is located over an aquifer that 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 SSC 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 Sections 3.2, “Criteria for Environmental Conditions and Natural Phenomena;” and 3.3, “Design Criteria for Safety Protection Systems,” address the structural and mechanical design criteria. The design criteria for the site include dead loads, live loads, seismicity, wind, tornado, tornado missiles, flood, snow and ice, explosion overpressure, fire, and site ambient temperature, humidity, solar radiation, and lightning. Information on the derivation of site-specific design criteria for the meteorology, hydrology, and seismology is contained in SAR Chapter 2, “Site Characteristics.”

The structural design loads for SSCs important to safety are provided in the following section of the SER. As identified, the SSCs 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 System are described in the HI-STORM 100 System FSAR. Table 4-6 of this SER identifies the HI-STORM 100 System design criteria in relation to the Diablo Canyon ISFSI design criteria. The staff review in this SER is limited to the evaluation of whether the HI-STORM 100 System design criteria and parameters envelope those for the Diablo Canyon ISFSI. The acceptability of the HI-STORM 100 System design criteria is discussed in the NRC HI-STORM 100 System SER.

Site-specific design criteria not enveloped by the HI-STORM 100 System criteria are identified in the Diablo Canyon ISFSI SAR Sections 3.2, "Criteria for Environmental Conditions and Natural Phenomena;" and 3.3, "Design Criteria for Safety Protection Systems." These criteria include the specific site data, storage system component affected, and the corresponding section in the SAR where it is addressed. Structural design criteria and radiological protection and confinement criteria are identified. The structural criteria are discussed in Chapter 4 of this SER. Radiological protection and confinement criteria are considered in Chapters 7, "Shielding Evaluation;" 8, "Criticality Evaluation;" 9, "Confinement Evaluation;" and 11, "Radiation Protection Evaluation" of this SER.

Seismicity

The staff reviewed the data presented in the SAR associated with seismic design criteria at the Diablo Canyon ISFSI in accordance with 10 CFR §72.102(f). SAR Section 3.2.3, "Seismic Design," gives the seismic design criteria, based on probabilistic site-specific seismology studies summarized in SAR Section 2.6, "Geology and Seismology." There are four site-specific design response spectra: design earthquake (DE), double design earthquake (DDE), Hosgri earthquake (HE), and the Long-Term Seismic Program (LTSP) earthquake. The DE and DDE correspond to the operational and SSE for the DCPP. The HE represents the maximum credible event on the Hosgri fault. This earthquake ground motion is the design-basis for the ISFSI, based on its proximity to the site. Based on work during the LTSP, another seismic design basis was defined for the DCPP for verification of the adequacy of seismic margins of certain plant SSCs. The response spectra for these design-basis events are given in SAR Figures 2.6-43 and 2.6-44. These response spectra have been accepted by the staff for the DCPP and are considered acceptable for the ISFSI because the ISFSI is close to the plant. Because ISFSI pad sliding, pad cutslope stability, transport route stability and transporter stability may be affected by a longer-period ground motion, additional design-basis response spectra were developed for the ISFSI that include ground motions out to periods of 10.0 seconds. This design-basis response spectrum includes fault normal and parallel directing response and fault fling. The response spectra for these design-basis events are given in SAR Figures 2.6-45 and 2.6-46. Spectra for the vertical and horizontal directions are identified for all design-basis ground motions. The site-specific seismic design criteria of the Diablo Canyon ISFSI are bounded by the HI-STORM 100 System seismic design criteria for an anchored cask. The staff assessment of the adequacy of the site-specific seismic design criteria is contained in Section 2.1.6 of this SER. The applicant's analysis of the HI-STORM 100 System 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 Diablo Canyon ISFSI and found that they are properly identified as required by 10 CFR §72.120(a) and §72.122(b).

Wind

Figure 6-1 in ASCE 7-98 (American Society of Civil Engineers, 2000) identifies a design-basis 3-second gust wind speed of 38 m/s [85 mph] for the region. Information provided in SAR Section 3.2.1, “Tornado and Wind Loadings,” identifies a maximum recorded wind gust speed at the DCPP site of 37.55 m/s [84 mph]. The design-basis wind speed of 35.76 m/s [80 mph] with a wind gust factor of 1.1 envelopes these values {35.76 m/s [80 mph] x 1.1 gust factor = 39.34 m/s [88 mph] > 38 m/s [85 mph]}. The staff reviewed the design-basis wind speed {35.76 m/s [80 mph]} for the Diablo Canyon ISFSI and found that it is consistent with that identified in ASCE 7-98 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 Diablo Canyon ISFSI design.

Tornado

The design-basis tornado wind loads for the ISFSI are the same as those identified in the DCPP site licensing-basis information provided in Section 3.3.2.1.1 of the DCPP FSAR update. The parameters for the tornado identified in the DCPP FSAR have been reviewed with respect to the 10 CFR Part 50 license and are consistent with those given in Regulatory Guide 1.76 (U.S. Atomic Energy Commission, 1974). 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 Diablo Canyon ISFSI design.

Tornado Missiles

The tornado missiles of concern to the ISFSI are identified in SAR Table 3.2-2. These tornado missiles are a compilation of those items specified as Spectrum II missiles in NUREG–0800, Section 3.5.1.4 (U.S. Nuclear Regulatory Commission, 1981), the items listed in the DCPP FSAR update, and missiles from the three 500-kV towers near the Diablo Canyon ISFSI site. These items are considered to be representative of potential missiles present at the site. As identified in the HI-STORM 100 System FSAR, the cask-specific tornado missiles correspond to Spectrum I missiles in NUREG–0800. Use of either Spectrum I or II missiles is considered acceptable by the NRC. The staff reviewed the design-basis tornado conditions for the Diablo Canyon ISFSI and found that the conditions are consistent with the 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

Based on the location of the Diablo Canyon ISFSI pad and the site surface hydrology, it is concluded in SAR Section 2.4, “Surface Hydrology,” that there is no potential for flooding in the vicinity of the ISFSI. The same conclusion is applicable to the HI-STORM 100SA Overpacks, which are stored on the pad, and the CTF, which is located close to the storage pads. In addition, the CTF is protected by a sump that can remove any significant accumulation of water in the vault. Therefore, the forces resulting from flood waters and flood protection measures do not need to be considered in the design of SSCs important to safety. The staff, therefore, concludes that the Diablo Canyon ISFSI design is consistent with the design criteria of NUREG–0800 and ASCE 7-98 to withstand floods as required by 10 CFR §72.120(a) and §72.122(b).

Snow and Ice

Figure 7-1 of ASCE 7-98 indicates that snow is not a concern below an elevation of 457.20 m [1,500 ft]. SAR Section 3.2.4, “Snow and Ice Loadings,” states that there is essentially no design ground snow load at the Diablo Canyon ISFSI. As identified in the HI-STORM 100 System FSAR, the overpack is designed for a 4.79-kPa [100-lb/ft^2] snow and ice load. This load bounds the Diablo Canyon ISFSI 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).

Fire

The applicant has identified a total of six fire events in SAR Section 8.2.5, “Fire” (Pacific Gas and Electric Company, 2003). A range of onsite fire scenarios has been evaluated. The applicant has demonstrated that the 189.21-L [50-gal] transporter fuel tank fire is the bounding condition. The proposed Diablo Canyon ISFSI Technical Specification 4.3.1.b will establish a 50 gallon limit for the transporter fuel tank to ensure that these levels are not exceeded (Pacific Gas and Electric Company, 2003b). The staff reviewed the fire considerations in the SAR and found that they are consistent with equipment to be used at the ISFSI and the operational restraints as required by 10 CFR §72.122(c). Appropriate design criteria are specified to ensure that SSCs important to safety will be designed and located so that they can perform their safety functions effectively during credible fire exposure conditions.

Explosion

The applicant has identified a total of seven explosion event categories (Events 1–7), in SAR Section 8.2.6, “Explosion,” that identify explosive hazards and potential impacts on cask transfer and storage activities. Explosion hazards were evaluated for the spent fuel in storage casks on the pad, and for the spent fuel in the transfer cask as it is transported from the FHB/AB to the storage pad.

Regulatory Guide 1.91, “Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants“ (U.S. Nuclear Regulatory Commission, 1978), provides criteria for determining acceptable standoff distances between explosion hazards and critical plant structures. Regulatory Guide 1.91 indicates that an acceptable method for establishing these standoff distances can be based on a level of peak positive incident overpressure below which no significant damage to these structures would be expected. For SSCs of concern, the guidance specifies an overpressure limit of 1 psi (approximately 7 kPa). In cases where the distances from the transportation route to the SSCs that must be protected are not great enough to demonstrate that the overpressure limit will not be exceeded, the guidance of Regulatory Guide 1.91 also allows for an analysis of the frequency of exposure of SSCs to potential explosion hazards. If conservative estimates are used and the cumulative exposure rate can be demonstrated to be less than 10^{-6} per year (or 10^{-7} per year, if realistic estimates are used); then the staff considers the hazard acceptably low. If neither of these first two criteria can be met, Regulatory Guide 1.91 specifies that an analysis of the explosion consequences on the affected SSCs must be performed. The NRC staff has applied the guidance of Regulatory Guide 1.91 in its evaluation of the explosion hazards analysis for the Diablo Canyon ISFSI.

PG&E developed a probabilistic risk analysis to estimate the annual frequency of exposure of SSCs important to safety from each potential source of explosion for those events that could result in exceeding the 6.9-kPa [1-psi] air overpressure limit. Based on its review of the applicant's probabilistic risk analysis for these events, as discussed in Section 15.1.2.4 of this SER, the staff concludes that no important to safety SSCs for the proposed facility will be subjected to explosion overpressures that exceed the 6.9 kPa [1 psi] threshold.

The Diablo Canyon ISFSI Technical Specifications will include requirements for a Cask Transportation Evaluation Program, which will specify administrative controls to ensure that any onsite vehicle movements are sufficiently restricted, consistent with the assumptions of the analysis, and that no additional hazards are introduced for the spent fuel while in the storage casks, CTF, or transfer cask.

The staff reviewed the explosion considerations in the SAR and found that the applicant has identified seven events. The applicant has characterized the credible events with respect to the resulting peak pressure. The analyses are consistent with standard design criteria, NFPA 30 (National Fire Protection Association, 1996) and Regulatory Guide 1.91, and meet the requirements of 10 CFR §72.122(c).

Lightning

During thunderstorms, a lightning strike is possible. However, the overhead transmission line and towers effectively prevent a direct lightning strike on any overpack or the CTF. The HI-STORM 100SA Overpack is designed for lightning protection. Any lightning strike on the cask will discharge through its steel shell to the ground, having no adverse impact on the cask or fuel. The staff reviewed the lightning design criterion as identified in the SAR and determined that it is acceptable for the design of SSCs important to safety as required by 10 CFR §72.122(b).

Ambient Conditions

The loads from environmental conditions and natural phenomena specified in the SAR are consistent with standard engineering practice, as identified in ASCE 7-98, which identifies the minimum design loads for buildings and other structures.

HI-STORM 100 System Design Criteria

The applicant has identified design criteria for each of the major systems at the ISFSI as provided in SAR Tables 3.4-2 through 3.4-5 and in Table 4-6 of this SER. The design criteria for the pressure vessel portions of the HI-STORM 100 System conform to standard engineering practice, as identified in the ASME International Boiler and Pressure Vessel Code (ASME International, 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 of the HI-STORM 100 System, as identified in SAR Table 3.4-2, the design criteria are based on the American Concrete Institute ACI 318-95 (American Concrete

Institute, 1995) and ACI 349-85 (American Concrete Institute, 1985). ACI 349-85 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.

Thermal, shielding and confinement, and criticality design criteria are discussed in Sections 4.1.3.3 through 4.1.3.5 of this SER.

Table 4-6. Design criteria for Diablo Canyon HI-STORM 100 System (Based on SAR Table 3.4-2)

Design Criterion	HI-STORM 100 System Design Value	Applicable Criteria and Codes
Structural Design		
HI-STORM 100 System Design Codes	Canister: Internals: Overpack: Transfer Cask: Holtec FSAR, as amended by LAR 1014-1 Tables 2.2.6, 2.2.7, 2.2.14, and 2.2.15.	ASME III-95, with 1996 and 1997 Addenda, Subsection NB ASME III, NG ASME III, NF, ACI-318 (95), ACI–349 (85) ASME III, NF, ANSI N14.6 (93) ASCE 7-88
Environmental Conditions and Natural Phenomena	See Diablo Canyon ISFSI SAR Table 3.4-1 (SAR Sections 3.2 and 3.3)	
Weights	Maximum loaded transfer cask handling weight = 113,400 kg [250,000 lb] Maximum loaded overpack weight = 163,300 kg [360,000 lb] Transporter weight = 77,100 kg [170,000 lb]	
MPC Internal Pressure	Normal/off-normal = 0.69 MPa [100 psig] Accident = 1.38 MPa [200 psig] (Holtec FSAR, Table 2.0.1 as amended by LAR 1014-1)	
Cask Loads and Load Combinations	Varies (Holtec FSAR Sections 2.2.1 through 2.2.3 and Tables 2.2.13 and 2.2.14 as amended by LAR 1014-1)	
Thermal Design		
Maximum Cask Heat Duty	Varies Maximum pressurized water reactor basket heat duty = 28.74 kW (Holtec FSAR, Section 4.4.2, as amended by LAR 1014-1, Section 4.4.2 and Table 4.4.28)	

Table 4-6. Design criteria for Diablo Canyon HI-STORM 100 System (Based on SAR Table 3.4-2) (continued)

Design Criterion	HI-STORM 100 System Design Value	Applicable Criteria and Codes
Peak Fuel Cladding Temperature Limits	Long-term (normal) limits = 400°C [752°F] Short-term (accident) = 570°C [1,058°F] (Holtec FSAR, Rev. 1, Tables 4.3.7 and 4.A.3 and HI-STORM 100 System FSAR Table 4.3.1)	
Other SSCs Temperature Limits	Varies by material (Holtec FSAR, Tables 2.0.1 through 2.0.3, as amended by Holtec LAR 1014-1)	
MPC Backfill Gas	99.995% pure helium (SAR Section 10.2.2.4 and Holtec CoC, Appendix A, Table 3-1, as amended by LAR 1014-1)	
Maximum Air Inlet to Outlet Temperature Rise	52.2 °C [126 °F] (Holtec CoC, Appendix A, LCO 3.1.2, as amended by LAR 1014-1)	
Radiation Protection and Shielding Design		
Storage Cask Dose Rate Objectives	0.6 mSv/hr [60 mrem/hr] on sides, top, and adjacent to air ducts (SAR Section 3.3.1.5.2 and Holtec FSAR Section 2.3.5.2 as amended by LAR 1014-1)	
Occupational Exposure Dose Limits	Total effective dose equivalent 50 mSv/yr [5 rem/yr] lens dose equivalent 15 mSv/yr [15 rem/yr] shallow-dose equivalent and extremities 500 mSv/yr [50 rem/yr]	10 CFR §20.1201
Restricted Area Boundary Dose Rate Limit	0.02 mSv/hr [2 mrem/hr]	10 CFR §20.1301
Normal Operation Dose Limits to Public	0.25 mSv/yr [25 mrem/yr] whole body 0.75 mSv/yr [75 mrem/yr] thyroid 0.25 mSv/yr [25 mrem/yr] other critical organ	10 CFR §72.104

Table 4-6. Design criteria for Diablo Canyon HI-STORM 100 System (Based on SAR Table 3.4-2) (continued)

Design Criterion	HI-STORM 100 System Design Value	Applicable Criteria and Codes
Accident Dose Limits to Public	50 mSv [5 rem] TEDE 150 mSv [15 rem] lens dose equivalent 500 mSv [50 rem] shallow dose equivalent to skin or extremity	10 CFR §72.106
Overpack Unreinforced Concrete	Various (Holtec FSAR, Appendix 1.D as amended by LAR 1014-1)	
Criticality Design		
Maximum Initial Fuel Enrichment	$\leq$5% (SAR Sections 3.3.1.4.1 and 3.1.1.1, Tables 10.2-1 through 10.2-5, Diablo Canyon ISFSI Technical Specification and Holtec CoC, Tables 2.1-1 and 2.1-2 as amended by LAR 1014-1	
Control Method (Design Features)	See SAR Table 3.4-2 (Diablo Canyon ISFSI Technical Specification 4.1.1)	
Control Method (Operational)	See SAR Table 3.4-2 (Diablo Canyon ISFSI Technical Specification 3.2.1)	
Maximum k_{eff}	<0.95 (SAR Section 3.3.1.4 and Holtec FSAR Table 2.0.1, as amended by LAR 1014-1)	
Confinement Design		
Confinement Method	MPC with redundant welds (Holtec FSAR, Section 2.3.2.1, and Chapter 7, as amended by LAR 1014-1)	
Confinement Barrier Design	MPC (Holtec FSAR, Sections 2.2.6 and 2.2.15, as amended by LAR 1014-1 and Diablo Canyon ISFSI Technical Specification 4.2)	ASME III, Section NB
Maximum Confinement Boundary Leak Rate	5.0×10^{-6} atm-cm^3/sec He (SAR Section 10.2.2.5)	

ISFSI Storage Pad Design Criteria

The applicant identified the design criteria for the storage pads, as indicated in SAR Table 3.4-3 and Table 4-7 of this SER. For concrete components of the storage pads, the design criteria are based on ACI 349-97 (American Concrete Institute, 1998) and Draft Appendix B. ACI 349-97 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 Draft Appendix B covers anchorage systems in concrete. Note that the anchorage system used at the Diablo Canyon ISFSI does not fall within the guidelines of this appendix, because the size and length of the anchors used at the ISFSI exceed those with supporting data in the Draft Appendix B. The applicant used the pullout strength of reinforced concrete to identify that the anchorage design has a ductile failure mode (i.e., failure of the anchorage steel before concrete).

Cask Transporter Design Criteria

The applicant identified the design criterion for the cask transporter, as indicated in SAR Table 3.4-4 and Table 4-8 of this SER. The item is to be purchased as a commercial-grade product and qualified by testing prior to use. In addition, NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) is identified for the design criteria of the cask transporter, for compliance with a single-failure-proof lift system. NUREG–0612 identifies controls for handling heavy loads at nuclear power plants.

Cask Transfer Facility Design Criteria

The applicant identified the design criteria for the CTF, as indicated in SAR Table 3.4-5 and Table 4-9 of this SER. The design criteria for the load support portions of the CTF 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 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. In addition, NUREG–0612 is identified for the design criteria of the CTF load support SSCs, for compliance with a single-failure-proof lift system.

Table 4-7. Design criteria for Diablo Canyon ISFSI storage pads (Based on SAR Table 3.4-3)

Design Criterion	Diablo Canyon ISFSI Design Values	Applicable Criteria and Codes
Design Codes	ACI-349 (97) and Draft Appendix B (10/01/01) (SAR Sections 3.3.2.3 and 4.2.1.1.2)	NUREG–1536
Design Life	40 years (SAR Section 3.3.2.3)	
Maximum Single Loaded Cask Weight	163,300 kg [360,000 lb] (Holtec FSAR Table 2.0.1)	
Transporter with Loaded HI-STORM 100 System	240,000 kg [530,000 lb] (Holtec FSAR Table 2.0.1 and assumed value)	
Maximum Number of Casks on a Single Pad	20 (SAR Sections 1.3 and 4.1)	
Maximum Number of Pads at the ISFSI	7 (SAR Sections 1.3 and 4.1)	
Operating Temperature Range	−17 to 38 °C [0 to 100 °F] (DCPP FSAR update, Section 2)	
Concrete Pad Strength	34.4 MPa [5,000 psi] at 90 days	DG-1098 (8/00) ACI-349 (97) and Draft Appendix B (10/01/01)
Pad Loads and Load Combinations	Various	NUREG-1536, Table 3-1
Cask Anchor Stud Loads and Load Combinations	Various (SAR Section 3.3.2.3.2)	ASME, Section III, Subsection NF and Appendix F
Environmental Conditions and Natural Phenomena	see SAR Table 3.4-1 (SAR Section 3.2 and 3.3)	

Table 4-8. Design criteria for Diablo Canyon ISFSI cask transporter (Based on SAR Table 3.4-4)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Transporter Design Codes	Purchase commercial grade and qualify by testing prior to use. (SAR Sections 3.3. and 4.3.2.1, and Diablo Canyon ISFSI Technical Specification 4.3.1)	NUREG–0612
Design Life	20 years (SAR Section 3.3.3.2.1)	
Maximum Payload	163,300 kg [360,000 lb] (Holtec FSAR Table 2.0.1)	
Transporter Weight	77,100 kg [170,000 lb] (assumed value)	
Maximum Loaded Travel Speed	0.64 kmh [0.4 mph] (assumed value)	
Minimum Uphill Grade Capability	5% (carrying a loaded overpack) 10% (carrying a loaded transfer cask) (assumed value)	
Maximum On-Board Fuel Quantity	189 L [50 gal] (SAR Section 2.2.2.3 and Diablo Canyon ISFSI Technical Specification 4.3.1)	
Maximum Hydraulic Fluid Volume	Unlimited (must be nonflammable) (SAR Section 3.3.3.2.2)	
Operating Temperature Range	−17 to 38 °C [0 to 100 °F] (DCPP FSAR update, Section 2)	
Redundancy and Safety Factors for Load Path Structures and Special Lifting Devices	Holtec FSAR Section 2.3.3.1	In accordance with applicable guidelines of NUREG–0612
Hoist Load Factor	15%	CMAA 70 (94)
Position Control Maintained with Loss of Motive Power	Stops in position	In accordance with applicable guidelines of NUREG-0612
Environmental Conditions and Natural Phenomena	see SAR Table 3.4-1 (SAR Section 3.2 and 3.3)	

Table 4-9. Design criteria for Diablo Canyon ISFSI Cask Transfer Facility (Based on SAR Table 3.4-5)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Design Codes	ASME III-95, with 1996 and 1997 Addenda, Subsection NF NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) NUREG–1536 ACI–349 (97) and Draft Appendix B (10/01/00) (SAR Sections 3.3.4 and 4.4.5.2)	
Design Life	40 years (HOLTEC FSAR Section 2.3)	
Design Payload for Lift System	163,300 kg [360,000 lb] (HOLTEC FSAR Table 2.0.1)	
Loads and Load Combinations	Various (SAR Section 3.3.4.2.7)	ASME, Section III, Subsection NF and Appendix F
Hoist Load Factor	15%	CMAA 70 (94)
Operating Temperature Range	−17 to 38 °C [0 to 100 °F] (DCPP FSAR update, Section 2)	
Redundancy and Safety Factors for Load Path Members Special Lifting Devices	HOLTEC FSAR Section 2.3.3.1	In accordance with the applicable guidelines of NUREG–0612
Load Travel on Loss of Power or Jack Malfunction	Stops in position	In accordance with the applicable guidelines of NUREG–0612
Environmental Conditions and Natural Phenomena	SAR Table 3.4-1	

For concrete components of the CTF, the design criteria are based on ACI 349-97 (American Concrete Institute, 1998). ACI 349-97 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.

Load Combinations

The load combinations identified in SAR Section 3.2.5, “Combined Load Criteria,” are used in the analysis of SSCs important to safety. The HI-STORM 100 System is designed for normal, off-normal, and accident conditions under the appropriate load combinations as identified in HI-STORM 100 System FSAR Sections 2.2.1 through 2.2.7. Load combinations for the concrete storage pad and HI-STORM 100SA Overpack anchor studs are given in HI-STORM 100 System FSAR Sections 3.3.2.3.1 and 3.3.2.3.2. The cask transporter meets the applicable load criteria for the CTF as identified in HI-STORM 100 System FSAR Section 2.3.3.1. Load combinations for the CTF steel structure and equipment are discussed in HI-STORM 100 System FSAR Section 2.3.3.1. Load combinations for the CTF concrete are discussed in HI-STORM 100 System FSAR Section 3.3.4.2.7.1.

These load combinations are based on the requirements of ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992) and ACI 349–97 (American Concrete Institute, 1998).

The staff reviewed the Diablo Canyon ISFSI documentation and determined that the load combinations design criteria are appropriately considered for the design of SSCs important to safety as required by 10 CFR §72.122(b). Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena are considered.

Structural Design Criteria Conclusion

The structural design criteria discussed previously represent the structural loads that may be present at the site. The Diablo Canyon ISFSI SSCs important to safety must be designed to withstand these structural loads, as applicable. The ability of the SSCs to perform their intended safety functions in accordance with the applicable structural design loads is evaluated in Chapters 5 and 15 of this SER.

The Diablo Canyon ISFSI site-specific structural design criteria are bounded by the applicable structural design criteria for the HI-STORM 100 System, except for the seismic design criteria. Thus, except for the seismic analysis, the structural analysis presented in the HI-STORM 100 System FSAR and the NRC structural evaluation as documented in the HI-STORM 100 System SER are valid for the Diablo Canyon ISFSI. Because the seismic design loads for the Diablo Canyon ISFSI are not enveloped by the seismic design loads for the HI-STORM 100 System, the applicant performed an analysis to demonstrate that the HI-STORM 100SA Overpack 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 Diablo Canyon ISFSI documentation and determined that the principal design criteria, given in SAR Sections 3.2.2 through 3.4 considered for the design of SSCs 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 Diablo Canyon ISFSI.

4.1.3.3 Thermal

The staff reviewed the discussion on thermal design criteria of SSCs 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 nuclear 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 SSC 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 environmental conditions and heat generated by the materials stored.

Ambient condition design criteria are based on site-specific meteorological conditions. The extreme minimum, annual average, and extreme maximum design temperatures identified for the site are −4, 13, and 40 °C [24, 55, and 104 °F]. The design temperatures are based on 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 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), and the criteria are detailed in Chapter 6 of this SER.

The site-specific maximum total insolation for a 24-hour period was 742 Whr/m^2 (766 g cal/cm^2). As identified in SAR Section 2.3.2, “Local Meteorology,” this value is based on a 13-year record from the database for California Polytechnic State University in San Luis Obispo, California. The HI-STORM 100SA Overpack has been evaluated for the solar insolation values specified in 10 CFR §71.71(c)(1), which are 774 Whr/m^2 (800 g cal/cm^2) for flat surfaces and 387 Whr/m^2 (400 g cal/cm^2) for curved surfaces. The Standard Review Plan for dry-cask storage systems, NUREG-1536 (U.S. 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 insolation, calculated by averaging over a 24-hour day the reported 10 CFR Part 71 values for insolation over a 12-hour solar day, in a steady-state calculation.” The staff concludes that both site-specific measurement and regional data are bounded by the design of the HI-STORM 100SA Overpack.

The storage systems are passive and incorporate passive heat removal. The thermal design criteria for the cask system were evaluated during licensing of the HI-STORM 100 System and documented in the NRC HI-STORM 100 System SER.

Design temperatures for various HI-STORM 100 System materials are identified in the FSAR and are in compliance with acceptable codes. ACI 349-97 specifies the maximum temperature for normal operation and accident conditions. Allowable temperature for the fuel cladding is based on Interim Staff Guidance 11, Revision 3 (U.S. Nuclear Regulatory Commission, 2003). The ASME Boiler and Pressure Vessel Code, Section II, Part D, Table 1A specifies a design temperature for steel casks under all load conditions (ASME International, 1999). The applicant has provided the thermal performance requirements for all materials in conformance with accepted standards, as required for compliance with 10 CFR §72.120(a). Cask-specific material properties given in the SAR are derived from the HI-STORM 100 System FSAR Table 2.2.3.

4.1.3.4 Shielding and Confinement

The staff reviewed the discussion on shielding and confinement design criteria of SSCs 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 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.

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

- 10 CFR §72.122(h)(1) requires that the spent nuclear 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 while in 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 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(d) requires that the ISFSI be designed to provide means to limit to as low as reasonably achievable the release of radioactive materials in effluents during normal operations and to control the release of radioactive materials under accident conditions. Analyses must show that releases to the general environment in 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 calibrating and testing their operability.

- 10 CFR §72.128(a) requires spent fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel must 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 testability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.

Criteria used in the design of cask radiological protection features and confinement design of the cask system are provided in the SAR and the HI-STORM 100 System FSAR and are summarized in Table 4-6 of this SER. The basic concept for the Diablo Canyon ISFSI shielding and confinement system is protection by multiple barriers and systems, as required by 10 CFR §72.104 and §72.126(a). The use of the HI-STORM 100 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). The bounding dose rate design criteria are consistent with the requirements in 10 CFR §72.104(a) and §72.106(b).

The staff reviewed the design criteria for spent nuclear fuel storage and handling and determined that the criteria are appropriately identified as required by 10 CFR §72.128(a). In this SER, the staff shielding evaluation is presented in Chapter 7; the confinement evaluation in Chapter 9; and the radiation protection evaluation in Chapter 11. 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 SSCs with respect to the following regulatory requirements:

- 10 CFR §72.124(a) requires that spent nuclear 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 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 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. Monitoring of dry-storage areas where special nuclear material is packaged in its stored configuration under a license issued under 10 CFR Part 72 is not required.

Criteria used in criticality design of the cask systems are provided in the Diablo Canyon ISFSI SAR and the HI-STORM 100 System FSAR and are summarized in this SER, Table 4-6. The staff 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), §72.124(b), and §72.124(c).

4.1.3.6 Decommissioning

The staff decommissioning evaluation of SAR Section 4.6, “Decommissioning Plan,” is presented in Chapter 13 of this SER.

4.1.3.7 Retrieval

The staff reviewed the discussion on retrieval design criteria of SSCs with respect to the following regulatory requirements:

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

- 10 CFR §72.128(a) requires that spent nuclear fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel must be designed to ensure adequate safety under normal and accident conditions.

The spent nuclear fuel will be stored in and handled with the HI-STORM 100 System, which has been approved for use under the general license provisions of 10 CFR Part 72. As discussed in the HI-STORM 100 System FSAR and the staff related SER, the HI-STORM 100 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 System FSAR. The design-basis loads considered in the HI-STORM 100 System FSAR bound the structural and thermal loads found at the Diablo Canyon ISFSI except for the seismic load (see SER, Sections 4.1.3.2 and 4.1.3.3). The applicant provided an analysis that demonstrated the HI-STORM 100SA Overpack would neither tip-over 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 System FSAR. The seismic analysis is evaluated in Chapters 5 and 15 of this SER.

Based on the previous discussion, there is reasonable assurance that the HI-STORM 100 System will provide adequate safety and maintain fuel retrievability during the Diablo Canyon ISFSI 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 SSCs not important to safety. Therefore, no evaluation findings are made in this section; only the information provided in the SAR is discussed. The design criteria for SSCs classified as not important to safety, but which have security or operational importance, are addressed in SAR Section 4.5.6, “Design Criteria for SSCs Not Important to Safety.” The SAR specifies that these SSCs will be designed to comply with 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 ISFSI SAR, the following evaluation findings are made regarding the proposed Diablo Canyon ISFSI:

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

- The staff finds that the SSCs 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 10 CFR §72.24(n). This list of SSCs is based on the definition in 10 CFR §72.3 of SSCs important to safety. The SAR appropriately specifies the design criteria for the SSCs 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 SSCs 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 Diablo Canyon ISFSI. These design criteria satisfy the requirements of 10 CFR §72.120(a) and §72.122(h). Additionally, the SSCs 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 §72.122(b). The seismic design criteria are in accordance with the site-specific seismic hazards analysis given in SAR Chapter 2, "Site Characteristics."

- The staff finds that the explosion hazards considerations in the SAR are consistent with standard design criteria specified by Regulatory Guide 1.91 and are in accordance with 10 CFR §72.122(c). Design peak incident pressures have been defined.

- The staff finds that the load combinations design criteria are adequately considered for the design of SSCs, 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.

- The staff finds that design criteria for criticality are identified in the SAR, as required by 10 CFR §72.124(a), §72.124(b), and §72.124(c).

- The staff finds that the Diablo Canyon ISFSI design, which includes use of the HI-STORM 100 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. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-85. Detroit, MI: American Concrete Institute. 1985.

American Concrete Institute. *Building Code Requirements for Structural Concrete.* ACI 318-95. Detroit, MI: American Concrete Institute. 1995.

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-97. Detroit, MI: American Concrete Institute. 1998.

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

American National Standards Institute/American Nuclear Society. *Radioactive Materials Special Lifting Devices for Shipping Containers Weighing 10,000* (4,500 kg). ANSI/ANS 14.6. La Grange, IL: American National Standards Institute. 1993.

American Society of Civil Engineers. *Minimum Design Loads for Buildings and Other Structures*. ASCE 7-98. New York City, NY: American Society of Civil Engineers. 2000.

ASME International. *ASME Boiler and Pressure Vessel Code, Section III, Division 1.* New York City, NY: ASME International. 1998.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D–Properties. 1998 Edition with 1999 Addenda.* New York City, NY: ASME International. 1999.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System Revision 1 (HI-STORM 100 Cask System).* Volumes I and II. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

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

Pacific Gas and Electric Company. *Dry Cask Storage System*. PG&E Specification 10012–N–NPG. Avila Beach, CA: Pacific Gas and Electric Company. September 2000.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update*. Revision 14. Avila Beach, CA: Pacific Gas and Electric Company. November 2001.

Pacific Gas and Electric Company. *Diablo Canyon ISFSI Safety Analysis Report*. Amendment 2. Docket No. 72-26. Letter DIL-03-012, October 3, 2003. Avila Beach, CA: Pacific Gas and Electric Company, 2003.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information regarding Cask Transporter Lateral Restraints for the Diablo Canyon ISFSI (TAC No. L23399).* Letter DIL–03–015, December 4, 2003. Avila Beach, CA: Pacific Gas and Electric Company, 2003a.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Amendment 2 to Proposed Technical Specifications and Technical Specification Bases for Diablo Canyon Spent Fuel Storage Installation (TAC No. L23399)*. DIL-03-013, October 10, 2003. Avila Beach, CA: Pacific Gas and Electric Company, 2003b.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information Regarding Implementation of Guidance Provided in ISG-11 (TAC No. L23399).* DIL-04-002, January 16, 2004. Avila Beach, CA: Pacific Gas and Electric Company, 2004.

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

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

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

U.S. Nuclear Regulatory Commission. Missiles Generated by Natural Phenomena. Revision 2, *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: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. July 1981.

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

U.S. Nuclear Regulatory Commission. 10 CFR Part 72 *Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report, Amendment 1*. Docket No. 72-1014. July 15, 2002b.

U.S. Nuclear Regulatory Commission. *Interim Staff Guidance-11: Cladding Considerations for the Transportation and Storage of Spent Fuel.* Rev. 3. Washington, DC: U.S. Nuclear Regulatory Commission. November 2003.