5 INSTALLATION AND STRUCTURAL EVALUATION

5.1 Conduct of Review

This chapter of the Safety Evaluation Report contains reviews of the information presented in the Diablo Canyon ISFSI Safety Analysis Report Chapter 4, "ISFSI Design." The objective of the installation design review is to ensure compliance with the site features and to support other evaluation areas. The objective of the structural evaluation review is to ensure structural integrity of structures, systems, and components (SSCs) with emphasis on those that are important to safety. The review also considers selected sections and documents referenced in SAR Chapters 1, 2, 3, and 8. These chapters discuss general information, site characteristics, principal design criteria, and accident analysis.

Spent nuclear fuel dry storage facilities are designed for safe confinement and storage of the spent nuclear fuel. The design of the proposed Diablo Canyon ISFSI is based on the use of the HI-STORM 100 System, which has been reviewed by the NRC and approved for general use under Certificate of Compliance (CoC) No. 1014, Amendment 1 (U.S. Nuclear Regulatory Commission, 2002a). The Diablo Canyon ISFSI SAR references the HI-STORM 100 System, as described in the HI-STORM 100 System Final Safety Analysis Report (FSAR) Revision 1 (Holtec International, 2002), for confinement and radiological safety. Where applicable, the NRC staff relied on its previous review of the HI-STORM 100 System SER, through Amendment 1 (U.S. Nuclear Regulatory Commission, 2002b). The major categories of safety protection systems discussed in the following sections include confinement SSCs, reinforced concrete structures, other SSCs important to safety, and other SSCs not important to safety.

The staff reviewed the Diablo Canyon ISFSI installation and structural evaluation with respect to the following regulatory requirements:

- 10 CFR §72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity. If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI and such other facility—including shared common utilities and services—must be evaluated.

- 10 CFR §72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

- 10 CFR §72.24(c) requires that the design of the ISFSI be described in sufficient detail to support the findings in §72.40, including (1) the design criteria for the ISFSI pursuant to subpart F of this part, with identification and justification for any additions to or departures from the general design criteria; (2) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including the design bases and the relation of the design bases to the design criteria. (3) requires that the design of the ISFSI be provided in sufficient detail to support the findings in Section 72.40, including information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin for safety. (4) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including applicable codes and standards.

- 10 CFR §72.24(d) requires an analysis and evaluation be provided of the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI and including determination of the (1) margins of safety during normal operations and expected operational occurrences during the life of the ISFSI and (2) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents, including natural and manmade phenomena and events.

- 10 CFR §72.24(i) requires the identification of any structures, systems, or components important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel or high-level radioactive waste for storage at the ISFSI.

- 10 CFR §72.106(a) requires that a controlled area must be established.

- 10 CFR §72.120(a) requires that, pursuant to the provisions of §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 §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(a) requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.

- 10 CFR 72.122(b) requires that (1) Structures, systems, and components important to safety must 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. (2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect (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 onto structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) 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 too minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

- 10 CFR §72.122(f) requires that systems and components that are important to safety be designed to permit inspection, maintenance, and testing.

- 10 CFR §72.122(g) requires that structures, systems, and components important to safety be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.

- 10 CFR §72.122(h)(1) 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. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR §72.122(h)(4) requires that the 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. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design 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, 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.

- 10 CFR §72.236(b) requires that design bases and design criteria be provided for structures, systems, and components important to safety.

- 10 CFR §72.236(c) requires that the spent fuel storage cask be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

- 10 CFR §72.236(e) requires that the spent fuel storage cask be designed to provide redundant sealing of confinement systems.

- 10 CFR §72.236(f) requires that the spent fuel storage cask be designed to provide adequate heat removal capacity without active cooling systems.

- 10 CFR §72.236(g) requires that the spent fuel storage cask be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required.

- 10 CFR §72.236(l) requires that the spent fuel storage cask and its systems important to safety be evaluated, by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

5.1.1 Confinement Structures, Systems, and Components

The discussion about confinement SSCs is presented in SAR Section 4.2.3, “Storage Cask Description;” and in Chapter 4 of the HI-STORM 100 System FSAR (Holtec International, 2002). The staff reviewed the discussion about confinement SSCs with respect to the applicable regulatory requirements as discussed next.

5.1.1.1 Description of Confinement Structures

The Diablo Canyon ISFSI confinement structure is a spent nuclear fuel canister, specifically the Holtec multi-purpose canister (MPC) of the HI-STORM 100 System. Detailed descriptions of the MPC are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff previously reviewed and found this description acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). SAR Section 4.2.3, “Storage Cask Description,” referenced the HI-STORM 100 System FSAR and provided a summary description of the confinement structure. The staff found the summary to be consistent with the information in the HI-STORM 100 System FSAR. The confinement structure was sufficiently described in accordance with 10 CFR §72.24 and §72.236.

5.1.1.2 Design Criteria for Confinement Structures

The design criteria for the MPC are presented in the HI-STORM 100 System FSAR (Holtec International, 2002) and evaluated in the related SER (U.S. Nuclear Regulatory Commission, 2002b). A summary of the design criteria is contained in the SAR Table 3.4-2. Design criteria for the MPC have been shown in Chapter 4 of this SER to be representative of the site.

The MPC confinement boundary is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-3200 (ASME International, 1995a). Fabrication of the MPC is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB–4000; and Subsection NG, Article NG–4000 (ASME International, 1995a). MPC inspection is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Articles NB–5000 and NG–5000 (ASME International, 1995a), and Section V (ASME International, 1995b).

MPC confinement boundary welding will be performed using welders and weld procedures that have been qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c); and Section III, Subsections NB and NG (ASME International, 1995a). Nondestructive examination of the MPC welds are specified in engineered drawings in the HI-STORM 100 System FSAR, Section 1.5 (Holtec International, 2002). MPC fabrication welds will be inspected using visual testing and radiographic testing or ultrasonic testing and penetrant testing in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB; Article NB–5300; and Subsection NG, Article NG–5300 (ASME International, 1995a).

Exceptions to the ASME and Pressure Vessel Code are provided in the CoC 1014-1, Appendix B, Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR Table 3.4-6 (Pacific Gas and Electric Company, 2002a).

The staff concludes that the design criteria of the confinement structures meet the requirements of the ASME Code, as applicable. The conclusions drawn in this section of the confinement structures design criteria are based on the evaluation findings made in Section 4.1.3 of this SER. The confinement structure design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

5.1.1.3 Material Properties for Confinement Structures

Materials Selection

The applicant provided a description of the HI-STORM 100 System including materials of construction, fabrication details, and testing in SAR Sections 4.2, "Storage System;" and 4.7, "Operating Environment Evaluation;" and Appendix A, "Materials" (Pacific Gas and Electric Company, 2002a). Engineering drawings and additional details of the storage system are included by reference to the Holtec HI-STORM 100 System FSAR Sections 3.3 and 3.4 (Holtec International, 2002). Technical specifications for the HI-STORM 100 System are included by reference to 10 CFR Part 72 CoC No. 1014-1 (U.S. Nuclear Regulatory Commission, 2002a) for the HI-STORM 100 System. The HI-STORM 100 System has been evaluated by the staff and approved for use for dry storage of spent nuclear fuel (U.S. Nuclear Regulatory Commission, 2002b). The staff reviewed the information contained in these documents to determine compliance of the proposed Diablo Canyon ISFSI with the requirements of 10 CFR §72.24(c)(3), §72.24(c)(4), §72.122(a), §72.122(b), §72.122(c), §72.122(h), §72.122(i), and §72.236(g).

The structural components of the MPC are constructed from Types 304, 304LN, 316, or 316LN austenitic stainless steel (Holtec International, 2002). Stainless steels were selected based on mechanical properties and corrosion resistance. Shielding is provided by additional material thickness of the MPC shell, baseplate, lid, and closure ring (Holtec International, 2002). Applicable codes for the material procurement, design fabrication, and inspection of the MPC are provided in HI-STORM 100 System FSAR Table 2.2.7 (Holtec International, 2002) and in the Proposed Technical Specifications for Diablo Canyon ISFSI (Pacific Gas and Electric Company, 2002b). Material procurement is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,e,f), and Section III, Subsection NB, Article NB-2000 and Subsection NG, Article NG–2000 (ASME International, 1995a). Materials for the MPC baseplate, lid, closure ring, port cover plates, and shell are examined in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB–2000 (ASME International, 1995a). No ductile-to-brittle transition temperature exists for the austenitic stainless steel structural materials, so the MPC is not susceptible to brittle fracture. The staff concludes that the selection of these materials is acceptable for the MPC.

Welds

The MPC welds are characterized in engineered drawing 3923 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002). The drawing includes standard welding symbols and notations in accordance with American Welding Society (AWS) Standard A2.4 (American Welding Society, 1998). The stainless steel materials for the MPC are readily weldable using commonly available welding techniques. MPC closure welds are inspected using visual testing and ultrasonic testing or multilayer penetrant testing. If multilayer penetrant testing is used, the examination will include penetrant tests for the root and final passes and for each approximately 0.95 cm [3/8 in] of weld depth consistent with the minimum flaw size for the MPC identified in Holtec position paper DS–213 (Holtec International, 1999). Additional details of the nondestructive examination of the lid-to-shell weld, structural and pressure tests, and hydrostatic testing of the MPC are provided in HI-STORM 100 System FSAR Section 9.1 and

Table 9.1.1 (Holtec International, 2002). The staff concludes that the welded joints of the MPC meet the requirements of the ASME and AWS codes, as applicable.

Mechanical Properties

Mechanical properties of the structural materials for the MPC, are provided in the Holtec HI-STORM 100 System FSAR Section 3.3 and Tables 3.3.1 and 1.A.1 (Holtec International, 2002) for stainless steels. The mechanical properties of the stainless steel structural materials, such as design stress intensity (S_m), tensile strength (S_u), yield strength (S_y), coefficient of thermal expansion (α), and coefficient of thermal conductivity (k), vary with stainless steel composition. Qualification of the MPC structure is accomplished using the least favorable mechanical and thermal properties of the entire group for all mechanical, structural, neutronic, radiological, and thermal conditions. Mechanical properties of the stainless steel structural materials are provided in the HI-STORM 100 System FSAR, Tables 3.3.1 and 1.A.1 (Holtec International, 2002). The values in these tables were obtained from ASME Code Section II Part D (ASME International, 1995f). The staff independently verified the temperature-dependent values for the stress allowables, ultimate strength, yield strength, modulus of elasticity, and coefficient of thermal expansion. The staff concludes that these material properties are acceptable and appropriate for the expected load conditions during the license period.

Coatings

No coatings are used on the MPC.

Chemical and Galvanic Reactions

Evaluation of possible chemical, galvanic, and other reactions among the materials in the range of possible exposure environments is included in SAR Section 4.7, “Operating Environment Evaluation,” and Table 4.7-1 (Pacific Gas and Electric Company, 2002a). The evaluation includes stainless steels used in the MPC. The staff concur that no adverse reactions are anticipated for stainless steels used in the MPC.

Based on the previous discussion of the materials selection welds, mechanical properties, coating, and chemical and galvanic reaction the staff concludes that the material selection for the confinement structures meets the requirements of the ASME codes, as applicable. The material properties for confinement structure have been identified in accordance with 10 CFR §72.24(c)(3), §72.120(a), §72.122, and §72.236(b).

5.1.1.4 Structural Analysis for Confinement Structures

The staff reviewed the discussion of the MPC design relative to the storage requirements of the Diablo Canyon ISFSI provided in SAR Section 4.2.3, “Storage Cask Description.” The Diablo Canyon ISFSI SAR provides a summary of the analysis performed in the HI-STORM 100 System FSAR (Holtec International, 2002). The Diablo Canyon ISFSI SAR Section 4.2.3.3.2, “Structural Design,” states that the MPC has the structural capability to withstand the loads created by all design basis normal, off-normal, and accident conditions and for the design basis natural phenomena. The following loads and combined loading conditions were considered.

- Dead and Live Loads (SAR Section 4.2.3.3.2.1)
- Internal and External Pressure Loads (SAR Section 4.2.3.3.2.2)
- Thermal Expansion (SAR Section 4.2.3.3.2.3)
- Handling Loads (SAR Section 4.2.3.3.2.4)
- Overpack/Transfer Cask Tip-Over and Drop (SAR Section 4.2.3.3.2.5)
- Tornado Winds and Missiles (SAR Section 4.2.3.3.2.6)
- Flood (SAR Section 4.2.3.3.2.7)
- Earthquake (SAR Section 4.2.3.3.2.8)
- Explosion Overpressure (SAR Section 4.2.3.3.2.9)
- Fire (SAR Section 4.2.3.3.2.10)
- Lightning (SAR Section 4.2.3.3.2.11)
- 500-kV Line Drop (SAR Section 4.2.3.3.2.12)

The detailed structural analysis of confinement structures is presented in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff previously reviewed this structural analysis and found it acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). As documented in that SER, the structural analysis shows the structural integrity of the HI-STORM 100 System is maintained under all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the confinement structures under the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code Section III (ASME International, 1995a) for the confinement structures materials.

The loading conditions at the ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR. The applicant did not perform any drop or tip-over analysis. Outside the Fuel Handling Building and Auxiliary Building (FHB/AB), tip-over of the HI-STORM 100SA Overpack is considered a noncredible accident. When not on the ISFSI pad, the system will either be in the Cask Transfer Facility (CTF) or attached to the Cask Transporter, both of which are designed, fabricated, inspected, operated, maintained, and tested in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). A hypothetical tip-over of a HI-STORM 100SA Overpack because of handling, a seismic event, or tornado winds with concurrent impact of the tornado-driven design missile (an automobile) at the top of the storage cask is noncredible. When on the storage pad, the casks are anchored to the pad. As demonstrated in the HI-STORM 100 System FSAR, an anchored cask will not tip-over, therefore, tip-over of the cask is noncredible.

Therefore, the staff conclusions in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b) with respect to the structural integrity of the MPC are valid for the Diablo Canyon ISFSI and, therefore, meets the Diablo Canyon ISFSI design criteria given in SAR Section 3.3, "Design Criteria for Safety Protection Systems." The confinement structure analysis has demonstrated compliance with 10 CFR §72.24(a), §72.24(d), §72.122(b), §72.122(c), §72.122(h), §72.128(a), and §72.236(l).

A site-specific analysis was performed to assess the vulnerability of the MPC at the CTF to a transmission tower collapse (Holtec International, 2001a). The transmission tower collapse looked at the vulnerability of the MPC lid during transfer at the CTF. The analysis results indicate there would be localized yielding of the MPC in the immediate vicinity of the impact site. Based on the results of the analysis the resultant material stress, away from the vicinity of the impact, does not exceed allowable values. Because the yielding is localized, there is no loss of shielding or confinement from the sealed MPC and no loss of retrievability of a spent nuclear fuel assembly from the MPC.

5.1.2 Pool and Pool Confinement Facilities

This provision is not applicable to 10 CFR Part 72 dry storage facilities.

5.1.3 Reinforced Concrete Structures

This section contains a review of SAR Section 4.2.1.1, “Cask Storage Pad.” The staff reviewed the discussion about reinforced concrete structures important to safety with respect to the applicable regulatory requirements, as discussed below.

5.1.3.1 Description of Reinforced Concrete Structures

There is one reinforced concrete structure in the Diablo Canyon ISFSI that has been classified as important to safety, the ISFSI storage pad with overpack anchorage hardware (QA Category B), with the design and analysis information given in SAR Section 4.2.1.1, “Cask Storage Pads.” The ISFSI storage pads are independent structural units constructed of reinforced concrete, designed in accordance with ACI 349-97 (American Concrete Institute, 1998) and Draft Appendix B, for the anchorage hardware. Figure 4.2-1 of the SAR shows a schematic of the storage pads. Each pad is 20.7 × 32 × 2.3 m [68 × 105 × 7.5 ft] and is capable of supporting 20 loaded HI-STORM 100SA Overpacks. The pad thickness may vary from 2.3 to 2.4 m [7.5 to 8 ft], based on the interface to the foundation rock and the 1 percent slope on the top surface for drainage. The size of the pad is based on a 5.2-m [17-ft] center-to-center spacing of the storage casks. It provides a level and stable surface for placement and storage of the storage casks. The ISFSI Storage Pad design is based on the maximum loaded weight of a storage cask of 163,300 kg [360,000 lb], the weight of the HI-STORM 100SA Overpack loaded with an MPC canister. The storage pad also provides the necessary embedment for the anchorage hardware for the HI-STORM 100SA Overpacks.

The overpack anchorage hardware is shown in the SAR Figures 4.2-1 and 4.2-2. The ISFSI storage pad is designed with a 50.8-mm-[2-in-] thick steel plate ring at the surface of the concrete that mates with the bottom of the cask. The base plates are designed to provide sufficient bearing area on the concrete to transfer loads. Each cask is attached to the pads using 16 studs that are threaded into a coupling steel block located on the underside of the embedment plate. Each stud is preloaded to approximately 71,200 kg [157,000 lb] by threading the stud into the coupling block, applying tension to the stud, and installing the nut on the stud to maintain the preload. The preloaded anchor studs are used to ensure that interface contact is maintained between the ISFSI pad embedment upper surface and the lower surface of the HI-STORM 100SA Overpack baseplate. Shear loads for each cask will be carried through the embedment plate/coupling block into the concrete. Long steel rods are embedded in the concrete to transfer the load from the cask to the concrete.

The SAR provides a design description of the ISFSI storage pads with overpack anchorage hardware in sufficient detail to support a detailed review and evaluation in accordance with 10 CFR §72.24(a) and §72.24(b).

Description of the ISFSI storage pads and associated operations procedures include consideration of inspection, maintenance, and testing as required in 10 CFR §72.122(f). The design of the reinforced concrete pads, a slab with an upper surface level with the surrounding grade, provides access to all locations and allows access to the storage casks in the event of emergencies. There are no barriers built into the ISFSI storage pads that would prevent access to any location on the pads adjacent to the storage casks. This design allows emergency response capability, as required in 10 CFR §72.122(g). This reinforced concrete slab embedded in a rock foundation also incorporates the capability for retrieving the spent nuclear fuel canisters. The cask transporter can drive onto the pad to access any storage cask and transport it back to the CTF. Because of the rock foundation, settlement of the pad is considered insignificant, as shown in this SER Section 5.1.3.4; stress in the rock under static and dynamic loads is significantly below allowables. Consequently, the storage cask can be retrieved from the storage pads in accordance with 10 CFR §72.122(l).

5.1.3.2 Design Criteria for Reinforced Concrete Structures

The design bases for the reinforced concrete ISFSI storage pads with overpack anchorage hardware are given in the SAR Section 3.3.2, "ISFSI Concrete Storage Pad;" and SAR Section 4.2.1.1, "Cask Storage Pads." Table 3.4-3 of the SAR identifies details of the Diablo Canyon ISFSI compliance with the general design criteria of 10 CFR Part 72, Subpart F. This conclusion is also supported by the structural analysis described in Section 5.1.3.4 of this SER. Design criteria for the storage pad have been shown in Chapter 4 of this SER to be representative of the site.

ISFSI storage pads are designed in accordance with ultimate strength design methods specified in ACI 349-97 (American Concrete Institute, 1998). The ACI 349-97 Code specifies the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. Draft Appendix B of ACI 349-97, identifies requirements for steel embedments. Appendix D of ACI 318-02 (American Concrete Institute, 2002) is the most recent revision to the design requirements for anchors in concrete to transmit structural loads. The design requirements identified are commonly accepted by the construction industry but have not been explicitly adopted by the NRC. The procedures are acceptable but the load cases must be supplemented by the load cases of American National Standards Institute (ANSI) 57.9 (American National Standards Institute/American Nuclear Society, 1992). A feature of the design requirements is a design that results in a ductile failure of the metal components of the anchorage system prior to brittle failure of the concrete. The applicant uses the guidance of Draft Appendix B to define the requirement for a ductile failure of the steel portion of the anchorage hardware prior to brittle failure of the reinforced concrete.

The design criteria for the ISFSI storage pads with overpack anchorage hardware establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for reinforced concrete storage pads. Additionally, the design criteria of the storage pads address site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. The conclusions drawn in this section on the storage pad design criteria are based on the evaluation findings made in Section 4.1.3 of this SER. The reinforced concrete structures design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

5.1.3.3 Material Properties for Reinforced Concrete Structures

The staff reviewed the materials of construction of the ISFSI storage pads, as identified in SAR Section 4.2.1.1. Material selection is based on concrete with a compressive strength of 34.5 MPa [5,000 psi] at 90 days following Chapter 5 of ACI 349-97 (American Concrete Institute, 1998). The reinforcing steel is specified to have a minimum yield strength of 414 MPa [60,000 psi] following Chapter 3 of ACI 349-97 (American Concrete Institute, 1998) and American Society for Testing and Materials (1990). Materials of construction of the overpack anchorage hardware, as identified in SAR Section 4.2.1.1, "Cask Storage Pads," include SA193–B7 studs, SA516 Grade 79 receptacles, and A–36 plates and bars. As identified in SAR Section 3.3.2.3, "Design Criteria," the exposed portions of this hardware will be coated to protect them from the environment.

Based on the review of information presented by Pacific Gas and Electric Company (PG&E), the staff concludes that materials to be used to construct the ISFSI storage pads with anchorage hardware have been adequately identified in accordance with 10 CFR §72.24(c)(3). The applicant has identified the appropriate codes and standards in accordance with 10 CFR §72.24(c)(4).

5.1.3.4 Structural Analysis for Reinforced Concrete Structures

The Diablo Canyon ISFSI reinforced concrete structures, as described in the SAR, are designed to meet the requirements of ACI 349-97 (American Concrete Institute, 1998). The staff accepts the strength design method, as presented in the ACI 349-97, for concrete structures important to safety. The reinforced concrete storage pads important to safety were analyzed for normal, off-normal, and accident-loading conditions. These analyses were carried out to ensure the storage pads would be able to perform their intended safety functions under the extreme environmental and natural phenomena, as specified in 10 CFR §72.122(b)(1) and §72.122(b)(2) and ANSI/ American Nuclear Society (ANS) 57.9 (American National Standards Institute/American Nuclear Society, 1992). The ultimate strength method of analysis is used with the appropriate load factors for the following loads:

- Dead loads (D)
- Live loads (L)
- Soil pressure loads (H)
- Temperature gradients (T)
- Wind loads (W)
- Earthquake loads (E)
- Accident (A) loads including explosion over pressure, drop/tip-over, accidental pressurization, fire, and aircraft impact
- Design-basis tornado wind loads and tornado-generated missile loads (W_t)
- Probable maximum flood loads (F)
- Lightning

The staff reviewed the SAR and found that the structural analysis procedures have been identified and are in conformance with standard engineering practice, as described in ACI 349-97 (American Concrete Institute, 1998). The relationship between the design criteria, identified in Chapter 3 of the SAR, and the analysis procedures was established in accordance with the requirements of 10 CFR §72.24(c)(2). The applicable codes and standards used in the analysis of the reinforced concrete structures also have been identified in the SAR, in accordance with the requirements of 10 CFR §72.24(c)(4).

Analysis of the stability of the subsurface materials under the storage pads and the potential for failure along the clay bed and tilting of the storage pads was reviewed in this SER Section 2.1.6.4, "Stability of Subsurface Materials." The slope stability of the material above the storage pads and the potential for it to impact and/or cover the storage casks on the storage pads was reviewed in the SER Section 2.1.6.5, "Slope Stability."

The SAR Section 8.2.1.2.3.1, "Cask and Anchorage Seismic Analysis," summarizes seismic analysis of the cask and anchorage system performed by Holtec. Although the Diablo Canyon site-specific seismic zero period accelerations (ZPAs) for all events are lower than those identified in Appendix B of the Holtec CoC (U.S. Nuclear Regulatory Commission, 2002a), Holtec International performed an analysis of the anchored HI-STORM 100SA Overpack at the Diablo Canyon ISFSI (Holtec International, 2001c). The primary reason for this analysis was a difference in the number of anchor rods identified for the Diablo Canyon ISFSI with respect to the design basis given for the HI-STORM 100 System (Holtec International, 2002). The Diablo Canyon ISFSI anchorage system uses sixteen 63.5-mm- [2.5-in] diameter rods, and the HI-STORM 100 System generic design calls for twenty-eight 50.8-mm [2-in] diameter rods, Section 3.4 of CoC No. 1014 (U.S. Nuclear Regulatory Commission, 2002a). The objectives of the analysis were to

(1) Demonstrate that the seismic events do not induce acceleration levels that exceed the design basis. Based on comparison of the maximum net shear at the base of the embedment plate to the bounding cask weight, the effective "g" is 1.43 g. The results indicate that the casks do not develop body decelerations that exceed the cask design basis of 45 g.

(2) Demonstrate that the seismic events do not induce stress in the preloaded anchor studs, cask flange, and shell that exceed the design basis ASME Code limits. This included a demonstration of all structural safety factors greater than 1 for ASME load Level A (preload) and Level D (Preload + Seismic Load). For the Level A load, a 72,600-kg [160,000-lb] load is assumed to act on the components that bound the actual 71,200-kg-[157,000 lb-] preload. For the Level D load, the maximum load per stud was 97,250 kg [214,400 lb], which is below the ultimate capacity of 97,800 kg [215,600 lb]. These loads were then applied to a local model of the cask flange and shell. Table 2 of Holtec International (2001c) identifies the factors of safety for these load conditions. The minimum factor of safety is 1.089, associated with a weld shear stress. In addition, the alternating stress intensity under the seismic loading must be sufficiently low that a factor of safety against fatigue failure is demonstrated. The estimated fatigue cycles required to fail the stud and sector lug are significantly higher than the number of cycles during a single earthquake event.

(3) Establish interface loads transferred to the ISFSI pad embedment. The peak interface loads at the lower surface of the embed plate are summarized in Table 3 of Holtec International (2001c). The resultant interface loads are identical to those identified in Table 2 of the storage pad seismic analysis (ENERCON Services Inc., 2001b).

The staff reviewed the design documentation to verify the assumptions, analysis procedures, modeling procedures, and summary of results. The dynamic analysis employed a three-dimensional model of the MPC, overpack, and anchoring system using VisualNastran 2001 (MSC Software Corporation, 2001). As identified in Table 1 of Holtec International (2001c), the input data for the model are consistent with the HI-STORM 100 System FSAR. Relative motion and impact between the MPC and overpack are properly modeled. The interface between the embedment and the ISFSI concrete is properly modeled by discrete springs to simulate the anchor rods and the compression only concrete. Of the four seismic events for the Diablo Canyon site, only the Long-Term Seismic Program (LTSP) and Hosgri Earthquake (HE) events were used for the dynamic simulation because they impart the highest acceleration and, therefore, highest loading to the anchorage components. The appropriate materials properties were identified and used. The staff concur with the results presented in SAR Section 8.2.1.2.3.1, "Cask and Anchorage Seismic Analysis."

SAR Section 8.2.1.2.3.2, "Storage Pad Seismic Analyses," identifies the analysis performed to ensure that the reinforced concrete pads and the anchored casks remain functional during all seismic conditions. Two analyses are covered in this section, a static analysis (ENERCON Services, Inc., 2001b) and a nonlinear pad sliding analysis (Pacific Gas and Electric Company, 2001d).

The static analysis (ENERCON Services Inc., 2001b) determined the storage pad size and thickness required to resist the loads resulting from seismic accelerations applied to the pad and resultant loads from the cask dynamic analysis (Holtec International, 2001c). The limiting parameter considered in this assessment is that the pad displacement under the pad dead weight and seismic loads be held to an acceptable value (displacement < length/320), ACI 349-97 (American Concrete Institute, 1998). The HE and LTSP spectra were used because these spectra produce the largest ZPAs. Load combinations for the sequencing of cask placement were considered. The ANSYS nonlinear static finite element model consisted of the pad, a portion of the underlying rock foundation, and the casks on top of the pad. The model was constructed of three-dimensional solid elements. All material properties were linear, and the concrete values are consistent with the facility design. The rock foundation Young's modulus was varied from 1,380 to 33,800 MPa [0.2×10^6 to 4.9×10^6 psi] (representing soft, hard, and very hard rock). The extent of the rock foundation was sufficient that the boundaries did not influence the response at the storage pad location. The casks are modeled up to the plane of their center of gravity. The Young's modulus for the cask was adjusted to force the fundamental frequency of the cask to match that obtained by Holtec International (2001c). Compression only gap elements were used to model the interface between the slab (target surface) and rock foundation (contact surface). The element stiffness and convergence parameters were computed from the geometry and material properties of the adjoining elements. Shear loads were transmitted from the pad into the rock foundations through constraint equations. The bounding load cases were derived from the results of the Holtec analysis (Holtec International, 2001c) and applied to the center of gravity of the storage casks. In addition to the cask loads, an inertia force was applied to the pad with reference to the ZPA of the seismic event. The Newmark 100-40-40 method was used to combine the three specified directions of the seismic load. A total of 19 discrete load steps were applied to the model.

The results of each load step were checked for equilibrium and found to be acceptable. The results were post-processed to obtain the maximum displacement of the pad, maximum displacement of the centerline of the casks on the perimeter of the array, vertical displacement of the pad for the soft rock model, max/min stresses in the X and Y horizontal directions, maximum principal stress (largest tensile stress), minimum principal stresses (largest compressive stress). The results are summarized in Tables 2 to 5 of the ENERCON calculation package (ENERCON Services Inc., 2001b). The pad and cask vertical displacement are small and within acceptable limits. The maximum tensile stress in the concrete is 2.48 MPa [359 psi] and is less than the tensile stress, 3.65 MPa [530 psi], that will cause cracking in the 34.5-MPa [5,000-psi] concrete. The maximum compressive stress is 5.56 MPa [806 psi], significantly less than the 34.5-MPa [5,000-psi] design value.

The analysis demonstrated that the greatest demand on the slab was for the HE seismic event. It was also determined that the mass of the pad and its inertial loads were as important to the overall response of the pad as the applied loads from the casks. Sections throughout the pad were isolated for the HE seismic event, and the internal forces acting upon them were computed. The results are summarized in Tables 6 to 10 of the ENERCON calculation package (ENERCON Services Inc., 2001b). The resulting internal forces for design purposes are given in Table 11 of the ENERCON calculation package (ENERCON Services Inc., 2001b). A factor of 1.15 will be applied to these loads to account for potential variations due to Poisson's ratio of the rock foundation.

The results were scanned to identify the stress in the rock below the pad. The applied gravity pressure was 0.11 MPa [2.36 ksf], which is less than the allowable bearing pressure of 1.91 MPa [40 ksf]. The maximum seismic pressure calculated was 0.33 MPa [6.95 ksf] for the HE load case and 0.31 MPa [6.42 ksf] for the LTSP load case, each is less than the allowable bearing pressure of 2.49 MPa [52 ksf]. A review of the stability against bearing capacity failure under static and dynamic loading is contained in this SER Section 2.1.6.4, "Stability of Subsurface Materials."

The reduced weight of concrete (90 percent of the design value) to account for construction variability resulted in displacements only 11 percent greater, still within the acceptable range, and lower stresses. The analyses for placement sequence configurations and cask extraction were shown to be bounded by the fully loaded analyses.

The results of the analysis were used in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) to evaluate the concrete per the design codes and to determine the size of the steel reinforcement in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998). The analysis (PGE–009–CALC–007) has been completed and is in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998).

A nonlinear analysis was performed to determine the extent of sliding at the pad/rock interface (Pacific Gas and Electric Company, 2001d). The ISFSI long period (ILP) time histories were used because the pad sliding may be sensitive to long-period ground motion. This analysis used a lumped mass model of 20 casks on a pad using SAP2000. The pad was modeled using a small lumped mass. The interface between the pad and the rock foundation was a biaxial friction element that has coupled fiction properties for the two shear deformation, post-slip stiffness in the shear directions, and gap behavior in the axial direction. The review of stability against sliding under dynamic loading is contained in this SER Section 2.1.6.4, "Stability of Subsurface Materials."

The anchorage system components for the HI-STORM 100SA Overpack provide a level surface for the casks to set on, 16 receptacles for the anchorage studs, and strength to transmit cask loads caused by external events to the concrete pads (ENERCON Services Inc., 2003b). The loads applied correspond to those calculated by Holtec (Holtec International, 2001c). The anchorage system was designed to meet the ductile anchorage provision of the Proposed Draft, Appendix B for ACI 349-97 (American Concrete Institute, 1998). The anchor bars are 63.5 mm- [2.5 in-] diameter A-36 steel bars to carry the applied load and have the appropriate stiffness, approximately 28,600 kg/mm [1.9×10^6 lb/in]. The A-36 steel was chosen because it has a well-defined yield plateau, which gives a ductile design strength of 1047 kN [235.63 kips]. The minimum yield strength of the anchor bars is more than 250 percent of the computed demand load, 276.1 kN [62.13 kips]. In addition, the anchorage system was designed so the anchor plates, attached to the bottom of the anchor bars, are of a sufficient size, 30.5 × 30.5 cm [12 x 12 in], to transfer the load by bearing on the concrete. The length of the coupler, 29.8 cm [11.72 in], is controlled by the engagement length for the threads for both the anchor studs and the round bar. The diameter is larger than the corresponding heavy hex bolts used for the round bars and has sufficient capacity to carry the applied loads. The geometry of the embedment plate is controlled by the geometry of the storage cask {inner diameter = 3.30 ± 0.01 m [130 ± 0.25 in], outer diameter = 3.78 ± 0.01 m [149 ± 0.25 in]} and the slope of the storage pad {50.8 mm [2 in] thick}.

To satisfy the requirements of Appendix B of ACI 349-97 (American Concrete Institute, 1998), the concrete breakout strength of anchor in tension and shear and pullout strength of anchor in tension must exceed the anchor bar ductile design strength of 1,625 MPa [235.63 kips]. The applicant has demonstrated the sizing and placement of reinforcing steel within the reinforced concrete storage pad are sufficient to ensure a ductile failure (ENERCON Services Inc., 2003b) The applicant has provided sufficient reinforcing steel to ensure that the failure cone for concrete pullout intersects sufficient rebar to prevent this brittle failure (ENERCON Services Inc., 2003b).

In response to requests for additional information (RAI), the applicant provided a calculation package that determines the shrinkage and thermal stress in the massive concrete storage pad (ENERCON Services Inc., 2003c). The forces and moments, together with the seismic forces and moments (ENERCON Services Inc., 2001b), were used to demonstrate that the design is compliant with the ACI code and to size the pad reinforcement (ENERCON Services Inc., 2003a). The shrinkage and thermal load data were provided to ENERCON Services, Inc. by Pacific Gas and Electric Company (2001a). The temperature data are internal concrete pad temperatures as a function of time that occur as the heat generated by cement hydration is dissipated through the structure to the surrounding environment. The heating transient caused by hydration lasted for 8 days. The loads applied consisted of incremental temperatures at various locations within the pad at the time of concern (ENERCON Services Inc., 2003c, Table 2). The shrinkage strains are a result of the moisture loss and the moisture gradient established in the concrete. The shrinkage data, provided in terms of microstrain through the pad thickness, are converted into temperatures for use in the model. The shrinkage process is very slow, reaching a maximum shrinkage in 117 days. Therefore, the two effects can be considered separately.

The analysis was performed using a three-dimensional ANSYS FEA model. Because the load conditions are symmetrical, one quarter of the rock foundation was modeled. The concrete pad and rock were modeled using eight-noded brick elements. Concrete materials properties were varied in the model, with respect to time, to account for the increase in strength (ENERCON Services Inc., 2003c, Table 5). A 0.6-m [2-ft] gap is provided around the pad to allow for some consideration for construction access. Only the stiff rock foundation, 13,800 MPa [2.0×10^6 psi], is considered because this will result in the maximum restraint of the pad. Constraint equations and contact elements are used to model the contact between the bottom of the pad and the rock foundation. The appropriate boundary conditions were applied to account for using a one-quarter symmetry model.

A check of the temperatures at the specified locations for the various load steps showed they were consistent with the load data (Pacific Gas and Electric Company, 2001a). In addition, a check showed that the overall equilibrium of the storage pad and rock foundation was maintained. The vertical displacements (ENERCON Services,, Inc., 2003c, Table 7) in the pad bottom, 15.3 mm [0.602 in], are significantly greater than for the seismic loads, 2.57 mm [0.101 in], but are still less than the deflections of 65 to 100 mm [2.55 to 3.9 in] (length/320) caused by dead weight. The internal stresses in the pad (ENERCON Services Inc., 2003c, Table 8) are such that the tensile stress is greater than the allowable; therefore, significant reinforcing will be required to prevent cracking. The pad internal forces and moments (ENERCON Services Inc., 2003c, Table 9) were used to size reinforcement and assess the level of expected cracking. These values are significantly greater than those for the seismic analysis and will therefore control the design. The results of the analysis were used in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) to evaluate the concrete according to the design codes and to determine the size of the steel reinforcement in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998).

These static and dynamic analyses confirm the structural adequacy of the reinforced concrete storage pad for supporting the storage casks when subjected to the design loading conditions. From the static and dynamic analyses, pad responses were obtained and then combined to give the maximum response values in accordance with the applicable load combinations. The combined response values were then used to check the structural adequacy of the concrete pad and the soil bearing and sliding stabilities. The structural analysis performed by PG&E demonstrated that the ISFSI storage pads are adequately designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events in accordance with the requirements of 10 CFR §72.122 (b)(1). The structural analysis carried out by the applicant demonstrates that the ISFSI Storage Pads are designed to withstand the effects of natural phenomena, such as earthquakes, without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

For the slab-on-grade design of the storage pads, the tornado winds will not exert any additional load to the structure. Additionally, the ISFSI storage pad will not be subjected to flood load because the storage pads will be above the maximum probable flood level. Moreover, lightning strikes will not affect the safety function of the pad because it is grounded. Therefore, the ISFSI Storage Pads are designed to withstand the effects of natural phenomena, such as tornadoes, lightning, and floods without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

The Diablo Canyon ISFSI concrete storage pads are massive reinforced concrete structures with noncombustible approach surfaces surrounded by an open gravel surface. The gravel surface will be kept free of growth so no combustibles will be present. The staff has reviewed the SAR and determined that the design of the ISFSI storage pads provides adequate fire protection. SAR Section 8.2.5.2, "Fire," indicates that the ISFSI system will not be affected by the postulated combustion of local fuel tanks, combustible materials outside the ISFSI storage pad perimeter or along the transport route, or an unsuppressed vegetation fire. Therefore, the requirements of 10 CFR §72.122(c) are satisfied.

5.1.4 Other Structures, Systems, and Components Important to Safety

This section contains a review of SAR Sections 4.2.1, "Structures;" 4.2.3, "Storage Cask Description;" 4.3, "Transport System;" 4.4, "Operating Systems;" and 8.2, "Accidents." The staff reviewed the discussion of other SSCs important to safety with respect to the applicable regulatory requirements.

5.1.4.1 Description of Other Structures, Systems, and Components Important to Safety

The following structures and components were identified in the SAR as other SSCs important to safety. The staff reviewed the description of SSCs important to safety with respect to the regulatory requirements of 10 CFR §72.24(b), §72.122(f), and §72.122(g).

<u>Fuel Basket and Damaged Fuel Container [Quality Assurance (QA) Category A] and Upper and Lower Fuel Spacer Columns and End Plates (QA Category B)</u>

The Diablo Canyon ISFSI MPC may include the Holtec fuel basket, SAR Section 4.2.3.2.1, "MPC;" damaged fuel container (DFC), SAR Section 4.2.3.2.2, "DFC;" and components of the HI-STORM 100 System. The fuel basket provides support for the fuel assemblies as well as the geometry and fixed neutron absorbers for criticality control. The end plates provide structural support for the fuel basket. The fuel spacer columns maintain the axial position of the fuel assemblies in the MPC basket. The DFC provides a basket to contain damaged fuel that can be placed in one of the positions in the fuel basket. Detailed descriptions of the fuel basket and DFC are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff previously reviewed and found these descriptions acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002a). The fuel basket and DFC structure were sufficiently described in accordance with 10 CFR §72.24(b).

<u>HI-TRAC 125 Transfer Cask (QA Category A)</u>

The HI-TRAC 125 Transfer Cask, as identified in SAR Section 4.2.3.2.4, "HI-TRAC 125 Transfer Cask" is a heavy-walled cylindrical vessel constructed of carbon steel with water for neutron and lead for gamma shielding. The transfer cask provides an internal cylindrical cavity of sufficient size for housing a HI-STORM 100 System MPC. The transfer cask is designed for transient use, to contain the MPC and to be submerged in the spent fuel pool (SFP) to support fuel loading. SAR Figures 4.2.8 to 4.2-10 show the major components of the transfer cask. Table 4.2-3 of the SAR identifies the physical characteristics of the HI-TRAC 125 Transfer Cask. Detailed design descriptions of the transfer cask are given in the HI-STORM 100 System FSAR (Holtec International, 2002).

The description of the transfer cask includes consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the transfer cask allows for control of loads in the event of emergencies. The HI-TRAC 125 Transfer Cask has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

<u>HI-TRAC Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM 100 System Lifting Brackets (QA Category A), Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, Transfer Cask Lift Slings (QA Category B)</u>

As identified in the SAR Sections 4.3.2.2, “Transfer Cask Horizontal Lift Ring;” 4.3.2.3, “Transfer Cask Lift Slings;” and 4.3.2.5 through 4.3.2.9, “HI-TRAC Lift Links;” “MPC Downloader Slings;” “MPC Lift Cleats;” “HI-STORM Lifting Brackets;” and “HI-STORM Lift Links,” the HI-STORM canister transfer equipment includes various lifting devices. The HI-TRAC lift links are attached to the HI-TRAC 125 Transfer Cask lifting trunnions and are used to raise and lower the transfer cask in a single-failure-proof mode. The MPC downloader slings are attached to the MPC and are used to raise and lower the MPC between the HI-TRAC 125 Transfer Cask and the HI-STORM 100SA Overpack in a single-failure-proof mode. The function of the MPC lift cleats is to provide a temporary means to lift the MPC. The function of the HI-STORM 100SA Overpack lifting brackets and lift links is to provide a means of lifting the storage cask. The transporter connection pins, as described in the SAR, Table 4.3-1, connect the transfer cask lift links or the overpack lifting brackets to the cask transporter lift links. The transfer cask horizontal lift rig transmits the load of the lifted transfer cask from the transfer cask lift slings to the cask transporter lift points. The transfer cask lift slings are used to support the weight of the loaded transfer cask and the cask transport frame during horizontal lifting by the cask transporter. Detailed design descriptions of the associated lifting devices are given in the HI-STORM 100 System FSAR (Holtec International, 2002).

Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the associated lifting devices allows for control of loads in the event of emergencies. The associated lifting devices have been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

<u>HI-STORM Mating Device Bolts and Shielding Frame (QA Category A) and HI-STORM Cask Mating Device (QA Category C)</u>

As identified in SAR Section 4.2.3.2.4, “HI-TRAC 125 Transfer Cask,” the cask mating device replaces the transfer lid on the HI-TRAC 125 Transfer Cask. The cask 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 at the CTF. The remainder of the cask mating device facilitates manipulation of the transfer cask bottom lid and is considered QA Category C. A drawing of the mating device is provided in SAR Figure 4.2-11. Design descriptions of the mating device are provided in the HI-STORM 100 System FSAR (Holtec International, 2002).

The description of the entire cask mating device includes consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the cask mating device allows for control of loads in the event of emergencies. The cask mating device has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

Cask Transporter (QA Category A)

SAR Section 4.3.2.1.1, “Description,” identifies the cask transporter as a self-propelled tracked vehicle used to move the transfer cask with loaded MPC from the FHB/AB to the CTF and the loaded storage cask between the CTF and the storage pad. The cask transporter, shown in SAR Figures 4.3-1 to 4.3-3 and Figures RAI 5-11-1 to 5-1-5, is a custom-designed, commercial-grade system that will be qualified upon receipt. The transporter was designed as a mobile single-failure-proof system in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) criteria. The transporter also was designed to preclude tip-over during a design-basis earthquake or impact by a design-basis tornado missile.

The description of the cast transporter includes consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the cask transporter allows for control of loads in the event of emergencies. The cask transporter has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

Lateral Restraints (QA Category A)

As identified in SAR Section 4.2.1.2, “CTF Support Structure,” four lateral ground restraints provide ground-level attachment points for restraint of the cask transporter during canister transfer operations. SAR Figure 4.2-4 shows the general layout of the lateral restraints. PG&E provided details of the position of the lateral restraints and how the maximum loads at each of the restraint locations were determined for design purposes in supporting calculations and responses to the staff’s request for additional information (Pacific Gas and Electric Company, 2001b, 2003). The restraints ensure that the transporter will remain stable and will not topple in the event of a design-basis event during the transfer operation. The staff determined that the information provided in the SAR and in the applicant's letter of December 4, 2003 adequately describes the configuration of the lateral restraints as required in 10 CFR §72.24(b).

HI-STORM 100SA Overpack (QA Category B)

As identified in the SAR Section 4.2.3, "Storage Cask Description," the storage cask is a steel and concrete cylindrical structure that serves as a missile barrier and radiation shield and provides flow paths for natural convective heat transfer and stability for the system. The storage cask is designed to meet ASME Boiler and Pressure Vessel Code, Section III, Subsection NF requirements (ASME International, 1995a). The unreinforced concrete infill material is not considered a structural material. Table 4.2-2 and Figure 4.2-7 of the SAR provide a summary of the physical characteristics of the storage cask. A complete design description of the storage cask system is provided in the HI-STORM 100 System FSAR (Holtec International, 2002).

The staff reviewed SAR Section 4.2.3, "Storage Cask Description," with respect to the description of the storage cask. These descriptions include consideration of inspection, maintenance, and testing. Components requiring inspection and maintenance were identified and operational procedures summarized. Inspection was limited to checks of the air vents to ensure that they are not blocked and assessment of the condition of the anchorage hardware. This design also allows for emergency access. Spacing of the storage casks on the reinforced concrete pads allows for access to critical locations and regions in the event of emergencies.

Additionally, the staff review of SAR Section 4.2.3, "Storage Cask Description," determined that the design features of the storage cask related to shielding and heat removal capability were appropriately described. A comprehensive shielding evaluation is contained in Chapter 7 of this SER. The design of the storage cask places the spent nuclear fuel in a sealed canister to limit the amount of radioactive waste generated at an ISFSI. A comprehensive waste confinement and management evaluation is contained in Chapter 14 of this SER. The HI-STORM 100SA Overpack has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g)

Cask Transfer Facility and Lift Jacks (QA Category B)

As identified in SAR Section 4.2.1.2, "CTF Support Structure," the CTF provides physical protection and shielding of the canisters during transfer from the transportation cask to the storage cask. SAR Figures 4.2-4 and 4.4-3 illustrate the layout of the CTF. The CTF is a cylindrical steel-lined structure embedded in rock and made of reinforced concrete slabs and walls. The reinforced concrete structure portion of the structure is designed in accordance with ACI 349-97 (American Concrete Institute, 1998). The ACI 349-97 code provides the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. The structural steel elements are designed in accordance with ASME Code, Section III Subsection NF (ASME International, 1995a). Components requiring inspection, testing, and maintenance are identified and adequately described in accordance with 10 CFR §72.122(f). Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. The design in accordance with ACI 349-97 and ASME Section III Subsection NF addresses these topics. The design of the CTF allows access to all locations and regions in the event of emergencies in accordance with the requirements of 10 CFR §72.122(g).

The SAR provides a design description of the CTF in sufficient detail to support a detailed review and evaluation. Consequently, the requirements of 10 CFR §72.24(a) and §72.24(b) have been satisfied. The staff, therefore, concludes that the design of the CTF complies with 10 CFR §72.24(c)(4).

Transfer Cask Impact Limiters (QA Category A) and Helium Fill Gas (QA Category B)

The transfer cask impact limiters are only associated with operations conducted in the FHB/AB under 10 CFR Part 50 and are not considered in this SER. Helium fill gas purity will be controlled under Technical Specification 5.1.3, "MPC and SFSC Loading, Unloading and Preparation Program," item (e), to ensure that the gas will perform its design functions of heat transfer and corrosion control.

5.1.4.2 Design Criteria for Other Structures, Systems, and Components Important to Safety

The design bases for the other SSCs important to safety are given in the SAR. Table 4.2-5 of the SAR identifies details of the Diablo Canyon ISFSI compliance with the general design criteria of 10 CFR Part 72, Subpart F. As identified in the SAR, the other SSCs important to safety are designed in accordance with the design criteria contained in Chapter 3 of the SAR. This conclusion is supported by the structural analysis performed as described in Section 5.1.4.4 of this SER. Design criteria have been shown in Chapter 4 of this SER to be representative of the site. A complete discussion of the design criteria for the transfer cask and associated lifting devices is given in Section 4.1.3, "Design Criteria for Structures, Systems, and Components Important to Safety," of this SER. As identified in Chapter 4 of this SER, the site-specific criteria are enveloped by the design criteria identified in the HI-STORM 100 System FSAR (Holtec International, 2002). The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for SSCs important to safety. The design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. The design criteria include the effects of natural phenomena and cover credible fires and explosions.

Fuel Basket, Damaged Fuel Container, Upper and Lower Fuel Spacer Columns, and End Plates

The design criteria for the MPC internals are presented in the HI-STORM 100 System FSAR (Holtec International, 2002) and evaluated in the related staff SER (U.S. Nuclear Regulatory Commission, 2002b). The design criteria are summarized in SAR Table 3.4-2. The MPC internals are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME International, 1995a). Fabrication of the MPC internals is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG–4000 (ASME International, 1995a). Inspection of MPC internals are in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG–5000 (ASME International, 1995a), and Section V (ASME International, 1995b). Exceptions to the ASME and Pressure Vessel Code are provided in the CoC 1014-1 Appendix B Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR Table 3.4-6 (Pacific Gas and Electric Company, 2002a). The lifting bolts of the damaged fuel container are designed in accordance with requirements for a single-failure-proof system identified in ANSI N14.6 (American National Standard Institute/American Nuclear Society, 1993) per the applicable guidelines of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980, Section 5.1.6).

The staff concludes that the design criteria of the MPC internals meet the requirements of the ASME Code, as applicable. The conclusions drawn in this section for other structures, systems, and components important to safety design criteria are based on the evaluation findings made in Section 4.1.3 of this SER. Details are contained within the HI-STORM 100 System FSAR (Holtec International, 2002), which the staff previously reviewed and accepted. The MPC internals design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

<u>HI-TRAC 125 Transfer Cask</u>

The structural steel elements of the HI-TRAC are designed in accordance with ASME Code, Section III, Subsection NF (ASME International, 1995a). The transfer cask is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF Article NF–3300 (ASME International, 1995a). Material procurement is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,f) and Section III, Subsection NF, Article NF–2000 (ASME International, 1995a). Metal components of the transfer cask are fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class 3 (ASME International, 1995b) or AWS D1.1 (American Welding Society, 2002).

Welding of the transfer cask structure will be performed using welders and weld procedures that have been qualified in accordance with the ASME and Pressure Vessel Code, Section IX (ASME International, 1995c) and Section III, Subsection NF (ASME International, 1995a). For nonNF welds, the AWS D1.1 code (American Welding Society, 2002) will be used for welders and weld procedures that have been qualified in accordance with the AWS requirements or in accordance with ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c). All welds require visual examination in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 9 (ASME International, 1995b), with acceptance criteria specified in Section III, Subsection NF, Article NF–5360 (ASME International, 1995a). As specified in engineered drawings 1880 and 2145 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002), the transfer cask structural welds of the outer shell, enclosure shell, and radial ribs will be inspected using liquid penetrant testing in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 6 (ASME International, 1995b) with acceptance criteria specified in Section III, Subsection NF, Article NF–5350 (ASME International, 1995a), or magnetic particle testing in accordance with ASME Boiler and Pressure Vessel Code, Section V, Article 7 (ASME International, 1995b), with acceptance criteria specified in Section III, Subsection NF, Article NF–5340 (ASME International, 1995a). Transfer cask inspection is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Article NF–5360 (ASME International, 1995a) and Section V (ASME International, 1995b). Exceptions to the ASME Boiler and Pressure Vessel Code are provided in the CoC 1014-1, Appendix B, Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR Table 3.4-6 (Pacific Gas and Electric Company, 2002a).

The HI-TRAC 125 Transfer Cask lifting trunnions and lifting trunnion blocks are also designed as special lifting devices in accordance with ANSI N14.6–1993 (American National Standard Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). ANSI N14.6–1993 sets forth the requirements for design, fabrication, testing, maintenance, and QA programs for special lifting devices used to handle containers with radioactive materials.

Details of the design criteria are contained within the HI-STORM 100 System FSAR (Holtec International, 2002), which the staff previously reviewed and accepted. The HI-TRAC 125 Transfer Cask design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

Transfer Cask Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM Lifting Brackets, HI-STORM Lift Links Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, and Transfer Cask Lift Slings

The lifts slings are designed in accordance with ASME B30.9 (ASME International, 2000) according to the guidance of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). All the other components include consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Specifics of the design bases for these components are given in SAR Section 4.2.3.3, “Design Bases and Safety Assurance.” A complete discussion of the design criteria for the transfer cask and associated lifting devices is provided in Section 4.1.3, “Design Criteria for Structures, Systems, and Components Important to Safety” of this SER.

Details are contained within the HI-STORM 100 System FSAR, which the staff previously reviewed and accepted. The associated lift hardware design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

HI-STORM Mating Device, Mating Device Bolts, Mating Device Shielding Frame

Explicit design criteria for the HI-STORM 100 System mating device are not contained within the SAR. Details are contained within the HI-STORM 100 System FSAR, which the staff previously reviewed and accepted.

Cask Transporter

The transporter is custom designed in accordance with the single-failure-proof criteria of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). The functional specification for the cask transporter is identified in Holtec International (2001i). The description of the cast transporter include consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612.

The cask transporter design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

Lateral Restraints

Explicit design criteria for the transporter lateral restraints are based on the seismic induced loads as identified in RAI response 3-3 (Pacific Gas and Electric Company, 2002d) and in subsequent PG&E response to staff's request for additional information (Pacific Gas and Electric Company, 2003). The final design of these elements will meet the stress limits of ASME Section III, Subsection NF (ASME International, 1995a). The attachment points will be designed in accordance with ACI 349-97 (American Concrete Institute, 1998).

HI-STORM 100SA Overpack

Design criteria for the cask systems are contained in the HI-STORM 100 System FSAR (Holtec International, 2002), which has been previously reviewed and accepted by the staff. The generic design base loadings specified in the HI-STORM 100 System FSAR for the anchored cask system envelop the Diablo Canyon ISFSI site parameters. A discussion of the design criteria for the storage cask is provided in Section 4.1.3, "Design Criteria for Structures, Systems, and Components Important to Safety" of this SER.

Applicable ASME Boiler and Pressure Vessel Code criteria for the overpack steel structure are provided in HI-STORM 100 System FSAR Table 2.2.7 (Holtec International, 2002). Material procurement for the overpack steel structure is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,e,f) and Section III, Subsection NF, Article NF–2000 (ASME International, 1995a). The overpack steel structure is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Article NF–3200 (ASME International, 1995a). Metal components of the overpack are fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class 3 (ASME International, 1995a) or AWS D1.1 (American Welding Society, 2002).

Welding of the overpack structure will be performed using welders and weld procedures that have been qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c) and Section III, Subsection NF (ASME International, 1995a). For non-NF welds, the AWS Code will be used for welders and weld procedures that have been qualified in accordance with the AWS D1.1 (American Welding Society, 2002) requirements or in accordance with ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995a). All welds require visual examination in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 9 (ASME International, 1995b), with acceptance criteria specified in Section III, Subsection NF, Article NF–5360 (ASME International, 1995a). As specified in engineered drawing 1495 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002), the overpack inner shell seam weld will be inspected using penetrant testing in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 6 (ASME International, 1995b) with acceptance criteria specified in Section III, Subsection NF, Article NF–5350 (ASME International, 1995a) or magnetic particle testing in accordance with ASME Boiler and Pressure Vessel Code, Section V, Article 7 (ASME International, 1995e), with acceptance criteria specified in Section III, Subsection NF, Article NF–5340 (ASME International, 1995a). Exceptions to the ASME Boiler and Pressure Vessel Code are provided in the CoC 1014-1, Appendix B Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR, Table 3.4-6 (Pacific Gas and Electric Company, 2002a). Overpack steel structure inspection is in accordance with ASME Boiler and Pressure

Vessel Code, Section III, Subsection NF, Articles NF–5350 and NF–5360 (ASME International, 1995a), and Section V (ASME International, 1995b). Material procurement for the overpack anchor studs is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,f) and Section III, Subsection NF, Article NF–2000 (ASME International, 1995a) and the anchor studs are designed in accordance with Subsection NF, Article NF–3300 (ASME International, 1995a).

The unreinforced concrete elements of the HI-STORM 100SA Overpack are designed in accordance with ACI 349-85 (American Concrete Institute, 1985), except for the allowable stress formulas and load combinations, which were determined in accordance with ACI 318-85.

The HI-STORM 100SA Overpack design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

Cask Transfer Facility and Lift Jacks

The design criteria for the CTF establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for this structure important to safety. A summary of the design criteria for the CTF is provided in SAR Table 3.4-5. Additional information is provided in the site-specific design criteria document (Holtec International, 2001d). The structural steel elements of the CTF are designed in accordance with ASME Code, Section III, Subsection NF (ASME International, 1995a). CTF concrete elements are designed in accordance with ultimate strength design methods specified in ACI 349-97 (American Concrete Institute, 1998). A complete discussion of the design criteria applicable to the CTF is provided in Section 4.1.3 of this SER.

The conclusions in this section regarding the CTF design criteria are based on the evaluation findings in Section 4.1.3 of this SER. The CTF design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

5.1.4.3 Material Properties for Other Structures, Systems, and Components Important to Safety

The staff reviewed the material properties for other SSCs important to safety with respect to the applicable regulatory requirements.

Fuel Basket, Damaged Fuel Container, and Upper and Lower Fuel Spacer Columns and End Plates

The applicant provided a description of the HI-STORM 100 System including materials of construction, fabrication details, and testing in SAR Sections 4.2, “Storage System;” 4.7, “Operating Environment Evaluation;” and Appendix A, “Materials” (Pacific Gas and Electric Company, 2002a). The structural components of the fuel basket and DFC are identical to those used for the MPC. Engineering drawings and additional details of the storage system are included by reference to the Holtec HI-STORM 100 System FSAR Sections 3.3 and 3.4 (Holtec International, 2002). Technical specifications for the HI-STORM 100 System are included by reference to 10 CFR Part 72 CoC No. 1014 (U.S. Nuclear Regulatory Commission, 2002a) for the HI-STORM 100 System and License Amendment Request 1014-1 (U.S. Nuclear Regulatory

Commission, 2002a). The HI-STORM 100 System has been evaluated by the staff and approved for use for dry storage of spent nuclear fuel (U.S. Nuclear Regulatory Commission, 2002b).

Mechanical properties of the structural materials for the fuel basket and DFC are provided in the Holtec HI-STORM 100 System FSAR Section 3.3 and Tables 3.3.1, 1.A.1, 3.3.2, 3.3.3, and 3.3.4 (Holtec International, 2002) for the major structural materials including stainless steels, carbon steel, low-alloy steels, nickel-alloy steel and nickel-base alloy. The values in these tables were obtained from ASME Code, Section II, Part D (ASME International, 1995f). The staff independently verified the temperature-dependent values for the stress allowables, ultimate strength, yield strength, modulus of elasticity, and coefficient of thermal expansion. The staff concludes that these material properties are acceptable and appropriate for the expected load conditions during the license period.

The materials of construction for the fuel basket and DFC are readily weldable using commonly available welding techniques. The fuel basket and DFC are constructed from stainless steel materials as specified in the HI-STORM 100 System FSAR Tables 3.3.1 and 1.A.1 (Holtec International, 2002). The fuel basket assemblies for the MPCs are shown in engineered drawings 3925 through 3928 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002). The drawings include standard welding symbols and notations in accordance with AWS Standard A2.4 (American Welding Society, 1998). The MPC baskets and basket supports are fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, and Section V (ASME International, 1995a,b). MPC welding will be performed using welders and weld procedures that have been qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c) and Section III, Subsections NB, and NG (ASME International, 1995a).

Criticality control in the MPC is achieved using a fuel basket structure of edge-welded composite boxes and Boral neutron poison panels (Holtec International, 2002). Boral has a long, proven history in the nuclear industry and has been used in other spent nuclear fuel storage casks. The Boral sheets are enclosed within the welded stainless steel cladding to minimize degradation as a result of environmental exposure.

Chemical reaction between the Boral and the borated water in pressurized water reactor spent nuclear fuel pools may produce small amounts of hydrogen gas during loading and unloading operations (Pacific Gas and Electric Company, 2002a). The safety hazards associated with ignition of this hydrogen gas are mitigated by monitoring for combustible gas concentrations prior to, and during, MPC lid welding operations. The space below the MPC lid will be exhausted or purged with inert gas prior to and during MPC lid welding operations to provide additional assurance that explosive gas mixtures will not develop in this space as indicated in the SAR Section 5.1.1.2, “MPC Loading and Sealing Operations,” and Table 4.7-1 (Pacific Gas and Electric Company, 2002a). In addition, the Boral will be passivated before installation in the fuel basket to minimize the amount of hydrogen released from the aluminum-water reaction to a noncombustible concentration during MPC lid welding or cutting operations. The staff concludes that the guidance in the generic procedures is adequate to prevent ignition of any hydrogen gas that may be generated during loading and MPC closure welding operations.

The staff concludes that the selection of these materials is acceptable for the DFC. The staff concludes that the material selection for the MPC internals meets the requirements of the ASME Codes, as applicable. The MPC internal materials have been identified in accordance with 10 CFR §72.24(c)(3), §72.120(a), §72.122, and §72.236(b).

HI-TRAC 125 Transfer Cask

The transfer cask is constructed from carbon steel, low-alloy steel, nickel-alloy steel, stainless steel, and nickel-base alloy as specified in the HI-STORM 100 System FSAR Table 2.2.6 (Holtec International, 2002). The inner shell, radial channels, enclosure shell panels, water jacket end plate, lower water jacket shell, top lid plates, pool lid plates, lid tongues, fill port caps, and top lid inner and outer rings are constructed from SA516 Grade 70 carbon steel. The top flange is constructed from SA350-LF3 low-alloy steel or SA203E nickel-alloy steel, and the bottom flange is constructed from SA350-LF3 low-alloy steel or SA516 Grade 70 carbon steel. The pool lid outer ring is constructed from either SA516 Grade 70 carbon steel, SA350-LF3 low-alloy steel, or SA203E nickel-alloy steel. Top lid studs are constructed from SA193-B7 low-alloy steel, and the top lid nuts are constructed from SA194-2H carbon steel. The pocket trunnion and the lifting trunnion block are constructed from SA350-LF3 low-alloy steel. The dowel pins are constructed from SA564-630 precipitation hardened stainless steel. Mechanical properties of the transfer cask structure are provided in the HI-STORM 100 System FSAR Tables 3.3.2, 3.3.3, and 3.3.4 (Holtec International, 2002). Applicable ASME Boiler and Pressure Vessel Code (ASME International 1995f) criteria for the transfer cask structure are provided in HI-STORM 100 System FSAR Table 2.2.7 (Holtec International, 2002).

The transfer cask fabrication welds are characterized in engineered drawing 3768, sheets 1-10, in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002). The drawings include standard welding symbols and notations in accordance with AWS Standard A2.4 (American Welding Society, 1998).

Structural steel components of the transfer cask are subject to brittle fracture at low temperatures. The lowest service temperature for the structural components of the transfer cask is specified as −18 °C [0 °F], which is above the ductile-to-brittle transition temperature for the structural steel components. A lowest service temperature of −18 °C [0 °F] is specified for all parts used to lift the transfer cask, which is above the ductile-to-brittle transition temperature for the pocket trunnions, lifting trunnions, and the lifting trunnion block.

The HI-TRAC 125 Transfer Cask uses lead encased between steel plates and water to provide gamma and neutron shielding in the radial direction. Layers of steel-lead-steel in the transfer cask lids provide shielding in the axial direction. The neutron shield material, Holtite-A, is used in the HI-TRAC 125 Transfer Cask lid. The HI-TRAC 125 Transfer Cask top lid also contain Holtite-A to provide gamma attenuation. Holtite-A is a high-hydrogen content, durable, fire-resistant material. A detailed description of Holtite-A is provided in the HI-STORM 100 System FSAR Appendix 1.B (Holtec International, 2002).

Possible chemical, galvanic, and other reactions among the materials in the range of possible exposure environments are evaluated in SAR Section 4.7, “Operating Environment Evaluation,” and Table 4.7-1 (Pacific Gas and Electric Company, 2002a). Steels used in the transfer cask are coated to prevent corrosion during exposure to water during loading operations. Threaded portions of the transfer cask that are not coated will be plugged or covered to prevent corrosion while immersed during loading operations. No adverse reactions were identified for coatings used on the transfer cask, or the elastomer seal used on the transfer cask during loading operations.

Material properties for the HI-TRAC 125 Transfer Cask are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The transfer cask materials have been identified in accordance with 10 CFR §72.24(c)(3).

<u>Transfer Cask Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM Lifting Brackets, HI-STORM Lift Links Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, and Transfer Cask Lift Slings</u>

Materials for the associated lifting devices are not explicitly identified in the SAR. As identified, they will be designed and fabricated in accordance with the applicable codes and standards. These standards identify the acceptable material characteristics. Additional details of the material properties for the associated lifting devices are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The associated lifting devices materials have been identified in accordance with 10 CFR §72.24(c)(3).

<u>HI-STORM Cask Mating Device and HI-STORM Mating Device Bolts and Shielding Frame</u>

Materials for the cask mating device, including bolts and shielding frame, are not explicitly identified in the SAR. As identified, it will be designed and fabricated in accordance with the applicable codes and standards. These standards identify the acceptable material characteristics. Additional details of the material properties for the cask mating device, including bolts and shielding frame, are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The cask mating device, including bolts and shielding frame, materials have been identified in accordance with 10 CFR §72.24(c)(3).

<u>Cask Transporter</u>

Materials for the cask transporter are not explicitly identified in the SAR. This is to be a custom-designed system that will be designed and fabricated in accordance with the applicable codes and standards. These standards identify the acceptable material characteristics. The staff concludes that use of the applicable codes and standards for the materials of construction is in accordance with 10 CFR §72.24(c)(3).

<u>Lateral Restraints</u>

Materials for the lateral restraints are not explicitly identified in the SAR; however, the applicant did provide sufficient design details in their response to the staff's request for additional information (Pacific Gas and Electric Company, 2003). This is to be a custom-designed system that will be designed and fabricated in accordance with the applicable codes and standards (ASME International, 1995a; American Concrete Institute, 1998). These standards identify the acceptable material characteristics. The staff concludes that the proper materials for construction will be selected in accordance with 10 CFR §72.24(c)(3).

HI-STORM 100SA Overpack

The overpack is constructed using carbon steel, low-alloy steel, nickel-alloy steel, and stainless steel as specified in the HI-STORM 100 System FSAR Table 2.2.6 (Holtec International, 2002). The inner and outer cylindrical shells, base plate, and lid are constructed from SA516 Grade 70 carbon steel. Lid studs are constructed from SA564-630 precipitation hardened stainless steel, and the nuts are SA194-2H carbon steel. The bolt anchor blocks are constructed from SA350-LF3 low-alloy steel or SA203E nickel-alloy steel. Material properties for the overpack are provided in the HI-STORM 100 System FSAR, Table 2.2.6 (Holtec International, 2002). Mechanical properties of the overpack structural materials are provided in the HI-STORM 100 System FSAR Revision 1, Tables 3.3.2, 3.3.3, and 3.3.4 (Holtec International, 2002). Structural steel components of the disposal overpack are subject to brittle fracture at low temperatures. The lowest service temperature for the structural components of the overpack is specified as −40 °C [−40 °F], which is above the ductile-to-brittle transition temperature for the structural steel components. A lowest service temperature of −18 °C [0 °F] is specified for all parts used to lift or anchor the overpack. The 0.76-m [30-in] annular space between the inner and outer shells is filled with unreinforced concrete for radiation shielding. Concrete is also used as a shielding material for the bolted lid. Material procurement for the concrete used for shielding meets the requirements of ACI 349, and the design of the overpack concrete meets the requirements of ACI 349-97 (American Concrete Institute, 1998), except for the allowable stress formulas and load combinations, which were determined in accordance with ACI 318-85. The staff concludes that the materials used to construct the disposal overpack are suitable for structural support, shielding, and protection of the MPC from environmental conditions. The staff concludes that the welded joints of the overpack meet the requirements of the ASME International (1995a) and American Welding Society (2002) Codes, as applicable.

All steel surfaces of the overpack are coated with either Thermaline 450, Carbozinc 11, or Carbozinc 11HS as indicated by the Holtec HI-STORM 100 System FSAR, Revision 1, Table 2.2.6 (Holtec International, 2002) and the response to RAI 18-8 (Pacific Gas and Electric Company, 2002c). The coatings are used to protect the steel overpack from oxidation and corrosion. External surface coatings will be maintained as indicated in Diablo Canyon ISFSI SAR Table 4.7-1. During review of the Diablo Canyon ISFSI SAR, staff identified the maximum normal operating temperature of the overpack may be 63 °C [145 °F], exceeding the maximum application temperature for Thermoline 450 {43 °C [110 °F]}. In response to RAI 18-8, the applicant indicated that the actual exterior surface temperatures are expected to be less than the calculated bounding value (Pacific Gas and Electric Company, 2002d). Repair of coatings on the exterior surfaces of the overpack will be made in accordance with the manufacturer's recommendations. Overpack surface temperatures will be checked with a pyrometer prior to coating repair. If necessary, the external surface will be cooled to below the maximum coating application temperature. The specified maximum surface temperature for application of Carbozinc 11 or Carbozinc 11HS is above the maximum normal operating temperature of the overpack (Pacific Gas and Electric Company, 2002d).

Exterior steel surfaces of the transfer cask are coated with Carboguard 890, and interior steel surfaces are coated with Thermaline 450 as indicated by the Holtec HI-STORM 100 System FSAR, Revision 1, Table 2.2.6 (Holtec International, 2002) and the response to RAI 18-8

(Pacific Gas and Electric Company, 2002d). The coatings are used to protect the steel surfaces from oxidation and corrosion. Thermaline 450 is used on the interior for its higher temperature ratings. Carboguard 890 is used on the exterior because of its decontamination characteristics. The coatings are not expected to be affected by short-term exposure to borated spent nuclear fuel pool water or the gamma radiation dose and neutron fluence. The transfer cask will be fully decontaminated, inspected, and recoated as necessary prior to its next use. Small nicks in the coatings are not expected to affect either the coatings or any of the exposed carbon steel.

The coatings are resistant to chemical attack in a variety of chemical environments, resistant to abrasion, and resistant to permeation. The coatings have been successfully used in other licensed nuclear power plants. The staff concludes that the applications of the epoxy paint to the exposed surfaces of the transfer cask and the epoxy paint and zinc coating applied to the carbon steel components of the overpack are acceptable.

The storage overpack uses concrete and steel for shielding in the radial direction and a thick circular concrete slab attached to the lid and a thick circular concrete pedestal to provide gamma and neutron attenuation in the axial direction. Additional steel plates and shell elements are used to provide additional gamma shielding in specific areas.

Material properties for the HI-STORM 100SA Overpack are provided in the HI-STORM 100 System FSAR. The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The overpack materials have been identified in accordance with 10 CFR §72.24(c)(3).

<u>Cask Transfer Facility and Lift Jacks</u>

Materials for the steel components and lift jacks used in the CTF and lift jacks are identified in SAR Figure 4.4-3. Additional details of the material properties are given in the CTF analysis document (Holtec International, 2001h). Structural steel elements will be constructed of SA516, Grade 70, SA36, and WELDOX 130 steel. Material properties for the lift jack screws are identified in Holtec, International (2001h). Concrete material selection is based on concrete with a compressive strength of 20.7 MPa [3,000 psi] at 28 days following Chapter 5 of ACI 349-97 (American Concrete Institute, 1998) as identified in the CTF analysis document (ENERCON Services, Inc,. 2001a). The reinforcing steel is specified to have a minimum yield strength of 414 MPa [60,000 psi] following Chapter 3 of ACI 349-97 (American Concrete Institute, 1998) and American Society for Testing and Materials (1990).

Based on the review of information presented by PG&E, the staff concludes that materials to be used to construct the CTF and lift jacks have been adequately identified in accordance with 10 CFR §72.24 (c)(3). The applicant identified the appropriate codes and standards in accordance with 10 CFR §72.24(c)(4).

5.1.4.4 Structural Analysis for Other Structures, Systems, and Components Important to Safety

The staff reviewed the SAR and found that the structural analysis procedures have been identified and are in conformance with standard engineering practice. Other SSCs important to safety were designed and analyzed to resist the loads and loading combinations specified in the design criteria. The analyses of other SSCs important to safety included loading conditions of dead and live loads, thermal loads, earthquake, tornado, wind, or tornado missiles, and fire, as applicable. The staff reviewed the structural analysis for other SSCs important to safety with respect to the regulatory requirements of 10 CFR §72.24 and §72.122.

Fuel Basket, Damaged Fuel Container, and Upper and Lower Fuel Spacer Columns and End Plates

Structural analyses of the fuel basket, DFC and upper and lower fuel spacer columns and end plates for the HI-STORM 100 System are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). The Diablo Canyon ISFSI will use these same components; therefore, no additional structural review of these components was performed for this SER.

HI-TRAC 125 Transfer Cask

Structural analysis of the HI-TRAC 125 Transfer Cask for the HI-STORM 100 System is provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in Section 3.5.5 of the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). The discussion below is based, in part, on the results presented in the HI-STORM 100 System FSAR and SER and summarized in the Diablo Canyon ISFSI SAR.

The HI-TRAC 125 Transfer Cask is designed to meet ASME Section III Subsection NF (ASME International, 1995a) stress limits for all loading conditions. The structural analysis for the HI-TRAC 125 Transfer Cask in the HI-STORM 100 System FSAR demonstrates that the transfer cask is designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions.

Fire loading conditions of the HI-TRAC 125 Transfer Cask are addressed in Section 11.2.4 of the HI-STORM 100 System FSAR and in Section 8.2.5, “Fire,” of the Diablo Canyon ISFSI SAR. As shown in Section 11.2.4 of the HI-STORM 100 System FSAR, fires near a loaded transfer cask would not produce any structural degradation, and only a small amount of neutron shielding material is lost or damaged. The FSAR indicates that fuel cladding, MPC, and transfer cask temperatures would remain below the design temperature limits.

A site-specific analysis was performed to assess the vulnerability of the transfer cask to site-specific tornado missiles (Holtec International, 2001e). The analysis demonstrated that the HI-TRAC 125 Transfer Cask provides effective missile barriers for the MPC. No missile strike compromises the integrity of the confinement boundary. In addition, global stress intensities arising from the missile satisfy ASME Code Level D limits for an ASME Section III, Subsection NF structure (ASME International, 1995a).

Site-specific evaluations of the risk that explosions could damage the HI-TRAC 125 Transfer Cask were performed (Holtec International, 2001b; Pacific Gas and Electric Company, 2002c). All evaluations concluded that these hazards were not credible.

The structural analysis performed by PG&E demonstrates that the HI-TRAC 125 Transfer Cask is adequately designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events in accordance with the requirements of 10 CFR §72.122(b)(1). Structural analysis carried out by PG&E demonstrates that the HI-TRAC 125 Transfer Cask is designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

Transfer Cask Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM Lifting Brackets, HI-STORM Lift Links Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, and Transfer Cask Lift Slings

Structural analysis of the associated lifting hardware is provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). No additional review was performed for this SER.

The lift links, slings, and rigs are designed as nonredundant lifting devices with a factor of safety of 10 or greater for material ultimate strength and 6 or greater for yield strength. A dynamic load increase factor of 10 percent has been applied to the lifting loads. Therefore, these elements meet the NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) stress limits for nonredundant special lifting devices.

The lift cleats and brackets and connector pins are designed with a minimum factor of safety of 3 for material yield strength and 5 for material ultimate strength, as well as a dynamic load increase factor of 10 percent. Multiple elements are used, and each can totally support the weight of the canister, thereby making them single-failure proof in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980).

HI-STORM Cask Mating Device and HI-STORM Mating Device Bolts and Shielding Frame

Structural analyses of the cask mating device, including bolts and shielding frame, for the HI-STORM 100 System are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). No additional review was performed for this SER.

Cask Transporter

The cask transporter is custom designed for the site-specific criteria (Holtec International, 2001i) in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Structural analysis to be completed by the applicant in accordance with the criteria on NUREG–0612 will demonstrate that the cask transporter is designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, in accordance with the requirements of 10 CFR §72.122(b)(1). The structural analysis will also demonstrate that the cask transporter is designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

The applicant provided site-specific analyses of the stability of the cask transporter (Holtec International, 2001f; ENOVA Engineering Services, 2002). In each of these analyses, the cask transporter is assumed to be rigid within the range of seismic excitation. Nonlinear analysis was performed to determine the extent of motion of the cask transporter in the direction of travel and perpendicular to the direction of travel. The Holtec analysis identified the response because of the site-specific design events using VisualNastran (MSC Software Corporation, 2001). To account for potential amplification of the seismic event because of soils, the ENOVA analysis considered a seismic event twice the design-basis event. Both sets of analysis demonstrated that the cask transporter would not tip-over or slide off the road. Both the analyses assume that the road surface remains stable. The staff reviewed this assumption in SER Section 15.1.2.6, based on the joint probability of the annual exposure probability for transport casks and the annual exceedance probability for the earthquake ground motion. Based on the review, the staff concludes that earthquake induced damage to the casks, while in transit from the power plant to the CTF, is not a credible hazard to the proposed facility.

Lateral Restraints

The applicant identified the basic configuration of the CTF seismic restraints (Pacific Gas and Electric Company, 2001b, 2003). The analysis identified the loads in the restraints as well as the resulting loads at the attachment points to the cask transporter and the rock foundation. These loads included the load of the transporter, 77,110 kg [170 kips]. The weight of either the MPC, 39,920 kg [88 kips] or the HI-STORM 100 System overpack 115,210 kg [254 kips] are not included. For a suspended load, the contribution of the lifted load to the horizontal and longitudinal forces are small (5 to 10 percent), as identified in ASME NOG–1 (ASME International 1995g). The amount of time that the MPC is supported by the cask transporter during the process of lowering the MPC into the storage cask is less than the transport time from the FHB/AB to the CTF. In addition, the amount of time it takes the cask transporter to raise the loaded storage cask the final 1.02 m [40 in] is less than the transport time from the FHB/AB to the CTF. The combined probability of these two events (when the cask transporter is supporting additional weight, combined with a seismic event) is comparable to the combined probability considered in SER Section 15.1.2.6 for cask transporter stability. Therefore, the staff concludes that the joint probability of these events (the annual exposure probability for the case where the cask transporter supports additional weight and the annual exceedance probability for the earthquake ground motion) is so low as to not pose a credible hazard to the proposed ISFSI.

PG&E provided further details on the design and structural analysis of the transporter lateral restraints in its response to additional staff questions (Pacific Gas and Electric Company, 2003). The cask transporter lateral restraint system will be designed to transfer any resultant loads to the ground adjacent to the CTF foundation, to prevent transporter lateral movement relative to the CTF during a postulated seismic event. Based on the design loads calculated by the applicant, the restraints will be steel struts or similar equipment designed to meet the stress limits of ASME Section III, Subsection NF including Appendix F. The surface level, in-ground portion of the restraints will be designed in accordance with ACI 349-97, including draft Appendix B, as clarified in NRC draft Regulatory Guide DG-1098. The staff finds that PG&E has established appropriate design criteria for the cask transporter lateral restraint system, considering the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, in accordance with the requirements of 10 CFR §72.122(b)(1). The applicant's analysis also indicated that the restraints will be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions, in accordance with the requirements of 10 CFR §72.122(b)(2).

HI-STORM 100SA Overpack

Structural analyses of the HI-STORM 100SA Overpack are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff's evaluation of the HI-STORM 100 System FSAR is documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). A limited confirmatory review was performed by the staff for this SER.

The Diablo Canyon ISFSI SAR provides a summary, in SAR Section 4.2.3, "Storage Cask Description," of the analyses performed in the HI-STORM 100 System FSAR. The loading conditions at the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). The following loads and combined loading conditions were considered:

- Dead and Live Loads (Diablo Canyon ISFSI SAR Section 4.2.3.3.2.1)
- Internal and External Pressure Loads (SAR Section 4.2.3.3.2.2)
- Thermal Expansion (SAR Section 4.2.3.3.2.3)
- Handling Loads (SAR Section 4.2.3.3.2.4)
- Overpack/Transfer Cask Tip-Over and Drop (SAR Section 4.2.3.3.2.5)
- Tornado Winds and Missiles (SAR Section 4.2.3.3.2.6)
- Flood (SAR Section 4.2.3.3.2.7)
- Earthquake (SAR Section 4.2.3.3.2.8)
- Explosion Overpressure (SAR Section 4.2.3.3.2.9)
- Fire (SAR Section 4.2.3.3.2.10)
- Lightning (SAR Section 4.2.3.3.2.11)
- 500-kV Line Drop (SAR Section 4.2.3.3.2.12)

The detailed structural analysis of the HI-STORM 100SA Overpack is presented in the HI-STORM 100 System FSAR (Holtec International, 2002). The design of storage casks to mitigate environmental effects is identified, and SAR Section 8.2, "Accidents," demonstrates the capability of SSCs important to safety to withstand postulated accidents and environmental conditions. The staff previously reviewed this structural analysis and found it acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). As documented in that SER, the structural analysis shows that the structural integrity of the HI-STORM 100SA Overpack is maintained under all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the overpack structures for the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code Section III (ASME International, 1995a) for the structure materials.

The applicant did not perform any additional drop or tip-over analysis. Outside the FHB/AB, tip-over of the HI-STORM 100SA Overpack is considered a noncredible accident. When on the ISFSI pad, the system will be anchored. As demonstrated in the HI-STORM 100 System FSAR, an anchored cask will not tip-over, therefore tip-over of the cask is noncredible. During transport, the overpack will be carried by the cask transporter designed to satisfy the single-failure-proof load criteria of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980).

A site-specific analysis was performed to assess the vulnerability of the HI-STORM 100SA Overpack on the storage pad to a transmission tower collapse (Holtec International, 2001a). The transmission tower collapse analysis evaluated the vulnerability of the overpack to impact by components of the transmission tower. The analysis results indicate there would be localized yielding of the overpack in the immediate vicinity of the impact site. Based on the results of the analysis, the material stress away from the vicinity of the impact does not exceed allowable values. Because the yielding is localized, there is no loss of shielding or confinement. In addition, there is no loss of retrievability of a spent nuclear fuel assembly.

A site-specific analysis was performed to assess the vulnerability of the overpack to site-specific tornado missiles (Holtec International, 2001e). The analysis demonstrated that the HI-STORM 100SA Overpack is an effective missile barrier. No missile strike compromises the integrity of the confinement boundary.

A site-specific evaluation of the risks that explosion could damage the HI-STORM 100SA Overpack was performed (Holtec International, 2001b; Pacific Gas and Electric Company, 2002c). All hazards evaluated resulted in a conclusion that these hazards were not credible.

Lightning is addressed in SAR Section 4.2.3.3.2.11, “Lightning.” Site-specific evaluations of the effects of lightning and a 500-kV line break were provided (Holtec International, 2001g). The HI-STORM 100SA Overpack is a large steel/concrete cask that will discharge lightning current through the steel shell of the overpack to the ground. The conductive carbon steel overpack outer shell will provide a direct path to the ground. Since the lightning current will discharge through the overpack, the MPC will be unaffected. The heat buildup in the material will be small and there may be some local spalling of materials. Therefore, the HI-STORM 100SA Overpack design meets the design criteria in Section 3.2.6 of the Diablo Canyon ISFSI SAR for lightning protection.

Based on a review of the PG&E site-specific loads as discussed above, the staff concludes that the Diablo Canyon ISFSI design criteria meet the loading conditions identified for the HI-STORM 100SA Overpack design. A discussion of the cask design relative to the storage requirements of the Diablo Canyon ISFSI is provided in SAR Chapter 4, “ISFSI Design.” The Diablo Canyon ISFSI SAR provides a summary of the analyses performed in the HI-STORM 100 System FSAR (Holtec International, 2002). The loading conditions at the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). Therefore, the staff conclusions in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b), with respect to the structural integrity of the HI-STORM 100SA Overpack, are valid for the Diablo Canyon ISFSI and, therefore, the system meets the Diablo Canyon ISFSI design criteria given in SAR Section 3.3, “Design Criteria for Safety Protection Systems.” The structural analysis has demonstrated compliance with 10 CFR §72.24(a), §72.24(d), §72.122(b), §72.122(c), §72.122(h), §72.128(a), and §72.236(l).

Cask Transfer Facility and Lift Jacks

The staff reviewed Section 4.2.1.2 of the SAR and found that structural analysis of the CTF to mitigate environmental effects has been performed by the applicant. The structural analysis under accident loads is given in SAR Section 8.2. The design criteria for the CTF were identified (Holtec International, 2001d). Additional information used as input into the analysis includes the bearing capacity of the rock surrounding the CTF (Pacific Gas and Electric Company, 2001c). The acceptability of these values is assessed in this SER Section 2.1.6.4.

The structural steel elements of the CTF are designed in accordance with ASME Code Section III Subsection NF (ASME International, 1995a). The analysis of the steel structures to demonstrate compliance with the material allowables was performed using a three-dimensional ANSYS finite element model of the systems (Holtec International, 2001h). This analysis addressed the following major structural elements: main shell, lifting jacks, jack support platform, CTF base support block, and lifting platform. Analyses were performed for lifting and lowering operations, MPC transfer operations, and seismic effects. The analysis evaluated the loads by considering force and moment equilibrium using the bounding values for weights. The appropriate spectral values are used to account for possible amplification of the horizontal accelerations of the stacked components. The stresses in the CTF structural components were evaluated under the combined action of the dead loads and the design basis seismic loads and then compared to the Level B allowables (ASME International, 1995a). The stresses in the CTF structural components were also evaluated under the combined action of the three Level D design-basis seismic loads and then compared to the Level D allowables. It was demonstrated that the factors of safety for all components and all load conditions are greater than 1.0. The adequacy of the structures has been demonstrated by the analysis results given in the SAR, as designed to satisfy the requirements of ASME Section III Subsection NF (ASME International, 1995a).

Loads from this analysis are also used in calculating the necessary thickness and reinforcement for the CTF concrete (ENERCON Services Inc., 2001a). The analysis determined the required size and general reinforcing requirements to resist the loads applied to the concrete structure. Additional specific reinforcing requirements will be developed during the course of preparing the construction drawings to address issues related to discontinuities, embedments, and cutouts in the concrete wall and extensions. The concrete structure is designed to withstand loads from both the CTF and the transporter. The only configuration considered is with the HI-STORM 100SA Overpack located at the bottom of the CTF. Using the controlling load combinations, finite element analyses carried out by Holtec identified the design loads. Steel reinforcement size and placement for the pad and wall were established based on these demands. The design of the concrete structure and its reinforcement are based on the requirements in ACI 349-97 (American Concrete Institute, 1998). The procedures for selection of the reinforcement and checks for axial, shear, moment, and torsional resistance of the elements are in conformance with standard engineering practice, as described in ACI 349-97. Results of the analysis indicate that the available design strength exceeds that required for the factored design loads.

The structural analysis performed by the applicant demonstrates that the structural elements of the CTF are designed to resist the seismic loads based on the site characteristics and environmental conditions in accordance with the requirements of 10 CFR §72.122(b)(1) and §72.122(b)(2). The PG&E analysis of the stability of the subsurface materials under the CTF and the potential for failure along the clay bed and movement of the CTF is reviewed in this SER Section 2.1.6.4, “Stability of Subsurface Materials.” The slope stability of the material above the CTF and the potential for it to encroach on the facility are reviewed in this SER Section 2.1.6.5, “Slope Stability.”

5.1.5 Other Structures, Systems, and Components Not Important to Safety

This section describes the design, design criteria, and design analysis for other SSCs not important to safety. There are no specific requirements identified in 10 CFR Part 72 for other SSCs not important to safety. Section 5.4.5, “Other SSCs,” of NUREG–1567 (U.S. Nuclear Regulatory Commission, 1998) cites the regulatory requirements that are applicable to other SSCs subject to NRC approval.

5.1.5.1 Description of Other Structures, Systems, and Components Not Important to Safety

As identified in SAR Section 4.5.6, the following SSCs are considered:

- Security Systems (Section 4.2.2)
- Fencing (Section 4.2.2)
- Lighting (Section 4.2.2)
- Electrical Power (Section 4.4.4)
- Communications Systems (Section 4.2.2)
- Cask Transport Frame (Section 4.3.2.4)
- CTF Drive and Control Systems (Sections 4.2.1.2 and 4.4.5)

Descriptions of the other SSCs are given in the SAR sections identified necessary to satisfy the requirement of 10 CFR §72.24(a) and §72.24(b). They are limited to a general description of the various systems. The majority of these systems will be based on commercially available systems that are designed, fabricated, constructed, tested, and maintained in accordance with approved engineering practices.

The HI-STORM 100 System is a completely passive system, and no electrical power is required to ensure safe, long-term storage of the spent nuclear fuel. The cask transport frame is not within the direct load path for handling the HI-TRAC 125 Transfer Cask and, therefore, is not considered important to safety. The CTF drive and control systems will be controlled manually during operations at the CTF. The CTF structural system would fail in place, and active control is not necessary to maintain public health and safety.

The following SSCs not important to safety are described in the HI-STORM 100 FSAR (Holtec International, 2002) and were approved by the staff in its SER for Amendment 1 to the HI-STORM 100 CoC (U.S. Nuclear Regulatory Commission, 2002b). The Diablo Canyon ISFSI will use these same SSCs; therefore, these items are not discussed further in this SER.

- Automated Welding System
- MPC Helium Backfill System
- MPC Forced Helium Dehydration System
- MPC Vacuum Drying System

5.1.5.2 Design Criteria for Other Structures, Systems, and Components Not Important to Safety

The design criteria for the various other SSCs not important to safety have been identified in the Diablo Canyon ISFSI SAR. Table 4.1-1 of the SAR identifies details of the Diablo Canyon ISFSI compliance with the general design criteria of 10 CFR Part 72 Subpart F. The design criteria identified for other SSCs are based on commonly used codes and standards. The design of the other SSCs not important to safety permits inspection, maintenance, and testing. The inspection, maintenance, and testing requirements are based on the appropriate codes and standards. This design also allows for emergency capability. The layout of the Diablo Canyon ISFSI allows areas to be reached in the event of an accident.

5.1.5.3 Material Properties for Other Structures, Systems, and Components Not Important to Safety

No specific material properties are identified in the SAR for the other SSCs not important to safety. PG&E will use materials that satisfy the code or standards used for the SSCs as required, and therefore, the requirement of 10 CFR §72.24(c)(3) is met.

5.1.5.4 Structural Analysis for Other Structures, Systems, and Components Not Important to Safety

Other SSCs not important to safety will be designed based on standard engineering practices that are in accordance with the applicable codes and standards. In most cases, other SSCs that are not important to safety are commercially available, and their design to standard industrial requirements is acceptable. This demonstrates compliance with the requirement of 10 CFR §72.24(d) and §72.24(i) and the applicable section of 10 CFR §72.122.

5.2 Evaluation Findings

The staff made the following determinations, based on its review of the Diablo Canyon ISFSI SAR and supporting documents:

- Information regarding HI-STORM 100 System specific structures, systems, and components important to safety is in the HI-STORM 100 System FSAR. The NRC approval of the HI-STORM 100 System is documented in CoC 1014 and the related HI-STORM 100 System SER.

- The Diablo Canyon ISFSI SAR, including the materials incorporated by reference, adequately describes the materials used for the structures, systems, and components important to safety and the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness.

- There will not be a pool or pool confinement system at the proposed Diablo Canyon ISFSI.

- The Diablo Canyon ISFSI SAR, as supplemented, adequately describes all structures, systems, and components that are important to safety, and provides drawings and text in sufficient detail to allow evaluation of their structural effectiveness to meet the requirements of 10 CFR §72.24(b) and §72.24(c). The structural analysis procedures used by PG&E have been identified. The relationship between the design basis and the design criteria is identified. The materials of construction are identified. The applicable codes and standards used in the analysis of the reinforced concrete structures are established.

- The structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety functions to be performed. The structures, systems, and components important to safety are classified based on their primary function and importance to overall safety. Therefore, the requirements of 10 CFR §72.122(a) are satisfied.

- The applicant has met the requirements of 10 CFR §72.122(a). The material properties of structures, systems, and components important to safety conform to quality standards commensurate with their safety function.

- The applicant has met the requirements of 10 CFR §72.104(a), §72.106(a), §72.124, and §72.128(a)(2). Materials used for criticality control and shielding are adequately designed and specified to perform their intended functions.

- The applicant has met the requirements of 10 CFR §72.122(h)(1) and §72.236(h). The design of the dry cask storage system and the selection of materials adequately protect the spent nuclear fuel cladding against degradation that might otherwise lead to gross rupture of the cladding.

- The structures, systems, and components important to safety are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events with an adequate margin of safety. The structural analysis performed by PG&E demonstrates that structures, systems, and components important to safety are designed to resist the loads based on the site characteristics and environmental conditions under normal operations and under postulated off-normal and accident events. The PG&E structural analysis demonstrates that structures, systems, and components important to safety are designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions. Stresses at critical locations of structures, systems, and components for bounding design loads are determined by analysis. The section properties are adjusted to ensure that the capacity of all structural elements at all locations exceeds the demand. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and found to be within the limits of applicable codes, standards, and specifications. The loads on the MPC and fuel assemblies remain bounded by the loads considered in the HI-STORM 100 System FSAR, and previously accepted by the staff in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). Therefore, the requirements of 10 CFR §72.122(b)(1) and §72.122(b)(2) are satisfied.

- PG&E specified that the reinforced concrete storage pads are designed in accordance with ACI 349-97 (American Concrete Institute, 1998). The final design that is covered in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) demonstrates compliance with the applicable codes and standards. Structural analysis completed by PG&E demonstrated that the reinforced concrete storage pads are designed to resist the loads based on the site characteristics and environmental conditions during normal operations and postulated off-normal and accident events in accordance with the requirements of 10 CFR §72.122(b)(1).

- The descriptions of structures, systems, and components important to safety include consideration of inspection, maintenance, and testing. Components requiring inspection and maintenance are identified, and operational procedures are summarized. Therefore, the requirements of 10 CFR §72.122(f) are satisfied.

- This design also allows for emergency capabilities because access to critical locations and regions in the event of emergencies is possible. In addition, the lifting components are designed to hold the load in the event of emergencies. Therefore, the requirements of 10 CFR §72.122(g) are satisfied.

- The applicant has met the requirements of 10 CFR §72.236(h) and §72.236(m). The material properties of structures, systems, and components important to safety will be maintained during normal, off-normal, and accident conditions of operation so the spent nuclear fuel can be readily retrieved for further processing or disposal without posing operational safety problems.

- The applicant has met the requirements of 10 CFR §72.236(g). The material properties of structures, systems, and components important to safety will be maintained during all conditions of operation so the spent nuclear fuel can be safely stored for a minimum of 20 years and maintenance can be conducted as required.

- The applicant has met the requirements of 10 CFR §72.236(h). The HI-STORM 100 System employs materials that are compatible with wet and dry spent nuclear fuel loading and unloading operations and facilities. These materials should not degrade over time or react with one another during any conditions of storage.

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