TABLE 4.3-1

IMPORTANT-TO-SAFETY COMPONENTS OF THE CASK TRANSPORTATION SYSTEM

Component	Function	Applicable Design Codes
Cask Transporter	Lift, handle, and transport a loaded HI-TRAC transfer cask or a HI-STORM 100SA overpack	Purchased commercial grade and tested prior to use in accordance with NUREG-0612
Transfer Cask Horizontal Lift Rig	Transmit the force of the lifted load to the cask transporter lift points from the transfer cask lift slings	ANSI N14.6 per NUREG-0612, Section 5.1.6
Transfer Cask Lift Slings	Transmit the force of the lifted transfer cask to the transfer cask horizontal lift rig devices during horizontal lifts.	ASME B30.9 Purchased commercial grade and tested prior to use in accordance with NUREG-0612
HI-TRAC Lift Links	Transmit the force of the lifted load from the transfer cask lifting trunnions to the cask transporter lift points during upending and vertical lifts	ANSI N14.6 per NUREG-0612, Section 5.1.6
MPC Downloader Slings	Transmit the force of the loaded MPC from the MPC lift cleats to the MPC downloader	ASME B30.9 Purchased commercial grade and tested prior to use in accordance with NUREG-0612
MPC Lift Cleats	Provide a lift point for raising and lowering the loaded MPC between the transfer cask and overpack	ANSI N14.6 per NUREG-0612, Section 5.1.6
HI-STORM Lifting Brackets	Transmit the force of the lifted load from the overpack lid studs to the cask transporter lift points during vertical lifts	ANSI N14.6 per NUREG-0612, Section 5.1.6
Connector Pins	Connect the transfer cask lift links or the overpack lifting brackets to the cask transporter lift links	ANSI N14.6 per NUREG-0612, Section 5.1.6
HI-STORM Lift Links	Transfer the force of the lifted load from the HI-STORM lifting brackets to the cask transporter lift points during vertical lifts under off-normal or accident conditions with a loaded overpack in the CTF.	ANSI N14.6 per NUREG-0612 Section 5.1.6

TABLE 4.5-1

QUALITY ASSURANCE CLASSIFICATION OF
MAJOR STRUCTURES, SYSTEMS, AND COMPONENTS

IMPORTANT TO SAFETY[a]	NOT IMPORTANT TO SAFETY
Classification Category A	
Multi-Purpose Canister	Security Systems
Fuel Basket	Fencing
Damaged Fuel Container	Lighting
Transfer Cask	Electrical Power
MPC Lift Cleats	Communications Systems
MPC Downloader Slings[b]	Automated Welding System (AWS)
Transfer Cask Impact Limiters	MPC Helium Backfill System
HI-STORM Lifting Brackets	MPC Forced Helium Dehydration System
HI-STORM Mating Device Bolts and Shielding Frame	MPC Vacuum Drying System
Cask Transporter[b]	Cask Transport Frame
Lateral Restraints[b] (HI-TRAC and transporter at CTF)	CTF Drive and Control Systems
HI-STORM Lift Links	
Transfer Cask Lift Links	
Classification Category B	
HI-STORM Overpack	
ISFSI Storage Pads	
Overpack Anchorage Hardware	
CTF (except jacks)	
CTF Jacks[b]	
Transfer Cask Horizontal Lift Rig	
Transfer Cask Lift Slings[b]	
Upper and Lower Fuel Spacer Columns and End Plates	
Transporter Connector Pins	
Helium Fill Gas[b]	
Classification Category C	
HI-STORM Cask Mating Device (except bolts and shielding frame)	

(a) Major cask system components are listed according to the highest QA category of any subcomponent comprising the major component. The safety classification of the subcomponents and the determination of the ITS category of each item is administratively controlled by PG&E via design and procurement control procedures with input from the storage cask vendor.

(b) Purchased commercial grade and qualified by testing prior to use.

TABLE 4.7-1

HI-STORM 100 SYSTEM MATERIALS SUMMARY

Material/Component	Fuel Pool (Borated Water)[a]	ISFSI Pad and CTF (Open to Environment)
Alloy X: - MPC fuel basket - MPC baseplate - MPC shell - MPC lid - MPC fuel spacers	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions or interactions with spent fuel.	The MPC internal environment will be an inert (helium) atmosphere and the external surface will be exposed to ambient air.
Boral - Neutron absorber in MPC fuel basket	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 non-combustible concentration during MPC lid welding or cutting operations. See Chapter 5 for additional requirements for combustible gas monitoring and actions for control of combustible gas accumulation under the MPC lid.	The MPC internal environment will be an inert (helium) atmosphere.
Steels (Transfer Cask): - SA350-LF2 - SA350-LF3 - SA203-E - SA515 Grade 70 - SA516 Grade 70 - SA193 Grade B7 - SA 106	All exposed steel surfaces (except seal areas, and lifting trunnions) will be coated with material specifically selected for performance in the operating environments. Lid bolts are plated and the threaded portion of the bolt holes are plugged or otherwise covered to seal the threaded area from exposure to borated water.	Exposed surfaces of the HI-TRAC transfer cask will be coated and maintained between uses.
Steels (Overpack): - SA516 Grade 70 - SA203-E - SA350-LF3	HI-STORM 100 storage overpack is not exposed to fuel pool environment.	Internal and external carbon steel surfaces will be coated (except for threaded bolts and holes). Accessible external surfaces, including cask anchor studs and, will be maintained with a fully coated nuts surface.

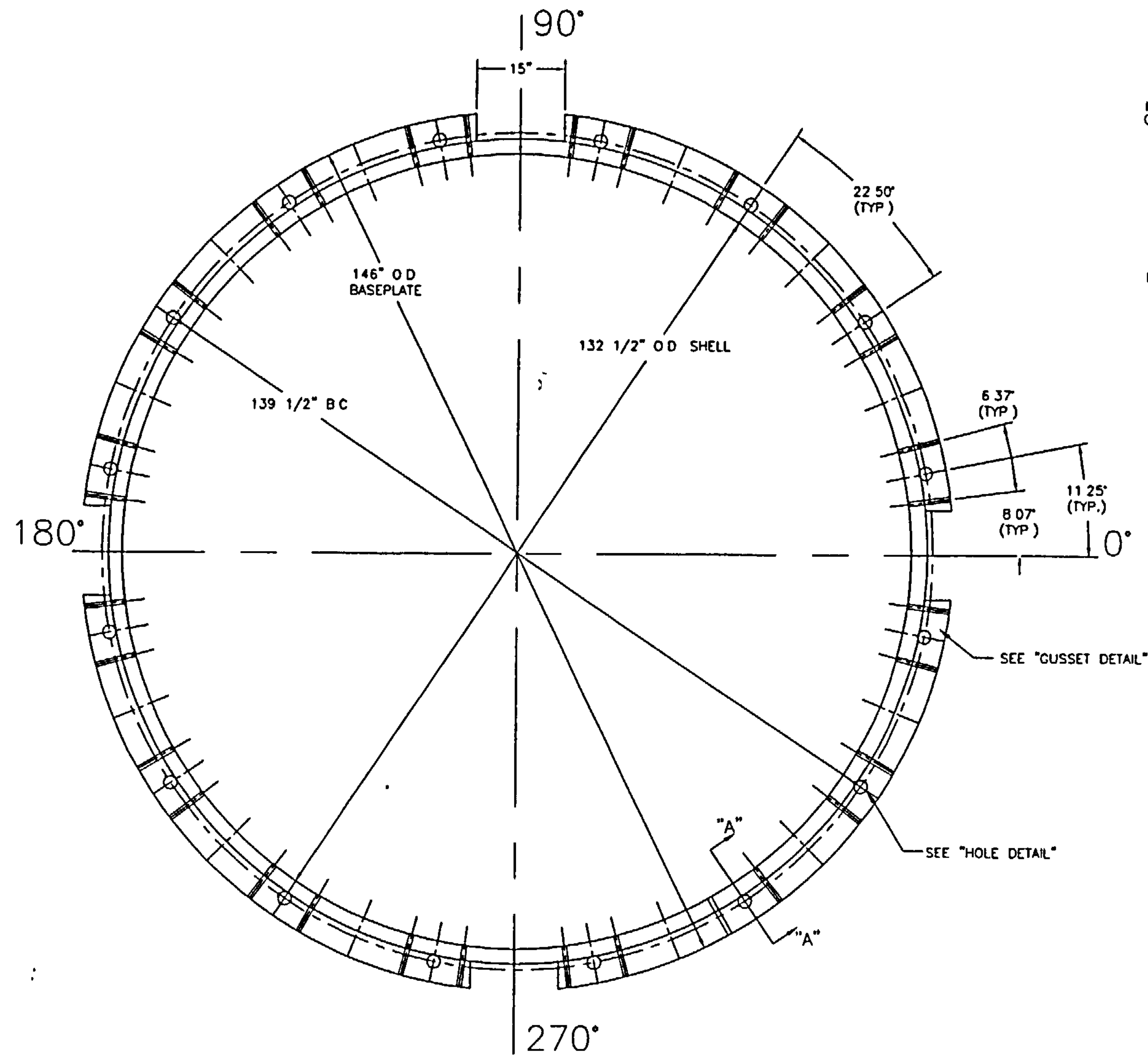
TABLE 4.7-1

Material/Component	Fuel Pool (Borated Water)[a]	ISFSI Pad and CTF (Open to Environment)
Stainless Steels (Misc.): SA240 304 MPC Fuel Spacer SA193 Grade B8 MPC Upper Fuel Spacer Bolt 18-8 S/S Transfer Cask Lid Washers	Stainless steels have been extensively used in spent fuel storage pools with borated and unborated water with no adverse reactions.	Stainless steel has a long proven history of corrosion resistance when exposed to the atmosphere. These materials are used for washers.
Nickel Alloy: SB637-NO7718 Transfer Cask Lifting Trunnions	No adverse reactions with borated water.	Short-term exposure to saline air environment.
Brass/Bronze: Transfer cask water jacket Pressure relief valve	Small surface of pressure relief valve will be exposed. No significant adverse impact identified.	Short-term exposure to saline air environment. Normal maintenance assures operability of valves.
Holtite-A: Solid neutron shield in transfer cask lid and cask transport frame bottom shield structure	The neutron shield is fully enclosed in the top lid and cask transport frame bottom shield structural steel. Neither the transfer cask top lid nor the cask transport frame are immersed in the spent fuel pool.	The neutron shield is fully enclosed in the top lid and cask transport frame bottom shield structural steel. Therefore, Holtite is not exposed to the environment.

TABLE 4.7-1

Material/Component	Fuel Pool (Borated Water)[a]	ISFSI Pad and CTF (Open to Environment)
Coatings: Carboline 890 Thermaline 450 Carbozinc 11/11HS Carboline 891 Bar-Rust 235 Exterior transfer cask and overpack carbon steel surface coatings	Carboline 890 used for HI-TRAC transfer cask surfaces other than the inner shell for good decontamination properties and acceptable temperature resistance for the application. Acceptable performance for short-term exposure in borated pool water. Thermaline 450 selected for HI-TRAC transfer cask inner shell surfaces for excellent high temperature resistance properties. Will be exposed to borated water during in-pool operations as annulus is filled with clean borated water prior to placement in the spent fuel pool, and the inflatable seal prevents contaminated fuel pool water in-leakage. Manufacturer's data confirms that these coatings will perform adequately in these environments.	Coating products are used in a variety of corrosive external environments, including chemical industry. Good for resistance to oceanside saline environment. Thermaline 450 or Carbozinc 11/11HS used for HI-STORM overpack surfaces exposed to the environment for high temperature resistance and weathering capability. Includes exposed portions of cask anchorage (e.g., bottom flange). Manufacturer's data confirms that these coatings will perform adequately in these environments. The Carboline 891 or Bar-Rust 235 will be applied to the exposed portions of the storage cask anchor studs, washers and nuts.
Elastomer Seals: Transfer cask pool	Gasket is compressed between pool lid and transfer cask bottom flange to prevent spent fuel pool water inleakage. Gasket will be inspected periodically and replaced as necessary.	Leakage prevention function not required after the transfer cask is removed from the spent fuel pool and the annulus is drained.
Lead: Transfer cask body and lid gamma shield	Enclosed by carbon steel in transfer cask body and pool lid. Lead is not exposed to spent fuel pool water.	Enclosed by carbon steel in transfer cask body and lid. Lead is not exposed to ambient environment.
Concrete: Overpack body, lid, and pedestal shield	Storage overpack is not exposed to fuel pool water.	Concrete is enclosed by carbon steel in lid, pedestal, and overpack body in final storage configuration and not exposed to ambient environment.

[a] HI-TRAC/MPC short-term operating environment during loading and unloading.



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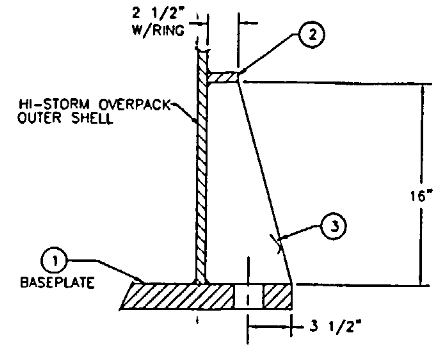

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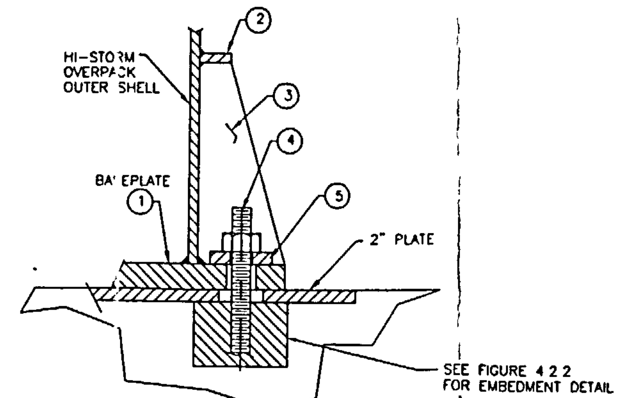
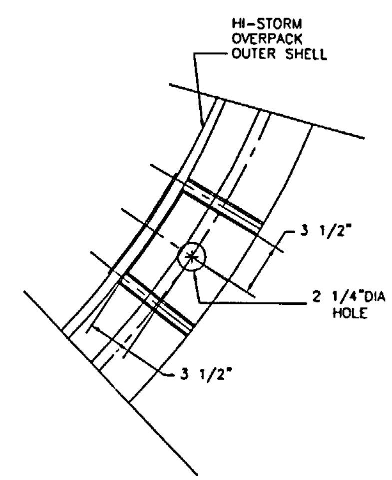


SECTION "A"–"A"



HOLE DETAIL

NOTES:
1. ALL DIMENSIONS ARE NOMINAL
2. THIS DWG. IS NOT TO SCALE
3. CHANGES TO THIS DRAWING MAY AFFECT DWG 3570

BILL OF MATERIAL

ITEM NO.	QTY.	MATERIAL	NOMENCLATURE
1	1	SA 516 GRADE 70	BASEPLATE
2	1	SA 516 GRADE 70	LUG SUPPT. RING
3	32	SA 516 GRADE 70	GUSSET
4	16	SA 193 B7	STUD WITH NUT
5	16	SA 516 GRADE 70	WASHER

FIGURE 4.2-6
Amendment 1 October 2002

REV	SUMMARY OF CHANGES/ AFFECTED ECO'S	PREP BY	APPROVAL DATE	VIR #
2	ADDED NOTE TO SECTION "A" CLARIFIED NOTE 3	JDA	9-26 2002	96268
1	REMOVED WELD DETAILS	JDA	12-07 2001	85580
0	FOR APPROVAL ORIGINAL ISSUE	JDA	11-06 2001	73908

THE DRAWINGS MUST BE USED IN CONJUNCTION WITH THE APPLICABLE DESIGN DRAWING PACKAGE COVER SHEET WHICH CONTAINS THE APPROPRIATE REVIEWS AND APPROVALS

UNLESS OTHERWISE SPECIFIED: ALL DIMENSIONS ARE IN INCHES. DIMENSIONS IN [ ] ARE IN MILLIMETERS

TOLERANCES

REMOVE ALL BURRS AND BREAK SHARP EDGES

– DO NOT SCALE OFF OF DRAWING –

PROJECT NO. 1073

P.O. NO. —

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT: DIABLO CANYON ANCHORED CASK HI-STORM 100SA

DESCRIPTION: LICENSING DRAWING FOR CASK ANCHOR STUD AND SECTOR LUG ARRANGMENT

SIZE: D DRAWING NO. 3769 SHEET 1 REV 2

CHAPTER 5

ISFSI OPERATIONS

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ISFSI OPERATIONS

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ISFSI OPERATIONS

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CHAPTER 5

ISFSI OPERATIONS

This chapter describes the operations associated with the Diablo Canyon ISFSI. Fuel handling and cask loading operations in the DCPP fuel handling building/auxiliary building (FHB/AB) will be performed in accordance with the DCPP 10 CFR 50 license. Transfer and storage activities associated with the ISFSI will be performed in accordance with the 10 CFR 72 Diablo Canyon ISFSI license. As indicated in previous chapters, the Diablo Canyon ISFSI, in its final storage configuration, is a totally passive installation. Periodic surveillance is required, by the Diablo Canyon ISFSI Technical Specifications (TS), to ensure the passive air-cooling system is properly operating. Maintenance is limited to minor, touch-up painting of the HI-STORM 100SA overpack and anchorage hardware. The operations described in this chapter relate to the loading and preparation of the multi-purpose canisters (MPCs), transport to the cask transfer facility (CTF) in the HI-TRAC transfer cask, transfer of the MPC from the transfer cask to the overpack at the CTF, and transport of the loaded overpack from the CTF to the ISFSI storage site. Also described is the process for off-normal event recovery, including unloading of fuel from a loaded overpack. An overview of activities occurring in the DCPP FHB/AB is provided. A detailed discussion of these activities is provided in the 10 CFR 50 license amendment request.

5.1 OPERATION DESCRIPTION

The methods and sequences described below provide an overview of the operational controls that the personnel performing spent fuel loading, cask transfer, and storage activities will implement to ensure safe, reliable, long-term spent fuel storage at the ISFSI storage site. Site-specific procedures will be used to implement these activities, including the use of existing procedures, revision of existing procedures, or the creation of new procedures. The specific number, wording, and sequence of site procedural steps may vary from the guidance provided here as long as the steps comply with assumptions and inputs in the governing, design-basis analyses.

Operations to load and place the HI-STORM 100 System at the storage location on the ISFSI pad will be performed both inside and outside the DCPP FHB/AB. MPC fuel loading and handling operations will be performed inside the FHB/AB using existing DCPP systems and equipment for heavy lifts, radiation monitoring, decontamination, and auxiliary support, augmented as necessary by ancillary equipment specifically designed for these functions. The implementing procedures will incorporate applicable 10 CFR 50 license conditions and commitments, such as those governing heavy loads. MPC transfer into the overpack at the CTF and movement of the loaded overpack to the storage location will be performed using procedures developed specifically for these operations.

5.1.1 NARRATIVE DESCRIPTION

The following discussion describes the specifics of the integrated operation, including fuel loading, MPC closure operations, transfer cask handling, overpack handling, and ISFSI pad placement. As described in the HI-STORM 100 System FSAR (Reference 1), as amended by Holtec License Amendment Request (LAR) 1014-1 (Reference 2), the MPC is loaded in a reusable HI-TRAC transfer cask in the spent fuel pool (SFP). The MPC is welded and prepared for storage while in the FHB/AB. The MPC and transfer cask are then transported to the CTF, located adjacent to the ISFSI storage site, where the MPC is transferred into an overpack for storage on the ISFSI pads. Section 5.1.1.1 describes loading operations for damaged fuel and fuel debris. Section 5.1.1.2 describes MPC loading and sealing operations. Section 5.1.1.3 describes the operations for transferring the loaded MPC to the ISFSI storage site and into the overpack for storage. Section 5.1.1.4 describes off-normal event recovery operations.

Specific procedures will identify and control the selection of fuel assemblies, and nonfuel hardware for loading into the HI-STORM 100 System. Candidate fuel assemblies will be selected based on their physical characteristics (for example, dimensions, enrichment, and uranium mass) to ensure they meet the requirements of the Diablo Canyon ISFSI TS and SAR Section 10.2. The selected fuel assemblies then will be classified as intact fuel, damaged fuel, or fuel debris, in accordance with the definitions in SAR Section 10.2. Once an assembly is found to be physically within the limits of the SAR Section 10.2 and correctly classified, the burnup, cooling time, and decay heat of the assemblies will be confirmed to be within SAR Section 10.2 limits using existing records. If any selected assemblies include nonfuel hardware, the particular type of nonfuel hardware also will be confirmed to meet SAR Section 10.2.

Fuel assemblies chosen for loading will be assigned a specific storage location in the MPC in accordance with the Diablo Canyon ISFSI TS and SAR Section 10.2. Criteria such as the classification of the assembly (that is, intact, damaged, or debris), the presence of nonfuel hardware in the assembly, and the use of a uniform or regionalized storage strategy (burnup, cooling time, decay heat) as defined in SAR Section 10.2 are used to determine the acceptable fuel storage locations for each assembly. Records will be kept that track the fuel assembly, and nonfuel hardware and its assigned MPC and specific fuel storage location. Videotape (or other visual record) will be used during fuel loading operations in the SFP to record fuel assembly and associated nonfuel hardware serial numbers and to provide an independent record of the MPC inventory.

Once the fuel inventory for an MPC is identified, the "time-to-boil" for that MPC is calculated based on the total decay heat rate of the fuel and the temperature of the SFP at the time of loading. This calculation establishes the time duration within which MPC sealing operations must reach the point where draining of the water in the MPC is complete and boiling of the water in the MPC is avoided. The commencement for time-to-boil starts when the MPC lid is installed in the SFP, effectively segregating the fuel in the MPC from the cooling provided by the SFP cooling system. The time-to-boil may be determined on an MPC-specific basis or a bounding time may be determined for a group of MPCs to be loaded, using a worst-case fuel decay heat value and initial water temperature. The methodology described in Section 4.5.1.1.5 of the HI-STORM 100 System FSAR shall be used to determine the time-to-boil.

Additional administrative controls will be used, as necessary, to govern the placement and use of impact limiters, special load-handling devices, allowable travel paths, and lift heights, both inside and outside of the FHB/AB, to ensure compliance with the DCPP and Diablo Canyon ISFSI licensing and design bases, as applicable.

The loading, unloading, and handling operations described in this section have been developed based on the Holtec International field experience in loading HI-STAR 100 dry cask storage systems at other ISFSIs. The equipment and operations used at these sites have been evaluated and modified, as necessary, based on this experience to reduce occupational exposures and further minimize the likelihood of human error in performing the activities needed to successfully deploy the HI-STORM 100 System at the Diablo Canyon ISFSI.

5.1.1.1 Damaged Fuel and Fuel Debris Loading

Damaged fuel containers (DFCs) are used to house damaged fuel assemblies and fuel debris in the MPC in accordance with the requirements of the Diablo Canyon ISFSI TS and SAR Section 10.2. Any qualified fuel assembly that is classified as damaged fuel may be loaded into an MPC-24E. Up to a total of four DFCs containing damaged fuel may be stored in an MPC-24E, with the balance being intact fuel assemblies. Fuel classified as fuel debris must be stored in a DFC and must be loaded into an MPC-24EF. The MPC-24EF may also be used to store damaged fuel. Up to a total of four DFCs containing either damaged fuel or fuel debris may be stored in the MPC-24EF, with the balance being intact fuel assemblies. The fuel assembly is placed in the DFC either before or after the DFC is placed into the MPC. Storage of damaged fuel and fuel debris in the HI-STORM 100 System is discussed, and the containers analyzed, in Section 2.1.3 and Appendix 3.AS, respectively, of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. Figure 2.1.2B in the HI-STORM 100 System FSAR, as amended by LAR 1014-1, shows the Holtec pressurized water reactor (PWR) DFC.

5.1.1.2 MPC Loading and Sealing Operations

This section describes the general sequence of operations to load and seal the MPC, including the movement of the transfer cask within the FHB/AB. Site-specific procedures will control the performance of the operations, including inspection and testing. At a minimum, these procedures will control the performance of activities and alert operators to changes in radiological conditions around the cask. As described in this section, several operational sequences have important time limitations including time-to-boil following MPC lid attachment, and evacuation and helium backfill time. These sequences are controlled by Diablo Canyon ISFSI TS and SAR Section 10.2.

Several components (that is, impact limiters, crane links, auxiliary lift component, and SFP frame) are used during the cask loading process. A discussion of these items is provided for the sole purpose of describing the loading process. These items, along with their design and use, are controlled under the DCPP Control of Heavy Loads Program.

A removable work platform is positioned in the cask washdown area to assist in transfer cask and MPC preparation and closure operations. The work platform also serves as a transfer cask seismic restraint.

For movements between the SFP and the cask washdown area, a removable impact limiter will be temporarily affixed to the base of the transfer cask. The impact limiter serves to limit loads on the cask system and loads imparted to the FHB/AB in the unlikely event of a vertical cask drop event.

During horizontal cask movements (that is, cask movements between the SFP and over the cask washdown area and movements between the cask washdown area and the cask transport frame), the crane is configured with a set of fixed length redundant load links (tension links). The tension links provide a redundant load path between the lift yoke and the crane eliminating the potential for cask drops as credible events during these cask handling evolutions.

Placement of loaded overpacks at the ISFSI is a cyclical process involving the movement of a loaded overpack to the ISFSI and returning with an empty transfer cask for the next loading process. The operations described herein start at the time the empty MPC is loaded into the transfer cask and is ready for movement into the FHB/AB.

Prior to bringing the transfer cask into the FHB/AB, the transfer cask is visually verified to have the pool lid bolted to the cask, and an empty MPC has been cleaned, inspected, raised, and inserted into the transfer cask. Alignment marks are checked to ensure correct rotational alignment between the MPC and the transfer cask.

The transfer cask containing an empty MPC is brought into the FHB/AB through the roll-up door in the horizontal orientation on a cask transport frame. Affixed to the bottom end of the transfer cask is a temporary shield. The transfer cask bottom shield is used during loaded transport operations to provide supplemental shielding to the operators. During transport of the empty transfer cask back to the FHB/AB, the bottom shield is used only as a spacer to ensure proper fit of the transfer cask in the cask transport frame. The cask transport frame is an L-shaped structure with front and rear saddles to support the transfer cask. The cask transport frame is used for horizontal transport of the transfer cask between the FHB/AB and the CTF and for cask upending and downending operations. (Upending is the process of rotating the cask from the horizontal to the vertical orientation. Downending is the process of rotating the transfer cask from the vertical to the horizontal orientation.) The cask transport frame is equipped with heavy-duty rollers that engage with a temporary track that runs from inside the FHB/AB to the access road located outside the FHB/AB roll-up door. The track and rollers are used because dimensional limitations of the FHB/AB roll-up door prevent access of the cask transporter inside the FHB/AB. The short side of the cask transport frame is designed to ensure that the transfer cask and cask transport frame rotate smoothly to the vertical orientation (without sudden load shifts normally experienced when a load's center of gravity traverses its corner). Heavy-duty rollers are affixed to the cask transport frame so the load will automatically position itself as it is lifted. The rollers also serve to strategically control the impact location should a hypothetical crane failure occur during cask upending or downending. An impact limiter is placed over the identified impact location (selected to be over a load-bearing wall). In the event of a crane failure, the transfer cask weight is directed through the upper saddle into the impact limiter and, in turn, into the strategic location on the floor.

After bringing the transfer cask into the FHB/AB, the transfer cask is positioned under the overhead crane, that is configured with the lift yoke. The lift yoke engages the transfer cask lifting trunnions, and the transfer cask and cask transport frame are tilted up slightly. A cask transport frame impact limiter is placed on the floor below the upper saddle portion of the cask transport frame. The transfer cask and cask transport frame are rotated integrally to the vertical position. The cask transport frame stabilizer is attached to secure the cask transport frame in the vertical orientation. Tension links are attached between the lift yoke and the auxiliary lift component to prevent a load drop during transfer cask horizontal movement. Bolts securing the transfer cask bottom shield to the transfer cask are removed and the straps securing the transfer cask to the cask transport frame are released. The transfer cask is moved horizontally from the frame. Specially designed bumpers are attached to the transfer cask prior to moving the transfer cask to the SFP. These bumpers are attached in eight locations (four at the top and four at the bottom) on the transfer cask using attachment holes fabricated on the transfer cask at 90-degree intervals around the cask body. Figures 4.2-9 and 4.2-10 show the bumper attachment configuration. The bumpers are employed to minimize swing-induced impacts of the transfer cask with the SFP seismic restraint structure.

The cask work platform main gate is opened to receive the transfer cask. A transfer cask impact limiter is positioned on the floor in the cask washdown area. The transfer cask then is positioned over the impact limiter. The main gate is closed and the cask work platform seismic restraints are closed. The tension links are disconnected and the transfer cask is lowered onto the transfer cask impact limiter (see Figure 4.4-1). Attachment bolts connect the transfer cask to the transfer cask impact limiter.

The annulus between the transfer cask and the MPC is filled with borated water, in accordance with the Diablo Canyon ISFSI TS and SAR Section 10.2, and a seal is installed in the top part of the annulus to minimize the risk of contaminating the external shell of the MPC. The MPC cavity is filled with water and borated in accordance with the Diablo Canyon ISFSI TS. MPC and annulus filling may occur in the cask washdown area, over the SFP, or any other intermediate location.

The seismic restraints are opened and the transfer cask, along with its attached impact limiter and empty MPC, are raised approximately 12 inches above the floor of the FHB/AB

(140 ft elevation). A second set of crane tension links are attached to provide the redundant load drop protection during horizontal movement over the SFP wall. The transfer cask is positioned adjacent to the SFP.

An annulus purge line is connected to the annulus drain port. The transfer cask is positioned over the cask recess area of the SFP and lowered using the FHB/AB crane auxiliary lift until the lower set of guides on the cask are engaged in corresponding guide channels of the SFP frame structure. The SFP frame provides lateral support of the cask during its vertical load movement in the cask recess area of the spent fuel pool frame structure. The transfer cask is lowered into the SFP, and an annulus purge of water is performed on the annulus through the annulus purge line. The annulus purge applies a slight overpressure to the annulus to protect the MPC external shell from contamination from the SFP water in the event there is a leak in the annulus seal. When the cask is fully lowered to the bottom of the cask recess area in the SFP, the lift yoke is remotely disconnected and removed from the SFP.

Fuel-loading and post-loading verification of fuel assembly identification is conducted in accordance with approved fuel-handling procedures. For damaged fuel assemblies and fuel debris, the assembly is loaded into the DFC, and the DFC is loaded into the MPC. Optionally, an empty DFC may be first loaded into the appropriate fuel storage location in the MPC and then the damaged fuel assembly or fuel debris loaded into the DFC.

The MPC lid, with the drain line and the lid restraint attached, are placed in position in the MPC after the completion of fuel loading, while the transfer cask is in the SFP. The MPC lid restraint is bolted on while the MPC is in the pool. The transfer cask and lift yoke are raised until the top of the MPC breaks the water surface. Rinsing of exterior surfaces is performed as the transfer cask emerges from the SFP. The transfer cask is raised completely out of the SFP to clear the SFP wall and the redundant crane tension links are attached. The annulus purge line is disconnected, the bumpers are removed, and the transfer cask is moved laterally (the crane tension links prohibit vertical movement and provide the necessary redundancy to make a drop event noncredible when installed) and positioned over the cask washdown area. The cask seismic restraints in the cask washdown area are positioned to prevent tipover if the cask should be dropped. The crane tension links are disconnected and the transfer cask is lowered into the cask washdown area. The eight guides are removed from the upper and lower gussets of the transfer cask. The cask seismic restraints are positioned for cask stability during a seismic event, the MPC lid retention device is removed, and the lift yoke is disconnected and removed from the area. Activities involving decontamination, water jacket filling, disconnection of cask rigging, and placement of auxiliary equipment may occur in parallel or in a different sequence based on cask-loading experience at DCPP.

The transfer cask water jacket is filled with water. A temporary shield ring may be installed in the area of the lifting trunnions to provide supplemental personnel shielding. Preparation for MPC sealing operations may now proceed. This may include the erection of scaffolding, staging of auxiliary equipment, additional cask decontamination, dose-rate surveys, and installation of temporary shielding.

As described above, fuel-assembly decay heat could eventually cause boiling of the water in the MPC after it is removed from the SFP. Therefore, MPC draining must be completed within the time-to-boil limit previously determined, which is measured beginning at the time the MPC lid is installed in the SFP and terminating at the completion of MPC draining. Should it become evident that the time-to-boil limit may be exceeded, a recirculation of the MPC water (borated as necessary in accordance with the Diablo Canyon ISFSI TS) will be performed to reduce the temperature of the water and allow a new time-to-boil value to be determined, if necessary. When the MPC water recirculation is complete, the time-to-boil clock is reset. This process may be repeated as necessary.

During welding operations, the MPC water volume is reduced to provide enough space between the water surface and the lid to avoid a water-weld interaction. Oxidation of Boral panels and aluminum components contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during, MPC lid welding operations. In addition, the space below the MPC lid shall 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. The automated welding system is installed. The MPC-lid welding, including nondestructive examinations, is completed.

Once the MPC-lid welding is complete, the MPC is filled with borated water, vented, and hydrostatically tested. After an acceptable hydrostatic test has been completed, a small amount of water is displaced with helium gas for leakage testing of the MPC lid-to-shell weld. MPC leakage testing is performed in accordance with ANSI N14.5 (Reference 4).

Following successful completion of the leakage testing, the remaining MPC water is displaced from the MPC by blowing pressurized helium gas into the vent port of the MPC, thus displacing the water through the drain line. The moisture removal system is connected to the MPC and is used to remove the remaining liquid water from the MPC and to reduce the moisture content of the MPC cavity to an acceptable level. This can be accomplished using a vacuum drying process (moderate burnup [that is, < 45,000 MWD/MTU] fuel only) or the forced helium dehydration (FHD) system (moderate or high burnup fuel). During the drying process, the annular gap between the MPC and the HI-TRAC will be continuously flushed with water.

Following the successful completion of moisture removal from the MPC, the MPC is backfilled with helium. If the vacuum drying process was used for moisture removal, no additional preparation of the MPC cavity is necessary prior to helium backfill operations. If the FHD system was used, the bulk residual gas must be evacuated from the MPC cavity to ensure the amount of helium being introduced into the MPC can be correctly determined. This evacuation (to 10 torr or less, where 760 torr equal 1 atmosphere) should be completed expeditiously to minimize fuel heatup, once completed, backfilling with helium must be initiated within 2 hours. If the 2-hour guideline is exceeded, the MPC should be refilled with helium and the pressure reduction process started again. Then, the helium backfill system

(HBS) is attached, and the MPC is backfilled with helium to within the required pressure range in accordance with the Diablo Canyon ISFSI TS. Helium backfill to the required pressure and purity level ensures that the conditions for heat transfer inside the MPC are consistent with the thermal analyses and provides an inert atmosphere to ensure long-term fuel integrity.

After successful helium backfill operations, the MPC vent and drain port cover plates are installed, welded, inspected, examined, and leak tested. The MPC closure ring is then installed, welded, and examined. The MPC closure ring provides a second welded boundary, in addition to the confinement boundary, and is described further in Section 3.3.1.1.1 with references to the design drawings in the HI-STORM 100 System FSAR for additional details.

The transfer cask water recirculation equipment is detached and remaining water in the transfer cask annulus is drained. The temporary shield ring is removed. The transfer cask and accessible portions of the MPC are checked to ensure any removable contamination is within applicable limits. Additional decontamination and surveys may be performed throughout the loading process. The MPC lift cleats are installed. The transfer cask top lid is installed and the fasteners are torqued.

The lift yoke is re-attached to the transfer cask, and the fasteners securing the impact limiter to the transfer cask bottom are disconnected. The transfer cask is raised and, while the transfer cask is maintained directly above the detached impact limiter, the crane tension links are attached. With the crane tension links attached and the cask suspended from the lift yoke, the bottom surface of the transfer cask is decontaminated using long-handled tools or other remotely-operated devices which do not require personnel to directly access the bottom of the transfer cask.

The seismic restraint is opened and the transfer cask is moved laterally away from the cask washdown area. The transfer cask is positioned in the bottom shield located in the transport frame (Figure 4.2-12). The transfer cask is fastened to the bottom shield and secured to the cask transport frame with straps. The cask transport frame impact limiter (Figure 4.4-2) is positioned on the floor in the same manner as described earlier to mitigate the effects on the transfer cask and building structure of an unrestrained tipover of the cask transport frame and cask. The cask and cask transport frame are supported by the crane, cask transport frame stabilizers, and the tension links. The tension links are disconnected and the cask transport frame stabilizers are removed. The crane hook is slowly lowered, causing the transfer cask and cask transport frame to gently roll, in its tracks, to the horizontal orientation. When the cask is about to contact the cask transport frame impact limiter, the impact limiter is removed and the cask transport frame is lowered to the full horizontal position. The loaded transfer cask is now positioned horizontally in the cask transport frame on the roller tracks.

If not performed earlier, the transfer cask and cask transport frame are surveyed to ensure that any fixed contamination is within acceptable limits. The loaded transfer cask and cask transport frame are then rolled out of the FHB/AB to the cask transporter.

5.1.1.3 Transfer to the ISFSI Storage Site

The cask transporter and associated ancillaries, described in Section 4.3, are positioned outside the FHB/AB doors to receive the horizontal transfer cask and cask transport frame. The transporter will undergo preoperational testing and maintenance and will be operated in accordance with the Cask Transportation Evaluation Program in the Diablo Canyon ISFSI TS, which evaluates and controls the transportation of loaded MPCs between the DCPP FHB/AB to the CTF and ISFSI. The transfer cask is positioned under the lift beam of the cask transporter and the transfer cask lift slings are rigged around the cask. The horizontal lift rig is attached to the slings and the transporter lift beam as described in Section 4.3. The horizontal lift rig supports the transfer cask directly and does not rely on the cask transport frame to support the cask. The transfer cask and cask transport frame are raised and secured within the transporter for the trip to the CTF. The transfer cask is transported to the CTF along the approved transportation route as described in Section 4.3.3 and shown in Figure 2.1-2.

In preparation for receiving the MPC, the overpack is positioned in the CTF and lowered to the full down position. The overpack lid is removed (if previously installed). The mating device (Figure 4.2-11) is secured to the overpack.

The cask transport frame is set down in the upending area near the CTF. The horizontal lift rig is disconnected, and the HI-TRAC lift links are attached. The HI-TRAC lift links are attached to the transfer cask lifting trunnions and the transfer cask is upended to the vertical orientation. Once vertical, the base of the cask transport frame is supported for stability. The cask transport frame straps are disconnected. A mobile crane attaches to the long leg of the cask transport frame. Fasteners connecting the long leg of the cask transport frame to its base are removed and the mobile crane removes the long leg of the frame. This step is performed to enable the transfer cask to be removed from the cask transport frame. The transfer cask bolts securing the transfer cask to the bottom shield are removed. The transfer cask is removed from the cask transport frame, and the transfer cask is aligned over the mating device. Restraints connect the cask transporter to the CTF pad. The transfer cask lift links are then disconnected. The MPC downloader slings are attached between the cask transporter towers and the MPC lift cleats, and the MPC is raised slightly to remove the weight of the MPC from the pool lid. The pool lid is supported by the mating device while the pool lid bolts are removed. The pool lid is removed from under the transfer cask.

The cask transporter towers are used to lower the MPC into the overpack. The MPC downloader slings are disconnected from the cask transporter and lowered onto the MPC lid. The pool lid is reinstalled. The HI-TRAC lift links are reconnected to the cask transporter and the cask transporter restraints are disconnected.

The transfer cask is lifted from the mating device and raised from the top of the overpack and placed back on the cask transport frame base and bolted to the bottom shield. The long leg of the frame is reattached. The cask transport frame straps are reinstalled. The cask and frame are lifted and the parabolic shapes are reinstalled. The cask is downended and placed beside the CTF. The lift cleats and MPC downloader slings are removed, and threaded inserts are installed in the MPC lid lift holes where the lift cleats were attached. The mating device containing the transfer cask pool lid is removed from the overpack and placed in a nearby location.

The overpack lid is installed. The overpack lifting brackets are attached. The cask transporter is positioned with its lift beam above the overpack. The overpack is raised to the up position in the CTF and the overpack lifting brackets are attached to the overpack. The overpack is lifted out of the CTF and moved to the ISFSI pad, where it is placed in its designated storage location. Once in position, the remaining overpack lid studs and nuts are installed and torqued. The cask transporter is disconnected from the overpack and driven away from the ISFSI pad. The grounding cables are attached to the overpack. The overpack duct photon attenuators (also known as gamma shield cross plates) are installed in the upper and lower air ducts and screens are secured.

5.1.1.4 Off-Normal Event Recovery Operations

The analysis of off-normal and accident events, as defined in ANSI/ANS-57.9 (Reference 5) and as applicable to the Diablo Canyon ISFSI, is presented in Chapter 8. Each postulated off-normal and accident event analyzed and discussed in Chapter 8 addresses the event cause, analysis, and consequences. Suggested corrective actions are also provided for off-normal events. The actual cause, consequences, corrective actions, and actions to prevent recurrence (if required) will be determined through the DCPP corrective action program on a case-specific basis. All corrective actions will be taken in a timely manner, commensurate with the safety significance of the event. Of primary importance in the early response to any event will be the verification of continued criticality prevention, the protection of fuel cladding integrity (that is, heat removal), and the adequacy of radiation shielding while longer-term corrective actions are developed. This may also involve the need for temporary shielding or cask cooling in accordance with the recommendations of PG&E technical staff personnel, based on the event conditions.

Should the need arise, the MPC can be returned to the SFP for unloading. To unload an overpack or transfer cask, the operations described above are effectively executed in reverse order from the point in the operation at which the event occurred. Once the transfer cask is back in the FHB/AB, the transfer cask top lid is removed, and preparations are made to reopen the MPC in the SFP. This involves first grinding out the welds and removing the MPC closure ring and vent and drain port cover plates. A sample of the gas inside the MPC may be drawn to determine the extent of fuel cladding failure, if any. Then, the helium cooldown system is connected and used to recirculate the helium in the MPC to cool it to a temperature at or below the maximum-allowed temperature for reflooding in accordance with the Diablo Canyon ISFSI TS and SAR Section 10.2. Cooling the helium allows the MPC to be reflooded with water (borated as necessary) with a minimal amount of flashing and the associated undesirable pressure spikes in the MPC cavity. Based on the time the cask has been in storage, a new time-to-boil may be determined using a lower decay heat value than was used when the cask was loaded. When the MPC has been reflooded, the time-to-boil clock is started. The weld removal system is used to cut the MPC lid weld, freeing the lid for subsequent removal.

Oxidation of Boral panels and aluminum components contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during, MPC lid cutting operations. In addition, the space below the MPC lid shall be exhausted prior to, and during, MPC lid welding operations to provide additional assurance that explosive gas mixtures will not develop in this space. When the lid weld has been successfully cut, the lid retention device and lift yoke are installed, and the transfer cask is returned to the SFP using the same procedures and equipment as used to remove the transfer cask from the SFP after fuel loading.

Once in the SFP, the MPC lid is removed, and the spent fuel assemblies are removed from the MPC and placed back into the wet storage racks. The time-to-boil consideration is stopped once the MPC lid is removed.

5.1.2 FLOWSHEETS

Figure 5.1-1 shows the operation sequence flowchart for cask system loading, sealing, testing, onsite transport, MPC transfer, and storage operations.

Figure 5.1-2 shows the operation sequence flowchart for overpack off-normal event recovery operations.

A detailed description of the operations is provided in Section 5.1.1. Radiation source terms are discussed in Chapter 5 of the HI-STORM 100 System FSAR for the generic cask analyses and in Section 7.2 of this SAR for site-specific dose analyses. Equipment descriptions, with dimensions, design and operating characteristics, materials of construction, special design features, and operating characteristics are provided in Sections 3.3, 4.2, 4.3, and 4.4. Generic cask component design drawings are found in Section 1.5 of the HI-STORM 100 System FSAR.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY AND RELIABILITY ANALYSIS

5.1.3.1 Criticality Prevention

A summary description of the principal design features, procedures, and special techniques used to preclude criticality in the design and operation of the HI-STORM 100 System is provided in Section 3.3.1.4. Additional detail on the criticality design of the storage cask is provided in Section 4.2.3.3.5.

5.1.3.2 Instrumentation

No instrumentation is required to detect off-normal operations of the HI-STORM 100 System while in its final storage configuration at the ISFSI storage site. The cask system is designed to maintain confinement integrity under all design-basis normal, off-normal, and accident conditions. Detection of degradation in the HI-STORM 100 heat removal system is accomplished by a Diablo Canyon ISFSI TS that requires periodic visual surveillance of the overpack inlet and outlet air ducts to ensure they remain free of blockage. If blockage is detected, action can be taken to remove the source of the blockage in a short time period, typically within one operating shift.

Examples of measuring and test equipment (M&TE) used during the preparation of the cask for storage operations are listed in Table 5.1-1. Additional, or different M&TE, may be used as determined through the development of site-specific operating procedures, including the revision of those procedures as experience in cask loading operations is gained and the state of the art evolves.

5.1.3.3 Maintenance Techniques

The HI-STORM 100 System is designed to safely store spent nuclear fuel with no regularly required maintenance. The only expected maintenance is to apply touch-up repair coatings to the overpack and/or the anchorage hardware due to exposure to the elements and normal wear and tear.

5.1.4 REFERENCES

1. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 0, July 2000.

2. License Amendment Request 1014-1, Holtec International, Revision 2, July 2001, including Supplements 1 through 4 dated August 17, 2001; October 5, 2001; October 12, 2001; and October 19, 2001; respectively.

3. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 Dry Cask Storage System, Holtec International, Revision 0, May 2000.

4. ANSI N14.5-1997, Leakage Tests on Packages for Shipment, American National Standards Institute.

5. ANSI/ANS-57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (dry type), American National Standards Institute.

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7.5 OFFSITE COLLECTIVE DOSE

The annual offsite dose is calculated for both direct radiation (neutrons and gammas) and from radionuclide releases from the MPC. Since the MPC is welded and designed to maintain confinement integrity under all normal, off-normal, and accident conditions of storage, there will not be any release of radionuclides during normal operation. Nonetheless, an analysis of the offsite dose consequences from a nonmechanistic confinement boundary leak from the ISFSI was calculated for normal, off-normal, and accident conditions. This section addresses doses for normal conditions. Off-normal and accident analyses are provided in Sections 8.1.3 and 8.2.7, respectively. The direct radiation dose from the ISFSI is the same for normal and off-normal conditions.

Since the loading of the MPC into the overpack occurs outside the FHB/AB at the CTF, the offsite dose due to loading operations was also calculated and included in the total annual dose estimate.

The controlled area boundary is located 1,400 ft (427 m) from the ISFSI. However, the nearest resident is located 1.5 mi (7,920 ft or 2,414 m) from the ISFSI. Therefore, consistent with ISG-13 (Reference 1), the occupancy time at the controlled area boundary for the dose calculation was assumed to be 2,080 hr based on a 40-hr work week and 52 weeks per yr while the occupancy time at the nearest resident location was assumed to be 8,760 hr (24 hr per day 365 days per yr).

7.5.1 DIRECT RADIATION DOSE RATES

Table 7.5-1 presents the dose rate and annual doses at the site boundary and the nearest residence from direct radiation from the Diablo Canyon ISFSI after it is completely filled with 140 overpacks loaded with the MPC-32 at design-basis burnup and cooling times. As described in Section 7.3.2.3, these dose rates and doses were calculated at distances that were perpendicular to the long side of the ISFSI and it was assumed that eight overpacks were loaded per year.

7.5.2 DOSE RATES FROM NORMAL OPERATION EFFLUENT RELEASES

The source term used for the offsite dose assessment from the effluent release from the MPC is discussed in Section 7.2.2. The dose assessment from effluent release was calculated for normal conditions. Effluent doses for off-normal operations are discussed in Section 8.1.3 of this SAR. Effluent doses for an accident condition are discussed in Section 8.2.7.

7.5.2.1 Release of MPC Contents Under Normal Occurrences

The MPC is designed to maintain confinement boundary integrity under all normal, off-normal, and accident conditions of storage. Nevertheless, a hypothetical, non-mechanistic confinement boundary leak was evaluated in the effluent dose analysis. For normal conditions, it was assumed that 2.5 percent of the total source term of each assembly is available for release to the MPC cavity. This was based on the assumption, from ISG-5 (Reference 2), that 1 percent of the fuel rods have ruptured. In addition to the 1 percent, it was assumed, consistent with ISG-11 (Reference 3), that an additional 3 percent of fuel rods had cladding oxide thicknesses greater than 70 micrometers and therefore had 50 percent of the source term in these rods available for release. The spent fuel is stored in a manner such that the spent fuel cladding is protected during storage against degradation that could lead to fuel cladding ruptures. The MPC cavity is filled with the inert gas helium after the MPC has been evacuated of air and moisture that might produce long-term degradation of the spent fuel cladding. The HI-STORM 100 System is additionally designed to provide for long-term heat removal to ensure that the fuel is maintained at temperatures below those at which cladding degradation occurs. It is therefore highly unlikely that a spent fuel assembly with intact fuel cladding will undergo cladding failure during storage, and the assumption that 2.5 percent of the source term is available for release is conservative.

The assumption that 10 percent of the fuel rods have ruptured was incorporated into the postulated pressure increase within the MPC cavity to determine a bounding pressure of the MPC cavity for effluent release calculations for the normal and off-normal cases. This pressure, combined with the maximum MPC cavity temperature was used to determine a postulated leakage rate. This leakage rate was based on an assumed leakage of 5.0×10^{-6} atm-cm^3/sec during the helium leak rate test and was adjusted for the higher temperature and pressure during the off-normal condition to result in a calculated leak rate of 7.37×10^{-6} atm-cm^3/sec.

The radionuclide release fractions, which account for the radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not releasable to the environment, were based on ISG-5 and are presented in Table 7.2-8. Additionally, only 10 percent of the fines released to the MPC cavity were assumed to remain airborne long enough to be available for release from the cask MPC (Reference 4). It was conservatively assumed that 100 percent of the volatiles, crud, and gases remain airborne and available for release. The release rate for each radionuclide was calculated by multiplying the quantity of radionuclides available for release in the MPC cavity by the leakage rate calculated above, divided by the MPC cavity volume.

7.5.2.2 Effluent Dose Calculations for Normal Conditions

The nearest distance from the ISFSI to the DCPP site boundary is 1,400 ft. A χ/Q value of 3.44×10^{-6} sec/m^3 (Reference 5) at the site boundary was used for this analysis. This χ/Q value is the highest χ/Q in any direction and is based on duration of an entire year. The dose conversion factors for internal doses due to inhalation and submersion in a radioactive plume were obtained from the EPA Federal Guidance Report No. 11 (Reference 6) and EPA Federal Guidance Report No. 12 (Reference 7), respectively. An adult breathing rate of 3.3×10^{-4} m^3/sec was assumed (Reference 2). For site boundary dose, an annual occupancy of 2,080 hr was assumed. For the nearest resident, full-time occupancy was assumed (8,760 hr).

The annual dose equivalent for the whole body, thyroid, and other critical organs to an individual at the DCPP site boundary as a result of a non-mechanistic normal effluent release were calculated for an ISFSI containing 140 overpacks, each loaded with an MPC-32. Table 7.5-2 summarizes the dose results for normal conditions. As can be concluded from Table 7.5-2, the estimated doses are a fraction of the limits specified in 10 CFR 72.104(a) for normal operations.

7.5.3 OFFSITE DOSE FROM OVERPACK LOADING OPERATIONS

The transfer of the MPC from the transfer cask to the overpack will occur outside the FHB/AB at the CTF. As a result, the impact of this operation on the offsite dose was considered. There are only two conditions that need to be considered in this analysis. The first is the condition of the MPC inside the transfer cask. The second condition is the MPC inside the overpack with the transfer cask no longer positioned above the overpack and the lid on the overpack not installed. Table 7.5-3 presents the results of these analyses.

7.5.4 TOTAL OFFSITE COLLECTIVE DOSE

Table 7.5-4 presents the annual dose at the site boundary and for the nearest resident from the combined dose rates from direct radiation and non-mechanistic effluent release for normal ISFSI operations and off-normal operations. The dose rates from other uranium fuel cycle operations (that is, DCPP) are also shown in this table to demonstrate compliance with 10 CFR 72.104. Table 7.5-4 demonstrates that the Diablo Canyon ISFSI will meet the 10 CFR 72.104 regulatory requirements. However, ultimate compliance with the regulations will be demonstrated through the DCPP environmental monitoring program.

The actual dose from the ISFSI will be considerably less than the conservatively estimated values in Table 7.5-4. The following are some of the conservative assumptions used in the calculating the dose rates presented.

- The design basis assembly and design basis burnup and cooling time were conservatively chosen.
- All fuel assemblies in the MPC are assumed to be identical with the design basis burnup and cooling time.
- BPRAs are assumed to be present in all fuel assemblies in all casks.
- The assumed ISFSI loading plan was conservatively chosen to result in the highest offsite dose rate.
- The dose rate was calculated at the most conservative location around the ISFSI.

7.5.5 REFERENCES

1. Real Individual, USNRC, Interim Staff Guidance Document-13, Revision 0, June 2000.

2. Normal, Off-Normal and Hypothetical Dose Estimate Calculations, USNRC, Interim Staff Guidance Document-5, Revision 1, June 1999.

3. Transportation and Storage of Spent Fuel Having Burnups in Excess of 45 GWD/MTU, USNRC, Interim Staff Guidance Document-11, Revision 1, May 2000.

4. Y.R. Rashid, et al, An Estimate of the Contribution of Spent Fuel Products to the Releasable source Term in Spent Fuel Transport Casks, SAND88-2778C, Sandia National Laboratories, 1988.

5. 1999 Annual Radioactive Effluent Release Report, PG&E Letter DCL-00-061, April 28, 2000.

6. Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, US EPA, Federal Guidance Report No. 11, DE89-011065, 1988.

7. External Exposure to Radionuclides in Air, Water, and Soil, US EPA, Federal Guidance Report No. 12, EPA 402-R-93-081, 1993.

TABLE 7.5-4

TOTAL ANNUAL OFFSITE COLLECTIVE DOSE (MREM) AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM THE DIABLO CANYON ISFSI

	Normal Operations				**Off-Normal Operations**		
Organ	**Effluent Release[c]**	**Direct Radiation[c]**	**Overpack Loading Operations[d]**	**Other Uranium Fuel Cycle Operations[a]**	**Effluent Release[d]**	**Total (normal + off-normal)**	**10 CFR 72.104 Regulatory Limit**
Site Boundary (1,400 ft / 427 m)							
Whole body ADE[b]	0.064	5.6	13.1E-02	4.357E-02	1.27E-03	5.84	25
Thyroid ADE	0.010	5.6	13.1E-02	1.260E-01	1.02E-04	5.87	75
Critical organ ADE (Max)	0.35	5.6	13.1E-02	5.590E-02	9.31E-03	6.15	25
Nearest Resident (1.5 miles / 7,920 ft / 2414 m)							
Whole body ADE	0.27	3.5E-04	13.1E-02	4.357E-02	5.33E-03	0.45	25
Thyroid ADE	0.043	3.5E-04	13.1E-02	1.260E-01	4.31E-04	0.30	75
Critical organ ADE (Max)	1.46	3.5E-04	13.1E-02	5.590E-02	3.92E-02	1.69	25

[a] Data for uranium fuel cycle operations were obtained from the DCPP FSAR Update, Rev. 11, Table 11.3-32. Table 11.3-32 was selected based on the highest dose values in the sectors at the site boundary (0.5 miles). These dose values for the site boundary were conservatively applied to the nearest resident. The critical organ dose listed was based on the total liver dose in Table 11.3-32. The values listed in Table 11.3-32 should bound the results calculated from effective dose equivalent methodology.

[b] ADE is annual dose equivalent.

[c] 140 casks

[d] Single cask

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This chapter describes the accident analyses for the Diablo Canyon ISFSI. Sections 8.1 and 8.2 evaluate the safety of the ISFSI under off-normal operations and accident conditions, respectively. For each event, the postulated cause of the event, detection of the event, and evaluation of the event effects and consequences, corrective actions, and radiological impact are presented. Unless otherwise identified in Chapter 8 or other SAR sections, the MPC 32 was evaluated as a bounding condition. The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events and accidents without affecting function and are in compliance with the applicable acceptance criteria. Section 8.3 summarizes site characteristics that affect the safety analysis.

8.1 OFF-NORMAL OPERATIONS

This section addresses events designated as Design Event II, as defined by ANSI/ANS-57.9 (Reference 1). The following are considered off-normal events for the Diablo Canyon ISFSI:

- Off-normal pressures
- Off-normal environmental temperatures
- Confinement boundary leakage
- Partial blockage of air inlets
- Cask drop less than allowable height
- Loss of electric power
- Cask transporter off-normal operation

For each event, the postulated cause of the event, detection of the event, an evaluation of the event effects and consequences, corrective actions, and radiological impact are presented. The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events without affecting function and are in compliance with the applicable acceptance criteria. The following sections present the evaluation of the HI-STORM 100 System for the design-basis, off-normal conditions that demonstrate that the requirements of 10 CFR 72.122 are satisfied and that the corresponding radiation doses satisfy the requirements of 10 CFR 72.104(a).

8.1.1 OFF-NORMAL PRESSURES

The HI-STORM 100SA overpack is a ventilated cask design. The sole pressure boundary of the storage system is the multi-purpose canister (MPC). The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure, variations in the helium temperature, and leakage of any gases contained within the fuel rods. The analyzed off-normal environmental temperature is 100°F and peak solar insolation is assumed. This bounds the Diablo Canyon ISFSI maximum off-normal site ambient temperature and solar insolation values. The MPC off-normal pressure evaluation includes the conservative assumption that 10 percent of the fuel rods rupture, allowing 100 percent of the fill gas and 30-percent of the fission gases from these fuel rods to be released to the MPC cavity. This assumption is consistent with the guidance in NUREG-1536 for the review of dry storage cask designs (Reference 2).

8.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to ensure long-term fuel cladding integrity during dry storage. The pressure of the gas in the MPC cavity is affected by the initial fill pressure, the MPC cavity volume, the decay heat emitted by the stored fuel, the presence of nonfuel hardware, fuel-rod gas leakage, ambient temperature, and solar insolation. Of these, the initial fill pressure, presence of non-fuel hardware, and MPC cavity volume do not vary with time in storage and can be ignored as a cause of off-normal pressure. The decay heat emitted by the stored fuel decreases with time and is conservatively accounted for in the analysis by using the highest rate of decay heat for a given fuel cooling time. Off-normal pressure is conservatively evaluated considering a concurrent non-mechanistic rupture of 10 percent of the stored fuel rods during a time of maximum off-normal ambient temperature (100°F) and full solar insolation.

8.1.1.2 Detection of Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to perform its design safety functions. No personal actions or equipment are required to respond to an off-normal pressure event. Therefore, no detection instrumentation is required.

8.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The evaluation of MPC pressure for this off-normal event was initially performed assuming normal ambient temperature (80°F), 10 percent of the fuel rods ruptured, peak insolation, maximum decay heat, and the effect of nonfuel hardware. The MPC-32 was used as the bounding MPC in this analysis because it provides the maximum internal pressure for all MPCs to be used at the Diablo Canyon ISFSI (see Section 4.2.3.3.2.2 for justification). The resulting pressure for MPC-32 with 80°F ambient temperature is 76.0 and 87.90 psig for the storage and transport conditions respectively. Using this initial pressure, the added effect of increasing the ambient temperature from 80°F to the maximum off-normal temperature of 100°F was conservatively evaluated using the Ideal Gas Law. Assuming the MPC cavity gas temperature increased by the full 20°F, the resulting absolute pressure P_2 for the storage condition is computed as follows:

$$P_2 = P_1 \times [(T_1 + \Delta T)/T_1]$$

Where,

P_1 = Absolute pressure at T_1 = 76.0 psig (90.7 psia)

T_1 = Absolute bulk temperature of the MPC cavity gas with design basis fuel decay heat = 513.6°K (Reference 4, Section 11.1.1.3)

ΔT = Absolute bulk MPC cavity gas temperature increase = 20°F, or 11.1°K

The resulting absolute pressure (P_2) was computed to be 92.7 psia, or 78.0 psig. Applying the same formula, the transport condition temperature can be calculated to be 89.84 psig. Both are below the normal/off-normal MPC internal design pressure of 100 psig.

8.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There are no corrective actions associated with off-normal pressure.

8.1.1.5 Radiological Impact from Off-Normal Pressure

The off-normal pressure event has no radiological impact because the confinement barrier and shielding integrity are not affected.

8.1.2 OFF-NORMAL ENVIRONMENTAL TEMPERATURES

The off-normal temperature ranges for which the HI-STORM 100 System is designed are summarized in the HI-STORM 100 System FSAR (Reference 3) Section 2.2.2. The off-normal temperature evaluation is described in HI-STORM 100 System FSAR Section 11.1.2, as amended by LAR 1014-1 (Reference 4). Off-normal environmental temperature ranges of –40 to 100°F (for the HI-STORM 100SA overpack and ISFSI storage pads) and 0 to 100°F (for the HI-TRAC transfer cask, cask transporter, and cask transfer facility) conservatively bound off-normal temperatures at the Diablo Canyon ISFSI site (24°F to 97°F). The off-normal environmental temperature ranges are used as the design criteria for the concrete storage pad, cask transporter, and CTF. The ranges of off-normal temperatures evaluated bound the historical temperature variations at the Diablo Canyon ISFSI.

This off-normal event is of a short duration. Therefore, the resultant fuel cladding temperatures for the cask evaluations are compared against the accident condition (short-term) temperature limits.

8.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by unusual weather conditions. To determine the effects of off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

8.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no personnel actions or equipment required for mitigation of an off-normal temperature event. Deleterious effects of off-normal temperatures on the cask transporter, CTF, and concrete storage pad are precluded by design. Administrative procedures based on Diablo Canyon ISFSI TS 5.1.3 will prohibit cask handling if temperatures fall outside the off-normal temperature limits. Ambient temperature is available from thermometers used for the DCPP site meteorological measurement program.

8.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

There are no adverse safety effects resulting from off-normal environmental temperatures on the cask transporter, CTF, or concrete storage pads, since they are designed for these temperature ranges.

The off-normal event, considering a maximum off-normal ambient temperature of 100°F, has been evaluated for the HI-STORM 100 System and is described in the HI-STORM 100 System FSAR Section 11.1.2.3, as amended by LAR 1014-1. The evaluation was performed for the loaded transfer cask and the loaded overpack, assuming design-basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 100°F environmental temperature was applied with peak solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design-basis temperatures are conservatively calculated at an environmental temperature of 80°F as an initial condition for this off-normal event. These temperatures (for MPC-32, MPC-24E, and the overpack) are shown in Tables 4.4.26, 4.4.27, and 4.4.36 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. The maximum off-normal environmental temperature is 100°F, which is an increase of 20°F over the normal design temperature. The resulting limiting component maximum off-normal temperatures are shown in Table 11.1.1 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. The temperatures are all below the applicable material short-term temperature limits.

The off-normal event considering a limiting low environmental temperature of -40°F and no insolation for a duration sufficient to reach thermal equilibrium has been evaluated with respect to overpack material brittle fracture at this low temperature. The overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the transfer cask is 0°F and the transfer cask is conservatively assumed to reach 0°F throughout the structure. This evaluation is discussed in the HI-STORM 100 System FSAR Section 3.1.2.3 and the results are acceptable. Administrative procedures based on Diablo Canyon ISFSI TS 5.1.3 prohibit cask handling operations at environmental temperatures below 0°F.

8.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. The cask, transporter, CTF, and ISFSI pad are designed for temperature ranges consistent with the dry storage cask components used at these facilities. Therefore, no corrective actions are required for off-normal environmental temperature conditions.

8.1.2.5 Radiological Impact of Off-Normal Temperatures

Off-normal environmental temperatures have no radiological impact as the integrity of the confinement barrier and shielding are unaffected by off-normal temperatures. The effect of elevated temperatures does not significantly increase the doses associated with the design-basis leak rate from the MPCs and is bounded by the results of the off-normal failure of fuel cladding event assessed in Section 8.1.3.

8.1.3 CONFINEMENT BOUNDARY LEAKAGE

The HI-STORM 100 System MPC has a welded confinement boundary to contain radioactive fission products under all design-basis normal, off-normal, and accident conditions. The radioactivity confinement boundary is defined by the MPC shell, baseplate, MPC lid, and vent and drain port cover plates. A non-mechanistic failure of fuel cladding in conjunction with allowable leakage in the MPC confinement boundary has been evaluated as both an off-normal and an accident condition. The difference between the two evaluations is in the radioactive source term, the bounding temperature and pressure determined in the thermal analysis of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, and the χ/Q value used for each of the two conditions. The analytical technique and assumptions used in both evaluations are consistent with Interim Staff Guidance (ISG) Document 5 (Reference 5). All other inputs to the confinement boundary leak dose analysis are identical for the off-normal and accident analyses. The accident condition is addressed in Section 8.2.7 of this SAR and is not discussed further here.

Since this event is applicable only to the MPC, the evaluation is applicable for all locations (that is, in the cask transporter, at the CTF, or on the ISFSI pad) and is independent of whether the MPC is inside the transfer cask or the overpack. Due to the close proximity of these three locations, the two χ/Q values used for the off-normal and accident condition evaluations are the same for all three postulated release locations.

8.1.3.1 Postulated Cause of Confinement Boundary Leakage

Based on the design of the MPC vessel and the protection provided by the transfer cask and the overpack, a leak in the MPC confinement boundary is not considered credible, so no cause is identified. Also, there is no credible mechanism for inducing the level of fuel failure assumed for this event. This off-normal condition is evaluated as a non-mechanistic event.

8.1.3.2 Detection of Confinement Boundary Leakage

The MPC is a welded cylindrical enclosure. There are no mechanical joints or seals in the confinement boundary. The confinement boundary is designed to maintain its integrity under all design basis normal, off-normal, and accident conditions. Therefore, leakage detection equipment is not required.

8.1.3.3 Analysis of Effects and Consequences of Confinement Boundary Leakage

The MPC confinement boundary is designed to remain intact under all design basis normal, off-normal, and accident conditions. However, as a defense-in-depth measure, the MPC closure ring, which provides a redundant weld for the MPC lid-to-shell weld and the vent and drain port cover plate welds, is designed to withstand full MPC cavity pressure. Therefore, the closure ring would provide the confinement boundary in this event. The dose consequences of a hypothetical, non-mechanistic confinement boundary leak are discussed in Section 8.1.3.5.

8.1.3.4 Corrective Action for Confinement Boundary Leakage

There is no corrective action required for the assumed leakage in the MPC confinement boundary because leakage in excess of allowable is not considered credible. Also, the assumed level of fuel failure is not considered credible.

8.1.3.5 Radiological Impact of Confinement Boundary Leakage

The dose consequences of a non-mechanistic leak in the MPC confinement boundary have been analyzed on a site-specific basis for the Diablo Canyon ISFSI using appropriate source terms, release fraction, leak rate, meteorology, breathing rate, and occupancy times. The analysis of this abnormal event considers the rupture of 10 percent of the stored fuel rods. The evaluation of this event under normal conditions is discussed in Section 7.5.2. The same methodology with the unique off-normal source is used here. Annual doses at the site boundary and nearest resident were calculated. The results are provided in Table 8.1-1 for the analysis of a single HI-STORM cask in the off-normal condition. The calculated doses are less than the regulatory limits in 10 CFR 72.104(a).

8.1.4 PARTIAL BLOCKAGE OF AIR INLETS

The HI-STORM 100 System overpack is designed with inlet and outlet air ducts, four each at the top and bottom of the overpack structure with the lid installed. Each duct opening includes a fine mesh screen across its outer face. These screens ensure the air ducts are protected from the incursion of foreign objects. Each set of four air inlet and outlet air ducts are spaced 90 degrees apart around the circumference of the overpack and it is highly unlikely that blowing debris during normal or off-normal operation could block all of the air inlet ducts. It is conservatively assumed, as an off-normal condition, that two of the four air inlet ducts are blocked. Blockage of the inlet air ducts is assumed to be thermally equivalent to blockage of the outlet air ducts. The evaluation of this off-normal event, as well as the blockage of three inlet ducts, is discussed in Section 11.1.4 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. The blocked air inlet ducts are assumed in the HI-STORM 100 System FSAR to be completely blocked, with an ambient temperature of 80°F, peak solar insolation, and maximum spent fuel decay heat values. The HI-STORM 100 System FSAR generic assumption of an annual average temperature of 80°F and peak solar insolation value of 800 g-cal/cm^2, respectively, bounds the Diablo Canyon site annual average temperature of 55°F and peak solar insolation value of 766 g-cal/cm^2.

8.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

It is conservatively assumed that the affected air inlet ducts are completely blocked, although the protective screens prevent foreign objects from entering into the ducts. The mesh screens are inspected periodically, as required by the Diablo Canyon ISFSI TS. Any duct blockage would be detected by visual inspection and removed to restore the heat removal system to full operational condition. Depending on the size and number of debris pieces, it is possible that blowing debris may simultaneously block two air inlet ducts of the overpack.

8.1.4.2 Detection of Partial Blockage of Air Inlets

Detection of partial blockage of air inlet ducts would occur during the routine visual surveillance of the storage cask air duct screens required by the Diablo Canyon ISFSI TS. The frequency of inspection is conservatively based on an assumed complete simultaneous blockage of all four air inlet ducts (Diablo Canyon ISFSI TS Bases).

8.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

Blockage of the overpack air inlet ducts can affect the heat removal process of the dry storage system. The magnitude of the effect is dependent upon the rate of decay heat emission from the stored fuel (itself dependent upon the fuel burnup and cooling time) and the ambient air temperature. Bounding evaluations were performed for the blockage of two and three inlet air ducts with the MPC-32 inside the overpack, at its maximum decay heat load at the ambient air temperature of 80°F. As stated above, the HI-STORM 100 System FSAR assumes an annual-average ambient air temperature of 80°F, which bounds the annual-average ambient air temperature for the Diablo Canyon Site of 55°F. The MPC-32 decay heat load bounds the MPC-24, MPC-24E, and MPC-24EF heat loads due to the presence of eight additional fuel assemblies. The largest component temperature rise for two ducts blocked is 25°F. The largest component temperature rise for three ducts blocked is 81°F. (Blocking of four ducts is treated as an accident in Section 8.2.15.) This maximum temperature rise was conservatively added to all cask component temperatures for comparison with the respective component short-term temperature limits. The results are shown in Table 11.1.2 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. All temperatures are less than the applicable component short-term temperature limits.

The MPC cavity pressure as a result of this limiting component temperature increase was also evaluated. An MPC cavity gas bulk temperature rise of 25°F was evaluated using the Ideal Gas Law method as described in Section 8.1.3 and the resulting MPC internal pressure was computed to be 78.4 psig, which is less than the normal condition MPC design pressure of 100 psig.

8.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet ducts is the removal of the cause of the blockage, and the cleaning, repair, or replacement, as necessary, of the affected mesh screens. After clearing of the blockage, the cask heat removal system is restored to its design condition, and temperatures will return to the normal range. Partial blockage of air inlet ducts does not affect the ability of the HI-STORM 100 System to safely store spent fuel for the long term.

Inspection of the overpack air duct screens is performed at a 24-hour frequency as required by the Diablo Canyon ISFSI TS. This inspection ensures blockage of air inlet ducts is detected and appropriately corrected.

8.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

For partial blockage of air inlet ducts, it is estimated that the removal, cleaning, and replacement of the affected mesh screens will take two people approximately 1 hour. The dose rate at this location is estimated to be 58 mrem/hr. The total exposure for personnel to perform these corrective actions is 0.116 man-rem.

8.1.5 CASK DROP LESS THAN ALLOWABLE HEIGHT

Cask drops outside the fuel handling building/auxiliary building (FHB/AB) are not credible due to the design of the cask transporter and the CTF, as discussed in Section 8.2.4. The structural load path members of both the CTF and the cask transporter used in Diablo Canyon ISFSI operations are designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612 (Reference 6). Therefore, a drop of the loaded MPC during inter-cask transfer operations is not a credible event. Although the cask and any ancillary components are lifted, handled, and moved in accordance with DCPP procedures and the DCPP Control of Heavy Loads Program, which provide assurance of safe heavy load handling, drop events inside the FHB/AB are nevertheless postulated and analyzed as described in the 10 CFR 50 license amendment request supporting the Diablo Canyon ISFSI license application, since the FHB/AB crane is not single failure proof.

8.1.6 LOSS OF ELECTRIC POWER

A total loss of external AC electric power is postulated to occur as a result of either a disturbance in the offsite electric supply system or the failure of equipment in the electrical distribution system feeding the ISFSI storage site and the CTF. A loss of electric power does not affect the cask transporter because all active functions of the transporter, such as cask lifting and MPC downloading, are driven from the onboard diesel engine.

8.1.6.1 Postulated Cause of Loss of Electric Power

Loss of the external power supply may occur as the result of natural phenomena, such as lightning strike or high winds, or as a result of undefined factors causing a disturbance in the offsite electrical grid. Loss of electrical power may also result from an electrical system fault or the failure of electrical distribution equipment such as a transformer.

8.1.6.2 Detection of Loss of Electrical Power

Loss of electrical power will be detected by the failure of electric-powered equipment.

8.1.6.3 Analysis of Effects and Consequences of Loss of Electrical Power

8.1.6.3.1 ISFSI Storage Site

There is no effect on the ability of the HI-STORM 100 System to safely continue storing the spent fuel at the ISFSI storage site during a loss of electric power event because the dry storage system is a completely passive design. No electric-powered equipment is used with the storage overpack while it is in its storage configuration on the concrete storage pads.

8.1.6.3.2 Cask Transfer Facility

The lift jacks of the CTF are the only functional component requiring electric power to operate. In a loss of electrical power, all lighting, power to the lift jacks, and any auxiliary power outlets will be unavailable. If the lift jacks are in operation at the time of the event, they will stop in place upon loss of power to preclude an uncontrolled lowering of the load. Upon restoration of power, the lift jacks will remain stopped by design of the electrical circuitry and will require manual action to restart.

8.1.6.4 Corrective Action for Loss of Electric Power

Corrective actions following a loss of electric power may vary widely, depending on the cause of the power loss. Restoration activities are generally straightforward. If electrical power to the CTF is lost with the loaded overpack in the lowered position, the overpack must be raised to grade level within 22 hours to ensure that the short-term, fuel-peak-cladding temperature limit is not exceeded. This is accomplished using the cask transporter and the HI-STORM lift links and lifting brackets.

8.1.6.5 Radiological Impact of Loss of Electric Power

The off-normal event of loss of electric power has no radiological impact because the MPC confinement barrier is not breached and shielding is not affected. The transfer cask is designed to provide adequate shielding and decay heat removal from the canisters. The sides of the transfer cask have both gamma and neutron shields, and the combination of the pool lid and bottom shield are designed to prevent excessive dose rates below the transfer casks. In the event the transfer operation is interrupted due to a loss of external power, operators would take measures as necessary to assure adequate distance and/or additional shielding between themselves and the transfer cask to minimize doses until electrical power is restored and the transfer process can resume.

8.1.7 CASK TRANSPORTER OFF-NORMAL OPERATION

Off-normal operation of the cask transporter includes postulation of the following human performance and active component failures during transport of the loaded transfer cask and the loaded overpack:

- Driver error
- Driver incapacitation
- Transporter engine failure
- Loss of hydraulic fluid

8.1.7.1 Postulated Cause of Cask Transporter Off-Normal Operation

Cask transporter driver error may be caused by driver inattentiveness, poor visibility, incorrect instructions, poor training, or any of several human performance-related causal factors. Driver incapacitation would be most likely caused by a sudden medical emergency. Transporter engine failure may be caused by a variety of mechanical problems typical of combustion engines. A loss of hydraulic fluid may be caused by a leak anywhere in the hydraulic system.

8.1.7.2 Detection of Cask Transporter Off-Normal Operation

Driver error or driver incapacitation would be detected by the support staff walking along with the transporter on the transport route observing the driver in distress or erratic transporter motion. Transporter engine failure would be detected by the halt of any engine-driven activity taking place at the time. A hydraulic fluid leak would be detected by the pressure instrumentation in the hydraulic system and possibly by visual observation of leaking fluid.

8.1.7.3 Analysis of Effects and Consequences of Cask Transporter Off-Normal Operation

In addition to the transporter driver, transport operations will be conducted with a support team consisting of security and other personnel affiliated with the fuel movement walking along with the transporter to ensure a safe and efficient move of the loaded cask from its point of origin to its destination. These personnel will be observing the movement of the transporter to ensure the designated travel path is being followed. Should the transporter start to veer from the travel path, the transporter will be stopped (either by the driver or by a support team member using either of two external stop switches mounted on the outside of the transporter), the cause investigated, and corrective actions taken to get the vehicle back on the correct path.

Incapacitation of the driver will be addressed by the design of an automatic shutoff control where the vehicle will stop whenever the control is released. The same control is used to move the transporter vehicle and operate the cask lifting apparatus integral to the transporter. A selector switch is used to ensure only one function can be performed by the transporter at a time. Also, either of two emergency stop switches, mounted on the outside of the transporter, can be operated to stop the transporter.

A transporter engine failure will result in the vehicle stopping or the hydraulic brakes engaging to stop any lift operations in progress.

A loss of hydraulic fluid will cause a loss of pressure in the hydraulic system that will engage the hydraulic brakes and stop movement of the lifting apparatus.

8.1.7.4 Corrective Action for Cask Transporter Off-Normal Operation

The corrective action for cask transporter off-normal operation will be developed and implemented based on the nature and safety significance of the problem. Corrective actions may include additional training for the driver, replacement of the driver, improved operating procedures, and repair or replacement of failed mechanical parts. The transporter is designed "fail-safe" to preclude uncontrolled lowering of the loaded transfer cask or overpack if a failure of an active component occurs, so no corrective actions related to the cask are necessary. If necessary, cribbing could be used to support the loaded transfer cask or overpack if the transporter needs to be replaced or detached from the load for repairs.

8.1.7.5 Radiological Impact of Cask Transporter Off-Normal Operation

The cask transporter off-normal event has no radiological impact since the confinement barrier is not breached and shielding is not affected.

REFERENCES

1. ANSI/ANS 57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (dry type), American National Standards Institute.

2. Standard Review Plan for Dry Cask Storage Systems, USNRC, NUREG-1536, January 1997.

3. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 0, July 2000.

4. License Amendment Request 1014-1, Holtec International, Revision 2, July 2001, including Supplements 1 through 4 dated August 17, 2001; October 5, 2001; October 12, 2001; and October 19, 2001; respectively.

5. Normal, Off-Normal, and Hypothetical Dose Estimate Calculations, USNRC, Interim Staff Guidance Document-5, May 2000.

6. Control of Heavy Loads at Nuclear Power Plants, USNRC, NUREG-0612, July 1980.

8.2 ACCIDENTS

8.2.1 EARTHQUAKE

An earthquake is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9 (Reference 1). The effects of seismic events on cask loading operations inside the fuel handling building/auxiliary building (FHB/AB) are discussed in the 10 CFR 50 License Amendment Request submitted in support of Diablo Canyon ISFSI licensing. This section addresses the effect of a seismic event on the operations related to the Diablo Canyon ISFSI that occur outside the FHB/AB. Cask handling activities outside the FHB/AB were reviewed to identify potential risk significant configurations during a seismic event. The seismic evaluations address the following potentially seismic risk significant configurations (all configurations are analyzed with an MPC loaded with spent fuel):

(1) HI-TRAC transfer cask suspended horizontally from the cask transporter on the transport route between the FHB/AB and the cask transfer facility (CTF).

(2) HI-TRAC transfer cask suspended vertically from the cask transporter at the CTF, prior to being placed atop the HI-STORM 100SA overpack.

(3) HI-TRAC transfer cask mounted atop the HI-STORM 100SA overpack at the CTF and the transporter restrained to the ground. The overpack is in the fully lowered position in the CTF.

(4) HI-STORM 100SA overpack being transported to the ISFSI storage pad, suspended vertically from the cask transporter. In terms of seismic stability, this configuration bounds configuration (2) because the HI-STORM 100SA overpack is heavier than the HI-TRAC transfer cask.

(5) HI-STORM 100SA overpack anchored to the ISFSI storage pad in its long-term storage configuration.

Additionally, the slopes above the ISFSI and transport route were analyzed for stability during a seismic event (see Section 2.6.5).

8.2.1.1 Cause of Accident

Earthquakes are natural phenomena caused by the movement of large geological plates under the earth's surface.

8.2.1.2 Earthquake Accident Analysis

Two methods were used for seismic analysis of SSCs, that is, equivalent static analysis load method and dynamic analysis method. These methods were used as follows:

Equivalent Static Analysis Method

(1) Design of CTF reinforced concrete support structure.

(2) Pad design.

(3) Design of CTF shell structural steel.

Dynamic Analysis Method

(1) Determination of slope stability.

(2) Determination of transporter stability while carrying a transfer cask or loaded overpack.

(3) Determination of ISFSI storage pad sliding.

(4) Design of storage cask anchorage to the pad.

As discussed in SAR Section 2.6.2.2, the design earthquake (DE), double-design earthquake (DDE), Hosgri earthquake (HE) and Long Term Seismic Program (LTSP) earthquakes are the DCPP seismic licensing basis. The DE and DDE spectra are defined for periods up to 1 second. The Hosgri spectra are defined for periods up to 0.8 seconds. The LTSP spectra are defined for periods up to 2 seconds.

The statistically independent free-field DE, HE and LTSP ground acceleration time histories in two horizontal and vertical directions were regenerated and updated based on the free-field response spectra and time histories from strong ground motion recorded at the Lucerne Valley site from the June 28, 1992 Landers magnitude 7.3 earthquake and from a rock site located approximately 8 km fault rupture distance from the September 20, 1999 Chi Chi magnitude 7.6 earthquake. These time histories are referred in this SAR as the DE, DDE, HE and LTSP time histories. The DDE is twice the DE. The regenerated DE, DDE, HE and LTSP free-field time histories meet the NRC Standard Review Plan (SRP) spectral matching criteria, Section 3.7.1 of NUREG-0800, (Reference 2) and the three components of the time-histories for each earthquake were verified to be statistically independent in accordance with ASCE 4-86 (Reference 3). The spectra generated from the time-histories were compared to existing DCPP DE, DDE, HE, and LTSP ground spectra. The regenerated DE, DDE, HE, and LTSP time histories were used in the seismic time history analysis of the cask anchorage; since the storage cask is anchored to the ISFSI storage pad long period energy will have a negligible impact on the analysis results.

As discussed in Section 2.6.2, PG&E developed the ISFSI Long Period (ILP) earthquake spectra to be used for the analyses of transporter stability, slope stability and ISFSI storage pad sliding to provide extra design margin since these analyses' results could be affected by long period energy. The ILP are 84th percentile spectras at damping values of 2 percent, 4 percent, 5 percent and 7 percent for the horizontal and vertical components that extend out to 10 seconds and which include near fault effects of directivity and fling. The ILP spectra envelops the DDE spectra at 2 percent and 5 percent damping, the Hosgri spectra at 4 percent, 5 percent, and 7 percent damping, and the LTSP spectra at 5 percent damping. Five sets of spectrum compatible time histories were generated from recordings of large magnitude earthquakes ($M > 6.7$) recorded at short distances (< 15 km from the fault), and they contain a range of characteristics of the near fault effects.

The modal damping ratios expressed as a percentage of critical damping for the seismic analyses are provided in Table 8.2-1. These damping values are from the DCPP FSAR Update (Reference 4). The analysis approach, results, and conclusions for each of the configurations are discussed separately below.

8.2.1.2.1 Seismic Evaluation of Operations Involving the Cask Transporter - Seismic Configurations 1, 2 and 4

This section discusses the seismic stability evaluation of the spent fuel cask transporter used at the Diablo Canyon ISFSI.

The HI-TRAC transfer cask, containing a loaded MPC, exits the FHB/AB on the cask transport frame in a horizontal orientation. The cask transporter lifts the HI-TRAC and the horizontal lift rig and moves along the road approximately 1.2 miles to the ISFSI storage site, in the process traversing an 8.5 percent (nominal) grade decline and climbing a 6 percent (nominal) grade incline. Figure 4.3-1 shows the cask transporter/transfer cask during this operational mode. At the CTF upending site, the transfer cask is rotated by the cask transporter to a vertical orientation and then moved to the CTF. Figure 4.3-2 shows the HI-TRAC transfer cask in the vertical orientation prior to mating to the overpack. After the MPC transfer operation is executed, the cask transporter carries the loaded overpack in a vertical orientation to its final position on the ISFSI storage pad. Figure 4.4-3 shows the loaded HI-STORM 100SA overpack en route to the ISFSI pad.

The transport route is approximately 1.2 miles long, approximately one third on bedrock and the remaining two thirds crossing surficial deposits over bedrock that would cause amplification of the ground acceleration (Section 2.6.2.4). The transporter has a minimum speed of 0.8 miles per hour. The time the transporter is on the transport route with a loaded cask, based on 8 transports per year times and 1-1/2 hour per transport, is 12 hours per year. Ground motions during the 12 hour cumulative annual transport time sufficient to result in a cask drop, overturn the transporter or cause the transporter to slide off the roadway is judged to not be credible.

A transporter stability analysis (Reference 31), described below, was performed for bedrock ground acceleration associated with the ILP earthquake. The analysis determined that the transporter would not overturn or leave the roadway. (Configurations 1, 2, and 4, and a portion of Configuration 1). Although the transporter route crossing surficial deposits would result in amplification of the ground acceleration, a significant margin of safety exists from transporter overturning or sliding off the roadway as discussed below.

Methodology - Stability on bedrock

The ILP seismic events for the Diablo Canyon ISFSI, described in Section 8.2.1.2, were evaluated and analyzed for the transporter stability analysis. Five sets of ILP time-histories were used to demonstrate transporter stability as it carries a loaded cask on the transport route. As discussed in Section 2.6.2.1, the ILP spectra and associated time histories are appropriate for use along the transport route.

VisualNastran 4-D (VN) (formerly Working Model 4-D) (Reference 5) serves as the simulation engine to obtain the response to the 3-dimensional seismic events. This computer code has previously been used in licensing the HI-STORM 100 System as described in the HI-STORM 100 System FSAR (Reference 6).

The time-domain dynamic simulations model the cask transporter, the HI-STORM 100SA overpack, the HI-TRAC transfer cask, the MPC (including the fuel basket, fuel, and lid), and the cask lids as rigid bodies. The mass of the MPC and the contained spent fuel is lumped in a free-standing rigid cylinder that, during the earthquake, is free to rattle in the cask cavity.

The cask transporter sits on grade that is subjected to a ground acceleration time history appropriate to the free field ILP event. The simulations use the Holtec generic model of the cask transporter with a track width identical to that planned for the Diablo Canyon cask transporter, but with a reduced track length. This ensures that the results from the dynamic simulations will conservatively bound the response of the real system using a transporter with a longer track length along the roadway.

Acceptance Criteria

The cask transporter plus its carried load must remain stable (not overturn) and remain on the travel path under all seismic events applicable to the Diablo Canyon ISFSI site. The minimum roadway width is 26 ft, which sets the allowable transporter lateral sliding distance. The maximum acceptable sliding movement along the roadway is limited to the DCPP cask transporter track length to ensure that the transporter will remain on the roadway after exiting a turn in the roadway.

Assumptions

The following key assumptions were employed to construct the models for the simulations:

(1) The time domain dynamic analyses of the transporter seismic stability simulate the modeled components (cask transporter, transfer cask, overpack and MPC) as rigid

bodies with specified geometry and bounding mass. The connections between the cask body and the lids were assumed to be rigid. These are conservative assumptions for the seismic analysis since energy dissipation in the dynamic system is neglected by virtue of the rigid body modeling.

(2) The time domain dynamic simulations model the MPC and the contained fuel by a solid cylinder with total mass that bounds the heaviest PWR MPC-32 (90,000 lb). This is conservative since all energy dissipation due to fuel assembly rattling inside the MPC is neglected and any reduction in amplitude due to chaotic fuel assembly motion over time is ignored.

(3) The analyses in time domain are simplified by assuming the rigid bodies to have uniform mass density when calculating their mass moments of inertia and mass center locations. Any shift in the centroid due to this assumption has a negligible effect on the results of the analysis.

(4) The coefficient of restitution for the internal contact surfaces (MPC/overpack) is set to zero. The coefficient of restitution between the transporter treads and the ground was set to 0.0 - 0.25 (the exact value has no influence on the solution when sliding motions predominate). For the coefficient of friction at the transporter tread/ground interface, an upper bound value of 0.8 was conservatively assumed to emphasize tipping action. A lower bound value for the tread/roadway surface of 0.4 was assumed to determine the sliding behavior of the transporter. The coefficient of friction between the MPC and the HI-TRAC transfer cask cavity side surfaces is set at 0.5. This is realistic because experience indicates a variation from 0.8 down to 0.2 for steel-on-steel depending on the relative velocity between the two surfaces.

(5) The time domain dynamic simulations use a generic model of the cask transporter with a track length that is shorter than the length of the Diablo Canyon cask transporter tracks. The analyses considered the stability of the cask transporter when supported by a horizontal ground surface.

(6) In all stability analyses, the positioning of the cask in the cask transporter is set slightly higher than the anticipated carry height to ensure that overturning moments are conservatively computed at each time point during the dynamic simulations.

(7) All bodies are assumed to be rigid for the global analysis. The cask transporter design specification includes a requirement that the transporter be designed such that its lowest global natural frequency is in the rigid range (>33 Hz).

Key Input Data

The key input data used in the cask transporter seismic analyses are shown in Tables 8.2-2 through 8.2-4. Input time histories used for the dynamic simulations are five sets of ILP design earthquake excitations. These seismic events are identified below with their duration:

Set 1: Lucerne Valley (48 sec)

Set 2a: Yarimca (40 sec)
Set 3: LGPC (22 sec)
Set 5: El Centro (40 sec)
Set 6: Saratoga (40 sec)

Results of Analyses, transporter on bedrock

A series of nonlinear dynamic simulations were performed using the VisualNastran 4-D computer code to assess the seismic stability of the cask transporter during the five ILP earthquakes. Table 8.2-5 lists the simulations performed for the stability evaluation. The combinations of grade, coefficient of friction, and seismic events have been chosen to be bounding for the site-specific conditions.

For each case considered, the loaded transporter was assumed to be on a flat or inclined surface with specified coefficients of friction. The simulations performed under Phase 1 serve to identify potentially bounding events from among the five candidate time histories. The choice of simulations for the remaining phases was based on the results from the simulations in Phase 1. The combination of grade and coefficient of friction were chosen to induce sliding as opposed to tipping.

Table 8.2-6 summarizes the estimates of the maximum transporter horizontal excursions in the transverse and longitudinal direction for each phase of the dynamic simulations performed. The reported maximum excursions are at the top of the transporter relative to the ground.

These results are bounding for all Diablo Canyon cask transporter operational modes and for all ILP earthquakes. The maximum value of 10.7 inches reported for the transverse excursion with a coefficient of friction of 0.4 demonstrates that in the event of seismic excitation, the transporter will not leave the road while moving from the FHB/AB to the Diablo Canyon CTF or while moving from the CTF to the ISFSI. The small relative movements reported for the case with friction coefficient of 0.8 demonstrate that overturning of the loaded cask transporter is not a credible event under the ILP seismic events. For the case where the transporter is on the 8.5 percent grade when the seismic event is postulated to occur, the results demonstrate that, the maximum sliding movement along the axis of the road (30.2 inches) is less than one transporter track length. In addition, the transverse movement of the transporter during a seismic event is small, 10.7 inches, compared to the distance between the edge of the transporter and the edge of the roadway (roadway minimum width is 26 ft and the width of the transporter from outside of track to outside of track is approximately 17.5 ft), provides additional margin of safety.

The time domain dynamic simulations of the cask transporter demonstrate that the cask transporter, carrying either a loaded HI-TRAC transfer cask in the horizontal orientation or a loaded HI-STORM 100SA overpack in the vertical orientation, will not overturn during a seismic event and will not leave the road while moving from the FHB/AB to the CTF or from the CTF to the storage pads. When the transporter is carrying a HI-TRAC horizontally, up or down the 8.5 percent grade, the magnitude of sliding displacement along the axis of the road is less than the length of the transporter track.

Cask drop during transport (seismic)

As discussed in Section 8.2.4, the load path portions of the cask transporter and the lifting devices attached to the cask components will be designed to preclude drop events, either through redundancy or enhanced safety factors. The design will include consideration of seismic loads. Therefore, a seismic event occurring during transport would not result in a cask drop. In addition, Holtec has qualified the HI-TRAC with an MPC for a horizontal cask drop of 42 inches (Section 3.4.9 of Holtec's Hi-STORM 100 System FSAR).

8.2.1.2.2 Seismic Analysis of Cask Transfer Facility Seismic Configuration 3

8.2.1.2.2.1 CTF Steel Structure

The CTF at the Diablo Canyon ISFSI is used in conjunction with the cask transporter to perform MPC transfers between the HI-TRAC transfer cask and the HI-STORM 100SA overpack. Prior to the transfer operation, the empty HI-STORM 100SA overpack is placed in the CTF. The overpack is lowered to the full down position in the CTF and a mating device is installed on the top of the overpack. This mating device serves as a structural connection and an alignment device between the top of the overpack and the bottom of the HI-TRAC transfer cask. The transfer cask is positioned over the overpack by the cask transporter, which remains in position during the transfer operation. Restraints are used to secure the cask transporter to ground during the MPC transfer operation.

The cask transfer facility is shown in Figure 4.4-3 and includes the following main structural components:

Main Shell – A cylindrical shell is positioned into a larger vertical hole in the rock with concrete backfill providing an interface connection with the rock walls of the hole. The bottom of the shell is anchored to a reinforced concrete base. This cylindrical shell serves as the cavity liner into which the overpack is lowered and provides the support for the lifting jacks and a set down location for the lifting platform when the lifting platform is fully lowered. Three vertical stiffening extensions (U-shaped) run the length of the cylinder shell and act as the main structural members that transfer the loads from the lifting jacks to the shell and down to the base. Restraints are installed at the top of the shell, which serve to restrain the cask under lateral loads from seismic events.

Lifting Jacks – Three lifting jacks are used to raise or lower the lifting platform. They are located in the three vertical stiffening extensions on the circumference of the main shell. The lifting jacks are supported at the top end and have traveling nuts that operate in unison to keep the platform level.

Jack Supports – Jack platform plates and gussets are welded to the top of the shell extensions to provide support for the lifting jacks.

Lifting Platform – A lifting platform of built-up plates provides vertical support of the HI-STORM 100SA overpack and transmits the load to the lifting jacks. During the lifting operation, a uniform loading of the lifting platform is afforded by the location and controlled movement of the lifting jacks. Support plates together with the top and bottom platform plates form the lifting platform structural frame. A cover plate covers the lifting platform plate and provides a base on which the overpack rests. The lifting platform has extensions that reach into each main shell stiffening extension to interface with the lifting jacks. Gussets are welded to the platform outer plates to provide a stiff structural member in the vicinity of the lifting jacks.

Reinforced Concrete Support Structure -The CTF steel structure is placed on a steel reinforced concrete foundation slab and surrounded by heavily reinforced concrete up to the surface. The concrete structure will carry all the compressive loadings on the base and the side-walls (cylindrical in shape) to the ground rock. The structure will have an adjoining gravity fed sump for drainage.

This section discusses the seismic structural analyses and evaluations of the CTF at the Diablo Canyon ISFSI. The capacity of the CTF structural components is evaluated including the lifting jacks, the jack supports, the shell extensions, and the lifting platform. The calculations provide the loads on the CTF base, CTF shell, and surrounding concrete under the specified ASME Section III (Reference 7), Subsection NF service (Level A and Level B) load conditions and Appendix F seismic (Level D) load conditions. A description of the analysis of the reinforced concrete support structure is also included.

<u>Methodology- Structural Analysis</u>

The analysis (Reference 32) evaluates the capacity of the CTF structural components under static loads (dead weight and factored dead load) and under static plus seismic and wind loads. Bounding values for the weights of the spent fuel casks and canisters are used to evaluate the dead loads applied on the CTF structure. In accordance with the HI-STORM 100 System FSAR (Reference 6), the dead loads incorporate an inertia amplification of 15 percent during the lifting operation (factored dead load). Quasi-static stability analyses provide the magnitudes of the seismic loads on the CTF steel structure during the governing LTSP earthquake excitation. The natural frequencies of the cask transporter, the HI-TRAC transfer cask, and the HI-STORM 100SA overpack stack was calculated. The actual horizontal spectral acceleration value corresponding to 19.85 Hz was used in the seismic analysis. Under vertical excitation, the ground vertical zero period acceleration value was used in the seismic analysis since the stacked configuration is rigid in the vertical direction. Examination of the response spectra for the four DCPP seismic events (DE, DDE, HE and LTSP) shows that the bounding spectral accelerations for CTF structural design are those from the LTSP spectra.

The analysis considers the most critical combinations of design loads for the loading scenario wherein a loaded HI-TRAC transfer cask is stacked on top of the HI-STORM 100SA overpack in the full down position (Configuration 3) (Figure 4.4-4).

The seismic analysis considers two critical combinations of the specified design earthquake components when the CTF structure is subjected either to upward vertical inertia forces or downward vertical inertia forces. The Newmark 100-40-40 Method is used to combine the three specified directions of the seismic load.

Using the calculated inertia loadings together with known dead loading, strength-of-materials solutions from the theory of elasticity are used to determine the stresses in the CTF structural components and weld connections. The ratio of the allowable stresses to the calculated stresses in the components and welds defines safety factors for service (Level A) and seismic (Level B and Level D) load conditions.

Acceptance Criteria

The stresses in the CTF structural components and welded connections under the service loads must be below the limits prescribed in ASME Section III, Subsection NF (Level A and Level B). The stresses in the CTF structural components and welded connections under the combination of dead plus seismic loads must be below the limits prescribed by ASME Section III, Appendix F (Level D).

The lifting jacks, as the primary load-bearing components, must meet the design criteria of Section 4.2 of ANSI N14.6 (Reference 8) and Section 5.1.6 of NUREG-0612 (Reference 9) applied to the lifted load, including any dynamic effects.

The seismic connectors at the CTF (cask transporter to ground, and between the transfer cask and the overpack) must have sufficient structural capacity to prevent extensive motions of the transfer cask during MPC transfer operations that could put the contained fuel at risk. The load capacity of all necessary connectors is designed to meet the applicable limits of ASME Section III, Subsection NF and Appendix F.

Assumptions

The following conservative assumptions are employed in the linear elastic structural analyses:

- The stability analysis of the CTF shell extensions conservatively neglects any contributory stiffening from the main shell and ignores the support from the concrete fill between the shell and the rock walls.
- The structural analysis of the lifting platform built-up plate structure is conservatively analyzed as a beam structure, thus neglecting any two-dimensional plate bending that would decrease the computed stress.

Key Input Data

The key input data used in the CTF seismic analyses are shown in Table 8.2-7. The seismic inputs for the analyses are obtained from ground acceleration response spectra for DCPP. The ZPAs for the vertical direction were used because the stacked casks in the CTF are rigid (>33 Hz) in the vertical direction. The spectral accelerations in the horizontal directions corresponding to 19.85 Hz were used. The ZPAs and spectral accelerations used in the analysis are shown in Table 8.2-8. Where load combinations are required for the strength evaluation, the Newmark 100-40-40 Method (for LTSP seismic event) is used to combine the three specific directions of the seismic load.

Results of Analyses

The results from the CTF structural analyses demonstrate that all structural members and welds stresses satisfy the condition that safety factors are greater than 1.0. Safety factors are defined as:

SF= (Allowable stress or load)/(Calculated stress or load).

In addition to the structural analysis of the CTF components, mandated by the appropriate design codes, analyses of the connector restraints (that inhibit relative movements between the cask transporter and ground) and the mating device (between the transfer cask and the overpack) will also be performed to ensure that any relative motion between the transfer cask and the overpack during the cask transfer operation will not compromise the integrity of the MPC. Load/stress limits on these ancillary items meet applicable requirements of Subsection NF and Appendix F. In order to optimize the design of connector restraints and mating device, it may be necessary to restrain the HI-TRAC transfer cask to ground.

8.2.1.2.2.2 CTF Reinforced Concrete Support Structure

Methodology - Structural Design/Analysis

A static analysis (Reference 30) was performed to appropriately size the base slab and the side cylindrical wall to accommodate the applied forces generated by the CTF as discussed in 8.2.1.2.2.1.

Acceptance Criteria

ACI-349 97 (Reference 10), in compliance with NUREG-1536 (Reference 11), concrete stress allowables and DG-1098, as applicable are used.

Assumptions

None

Key Input Data

The surrounding rock properties and the functional requirements of the CTF steel structure (as described earlier in this section) and the loads developed in the CTF analysis (Section 8.2.1.2.2) are the key input parameters.

Results

The reinforced concrete structure meets the stress requirements of ACI 349-97 and the functional requirements of the facility.

8.2.1.2.3 Seismic Analyses of the HI-STORM 100SA Overpack Anchored to the ISFSI Storage Pad in its Long-Term Storage Configuration Seismic Configuration 5

8.2.1.2.3.1 Cask and Anchorage Seismic Analysis

The HI-STORM 100SA overpack design differs from the HI-STORM 100S only in that it includes an extended bottom flange and gussets that enhance the structural resistance of the flange/shell around the bottom periphery of the overpack (see Figure 4.2-7). This flange includes a bolt circle to permit structural "mating" of the overpack to the ISFSI storage pad steel embedment plate by 16, 2-inch diameter, SA193-B7 preloaded cask anchor studs. The preloaded cask anchor studs are threaded into compression/coupling blocks to ensure a continuous compressive state of stress at the interface between the lower surface of the HI-STORM 100SA overpack and the top surface of the embedment plate. The continued contact ensures development of interface friction forces sufficient to resist lateral movement of the overpack base relative to the embedment plate. It also ensures that the ISFSI storage pad embedment structure provides the resisting moment to stabilize the system under seismic loading. The cask anchor studs are threaded into compression/coupling blocks that bear against the lower surface of the embedment plate from the action of the preload. The embedment plate is held to the concrete by 16 longer embedment anchor rods that are threaded into the same compression/coupling blocks, but are not preloaded. The embedment anchor rods are only loaded, as the seismic event proceeds, to the extent necessary to maintain vertical force and moment equilibrium. Oscillations in the cask anchor stud load are minimized due to the presence of the initial preload. Figure 4.2-2 shows a section depicting the embedment plate, the compression block, the cask anchor studs and the embedment anchor rods. The cask is not shown in this figure.

The cask and anchorage seismic analyses are not sensitive to long period ground motion. Therefore, these analyses were performed using the four DCPP seismic events (DE, DDE, HE, and LTSP). The DE, DDE, HE, and LTSP are characterized by free-field acceleration time-histories, in each of three orthogonal directions, with durations of 41 seconds for the DE and DDE cases and 48 seconds for the HE and LTSP cases. The HE and LTSP events have the highest, zero-period accelerations, and the largest, free-field excursions. Therefore, the results from these events are bounding and the dynamic simulations to obtain time-history behavior of the system are performed using the VisualNastran (VN) simulation code described previously only for these two events.

Methodology

The dynamic model of the HI-STORM 100SA overpack in VN consists of the following major components:

(1) The HI-STORM 100SA overpack plus the embedment plate is modeled as a six degree-of-freedom (rigid body) component.

(2) The loaded MPC is also modeled as a six degree-of-freedom (rigid body) component that is free to rattle inside the overpack shell. Gaps between the two bodies reflect the nominal dimensions from the design drawings in Reference 12.

(3) The embedment anchor rods provide the vertical connection between the embedment plate and ISFSI slab. The embedment anchor rods are modeled as individual linear springs connecting the periphery of the extended baseplate to the ISFSI storage pad section. The concrete pad/embedment compression resistance at the interface is simulated with compression-only stiffness elements around the periphery.

(4) For the global dynamic analysis of the anchored cask, the slab section under the cask is assumed rigid and the three components of acceleration and time-history are applied simultaneously at the base of the slab. Since the HE and LTSP events provide the bounding loads to the anchorage, the importance of directional effects on the responses is evaluated for both the HE and LTSP events by repeating the simulations with the only change being the negative of the vertical seismic time history is used in conjunction with the specified horizontal time histories.

(5) The contact between the MPC and the overpack is simulated by a classical impulse-momentum equation. The coefficient of restitution (COR) is set to 0.0 reflecting the large contact areas involved and the coefficient of friction is set to 0.5, which is representative of steel-on-steel. This is a realistic assumption and allows for energy loss during contact between the two, large rigid bodies.

(6) The interface contact between the base of the overpack and the ISFSI storage pad embedment is modeled by discrete linear springs to simulate the embedment anchor rods and by compression-only elements to simulate the balancing force from the embedment. The spring rates are computed using established methodology for embedment anchor components. Damping is consistent with that specified for steel and concrete components in Table 8.2-1. These are realistic assumptions that appropriately model the expected interface behavior.

(7) Bounding (high) weights for the cask components are used for conservative results; inertia properties are computed consistent with these bounding weights.

Each VN dynamic simulation produces time-history results for the tensile loads in each of the 16 embedment anchor rods, as well as time-history results for the total interface compression load between the base of the embedment plate and the ISFSI pad concrete. The results of the VN-time-history analyses are stored in spreadsheet form and a FORTRAN computer code is used to post-process the results to determine vertical-load and overturning-moment time-histories for subsequent structural-integrity evaluation. Figure 8.2-1 shows an expanded model of the components (excluding the 16 anchor rods) that make up the dynamic model.

To ensure the capture of all energy from a seismic event, while at the same time eliminating high frequency components not pertinent to satisfying Code requirements in a structural evaluation, the filtering frequency for processing the "raw" numerical results is set as 40 Hz. The use of filtering of dynamic results in cask structural integrity analysis has been previously licensed for the HI-STORM 100 System as described in the HI-STORM 100 System FSAR.

<u>Acceptance Criteria:</u>

The design criteria for the HI-STORM 100 SA overpack are discussed in Chapter 2 of the HI-STORM 100 System FSAR, as amended by Holtec LAR 1014-1 (Reference 12). The anchorage system, being an integral part of the overpack structure, is subject to the same design requirements. The anchorage (cask anchor studs, sector lugs, and adjacent shell structure) is designed to meet the static stress limits of ASME Section III, Subsection NF and Appendix F (Reference 7).

Two conditions for analysis are defined as follows:

(1) Level A (Preload) - The anchor stud preload is established at approximately 157 kips in each stud. Under this load and the corresponding balancing load from the ISFSI storage pad, the sector lug structural components must meet the allowable stress limits for plate and shell structures given in Article NF-3200. The stress limits at 200°F for SA-516, Grade 70 material (used for the sector lugs) listed in Table 3.1.10 of the HI-STORM 100 System FSAR are used in the acceptance evaluation.

(2) Level D (Preload plus Seismic Load) - In accordance with Appendix F of ASME Section III, the tensile stress in the stud, averaged through the cross-section is limited to 70 percent of the ultimate strength of the stud material. The extreme fiber stress in the stud is limited to ultimate strength per F-1335.1. The design criteria and stress intensity limits for the sector lug components are given in Chapter 2 and Table 3.1.12, respectively, of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. The stud alternating stress intensity under the dynamic loading induced by the seismic event must be sufficiently low to ensure a safety factor greater than 1.0 against fatigue failure for the number of stress intensity cycles associated with the seismic event.

In addition to the above anchorage acceptance criteria, it is required to demonstrate that the seismic events do not induce acceleration levels in the body of the cask that exceed the cask design basis (45 g) as defined in the HI-STORM 100 System FSAR (Reference 6).

Assumptions

The key assumptions used in the dynamic model are listed and explained within the methodology description given above.

Key Design Inputs

Bounding weights of 270,000 lb for an empty HI-STORM 100SA and 90,000 lb for a loaded MPC are used in the analyses (References 6 and 12, Table 3.2.1). SA193-B7 material is used for the anchor stud material. For the dynamic analyses, anchor stud minimum yield and ultimate strengths of 105 ksi and 125 ksi, respectively, are used. Dimensions for the two cask bodies are taken from Drawing 3187 in Reference 12. Mass moment of inertia properties are determined based on cylindrical body assumptions with the specified mass uniformly distributed.

The spring rate of the embedment anchor rods is equivalent to a 2-inch diameter carbon steel rod, 48 inches long.

Seismic inputs for the dynamic analyses are obtained from acceleration time histories developed from the response spectra for each of the DCPP earthquakes.

Results of Analyses

The results from the series of analyses performed for the anchored cask can be summarized as follows:

(1) The anchored HI-STORM 100SA overpacks do not exceed the generic cask design basis deceleration limit of 45 g under any of the seismic events.

(2) The state of stress in the cask anchor studs and in the overpack bottom flange, gussets, and the shell structure remain below the stress limits of ASME Section III, Subsection NF and Appendix F under all seismic events.

(3) The interface loads on the embedment structure determined for the ISFSI pad structural qualification are summarized in Table 8.2-9. The peak values are obtained from the filtered, time-history results for embedment anchor rod tension and for interface compression from the dynamic simulations.

A finite element analysis of the sector lug was performed using as input the tensile load in the cask anchor stud. Structural integrity evaluations were performed for both Level A (where the preload is balanced by compression between the extended flange and the embedment plate) and for Level D conditions (where local lift-off of the flange is assumed and the stud maximum load capacity is conservatively assumed). The results from the finite element analyses are reported in Table 8.2-10.

The maximum values obtained for the interface loads at the embedment structure are summarized in Table 8.2-9 and form the input for the structural integrity evaluation of the ISFSI pad.

The bounding cask weight is 360 kips. Using the maximum net shear force result from Table 8.2-9 and dividing by the cask weight provides the effective "g" loading on the cask as 1.43 g. This demonstrates that the cask design basis deceleration level (from the HI-STORM 100 System FSAR) of 45 g is not exceeded with a large margin of safety.

The results summarized in Table 8.2-9 provide the information needed to determine the coefficient of friction required at the cask/embedment plate interface to ensure that there is no relative sliding at that location. These results are obtained by dividing the net filtered shear force by the filtered normal force at each instant of time through the simulation. From the simulations performed, the largest required value for the coefficient of friction is 0.18. In accordance with the ASME Code (NF-3324.6, Table-3324.6(a)(4)-1), a minimum coefficient of friction of 0.25 may be assumed to exist at the interface when preload is used. Therefore, the minimum safety factor against sliding of the cask relative to the embedment plate is 1.39 and the desired benefit of the preload is assured.

To evaluate the propensity for a failure by fatigue in the sector lug, the results from the finite element stress analysis of the sector lug under the limiting tensile load was used. Using the recommended methodology for fatigue analysis as outlined in ASME Section III and determining the likely number of stress cycles by using the results from the dynamic analyses, large margins of safety against a fatigue failure during a single seismic event were obtained. Therefore, fatigue failure of the overpack anchorage is not credible at the Diablo Canyon ISFSI.

To ensure continued maintenance of the design bases assumptions for preloading of the anchorage connections, PG&E will develop an inspection program that periodically visually checks a sampling of the exposed portions of the anchor studs, washers, nuts, and storage cask baseplate surrounding the nuts to note any degradation or relaxation of these connections. This program will verify that the studs, washers, and nuts have not turned from their as-left preloaded position, are not loose to the touch, and that visually their mating surfaces remain engaged. This verification will be performed as part of the 10 CFR 50 DCPP Maintenance Rule Program developed for compliance to the maintenance rule and will have similar periodic inspection requirements.

8.2.1.2.3.2 Storage Pad Seismic Analyses

The objective of the seismic analyses of the concrete pad is to ensure that the steel reinforced concrete pads and the anchored casks remain functional during all seismic conditions. A static analysis was performed to determine the storage pad size and thickness required to resist the loads resulting from seismic accelerations (DE, DDE, HE, and LTSP ground zero period acceleration [ZPAs]) applied to the pad, in addition to the resultant loads from the cask dynamic analysis (Section 8.2.1.2.3.1). Also, a nonlinear time history analysis of the cask/pad set-up was performed to determine the extent of sliding that occurs at the pad/rock interface.

Pad Static Analysis

Methodology

The analysis is a nonlinear static finite element analysis (FEA), Using the ANSYS computer code. The storage pad size and thickness analysis is not sensitive to long period ground motion. Therefore, this analysis (Reference 29) was performed using the four DCPP seismic events (DE, DDE, HE and LTSP). The seismic inputs used for this analysis were HE and LTSP ZPAs. The HE and LTSP spectra were used since these spectra produce the largest ZPAs and the cask/pad interfaces are not sensitive to longer period ground motion. The concrete slab was allowed to lift off the rock support if the loads and geometry dictate that liftoff should occur. All material properties are linear. Compression only gap elements are used at the interface between the slab and the rock. This is the only nonlinear modeling feature in the analysis.

The FEA model consists of the pad, portion of the underlying rock, and elements representing the cask on top of the pad. The casks are modeled up to a plane, 118.5 inches above the slab. This is the location of the center of gravity of the casks and is, therefore, where the loads are applied. The pad uplift and concrete stresses are determined by the FEA analysis. The steel embedment/anchorage structure is designed to meet the ductile anchorage provisions of the proposed new draft Appendix "B," dated October 01, 2000, to ACI-349-97. Other provisions of Appendix B are not applicable due to the thickness of the pad and length of the rod. Specifically, design strength capacity of the embedded base plate; concrete bearing and diagonal tension shear capacity computed must be more than the required ductile design strength of the embedded rod/stud. The Newmark 100-40-40 Method is used to combine the three specified directions of the seismic load.

Acceptance Criteria

Concrete and the embedded steel structures, are designed to the requirements of ACI-349-97 and ductility provisions of Draft Appendix "B" dated October 01, 2000, and NUREG-1536 (Reference 13).

Assumptions

Normal engineering assumptions associated with developing FEA models (for example, boundary conditions, modeling techniques). The anchorage evaluation methodology used assumes the loading imposed on the pad embedment structure is similar to an inverted column and as such diagonal shear provisions of the ACI, Section 11.3, were followed.

Key Input Data

Table 8.2-9 shows the resultant cask loading on the pads. The underlying rock material properties have an impact on the analysis. The rock's Young's modulus range of 0.2 x 10^6 psi to 2.0 x 10^6 psi were considered in the analysis to account for variability of the rock types.

Results

The maximum pad stresses and the embedded steel ductility requirements meet the ACI 349 code requirements. The yield strength of the embedded studs is greater than 250 percent of the computed demand load on these studs. The maximum potential uplift on an edge of the pad is less than 1/32 inch to 1/8 inch, depending on the variation in the rock properties.

Pad Sliding Dynamic Analysis

Methodology

A nonlinear time history analysis of the cask/pad structure sliding at the rock/pad interface was performed (Reference 28). Analyses were performed with the five sets of ILP time histories. The ILP time histories were used since the pad sliding analysis may be sensitive to long period ground motion and the use of ILP time histories produces bounding results.

A nonlinear stick model is developed for the purposes of these analyses. A lollypop stick model representing the cask behavior represents the set of 20 casks on a pad. The pad is represented by its mass only. The interface between the rock and the pad surface is modeled using SAP2000N's NLLINK element with friction properties. This element is a biaxial friction element that has coupled friction properties for the two shear deformations, post-slip stiffness in the shear directions, gap behavior in the axial direction. The cask superstructure stick is modeled such that it represents the dynamic properties of the anchored cask. [The cask and anchorage seismic analysis described in Section 8.2.1.2.3.1 models the anchored cask (in the absence of sliding of the pad) and perform dynamic analysis to predict the cask/pad interface design shears, moments, tension, and compression forces to be used in the pad design.] The fundamental frequency of the cask superstructure in sliding analyses is based on best estimate of the rocking frequency of the anchored cask. In the absence of local nonlinearities, it is expected that the fixed base model (no pad sliding) of the cask will yield slightly more conservative results than Section 8.2.1.2.3.1 results. The same model when mounted on the friction element is called the sliding model. The relative ratio of peak response between the sliding model and the fixed base model will yield an adjustment factor, which if found to be greater than unity, would have to be applied to the design shears and moments predicted by the analysis described in Section 8.2.1.2.3.1. This approach identifies any potential increases in design responses due to sliding.

For the vertical direction, the tensile component of cask/pad reactions is studied. This component is judged to be an important parameter that controls the normal resisting force at the interface, thus affecting the sliding displacement during a seismic event.

All analyses are performed based on the nonlinear time-history analysis option using Fast Nonlinear Analysis (FNA) approach of SAP2000N computer FEA program.

<u>Acceptance Criteria</u>

The pad must maintain its ability to perform its functional requirements with insignificant impact on the cask design qualifications.

<u>Assumptions</u>

Net Vector sliding is conservatively calculated assuming simultaneous peak X and Y horizontal sliding displacements.

<u>Key Input Data</u>

The analysis was performed assuming two pad-to-rock interface sliding friction coefficients $\mu = 1.19$ corresponding to a friction angle of 50 degrees, and $\mu = 0.73$ corresponding to a friction angle of 36 degrees. This represents the range of the sliding friction coefficient expected at this interface.

Cask Weight:	W = 360 kips
No. of Casks on a pad	20

<u>Results</u>

Based on the results of these analyses, the following is concluded:

(1) The best estimate of maximum pad sliding for a lower bound friction coefficient of 0.73 corresponding to a rock friction angle of 36 degrees is estimated as 1.21 inches.

(2) The best estimate of maximum pad sliding for an upper bound friction coefficient of 1.19 corresponding to a rock friction angle of 50 degrees is estimated as 0.41 inches.

(3) The above pad sliding displacements are considered small and not large enough to cause any damage to the pad or the casks. The acceptance criteria for pad sliding is defined as whether pad sliding results in increased design shears and moments at the cask-to-pad interface, which is discussed further below.

(4) After pad sliding is considered, it is concluded that the cask design shear of 515 kips (load on to the pad) remains valid for design. The best estimate of the adjustment factor to account for the effects of pad sliding is calculated as 0.95 for a friction coefficient of 1.19, and 0.90 for a friction coefficient of 0.73. Both of these ratios are

below unity, as such the design shear of 515 kips (and associated moments) remains valid for design.

(5) The best estimate of maximum vertical tensile load after sliding remains unchanged. Thus the design axial bolt tensions of the analysis described in Section 8.2.1.2.3.1 remain valid.

(6) The response spectra comparison plots of the rock versus pad sliding indicate that the responses at the cask-to-pad interface generally do not vary up to about 16 Hz. However, above this frequency some differences in the responses are seen as a result of sliding. An evaluation by the cask supplier determined that there were no components inside the cask are sensitive to changes in input motion in this higher frequency range. The highest peak spectral ordinate associated with change in motion as a result of pad sliding is 4.1 g at approximately 26 Hz and 5 percent critical damping well below the cask qualifications.

(7) Given that the base shear (and therefore base moments) and axial tension do not change as a result of pad sliding, it is concluded that analyses described in Section 8.2.1.2.3.1 remain valid.

8.2.1.3 Earthquake Accident Dose Calculations

The HI-STORM 100SA overpack and the HI-TRAC transfer cask were explicitly analyzed for, and shown to withstand the seismic ground motion during transport to the CTF, during activities conducted at the CTF, during movement from the CTF to the storage pads, and during storage operations, as applicable. The seismic ground motion does not cause stresses above allowable limits in the MPC confinement boundary, the transfer cask, or the storage overpack during canister transport, transfer, or storage operations. The CTF and cask transporter structures are also designed to withstand the DCPP ground motion. No radioactivity would be released in the event of an earthquake and there would be no resultant dose.

8.2.2 TORNADO

A tornado is classified as a natural phenomenon Design Event IV, as defined in ANSI/ANS-57.9. This event involves the potential effects of tornado-induced wind, differential pressure, and missile impact loads on the ISFSI SSCs that are important to safety. The design basis wind and tornado evaluation is provided in Reference 27.

8.2.2.1 Cause of Accident

The cause of this event is the occurrence, at or near the ISFSI site, of meteorological conditions that are favorable to the generation of a tornado. The design-basis tornado wind speed for the ISFSI is based on a conservative estimate appropriate for DCPP (annual probability of 10^{-7}), which was developed by the NRC (SSER No. 7). The specific topography associated with the plant site indicates that the postulated tornado event is unlikely. However, it has been included in the ISFSI design basis as a potential accident event.

8.2.2.2 Accident Analysis

The accident analysis for tornado effects involves evaluation of the loaded transfer cask during transport to the CTF, MPC transfer activities at the CTF, transport of a loaded HI-STORM 100SA overpack to the ISFSI pad, and long-term storage of the loaded overpack at the ISFSI pad. As discussed in Section 3.2.1 and 4.2.3.3.2.6, tornado-wind and missile design criteria are a combination of Diablo Canyon site-specific winds and missiles and the design-basis missiles described in the HI-STORM 100 System FSAR. In the evaluation of the Diablo Canyon ISFSI for tornado effects, the missiles were categorized as large, intermediate, or small missiles and were compared with those missiles for which the HI-STORM 100 System was generically designed to withstand. The description, mass, and velocity of all missiles considered for evaluation are listed in Table 3.2-2. As noted in Table 3.2-2, some of the additional Diablo Canyon ISFSI missiles were conservatively evaluated for the generic Region II missile velocities described in NUREG-0800, Section 3.5.1.4. The 1800 kg automobile and the 4 kg, 1-inch-diameter steel rod were determined to be the bounding large, and small missiles, respectively. For the intermediate missile category, the 500-kV insulator string was found to be bounding for penetration resistance and the 8-inch-diameter steel rod was determined to be bounding for the global stress evaluation.

The bounding large and intermediate (for penetration only) missiles were chosen by comparison of the kinetic energies of the missiles. The small missile was chosen based on the guidance of NUREG-0800, Section 3.5.1.4, for selecting a missile that can pass through an opening in a protective barrier. For the global stress evaluation of the intermediate category missile, the bounding missile was chosen based on a comparison of safety factors (SF), the missile producing the lower SF being bounding. If the generic analysis described in the HI-STORM 100 System FSAR was bounding, no additional evaluation was performed. If a DCPP site or Diablo Canyon ISFSI-specific missile was bounding, an analysis was performed for the applicable component (that is, the overpack and/or the transfer cask). The following is a summary of the evaluations performed for the four operating ISFSI configurations: transport to the CTF, MPC transfer activities at the CTF, transport to the ISFSI pad, and long-term storage at the ISFSI pad.

The missile impacts are analyzed using formulas from Bechtel Power Corporation Topical Report BC-TOP-9A (Reference 14), ORNL Report TM-1312 (Reference 33), and energy balance methods. In all cases, at all locations away from the impact locations, missile-induced stresses in the transfer cask and overpack are below ASME Level D stress intensity limits.

Another possible consequence of a tornado is to cause the collapse of a nearby 500-kV transmission tower. This event is discussed in Section 8.2.16.

8.2.2.2.1 Transport to the CTF

The transfer cask is transported between the DCPP FHB/AB and the CTF in a horizontal position. Section 3.4.8.2.2 of the HI-STORM 100 System FSAR discusses the side impact from a large missile and concludes loads are below ASME Level D stress intensity limits. The small missile is bounded by the intermediate missile. The evaluations for the side, top, and bottom impact from an intermediate missile (344.7 kg insulator string traveling at 157 mph) are as follows.

- For the side impact, conservatively neglecting the water jacket and the lead shielding, the intermediate missile will penetrate the outer steel shell, but will not penetrate the 3/4-inch inner shell of the transfer cask. Using this conservative model, the minimum inner shell thickness required to withstand the missile impact is 0.266 inch. The design inner shell thickness is 0.75 inch.

- A bottom shield is attached to the transfer cask while suspended horizontally in the cask transporter. On the bottom of the transfer cask, the missile impact occurs on the bottom shield, which covers the pool lid. The HI-STORM 100 System FSAR contains an evaluation for the impact of the intermediate missile on the HI-TRAC transfer lid door. The analysis shows that the intermediate missile would not penetrate the 2-1/4-inch, carbon-steel top plate of the transfer lid door. The minimum required steel thickness to withstand the missile impact is 0.619 inch. This evaluation is conservative for the configuration used at the Diablo Canyon ISFSI, which includes the pool lid (3 inches of steel) and the bottom shield (7-1/4 inches of steel).

- On the top of the transfer cask, the top lid has a hole for rigging, lowering, and raising the MPC during transfer of the canister between the transfer cask and the overpack. While suspended horizontally, this hole is shielded from tornado missiles by the cask transporter body. Conservatively neglecting credit for the missile protection provided by the transporter, an analysis was performed for the 500-kV insulator string intermediate missile entering the transfer cask through the hole in the top lid and impacting the MPC lid. If the insulator string missile directly impacts the MPC, it will not penetrate the 9-1/2-inch-thick, stainless-steel lid. The global stress analysis of the 8-inch steel cylinder missile impacting the MPC lid yielded a safety factor against failure of the peripheral MPC lid-to-shell (LTS) weld of 1.23 versus a safety factor of 7.1 for the insulator string.

8.2.2.2.2 Transfer Operations at the CTF

During MPC transfer operations at the CTF, the transfer cask and the overpack are oriented vertically with the transfer cask stacked on top of the overpack. All but approximately the top 3 ft of the overpack are below grade and not susceptible to tornado missile strikes. The top of the overpack is shielded by the transfer cask until the transfer cask is removed to allow installation of the HI-STORM lid. As discussed in Section 8.2.3.1, cask transport and transfer operations will not be conducted during severe weather. The top of the MPC will only be exposed for a short duration (nominally less than 4 hours). Therefore, in the configuration with the lid removed, a tornado missile impact is not credible. With the top of the MPC exposed during this time, the evaluation of an intermediate missile impact on the MPC lid, described in Section 8.2.2.2.1 ensures the MPC integrity is maintained.

In the vertical orientation, the top of the transfer cask is not subject to direct impacts from these missile strikes and the bottom of the transfer cask is not exposed to tornado-missile strikes. The evaluation of the missile strike on the side of the transfer cask described in Section 8.2.2.2.1 is applicable for this configuration.

8.2.2.2.3 Overpack Transport to the ISFSI Pad

The effect of tornado missiles impacting the transporter while carrying an overpack during transport to the ISFSI pad was evaluated for a horizontal large tornado missile. The transporter with overpack will not turnover from the impact.

Tornado wind effects are enveloped by the HI-STORM 100 System FSAR analysis of a freestanding HI-STORM on a pad. The overpack is lifted only to those heights necessary to travel from the CTF to the ISFSI storage pad. Typically, this is only several inches. This small lift height eliminates tornado missiles striking the bottom of the cask as a credible event.

8.2.2.2.4 Long-Term Storage at the ISFSI Pad

The HI-STORM 100 and 100S free-standing overpack designs have been analyzed for steady state tornado wind loads with a concurrent, large-missile impact, as well as intermediate and small-sized missiles for penetration, as described in Appendices 3.C and 3.G of the HI-STORM 100 System FSAR. The anchored version of the HI-STORM 100S overpack (HI-STORM 100SA) to be deployed at the Diablo Canyon ISFSI is bounded by the free-standing analysis because the anchorage provides additional protection against overturning. The wind loading evaluated in the HI-STORM 100 System FSAR bounds the maximum wind loading at the Diablo Canyon ISFSI site (Table 3.2-1). The loads on the MPC confinement boundary due to the design-basis, 3.0-psi pressure differential are bounded by the 100-psi normal design internal pressure for the MPC described in Section 3.4.4.3.1.2 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. The HI-STORM 100SA overpack is a ventilated design that includes four air inlet ducts and four air outlet ducts at the bottom and top, respectively. Therefore, no tornado-induced pressure differential analysis was performed for the overpack.

The HI-STORM 100SA overpack is generically designed to withstand three types of tornado missiles in accordance with Section 3.5.1.4 of NUREG-0800.

Sections 3.4.8 and 3.4.8.1, as well as Appendices 3.C and Appendix 3.G of the HI-STORM 100 System FSAR, provide discussions of the generic design criteria and the effects of the large (automobile), intermediate (rigid cylinder) and small (sphere) tornado missiles on the overpack. The Diablo Canyon ISFSI-specific intermediate missile (344.7-kg insulator string) is a more limiting design-basis missile for penetration and was evaluated for penetration after impacting the outer shell and the top lid of the overpack at design-basis velocity. The 8-inch-diameter steel cylinder was evaluated generically for global stresses induced after a strike on the top lid of the overpack. The Diablo Canyon ISFSI-specific small missile (1-inch-diameter steel rod) was evaluated for puncture and whether it will enter the overpack air ducts and impact the MPC at design-basis velocity.

The small missile, while less energetic than the intermediate missile, was analyzed specifically due to its unique ability to travel through one of the overpack air inlet ducts and directly impact the MPC pedestal. The evaluations of the effects of the large, intermediate, and small categories of missiles impacting the overpack are described below.

- The free-standing overpack is capable of withstanding the combination of tornado wind (or instantaneous pressure drop) and a large-missile-load impact with a conservative safety factor against overturning of greater than two. The anchored cask system, which provides additional resistance to overturning, is bounded by the free-standing overpack analysis. Local damage to the cask surface by a large-missile impact is bounded by the small and intermediate category missiles.

- Conservatively neglecting the concrete in the overpack, the 500-kV insulator string intermediate missile will penetrate the outer shell of the overpack, but will not penetrate the 1-1/4-inch inner shell of the overpack or result in loss of MPC retrievability. Using this conservative model, the minimum inner shell thickness required to withstand the missile impact is 0.619 inches.

- The 500-kV insulator string intermediate missile will not penetrate the 2-inch top lid of the overpack. The minimum required thickness to withstand the missile impact is 1.089 inches.

- The 8-inch steel cylinder intermediate missile will not cause an over-stress condition on the overpack lid. The factor of safety is 1.4 for this event. The factor of safety for the 500-kV insulator string for this event is 1.6.

- The 1-inch diameter steel rod (that is, small missile) is postulated to enter an overpack inlet duct and impact the pedestal shell. The analysis shows that the rod will pierce the shell and penetrate the concrete to a depth of 6.179 inches, which is significantly less than the radius of the pedestal shield. The damage to the concrete pedestal shield does not affect the confinement boundary or the ability of the MPC to remain standing on the pedestal, nor does it affect the retrievability of the MPC.

The effects of large and small missiles on the free-standing HI-STORM 100 overpack, which were determined in the generic evaluations, are applicable to and bounding for the anchored HI-STORM 100SA overpack to be deployed at the Diablo Canyon ISFSI. The Diablo Canyon ISFSI-specific intermediate missile has been evaluated for penetration and found to have acceptable consequences. The effect of the intermediate missile impact on the overpack lid has been evaluated and found to have acceptable consequences.

8.2.2.3 Conclusions

The above discussion demonstrates that the HI-STORM 100SA overpack and the HI-TRAC transfer cask provide effective missile barriers for the MPC. No missile strike will cause instability of the overpack, compromise the integrity of the confinement boundary or jeopardize retrievability of the MPC. In addition, global stress intensities arising from the missile strikes satisfy ASME Code Level D limits for an ASME Section III Subsection NF structure. For the case where the transfer cask is being transported to the CTF in the horizontal position, the MPC top lid has been evaluated for an intermediate missile strike. The stress intensities from this missile strike satisfy the ASME Section III Subsection NB Level D limits. Therefore the requirements of 10 CFR 72.122(b) are met with regard to tornadoes.

The cask transporter has redundant drop protection by design (Section 3.3.3). Therefore, a loss of load due to a direct missile strike on the transporter is not credible. Since the CTF structure at DCPP is underground, it is not exposed to missile impacts (Section 3.3.4).

8.2.2.4 Accident Dose Calculations

Extreme winds in combination with tornado missiles are not capable of overturning a storage cask or of damaging an MPC within a storage cask resulting in a loss of shielding. Therefore, no radioactivity would be released due to tornado effects on the overpack in the event of a tornado. Dose rates at the controlled area boundary and onsite would not be affected by the minor damage to the transfer or storage cask from tornado-driven missile strikes.

8.2.3 FLOOD

A flood is classified as a natural phenomenon Design Event IV in accordance with ANSI/ANS 57.9.

8.2.3.1 Cause of Accident

The probable maximum flood is classified as a severe natural phenomenon. In general, floods are caused by extended periods of rainfall, tsunamis, storm surges, or structural failures, such as a dam break.

The Diablo Canyon ISFSI storage pads are located at an elevation of over 300 ft mean sea level (MSL). The Diablo Canyon ISFSI site surface hydrology is described in Section 2.4. It is concluded in Section 2.4 that there is no potential for flooding in the vicinity of the ISFSI storage pads. Therefore, flooding is not a consideration for ISFSI operations or on the capability of the dry storage cask system to safely store the spent fuel. Likewise, due to the elevation of the ISFSI site, a tsunami (about 35 ft MSL) as discussed in the DCPP FSAR Update (Reference 4), Section 2.4.6, is not a threat to the HI-STORM 100 Systems being stored on the pad. Since the CTF is located adjacent to the ISFSI pads, it is similarly concluded that there is no potential flooding and tsunami impact on the CTF.

Floods are generally predictable events. As such, administrative controls contained in ISFSI operating procedures will be used to preclude transport of the MPC in a transfer cask, CTF MPC handling activities, and transport of a loaded overpack between the CTF and storage pads during severe weather. Therefore, flooding during these configurations is also not considered credible. Also, the minimum elevation of the transport route (about 82 ft MSL) precludes a tsunami flooding the transport route while in use.

The potential for flooding at the CTF is further reduced by the CTF having a removable cover that is installed when the CTF is not in operation. As a further precautionary measure, the CTF is equipped with a sump as described in Section 4.4.5.

8.2.3.2 Accident Analysis

The HI-STORM 100 System is designed to withstand the pressure and water forces associated with a flood. The design criteria for a flood are discussed in Section 2.2.3.6 of the HI-STORM 100 System FSAR. The flood is assumed to submerge the HI-STORM 100 System to a depth of 125 ft with a water velocity of 15 ft/sec (HI-STORM 100 System FSAR, Table 2.2.8).

No additional flooding analyses have been performed for the Diablo Canyon ISFSI because flooding of the ISFSI is not considered credible.

8.2.3.3 Accident Dose Calculations

Flooding is not a credible event for the Diablo Canyon ISFSI because of the elevation of the ISFSI site. There will be no releases of radioactivity and no resultant doses.

8.2.4 DROPS AND TIP-OVER

The hypothetical drop/tip-over of a storage cask is classified as Design Event IV, as defined by ANSI/ANS-57.9. The design for the Diablo Canyon ISFSI, as explained below, eliminates the need to postulate and analyze cask drop and non-mechanistic tip-over events. The load path portions of the cask transporter and the lifting devices attached to the cask components (that is, the HI-TRAC lifting trunnions and the overpack lift bolt anchor blocks) are designed to preclude drop events, either through redundancy or enhanced safety factors. Table 2.2.6 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, discusses the design codes and standards applicable to the transfer cask and the overpack. Sections 3.3.3, 4.3, and 8.2.1 discuss the design criteria, applicable codes and standards, and design features of the cask transporter that demonstrate that the transporter will not leave the transport route, tip over, or drop the loaded transfer cask or overpack under all design basis conditions, including natural phenomena. Since the CTF lifting devices are designed, fabricated, inspected, maintained, operated, and tested in accordance with applicable guidelines of NUREG-0612, a drop of the transfer cask and MPC will not occur.

Section 8.2.1 describes the analysis of a seismic event, verifying that the CTF and the cask transporter will not drop a loaded transfer cask or overpack, and the cask transporter will remain stable on the transport route for the duration of the earthquake. Therefore, transfer cask and overpack drop events are not analyzed outside the FHB/AB, nor are maximum lift heights established for handling the casks. Administrative controls in operation procedures will ensure the casks are lifted only to those heights necessary to complete the required activities for cask loading and unloading.

The design of the Diablo Canyon ISFSI also includes a requirement to anchor the overpack to the concrete ISFSI pad. This design concept is necessary to accommodate a design-basis seismic event at the site without the cask sliding or tipping over. The anchored overpack concept eliminates the need to postulate a non-mechanistic tip-over of the loaded overpack when anchored to the ISFSI storage pad. Section 8.2.1 describes the analysis that verifies the anchored overpack will not slide or tip over during a seismic event. Section 8.2.2 describes the analysis that demonstrates that the overpack will not tip over as a result of tornado wind concurrent with a large tornado missile impact.

8.2.4.1 Cause of Accident

Cask drop or tip-over is not a credible event outside the DCPP FHB/AB as discussed above. Cask drop events have been postulated as part of the 10 CFR 50 licensing basis inside the FHB/AB due to the nonsingle-failure-proof design of the FHB/AB crane, which will be used to lift and move the unloaded and loaded transfer cask. The description of the drop events, necessary ancillary equipment (that is, impact limiters), and the analyses performed to show the cask and building structure remain within acceptable limits are included in the 10 CFR 50 license amendment request supporting the Diablo Canyon ISFSI license application.

At the Diablo Canyon ISFSI, transfer of the loaded MPC between the transfer cask and the overpack is accomplished at the CTF using the cask transporter to lift the transfer cask to the height necessary to accomplish this objective. The CTF and the cask transporter used in Diablo Canyon ISFSI operations are designed, fabricated, operated, inspected, maintained, operated, and tested in accordance with the applicable guidelines of NUREG-0612. Therefore, a drop of the loaded MPC during inter-cask transfer operations is not a credible event.

8.2.4.2 Accident Analysis

As discussed above, cask drop or tip-over or MPC drop are not credible events outside the FHB/AB.

8.2.4.3 Dose Calculation for MPC Drop Event

Cask drop or tip-over or MPC drop are not credible events. Thus, there is no breach of the MPC confinement boundary and no release of radioactivity.

8.2.5 FIRE

Fires are classified as human-induced or natural phenomena design events in accordance with ANSI/ANS 57.9 Design Events III and IV. To establish a conservative design basis, the following fire events are postulated:

(1) Onsite transporter fuel tank fire

(2) Other onsite vehicle fuel tank fires

(3) Combustion of other local stationary fuel tanks

(4) Combustion of other local combustible materials

(5) Fire in the surrounding vegetation

The potential for fire is addressed for both the overpack and the transfer cask. Locations where the potential for fire is addressed include the ISFSI storage pads, the area immediately surrounding the ISFSI storage pads, including the CTF, and along the transport route between the DCPP FHB/AB and the ISFSI storage pads. The evaluations performed for these postulated fire events are discussed in the following sections.

8.2.5.1 Cause of Accident

Multiple causes, both human-induced and natural, are assumed for each of the fire events postulated above. For the purposes of this SAR, all conservatively postulated fire events are classified as ANSI/ANS 57.9, Design Event IV, events that are postulated because they establish a conservative design basis for important-to-safety SSCs.

There are several potential mechanisms for the initiation of Events 1 through 4, listed above, including both human-induced (electrical shorts, vehicle accidents, transmission line strikes, etc.) and natural (lightning strikes, tornado missiles, etc.) phenomena. While the probability of occurrence of these mechanisms would be very low, the classification of these fire events as ANSI/ANS 57.9, Design Event IV, requires performing an evaluation.

The postulated fire in the vegetation surrounding the ISFSI storage pad (Event 5) could be caused by the spread of an offsite fire onto the site or as the result of natural phenomena such as a lightning strike or a transmission line strike. Unlike the other fire events, it is reasonable to expect that some type of vegetation fire will occur during the ISFSI license period. While plant personnel would quickly act to suppress or control vegetation fire, it is postulated that no fire suppression activity occurs. Thus, this fire event is conservatively classified as an ANSI/ANS 57.9, Design Event IV.

8.2.5.2 Accident Analysis

For the evaluation of the onsite transporter and other onsite, vehicle-fuel-tank fires (Events 1 and 2), it is postulated that the fuel tank is ruptured, spilling all the contained fuel, and the fuel is ignited. The fuel tank capacity of the onsite transporter is limited by the Diablo Canyon ISFSI TS to a maximum of 50 gallons of fuel. The maximum fuel tank capacity for other onsite vehicles in proximity to the transport route and the ISFSI storage pads is assumed to be 30 gallons. The location of any transient sources of fuel in larger volumes, such as tanker trucks, will be administratively controlled to provide a sufficient distance from the ISFSI storage pads (at all times), the CTF (while transferring a MPC), and the transport route during cask transport to ensure the total energy received is less than the design-basis fire event.

Section 11.2.4 of the HI-STORM 100 System FSAR presents an evaluation of the effects of an engulfing 50-gallon fuel fire for both overpack and transfer cask. Results of these analyses indicate that neither the storage cask nor the transfer cask undergoes any structural degradation and that only a small amount of neutron shielding material (concrete, Holtite-A, and water) is damaged or lost. This analysis bounds the 30-gallon, onsite, vehicle-fuel-tank fire (Event 2).

Administrative controls are imposed to ensure no combustible materials are stored within the protected area fence around the ISFSI storage pads. Prior to any cask transport, a walkdown will be performed to ensure all local combustible materials (Event 4), including transient combustibles, are controlled in accordance with ISFSI fire protection requirements. All stationary fuel tanks (Event 3) are at least 50 ft from the ISFSI storage pad security fence and at least 100 ft from the transport route and the CTF. These existing stationary tanks have been evaluated. Due to their distances to the transport route or the ISFSI pad, the total energy received by the storage cask or the transporter is insignificant compared to the design-basis fire event.

The native vegetation surrounding the ISFSI storage pad is primarily grass, with no significant brush, and no trees. Maintenance programs prevent uncontrolled growth of the surrounding vegetation. As previously stated, no combustible materials will be stored within the ISFSI protected area. A conservative fire model was established for evaluation of grass fires. Analysis has demonstrated that grass fires are bounded by the 50-gallon, transporter-fuel-tank fire evaluated in the HI-STORM 100 System FSAR (Event 5).

In summary, the fire evaluations performed generically in the HI-STORM 100 System FSAR, the physical layout of the Diablo Canyon ISFSI, the fire analysis for the surrounding vegetation, and the administrative controls on fuel sources ensure that the general design criteria related to fire protection specified in 10 CFR 72.122(c) are met.

8.2.5.3 Accident Dose Calculations

The effects of an onsite transporter, or other onsite vehicle-fuel-tank fire postulated for the Diablo Canyon ISFSI, are enveloped by the design basis transporter fire evaluated in the HI-STORM System FSAR. Section 11.2.4 of the HI-STORM 100 System FSAR describes how the MPC confinement boundary remains intact after a design basis fire for both the overpack and the transfer cask. Therefore, there is no release of the contained radioactive material from the MPC and no dose consequences in this regard. The shielding implications of a design basis fire for each of these components are discussed below.

8.2.5.3.1 HI-STORM 100 Overpack

Section 11.2.4.2.1 of the HI-STORM 100 System FSAR discusses the fire analysis for the overpack, including radiological implications. The design-basis fire for the HI-STORM 100 overpack causes a small reduction in the shielding provided by the concrete. No portions of the steel structure of the overpack experience temperatures exceeding the short-term temperature limits. While the temperature in the outer 1-inch of concrete is shown to exceed the material short-term temperature limit, there is no significant reduction in the shielding provided by the overpack. All MPC component and fuel assembly temperatures remain within their short-term temperature limits.

8.2.5.3.2 HI-TRAC Transfer Cask

Section 11.2.4.2.2 of the HI-STORM 100 System FSAR discusses the fire analysis for the transfer cask. The elevated local temperatures due to the fire will cause approximately 11 percent of the water in the water jacket to boil off and relieve as steam through the relief valves on the water jacket. However, it is conservatively assumed for the dose calculations that all of the water in the water jacket is boiled off. The fire could also heat the Holtite-A shielding material in the transfer cask top lid and bottom shield above its temperature limit. Therefore, it is conservatively assumed in the dose calculations that all of the Holtite-A in the transfer cask is lost.

The postulated losses of all neutron shielding, due to the loss of water in the water jacket and all Holtite-A in the transfer cask top lid and bottom shield, will not exceed the 10 CFR 72.106 dose limits at an assumed controlled-area boundary located 100 meters from the ISFSI pad for the 30-day duration of the accident, as discussed in Section 5.1.2 of the HI-STORM 100 System FSAR. The nearest controlled area boundary at Diablo Canyon is approximately 1,400 ft from the ISFSI storage pads, which would further decrease the estimated accident dose to well below the 10 CFR 72.106 limit.

Also, as discussed in Section 8.2.11.2, the increase in fuel cladding and component material temperatures due to the loss of water in the water jacket do not cause the short-term fuel cladding or material temperature limits listed in the HI-STORM 100 System FSAR Table 2.2-3 to be exceeded. The internal MPC pressure also remains below the 200-psig accident design limit. Thus, there is no effect on the integrity of the MPC confinement boundary.

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, there are no dose consequences beyond the 10 CFR 72.106 limits for these postulated events.

8.2.6 EXPLOSION

Explosions are classified as human-induced or natural phenomena design events in accordance with ANSI/ANS 57.9 Design Events III and IV. The following explosion event categories have been evaluated for the Diablo Canyon ISFSI:

(1) Detonation of a transporter or onsite vehicle fuel tank

(2) Detonation of a propane bottle transported past the ISFSI storage pad

(3) Detonation of an acetylene bottle transported past the ISFSI storage pad

(4) Detonation of large stationary fuel tanks in the vicinity of the transport route

(5) Detonation of mineral oil from the DCPP Unit 2 main bank transformers

(6) Explosive decompression of a compressed gas cylinder

(7) Detonation of the bulk hydrogen storage facility

(8) Detonation of acetylene bottles stored on the east side of the cold machine shop

Events 1, 2, 3, and 6 are assumed to occur in the vicinity of the ISFSI storage pads, CTF, or transport route; and potentially affect both the overpack and the transfer cask. The assumed distance between the source of detonation and the nearest overpack is 50 ft. This is based on: (a) no gasoline-powered vehicles being allowed within the ISFSI protected area, and (b) the minimum distance between the storage casks and the north side of the ISFSI protected area fence (where the road is) being 50 ft. Detonation sources in the vicinity of the CTF or transporter during fuel transportation or storage operations will be controlled by administrative procedures to provide a sufficient separation distance. Events 4 through 7 occur in the vicinity of the transport route and affect only the transfer cask.

As a result of their physical properties, diesel fuel and mineral oil do not pose any real explosion hazard. The pertinent material property for this determination, the flash point, is defined as the lowest temperature at which the vapor pressure of a liquid is sufficient to produce a flammable vapor/air mixture at the lower limit of flammability. In other words, a combustible liquid cannot vaporize sufficiently to detonate if the ambient temperature is below the flash point. Such materials could conceivably burn, but would be incapable of detonation.

The flash points of diesel fuel and mineral oil are 125°F and 275°F, respectively. To be classified as flammable, the flash point of a liquid must be less than 100°F as discussed in the National Fire Protection Association Handbook (Reference 15). The highest ambient temperature predicted for the Diablo Canyon ISFSI site (5- to 10-year recurrence interval) is 104°F and would normally (99 percent of the time) be no more than 85°F; and the normal operating temperature of the 13,000 gallons of mineral oil in each of the DCPP Unit 2 main bank transformers is approximately 160°F. These temperatures are considerably less than the respective flash points of either diesel fuel or mineral oil. Therefore, under ambient or normal operating temperature, these materials do not represent a credible explosive hazard. However, if an electrical fault were to occur in the transformers, an explosion could occur. The probability of this event occurring while the transfer cask is in proximity was evaluated. The potential risk is insignificant using the Regulatory Guide 1.91 (Reference 16) risk acceptance criteria. Therefore, Event 1 for vehicles containing diesel fuel and Event 5 for main bank transformers is excluded from further consideration.

Since the cask transporter is powered by diesel fuel, which cannot detonate as discussed above, explosion Event 1 is reduced to the explosion of onsite, gasoline-powered vehicles. The fuel tank capacity of these vehicles is limited to 30 gallons by administrative controls and the vehicles are not allowed within the perimeter of the ISFSI site. Administrative controls will be used to keep onsite, gasoline-powered vehicles and tanker trucks carrying flammable liquids either: (a) at sufficient distance from the ISFSI storage pad (at all times), the CTF (while transferring an MPC), and the transport route during cask transport to ensure the total explosion overpressure is less than 1 psig (b) a risk assessment will be performed using Regulatory Guide 1.91 risk acceptance criteria, or (c) diesel-powered vehicles will be used. An exception to the distance criteria is when the 2,000-gallon gasoline tanker truck is using the transport route near the ISFSI pad. The truck will only be in this area momentarily while passing by the ISFSI pad on its way to and from the vehicle maintenance shop that is located approximately 2,000 ft northeast of the ISFSI pad. A probabilistic risk analysis was performed, and it was determined the risk is insignificant using the Regulatory Guide 1.91 risk acceptance criteria.

Explosion Events 2 and 3 include valid sources of detonation evaluated in Section 8.2.6.2.1 below.

The large fuel tanks referred to in Event 4 are located along the main plant access road from the Avila Gate, approximately 1,200 ft from the onsite transport road at the closest point. The tanks include a 250-gallon propane tank, a 2,000-gallon diesel fuel tank and a 3,000-gallon gasoline tank. The diesel fuel cannot detonate, so Event 4 is limited to the detonation of the 250-gallon propane and 3,000-gallon gasoline tanks.

Event 6 is a credible explosive decompression event for a compressed gas cylinder. The cylinder is evaluated as a projectile, similar to a tornado-borne missile and is discussed in Section 8.2.6.2.2 below.

Event 7 includes a valid source of detonation and is discussed in Section 8.2.6.2.3.

Event 8 is a credible source of detonation and is discussed in Section 8.2.6.2.1.

8.2.6.1 Cause of Accident

There are several potential mechanisms for the initiation of the postulated explosion events listed above, including both human-induced (electrical shorts, vehicle accidents, transmission line strikes, etc.) and natural (lightning strikes, tornado missiles, etc.) phenomena. While the probability of occurrence of these mechanisms is expected to be very low, the credible explosion events are classified as ANSI/ANS 57.9, Design Event IV, and are evaluated.

8.2.6.2 Accident Analysis

8.2.6.2.1 Explosive Overpressure Due to Detonation Events

During a detonation event, the overpack and/or transfer cask would be subjected to an external overpressure. Regulatory Guide 1.91 states: "...for explosions of the magnitude considered in this guide and the structures, systems, and components that must be protected, overpressure effects are controlling." The magnitude of the overpressure would be a function of the calorific energy released and the distance between the overpack/transfer cask and the explosion source. Due to the extremely short duration of explosion events, any heat input to the casks would be negligible (fires are evaluated in Section 8.2.5).

Events 1 through 4 and Event 8 are evaluated under the following assumptions:

(1) The fuel tank or gas bottles are ruptured, releasing all contained flammable material, and all spilled flammable liquids are completely vaporized.

(2) The flammable gas or vapor is mixed with air at the lower flammability limit of the material.

(3) The flammable fuel/air mixture is detonated, releasing a portion of the total heating value as a hemispherical overpressure wave front. The fraction of the available energy that contributes to the overpressure, called the explosive yield, is between 3 percent and 6 percent for hydrocarbon/air mixtures, as discussed in the Handbook of Chemical Hazards Analysis (Reference 17).

To determine the magnitude of the explosive overpressure incident on the overpack and transfer cask, the energy released during detonation is converted to an equivalent weight of trinitrotoluene (TNT). This is accomplished by dividing the explosion energy by the detonation energy of TNT, which is 4.5 megajoules per kilogram as discussed in Perry's Chemical Engineers' Handbook (Reference 18).

Once the equivalent weight of TNT is known, the explosive overpressure can be determined as a function of the separation distance between the explosion and the cask systems using a methodology developed by the U.S. Army (Reference 19) and endorsed by the NRC through Regulatory Guide 1.91. This methodology requires the calculation of a scaled ground distance, Z_G, which is the ratio of the physical separation distance divided by the cube root of the equivalent weight of TNT and has units of ft/lb$^{1/3}$. The incident overpressure at a given scaled ground distance is then obtained directly from Figure 2-15 of Reference 19.

Based on the minimum physical separation distances (50 ft for Events 1 through 3 and 1,200 ft for Event 4) and the maximum quantities of flammable material, the equivalent weight of TNT, and the scaled ground distance, the incident overpressure for the explosive overpressure incident on the overpack and/or transfer cask for the credible detonation events are calculated. These results are shown in Table 8.2-11.

The maximum calculated overpressure from these four explosion events is 9.19 psig. Section 3.4.7.2 of the HI-STORM 100 System FSAR evaluates the effects of a 10-psig overpressure for overturning of a free-standing overpack for a duration of 1 sec on an overpack. Results of this analysis indicate that overturning of the overpack will not occur and that no shielding material is damaged or lost, meeting the licensing basis acceptance criteria for the casks. Due to the anchored design, the margin for overturning for the DCPP storage cask is much higher. The MPC is designed for a 60-psig overpressure (HI-STORM 100 System FSAR, Table 2.2.1). A comparison of the overpack and MPC design overpressures from the HI-STORM 100 System FSAR with the maximum calculated overpressure evaluated for the site-specific Diablo Canyon ISFSI detonations indicates that the HI-STORM 100 System FSAR generic design basis bounds the site-specific explosion accidents and 10 CFR 72.122(c) is met.

Section 3.4.9 of the HI-STORM 100 System FSAR presents an evaluation of the effects of a handling accident (a 45-g deceleration during a side drop) on a transfer cask. During this event, the transfer cask shell is exposed to a one-sided force of at least 7.2×10^6 lb when the MPC weight is neglected. Applied evenly over the projected area of the pressure-retaining surface of the transfer cask, this load corresponds to a minimum pressure of approximately 384 psig. Results of this analysis indicate that the structural integrity of the transfer cask is not degraded and that only a small amount of neutron shielding material (water) is lost, meeting the licensing basis acceptance criteria for the casks.

Event 8 was evaluated by determining the number of acetylene bottles that would have to be stored on the east side of the cold machine shop – and detonate – to degrade the structural integrity of the transfer cask. Approximately 16,000 acetylene bottles would be required to detonate at this location to develop an overpressure at the passing transfer cask greater than 384 psig. The number of required bottles far exceeds the available bottle storage space at the cold machine shop. Thus, detonation of acetylene bottles stored on the east side of the cold machine shop would not degrade the structural integrity of the transfer cask.

The site-specific explosive overpressures caused by detonation events are bounded by the generic design basis described in the HI-STORM 100 System FSAR. Therefore, 10 CFR 72.122(c) is met.

8.2.6.2.2 Missiles Due to Explosive Decompression of a Compressed Gas Cylinder

The missile created by the explosive decompression of a gas cylinder (Event 6) is evaluated assuming that a compressed gas cylinder under high-pressure is damaged such that the valve assembly located at the top of the cylinder breaks off. Expansion of the high-pressure compressed gas out of the hole in the cylinder accelerates the cylinder or valve assembly toward the cask systems, resulting in an eventual impact. Cylinders filled with acetylene, air, argon, helium, nitrogen, oxygen, and propane are evaluated.

The acceleration of the cylinder is dependent on the thrust force generated by the escaping high-pressure gas, which reduces over time as the cylinder internal pressure decreases. The thrust force as a function of time is determined from principles of compressible flow, which state that the thrust force is the product of the mass flow and velocity of the gas escaping through the hole in the cylinder wall. While the internal pressure of the cylinder is sufficiently high (that is, greater than the critical pressure), the velocity of the gas is limited to the speed of sound (that is, sonic or choked flow). As the pressure falls below the critical pressure, the velocity becomes subsonic, and eventually reaches zero when the cylinder internal pressure is equal to the atmospheric pressure.

Conservatively neglecting aerodynamic drag (which would decrease the maximum velocity of the cylinder by opposing the thrust force), and assuming bounding discharge coefficients, the cylinder is determined to accelerate from rest to a maximum of approximately 109 mph as the internal pressure drops toward ambient pressure (propane gas). The detached valve assembly is determined to accelerate to a maximum of approximately 342 mph (all gases equal).

Section 8.2.2 of this SAR presents evaluations of the impact of tornado missiles on both the loaded overpack and the transfer cask. Using the same energy method employed in Section 8.2.2, the effects of the impact of cylindrical missiles are evaluated. The maximum penetration into a steel target for the cylinder and valve assembly missiles is less than 1/4 inch. These penetrations are insufficient to completely penetrate either a storage overpack or a transfer cask, thereby precluding damage to the MPC confinement boundary. These missile evaluations conclude that neither the loaded overpack nor the transfer cask undergoes any significant reduction of structural integrity and no shielding material (concrete and water) is damaged or lost, such that the licensing basis acceptance criteria for the casks is met.

8.2.6.2.3 Potential Explosion Event at the Bulk Hydrogen Facility

A bulk hydrogen facility is located east of the FHB/AB. This facility contains 6 tanks for a total of about 300 cubic ft and is near the transport route (approximately 15 ft) from where the transfer cask enters and leaves the Unit 1 FHB/AB. These tanks are refilled approximately twice a month. They are held in a seismic-qualified rack, which is enclosed, in a seismic-qualified vault. The vault is only open on the side toward the FHB/AB and is provided with a 12-inch-diameter top vent to ensure no possible buildup of gas from leakage. This facility is designed to protect against over pressurization, excessive flow, and vehicle (delivery truck) damage during filling. The transporter will only be in this area for a very short period of time, and during this time, all filling of tanks will be suspended and all vehicle movement will be administratively controlled in accordance with the cask transportation evaluation program. A probabilistic risk assessment was performed in accordance with the Regulatory Guide 1.91 methodology. Due to the noncredible nature of an explosion and the limited exposure to the transporter, the event is not risk significant using the Regulatory Guide 1.91 acceptance criteria and is considered acceptable.

8.2.6.3 Accident Dose Calculations

As discussed above, the effects of the Diablo Canyon site explosion events involving detonation (Events 1, 2, 3, 4, and 8) are enveloped by the design-basis accident conditions (explosion and transfer cask side drop) in the HI-STORM 100 System FSAR. The missile evaluation for Event 6 concludes that only a small amount of the shielding materials may be damaged or lost. The structural evaluations in Chapter 3 of the HI-STORM 100 System FSAR confirm that the MPC confinement boundary remains intact and the shielding effectiveness of the HI-STORM 100 System is not significantly affected by these explosion and missile events. The radiological evaluations presented in Chapter 11 of that document also conclude that the loaded overpack and transfer cask continue to meet the accident dose limits of 10 CFR 72.106 at the controlled area boundary after these events.

8.2.7 LEAKAGE THROUGH CONFINEMENT BOUNDARY

The hypothetical leakage of a single, loaded MPC-32 under accident conditions, where the cladding of 100 percent of the fuel rods is postulated to have ruptured, is described in this section.

8.2.7.1 Cause of Accident

The analyses presented in Chapters 3 and 11 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, demonstrate that the MPC confinement boundary remains intact during all hypothetical accident conditions, including the associated increased internal temperature and pressure due to the decay heat generated by the stored fuel.

This section evaluates the consequences of a non-mechanistic, 100 percent, fuel-rod rupture and confinement boundary leak. The breach could result in the release of gaseous fission products, fines, volatiles, and airborne crud particulates to the MPC cavity. Doses resulting from the canister leakage under hypothetical accident conditions were calculated in accordance with Interim Staff Guidance (ISG) Document 5 (Reference 20), ISG 11 (Reference 21) and NUREG/CR-6487 (Reference 22).

8.2.7.2 Accident Analysis

8.2.7.2.1 Confinement Vessel Releasable Source Term

The MPC-32, which holds 32 PWR fuel assemblies, is used in the confinement analysis because it bounds the other, lower-capacity Holtec PWR MPCs for the total quantity of radionuclides available for release from a single cask. The methodology for calculating the spent fuel isotopic inventory for an MPC-32 is detailed in Section 7.2.2. A summary of the isotopes available for release is provided in Table 7.2-8.

8.2.7.2.2 Release of Contents under Accident Conditions of Storage

In this hypothetical accident analysis, it is assumed that 100 percent of the fuel rods have developed cladding breaches, even though, as described below, the spent fuel is stored in a manner such that the spent fuel cladding is protected against degradation that could lead to fuel rod cladding ruptures. The MPC cavity is filled with helium after the MPC has been evacuated of air and moisture that might produce long-term degradation of the spent fuel cladding. Additionally, the HI-STORM 100 System is designed to provide for long-term heat removal capabilities to ensure that the fuel is maintained at a temperature below those at which cladding degradation occurs. It is, therefore, highly unlikely that a spent fuel assembly with intact fuel rod cladding will undergo cladding failure during storage, and the assumption that 100 percent of the fuel rods have ruptured is extremely conservative.

The assumption that 100 percent of the fuel rods have ruptured is incorporated into the postulated pressure increase within the MPC cavity to determine the maximum possible pressure of the MPC cavity. This pressure, combined with the maximum MPC cavity temperature under accident conditions, is used to determine a postulated leakage rate during an accident. This leakage rate is based on the SAR Section 10.2 leakage rate limit of 5.0×10^{-6} atm-cm^3/sec for the helium-leak-rate test, and is adjusted for the higher temperature and pressure during the accident to result in a hypothetical accident leak rate of 1.28×10^{-5} cm^3/sec.

The radionuclide release fractions, which account for the radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not releasable to the MPC cavity from the fuel cladding, are based on ISG-5. Additionally, only 10 percent of the fines released to the MPC cavity are assumed to remain airborne long enough to be available for release through the confinement boundary based on SAND88-2778C (Reference 23). It is conservatively assumed that 100 percent of the volatiles, crud, and gases remain airborne and available for release. The release rate for each radionuclide was calculated by multiplying the quantity of radionuclides available for release in the MPC cavity by the leakage rate calculated above, divided by the MPC cavity volume. No credit is taken for any confinement function of the fuel cladding or the ventilated overpack.

8.2.7.3 Dose Calculations for Hypothetical Accident Conditions

Doses at the Diablo Canyon ISFSI site boundary resulting from a postulated leaking MPC-32 were calculated using an inhalation and submersion pathway. An ingestion pathway is not included because of the lack of broadleaf vegetation within 4 miles of the site boundary; the lack of fresh surface water; the lack of milk animals or a credible meat pathway within 800 meters of the ISFSI site; and the very low population within a 6-mile radius of the site. The nearest distance from the ISFSI to the DCPP is 1,400 ft. A χ/Q value of 4.50×10^{-4} s/m^3 was assumed. This χ/Q value is conservative because it is based on a 1-hour release period, whereas the hypothetical accident duration is 30 days per ISG-5. The dose conversion factors for internal doses due to inhalation and submersion in a radioactive plume were taken from EPA Federal Guidance Report No. 11 (Reference 24) and EPA Federal Guidance Report No. 12 (Reference 25), respectively. An adult breathing rate of 3.3×10^{-4} m^3/s was assumed.

Doses to an individual present continuously for 30 days were calculated assuming a release from a single cask with the wind blowing constantly in the same direction for the entire duration. The following 30-day doses were determined:

- The committed dose equivalent from inhalation and the deep dose equivalent from submersion for critical organs and tissues (gonad, breast, lung, red marrow, bone surface, thyroid)

- The committed effective dose equivalent from inhalation and the deep dose equivalent from submersion for the whole body

- The lens dose equivalent for the lens of the eye

- The shallow dose equivalent from submersion for the skin

- The resulting total effective dose equivalent and total organ dose equivalent.

The doses were calculated, as appropriate, for both inhalation and submersion in the radioactive plume. Doses due to exposure to soil with ground surface contamination and contamination to a depth of 15 cm have been evaluated generically for the HI-STORM 100 System. The dose due to ground contamination was found to be negligible compared to those resulting from submersion in the plume and are not reported here (HI-STORM 100 System FSAR, Section 7.2.8).

Table 8.2-12 summarizes the accident doses for a hypothetical confinement boundary leak. The estimated doses are a fraction of the limits specified in 10 CFR 72.106(b).

8.2.8 ELECTRICAL ACCIDENT

Electrical accidents considered include a lightning strike and a 500-kV transmission line drop. Both events are postulated to apply high voltage electrical current through the overpack or the transfer cask. These events are classified as natural phenomena, Design Event IV, in accordance with ANSI/ANS 57.9.

8.2.8.1 Cause of Electrical Accident

Lightning strikes are natural phenomena caused by meterological conditions conducive to the discharge of large amounts of static electricity to ground. The 500-kV transmission line drop is postulated as a result of a transmission tower collapse or transmission line hardware failure near the ISFSI storage site and the CTF. The worst-case fault condition for a cask is that which places a cask in the conduction path for the largest current. This condition is the line drop of a single conductor of one phase with resulting single, line-to-ground fault current and voltage-induced arc at the point of contact.

A number of transmission line failure modes were postulated. These included the break or drop of: a single conductor of one phase, both conductors of a single phase, and all three phases. The failure modes considered are:

(1) Three-phase drop onto cask structures - The fault would be balanced, most current would return through the phase conductors and only a small amount would pass through the casks and into the earth.

(2) Both conductors of one phase fall onto one cask - The single line-to-ground fault would split evenly between the two conductors (spaced at 18 inches) and effectively reduce the energy at the point of contact by a factor of two. Therefore, it would create two points of contact, each dissipating half the energy.

(3) One conductor of one phase breaks into two and each end falls onto separate casks or onto different points of the same cask - The single, line-to-ground fault would split between the two points of contact reducing the energy at each point of contact.

(4) One conductor falling while remaining intact - The single, line-to-ground fault would be forced into one point of contact, through the cask, and into the earth/ground grid. All energy would be forced to dissipate at this one point. This would be the worst-case for the cask systems.

Protective relaying is assumed to actuate on arc initiation. The time duration from relay actuation to breaker opening is assumed to be 0.1 sec (6 cycles).

8.2.8.2 Electrical Accident Analysis

The overpack and the CTF are sited beneath a 500-kV transmission line. The transmission line connects the Unit 1 main generator to the 500-kV switchyard. The transmission line is protected from direct lightning strikes by two shield wires installed above the line. Similarly, the transmission conductors provide lightning protection for the overpack and the CTF. The transmission lines themselves act as shield wires for metal objects located below them and within their effective shield angle. Inside this effective shield angle, the distance from the lightning arc to the line will be less than from the lightning arc to the top of the cask, and all lightning within this zone will hit the transmission line instead of the cask. Outside of this effective shield angle, the lightning will be so close to the ground that it will directly hit the ground before it strikes any metal object. Thus, the overhead transmission line prevents a direct lightning strike on any overpack or the CTF. Even so, the effects of a lightning strike are evaluated.

The cask transporter provides protection for the transfer cask from direct lightning strikes and transmission line drops. The gantry and rigging metal is sufficiently above the cask material that any line drop would be effectively deflected by this metal before it is able to contact the cask surface.

For the evaluation of the lightning strike, direct atmospheric lightning strikes on the overpack and the transfer cask are postulated. The lightning strike, defined by a current versus time profile, is defined by standard industry practice as a peak current of 250 kiloamps for 260 microseconds followed by a continuing current of 2 kiloamps for 2 additional seconds.

For the evaluation of the 500-kV transmission line drops for both the overpack and the transfer cask, it is postulated that while both DCPP units are operating at full power a single overhead transmission conductor falls onto a cask. The 500-kV system is operated at a nominal voltage of 525-kV phase to phase. The line-to-ground voltage is 303-kV. The transmission line drop sequence of events is defined in three distinct time periods as follows:

- Period 1 free air arc (wire falling but not yet touching cask) - voltage drops from 303 kV to 1 kV and current rises from 0 kiloamps to 18.6 kiloamps over a 0.05 second arc duration.

- Period 2 prior to breaker trip (wire in solid contact with the cask but breaker not yet fully open) - voltage and current are constant at 1 kV and 18.6 kiloamps, respectively, over a 0.05 second breaker trip duration.

- Period 3 during generator coast-down (all breakers open, faulted generator still contributing fault current) - voltage and current are constant at 0.2 kV and 5.08 kiloamps, respectively, over a generator coast-down duration of 3.9 seconds.

Both electrical events result in an electrical discharge that travels along the least resistive path through the cask to the ground. Both the lightning strike and the transmission line drop originate external to the casks, so the least resistive path for both the overpack and the transfer cask will be through the outermost shell (that is, overpack outer shell and transfer cask enclosure shell). The MPC contained within an overpack or transfer cask will, therefore, be protected from any electrically-induced damage.

For the postulated lightning strike, the electrical discharge deposited into the cask and conducted to ground must overcome the inherent electrical resistance of the conducting material. This resistance to current flow generates heat, called resistance or Joulean heating, and is governed by the following formula:

$$E = I^2 \times t \times R$$

where E is the resistance heat energy, I is the current, t is the current duration and R is the material resistivity. The heat generated by resistance heating must be absorbed by sensible heating of the affected cask component, governed by the following equation:

$$E = m \times c_p \times \Delta T$$

where m is the mass of the cask component, c_p is the material heat capacity and ΔT is the component temperature rise. These two equations can be used to determine the cask component temperature rise for each cask, the results of which are contained in Table 8.2-13.

All of the computed, electrically-induced, temperature-rise values are less than 1°F. The HI-STORM 100 System FSAR contains evaluations of both the overpack and the transfer cask under normal temperature conditions. The increase in outer shell temperature for both structures is well below the normal temperature condition limits. Accident condition temperature limits for the outer shells of both casks are significantly higher than the normal condition limits. It is therefore concluded that the postulated lightning strike will not cause the affected cask components to exceed either normal or accident condition temperature limits and do not adversely affect the performance of either system.

For the postulated transmission line break, because of the significant influence of the time-varying voltage and the longer time periods involved, a slightly different method of calculating the energy input is used. The electrical energy is governed by the following formula:

$$E = \int_t V(t) \times I(t) dt$$

where V(t) is the time-varying voltage function, I(t) is the time-varying current function and t is the independent time variable. The electrical energy is calculated separately for each time period of the postulated electrical profile.

As the transmission line drops onto a cask, the predominant portion of arc energy is dissipated to the atmosphere, with the remaining portions heating the cask and vaporizing a portion of the steel outer shell. During the arc phase (Period 1) of the postulated accident, it is conservatively assumed that 10 percent of the total energy is dissipated in sublimating (vaporizing) steel at the point of arc, 40 percent of the total energy is dissipated in resistance heating of the affected cask component, and the balance of the arc energy is dissipated to the environment. During the breaker trip and generator coast-down periods (Periods 2 and 3) of the postulated accident, it is conservatively assumed that all energy is dissipated in resistance heating of the affected cask component. The results of these evaluations are contained in Table 8.2-14.

With respect to the computed, electrically-induced, temperature rise values, the HI-STORM 100 System FSAR contains evaluations of both the overpack and the transfer cask under normal temperature conditions. Again, the increase in the outer shell temperature of both structures is well below the normal condition temperature limits. Accident condition temperature limits for these components for both casks are significantly higher than the normal condition limits.

The sublimated hole diameters are calculated assuming that a cylindrical plug of material, with a length equal to the thickness of the component material, is vaporized. Even if a hole is sublimated in the overpack outer shell, there are no negative thermal consequences. Behind the steel outer shell is a thick concrete layer that is unlikely to be significantly affected given the rapidity of the event and the low thermal diffusivity of concrete. Experience with high-fault currents has shown that spalling and crystallization of the concrete surface would be expected at the point of contact of the fault. The maximum depth of the concrete plug affected would be less than the diameter of the surface hole. It should also be noted that the existence of a hole in the overpack outer shell was postulated and evaluated in Section 8.2.2. The cause of the hole in that section was due to a hypothesized tornado missile. Should a hole be formed in the transfer cask, the water jacket used to provide shielding and to help maintain cool conditions inside the MPC could be drained. This condition has an insignificant thermal impact, and the shielding impact is already addressed in Section 8.2.11 and was found to be acceptable. Section 8.2.11 considers a loss of water jacket without considering any specific cause.

These results are considered bounding for the design life of the ISFSI. Even if the fault current increases over the life of the facility, the results remain valid because the resulting damage increase would not be significant. The line-to-ground voltage is the predominant factor in arc ignition. An increase in fault current would have minimal consequences. A larger hole size does not change the radiological dose consequences because there is minimal damage to the concrete shielding in the overpack, no damage to the lead shielding in the transfer cask, and no damage to the inner steel liners in both the overpack and the transfer cask.

It is concluded that the postulated transmission line break will not cause the affected cask components to exceed either normal or accident condition temperature limits and that localized material damage at the point of arc is bounded by accident conditions discussed in Sections 8.2.2 and 8.2.11. As a result of these considerations, it is concluded that the postulated transmission line drop does not adversely affect the thermal performance of either system.

8.2.8.3 Electrical Accident Dose Calculations

The postulated electrical events are shown to result in a negligible increase in the temperatures of the affected components and damage to a small amount of material in the localized area of arc. The resulting temperatures would remain bounded by both the normal and accident condition temperature limits.

The small loss of material is negligible compared to the total mass of shielding materials, so there would be no significant increase in overall cask dose rates. As noted above, the concrete behind the overpack outer shell would not likely be affected. Thus, the change in shielding would be negligible. In any event, a more limiting condition is evaluated in Section 8.2.2.

In the case of the transfer cask, there would be an increase in radiation doses adjacent to the cask should the shielding water in the water jacket be lost. The loss of neutron shielding is evaluated in Section 8.2.11. The addition of a hole in the transfer cask outer shell would have a negligible impact on dose. The impact on personnel exposures is considered to be negligible.

The MPC is protected from electrical damage by the overpack. Thus, there is no release of the contained radioactive material from the MPC. Doses to persons located offsite are not affected by these events.

8.2.8.4 Conclusions

The postulated electrical events may possibly result in a small hole in either the overpack or the transfer cask. Both conditions are conservatively bounded by previously analyzed events in Sections 8.2.2 and 8.2.11.

8.2.9 LOADING OF AN UNAUTHORIZED FUEL ASSEMBLY

The Diablo Canyon ISFSI TS and SAR Section 10.2 specify limiting values for the initial enrichment, burnup, decay heat, and cooling time after reactor discharge for the fuel assemblies to be placed into the MPCs The possibility of storing a fuel assembly that does not meet the Diablo Canyon ISFSI TS and SAR Section 10.2 has been considered.

8.2.9.1 Cause of Loading an Unauthorized Fuel Assembly

Procedures will be used to administratively control and document the planning and loading of all DCPP fuel assemblies to be stored in each overpack. The cause of this event is postulated to be an error during spent fuel planning or loading operations (for example, a planning error occurs in selecting the fuel assembly to be stored or the wrong fuel assembly is loaded into an MPC).

8.2.9.2 Analysis of the Loading of an Unauthorized Fuel Assembly

The chance of loading of an unauthorized fuel assembly is greatly minimized because of the multiple administrative controls imposed via procedures to ensure a fuel planning or loading error does not remain undetected. These procedures prescribe how the planning is performed and verified to ensure the characteristics of selected fuel assemblies are within the applicable Diablo Canyon ISFSI TS and SAR Section 10.2 limits. Likewise, the spent fuel loading procedures require that a final verification of the identity and location of fuel assemblies be performed prior to placing the lid on the MPC. These procedures are part of the ISFSI operational procedures described in Section 9.4.1.1.4.

The loading of an unauthorized fuel assembly has no consequence while the transfer cask/MPC assembly remains in the spent fuel pool (SFP) as explained below. The borated water in the SFP provides adequate protection against a criticality event, and also provides shielding and heat removal. Loading of an unirradiated fuel assembly will not cause a criticality event because the MPC design precludes criticality assuming all loaded fuel assemblies are unirradiated (that is, no burnup credit taken). Loading of a fuel assembly with gross cladding defects will not cause further damage to the cladding or result in the release of radioactive material. Loading of a fuel assembly with structural defects will likely be detected during placement into the MPC. These events will not go undetected because fuel condition will be verified as part of the loading process.

8.2.9.3 Conclusion

As discussed above, the use of procedures, which prescribe and verify the rigorous planning and loading activities, provides reasonable assurance that only fuel assemblies meeting Diablo Canyon ISFSI TS and SAR Section 10.2 requirements will be loaded for storage.

8.2.10 EXTREME ENVIRONMENTAL TEMPERATURE

Extreme environmental temperature is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9. The extreme environmental temperature accident involves the postulation of an unusually high ambient temperature at the Diablo Canyon ISFSI site. Unlike the off-normal high temperature evaluated in Section 8.1.2, the postulated, extreme-high temperature is beyond what can be reasonably expected to occur over the life of the ISFSI and represents a bounding, worst-case scenario.

8.2.10.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature event for the HI-STORM 100 System is analyzed at an environmental temperature of 125°F in the HI-STORM 100 System FSAR, as amended by LAR 1014-1, Section 11.2.15, and -40°F in LAR 1014-1, Section 4.4.3. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

8.2.10.2 Extreme Environmental Temperature Analysis

8.2.10.2.1 Upper Temperature Limit

The accident condition considered in the HI-STORM 100 System FSAR, as amended by LAR 1014-1, assumes an extreme environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium. This bounds the extreme-maximum-site ambient temperature for the Diablo Canyon ISFSI site of 104°F (Section 3.4.). This condition is evaluated with respect to accident condition component design temperatures listed in Table 2.2.3 of the HI-STORM 100 System FSAR. The evaluation was performed with the HI-STORM 100 System FSAR design-basis fuel with the maximum decay heat and the most restrictive thermal resistance. The HI-STORM 100 generic evaluation of a 125°F environmental temperature is applied with the peak solar insolation as described in the HI-STORM 100 System FSAR. The solar insolation assumed in the generic analysis bounds that for the Diablo Canyon ISFSI site.

The HI-STORM 100 System maximum temperatures for components close to the design-basis temperatures are discussed in the HI-STORM 100 System FSAR, Section 4.4. These temperatures are calculated at a normal environmental temperature of 80°F. The extreme environmental temperature is 125°F, which is an increase of 45°F. This event is simplistically evaluated by adding the 45°F difference to each of the limiting normal component temperatures. This yields conservatively bounding temperatures for all of the HI-STORM 100 System components because the thermal inertia of the HI-STORM 100 System is not credited. The resulting component temperatures under extreme environmental temperature condition are reported in the HI-STORM 100 System FSAR, Table 11.2.7, as amended by LAR 1014-1. As illustrated by the table, all the temperatures are well below the accident-condition, design-basis component temperatures. Since the extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely), the resultant temperatures are evaluated against short-term accident condition temperature limits. Therefore, the HI-STORM 100 System component temperatures meet design requirements under the extreme environmental temperature condition.

Additionally, the effect of extreme environmental temperature on MPC internal pressure was evaluated. The resultant pressure was bounded by the pressure calculated for complete blockage of the inlet duct. In the case of complete duct blockage, the calculated temperatures are much higher than the temperatures that result from the extreme environmental temperature. The accident condition pressure for the bounding MPC (MPC-32) was determined for concurrent 100 percent fuel rod rupture and was found to be below the accident design pressure of 200 psig.

8.2.10.2.2 Lower Temperature Limit

The HI-STORM 100 System was also evaluated for a -40°F extreme low ambient temperature condition, as discussed in Section 4.4.3 of the HI-STORM 100 System FSAR. Zero decay heat generation from spent fuel and no solar insolation were conservatively assumed. All materials of construction for the MPC and overpack will perform their design function under this extreme cold condition. Since the minimum temperature at the Diablo Canyon ISFSI is greater than or equal to 24°F (Table 3.4-1), the extreme low ambient temperature evaluation in the HI-STORM 100 System FSAR bounds the conditions at the Diablo Canyon ISFSI.

8.2.10.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature range at the Diablo Canyon ISFSI will not cause the overpack concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete shielding effectiveness. The extreme temperature range will not cause a breach of the confinement system and the short-term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact on the HI-STORM 100 System for the extreme environmental temperature range, and the dose rates under this accident condition are equivalent to the normal condition dose rates.

8.2.10.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

8.2.11 HI-TRAC TRANSFER CASK LOSS-OF-NEUTRON SHIELDING

This accident event postulates the loss-of-neutron shielding provided by the transfer cask water jacket and the Holtite-A solid neutron shielding in the transfer cask top lid and bottom shield. A loss-of-neutron shielding is classified as a Design Event IV, as defined in ANSI/ANS-57.9.

8.2.11.1 Cause of Loss-of-Neutron Shielding

Throughout all design-basis-accident conditions, the axial location of the fuel will remain fixed within the MPC because of the upper and lower fuel spacers. Chapter 3 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, shows that the fuel spacers, transfer cask inner shell, lead, and outer shell remain intact throughout all design-basis normal, off-normal, and accident loading conditions. (The 10 CFR 50 LAR in support of the Diablo Canyon ISFSI addresses the effect of lead slump on the transfer cask shielding after a vertical drop inside the FHB/AB.) Localized damage of the transfer cask outer shell could be experienced, but no loss of shielding results.

Two potential causes for the loss of neutron shielding provided by the transfer cask are:

(1) Elevated temperatures as a result of a fire accident could result in the temperature of the Holtite-A exceeding the design-accident temperature. The pressure of the water jacket could also increase due to a fire, to the point where the overpressure relief valve on the water jacket would vent steam and water to the atmosphere. This would result in the loss of some amount of the water used for neutron shielding.

(2) Puncture of the transfer cask outer neutron shield jacket by a small object traveling at high speed, such as a tornado-borne missile, would cause the shield water to drain out at the point of puncture.

Other shielding credited in the shielding analyses includes the steel transfer cask and overpack structures, concrete, and lead. There are no credible events that could cause a significant degradation or loss of these solid forms of shielding.

8.2.11.2 Loss-of-Neutron Shielding Analysis

In the transfer cask, which uses Holtite-A in the top lid and bottom shield for neutron shielding, a fire could cause the Holtite-A to exceed its design-accident-temperature limit. For the dose analysis, it is conservatively assumed that all of the Holtite-A in the transfer cask top lid and bottom shield is lost. The potential reduction in shielding effectiveness of the Holtite-A in the transfer cask top lid results in a dose rate that is bounded by the normal dose rates in the area of the access hole in the transfer cask top lid. Therefore, no additional evaluation of this scenario is required. The accident condition dose rate through the transfer cask bottom shield with no Holtite-A is bounded by the accident dose rate at the side of the transfer cask with an assumed loss of all water in the water jacket, as discussed below. This is based on the accident dose rate adjacent to the empty water jacket being greater than the normal condition dose rate adjacent to the transfer cask pool lid without the bottom shield installed as discussed in the HI-STORM 100 System FSAR, as amended by the LAR 1014-1, Tables 5.1.8 and 5.1.10.

The bounding consequence that affects the shielding materials of the transfer cask is the potential for damage to the water jacket shell and the loss of all of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void. The assumed loss of all water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Section 5.1.2 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, demonstrate that the dose limits of 10 CFR 72.106 are not exceeded if all of the water in the water jacket is lost. It is shown in Section 11.2.4 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, that the increase in fuel cladding and component material temperatures due to the loss of water in the water jacket do not cause the short-term fuel cladding or material temperature limits listed in the HI-STORM 100 System FSAR Table 2.2.3 to be exceeded. The internal MPC pressure also remains below the 200-psig-accident design limit. Therefore, there is no affect on the integrity of the MPC confinement boundary.

8.2.11.3 Loss-of-Neutron Shield Dose Calculations

The complete loss of the transfer cask neutron shield along with the water-jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded transfer cask in Section 5.1.2 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. As shown therein, the complete loss of the transfer cask neutron shield significantly affects the dose rate at mid-height of the transfer cask, and the accident dose rate (calculated using the burnups and cooling times that produce the highest dose rates) is 1.47 mrem/hr at an assumed distance of 100 meters from the ISFSI storage pad. For the 30-day duration of the event, the total dose at this location is 1.058 rem, which is less than the accident dose limit in 10 CFR 72.106. The minimum distance to the controlled-area boundary at the Diablo Canyon ISFSI is approximately 1,400 ft (430 m). Therefore, the generically-calculated doses for this accident from the HI-STORM 100 System FSAR bound those for the Diablo Canyon ISFSI site.

Doses to onsite personnel will be monitored after a loss-of-neutron shielding event and temporary shielding may be employed at the discretion of the DCPP radiation protection organization.

8.2.12 ADIABATIC HEAT-UP

This noncredible accident event postulates that the loaded overpack is unable to reject heat to the environment through conduction, convection, or radiation. This is classified as a Design Event IV, as defined by ANSI/ANS 57.9.

8.2.12.1 Cause of Accident

There is no credible accident that could completely stop heat transfer from the overpack to the environment. Even if the overpack were to be completely buried, with the inlet and outlet vent ducts blocked, some heat transfer would occur via conduction through the overpack structure and the material covering the overpack, and through convection at the surface of the outer material. The Diablo Canyon ISFSI site is located where a portion of the hill has been excavated (Figure 2.1-2). The slope protection of the hill adjacent to the storage pads (Section 4.2.1.1.9) precludes a landslide that completely covers one or more casks on the ISFSI pads. Should a slide occur, minor amounts of material could be removed before excessive heat up would occur. Also, there are no sources of volcanic activity or large amounts of debris located above, and sufficiently close to, the ISFSI site that could cause a complete covering of one or more casks on the ISFSI pads. This is a non-mechanistic accident and is evaluated to yield the most conservative response of the HI-STORM 100 System.

8.2.12.2 Accident Analysis

Section 11.2.14 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, discusses the "Burial-Under-Debris" accident, which is modeled as an adiabatic heat-up event. The analysis of this event is summarized below.

Burial of the loaded overpack does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for the other accidents analyzed. The debris would provide additional shielding to reduce radiation doses. The accident external pressure encountered during the flooding accident (Section 8.2.3) bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. The insulating effect will cause the HI-STORM 100 System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short-term, accident-condition temperature limit during a burial under debris accident.

To demonstrate the inherent safety of the HI-STORM 100 System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM 100 System will undergo a transient heat up under adiabatic conditions. The minimum time required for the fuel cladding to reach the short-term, design, fuel-cladding-temperature limit depends on the amount of thermal inertia of the cask, the cask initial conditions, and the spent fuel decay heat generation.

Figure 11.2.6 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, shows that the time to reach the short-term, fuel-cladding-temperature limit varies from approximately 45 hours at a total cask heat load of 30 kW (higher than the maximum authorized cask heat load) to more than 130 hours at a cask heat load of 10 kW.

8.2.12.3 Accident Dose Calculations

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event. As discussed in burial-under-debris analysis, the shielding is enhanced while the HI-STORM 100 System is covered. The elevated temperatures will not cause the breach of the confinement system and the short-term, fuel-cladding-temperature limit is not exceeded. Therefore, there is no radiological impact.

8.2.13 PARTIAL BLOCKAGE OF MPC VENT HOLES

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. These holes facilitate the natural circulation of helium inside the MPC for convection heat transfer. The partial blockage of the MPC basket vent holes accident has been evaluated to determine the effects on the HI-STORM 100 System due to the reduction in the size of the vent openings. This accident condition is discussed in Section 11.2.5 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1.

8.2.13.1 Cause of Partial Blockage of MPC Vent Holes

After the MPC is loaded with spent nuclear fuel, the MPC cavity is drained, dried, and backfilled with helium. There are three possible sources of material that could block the MPC basket vent holes. These are the fuel cladding, fuel pellets, and crud. Gross fuel cladding rupture is precluded by design in accordance with 10 CFR 72.122(h)(1). Due to the maintenance of relatively low cladding temperatures during storage, it is not credible that the fuel cladding would rupture and that fuel cladding and fuel pellets would fall to block the basket vent holes. Damaged fuel and fuel debris are stored in damaged fuel containers, which have screens to minimize the dispersal of gross particulates. However, it is conceivable that a percentage of the loose crud deposited on the external surfaces of the fuel rods may fall away and deposit at the bottom of the MPC.

Helium in the MPC cavity provides an inert atmosphere for storage of the fuel. During normal storage operations, the design of the HI-STORM 100 System maintains the peak fuel rod cladding temperature below the required long-term storage limits. There are no credible, design-basis accidents that cause the fuel assembly to experience a deceleration loading greater than the limits established in the HI-STORM 100 System FSAR, Section 3.5. (As discussed in Section 8.2.4, the load portions of the transporter and the lifting devices attached to the transfer cask and overpacks are designed to preclude drop events.)

Crud can be made up of two types of layers, namely, loosely-adherent and tightly-adherent. The fuel assembly movement from the fuel racks to the MPC, and subsequent movement of the MPC during cask loading, transfer, and transport operations, may cause a portion of the loosely-adherent crud to fall away. The tightly-adherent crud remains in place during ordinary fuel handling operations.

8.2.13.2 Analysis of Partial Blockage of MPC Vent Holes

The MPC vent holes that act as the bottom plenum for the MPC internal helium circulation are of an elongated, semi-circular design to ensure that the flow passages will remain open under a hypothetical shedding of the crud on the fuel rods. For conservatism, only the minimum semi-circular hole area is credited in the thermal models (that is, the elongated portion of the hole is completely neglected).

The amount of crud on fuel assemblies varies greatly from plant to plant. The maximum crud depths calculated for each of the MPCs is listed in Table 2.2.8 of the HI-STORM 100 System FSAR. The maximum amount of crud was assumed to be present on all fuel rods within the MPC. Both the tightly- and loosely-adherent crud was conservatively assumed to fall off of the fuel rods. The assumed crud depth does not totally block any of the MPC basket vent holes as the crud accumulation depth is less than the elongation of the vent holes. Therefore, the remaining cross-sectional flow area through the vent holes area is greater than that used in the thermal models.

The partial blockage of the MPC basket vent holes has no effect on the structural, confinement, and thermal analysis of the MPC. There is no significant effect on the shielding analysis because the source term from the crud is enveloped by the source term from the fuel and the activated nonfuel hardware of the fuel assemblies. As the MPC basket vent holes are not completely blocked, preferential flooding of the MPC fuel basket is not possible during draining operations and, therefore, the criticality analyses are not affected.

8.2.13.3 Dose Calculations for Partial Blockage of MPC Vent Holes

Partial blockage of basket vent holes will not result in a compromise of the confinement boundary because the thermal model accounts for the partial blockage. Fuel decay heat, burnup, and cooling time limits in SAR Section 10.2 are determined accordingly to ensure that the cask heat transfer remains within the limits of the licensing analysis. Therefore, there will be no loss of confinement or radioactive material release.

Any increase in dose rate through the bottom of the cask due to crud accumulation is inconsequential for several reasons. The total amount of source in the cask is not increased; it is simply relocated by the distance between where the crud particle was located on the fuel assembly and the bottom of the MPC. Any minimal dose increase at the bottom of the cask is inconsequential while the cask is on an ISFSI pad because the bottom of the cask (being flush against the pad surface) is not a source of exposure during storage operations. During vertical handling operations, the overpack and transfer cask are lifted only to those heights necessary to facilitate required cask movements. These heights are typically low enough to physically prevent personnel access. Administrative controls related to prudent, heavy-load movement will preclude personnel from access underneath the lifted cask inside the FHB/AB. During horizontal transportation of the transfer cask between the FHB/AB and the CTF, the additional dose is negligible due to the shielding provided by the bottom of the MPC, the pool lid, and the supplemental transfer-cask bottom shield.

8.2.14 100 PERCENT FUEL ROD RUPTURE

This accident event postulates that all of the fuel rods in a sealed MPC rupture and that fission-product gases and fill gas are released from the fuel rods into the MPC cavity.

8.2.14.1 Cause of Accident

Through all credible accident conditions, the HI-STORM 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel-cladding temperature below the short-term temperature limits, thereby ensuring fuel-cladding integrity. Although rupture of all the fuel rods is assumed, there is no credible cause for 100 percent fuel rod rupture. This accident is postulated to evaluate the MPC confinement boundary for the maximum possible internal pressure based on the non-mechanistic failure of 100 percent of the fuel rods.

8.2.14.2 Accident Analysis

The 100 percent fuel-rod-rupture accident has no thermal, criticality, or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, which is being shielded, the shielding capacity, or the criticality control features of the HI-STORM 100 System. It only has the potential for affecting the internal pressure of the MPC and the leakage from the MPC. The determination of the maximum accident pressure due to a hypothetical 100 percent fuel rod rupture accident was evaluated for the MPC-32 as a bounding case for all MPCs that are licensed for use at the Diablo Canyon ISFSI.

The MPC-32 internal cavity pressure was calculated for the 100 percent rod rupture accident using the methodology from the HI-STORM 100 System generic analysis documented in Section 4.4.4 of the HI-STORM 100 System FSAR. Limiting input values were assumed for initial fuel rod fill pressure (715 psia), fuel burnup (70,000 MWD/MTU), decay heat load (28.74 kW) and minimum MPC cavity volume. The presence of nonfuel hardware and the release of fission gases from the BPRAs was also accounted for. These assumptions bound the characteristics for fuel to be loaded in any MPC to be deployed at the Diablo Canyon ISFSI. The computed MPC internal pressure from the 100 percent rod rupture accident is 185.5 psia (170.8 psig), which is less than the MPC accident design pressure of 200 psig (Reference 12, Table 2.0.2).

8.2.14.3 Accident Dose Calculations

There is no effect on the shielding performance or criticality control features of the system as a result of this event. There is no effect on the confinement function of the MPC as a result of this event. All stresses remain within allowable values, ensuring confinement boundary integrity. Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

The MPC confinement boundary maintains its integrity for this postulated event. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. However, the radiation source could redistribute within the sealed MPC cavity causing a slight change in the radiation dose rates at certain locations. In that case though, the radiation dose at the ISFSI site boundary would not be affected. There is no release of radioactive material or significant increase in radiation dose rates.

8.2.15 100 PERCENT BLOCKAGE OF AIR INLET DUCTS

This accident postulates the complete blockage of all four inlet air ducts of the overpack. Blockage of the inlet air ducts is equivalent to the condition where all four outlet air ducts are blocked because either scenario stops air flow through the overpack. While a small amount of warmed air may exit the outlet air ducts and be replaced with cooler ambient air, this mechanism is of second order compared with the heat redistribution effect of the buoyancy-driven, natural-convection circulation that is established in the annular space between the MPC and overpack. As the dominant natural convection circulation is identical for either the inlet or outlet air ducts blockage, the following evaluation is applicable to both conditions. The loss of the small, second-order, air-exchange effect should the top ducts be blocked would be a lesser magnitude than the inherent conservatisms in the analysis resulting from the assumptions of complete blockage, maximum decay heat load, high ambient temperature, conservative conductivity modeling, and conservative solar heat. The complete blockage of air inlet ducts is classified as Design Event IV as defined by ANSI/ANS-57.9.

8.2.15.1 Cause of 100 Percent Blockage of Air Inlet Ducts

In Section 11.2.13 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, the 100 percent blockage of all overpack air inlet ducts is postulated to occur due to an environmental event such as flooding, snowfall, tornado debris, or volcanic activity. Of these, only blockage by tornado debris is credible at the Diablo Canyon ISFSI (Chapter 2). The slope protection of the hill adjacent to the storage pads (Section 4.2.1.1.9) precludes a landslide that completely covers all air inlet ducts. Should a slide occur, minor amounts of material could be removed before excessive heatup would occur. There is no credible, design-basis event at the Diablo Canyon ISFSI that could completely block all four air inlet ducts for an extended period of time where corrective action could not be taken in a timely manner to remove the blockage.

8.2.15.2 Analysis of 100 Percent Blockage of Air Inlet Ducts

The immediate consequence of a complete blockage of the air inlet ducts is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC, and the stored fuel assemblies will rise as a function of time.

As a result of the large mass, and correspondingly large thermal capacity, of the storage overpack (in excess of 170,000 lb), it is expected that a significant temperature rise is only possible if the completely blocked condition is allowed to persist for a number of days. This accident condition is, however, a short-duration event that will be identified and corrected through the performance of daily surveillance inspections required by the Diablo Canyon ISFSI TS.

There is a large thermal margin between the maximum-calculated, fuel-cladding temperature with design-basis fuel decay heat (HI-STORM 100 System FSAR Tables 4.4.9, 4.4.26, and 4.4.27) and the short-term, fuel-cladding-temperature limit (1,058°F), to accommodate this transient, short-term, fuel-cladding temperature excursion. The fuel stored in a HI-STORM 100 System can heat up by over 300°F before the short-term temperature limit is reached. The concrete in the overpack has a smaller, but nevertheless significant, margin between its calculated, maximum, long-term-temperature and its short-term-temperature limit, with which to withstand the temperature rise caused by this accident.

A detailed discussion of the analysis of this accident is provided in Section 11.2.13.2 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1. This accident has been generically analyzed both with and without considering the effect of the thermosiphon convection heat transfer phenomenon inside the MPC. Since the limiting decay heats, burnups, and cooling times for the DCPP spent fuel authorized for loading into the HI-STORM 100 System are based on credit for thermosiphon convection in the MPC, the convection-based analysis is applicable to the Diablo Canyon ISFSI.

The results of the analysis without thermosiphon bound the Diablo Canyon ISFSI design-basis analysis with thermosiphon and show that the concrete section average (that is, through-thickness) temperature remains below its short-term-temperature limit for the 72-hour duration of the accident. Both the fuel-cladding and the MPC-confinement boundary temperatures remain below their respective short-term-temperature limits at 72 hours, the fuel cladding by over 150°F, and the confinement boundary by almost 175°F. Table 11.2.9 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, summarizes the temperatures at several points in the HI-STORM 100 System at 33 hours and 72 hours after complete, inlet-air-duct blockage.

The thermosiphon effect is credited in the determination of the maximum allowable fuel heat emission rates (via maximum burnup, maximum decay heat, minimum cooling time limits) in SAR Section 10.2. Incorporation of the MPC thermosiphon internal convection phenomenon, as described in Chapter 4 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1, enables the maximum, design-basis, PWR-decay-heat load to rise to about 29 kW. The thermosiphon effect also shifts the highest temperatures in the MPC enclosure vessel toward the top of the MPC. The peak, MPC-lid, outer-surface temperature, for example, is computed to be about 450°F in the thermosiphon-enabled solution compared with about 210°F in the thermosiphon-suppressed solution, with both solutions computing approximately the same peak cladding temperature. In the 100 percent, inlet-duct-blockage condition, the heated MPC lid and MPC shell become effective heat dissipaters because of their proximity to the overpack outlet ducts and because the thermal radiation heat transfer rises at the fourth power of absolute temperature. As a result of this increased heat rejection from the upper region of the MPC, the time limits for reaching the short-term peak fuel-cladding temperature limits calculated without thermosiphon (72 hours) remains bounding.

Under the complete, air-inlet-duct-blockage condition, it must also be demonstrated that the MPC internal pressure does not exceed its design-basis accident limit. The bounding MPC internal pressure calculated at an ambient temperature of 80°F, 100 percent fuel rods ruptured, design-basis insolation, and maximum decay heat is 185.5 psia, as discussed in Section 8.2.14.2. This calculated pressure is for an MPC cavity bulk gas temperature of 513.6°K. Using this initial pressure, a bounding increase in the MPC cavity temperature of 184°F (102.2°K, maximum of MPC shell or fuel cladding temperature rise 33 hours after blockage of all four ducts; see HI-STORM 100 System FSAR Table 11.2.9), the reduction in the bulk average gas temperature due to increased MPC heat dissipation at higher pressure of 62.1°F (34.5°), and the Ideal Gas Law, the resultant MPC internal pressure is calculated to be 209.9 psia (195.2 psig), which is less than the accident design pressure of 200 psig (HI-STORM 100 System FSAR Table 2.2.1). The HI-STORM 100 System FSAR generic assumption of an annual average temperature of 80°F bounds the Diablo Canyon site annual-average temperature of 55°F. The HI-STORM 100 System FSAR uses 800 g-cal/cm^2 per day for the full insolation level as recommended in 10 CFR 71 (averaged over a 24-hour period as allowed in NUREG-1567). The maximum insolation values for the ISFSI site are estimated to be 766 g-cal/cm^2 per day for a 24-hour period and are therefore bounded by the analysis in the HI-STORM 100 System FSAR, as amended by LAR 1014-1.

8.2.15.3 Dose Calculations for 100 Percent Blockage of Air Inlet Ducts

As shown in the analysis of the 100 percent blockage of air inlets accident in the HI-STORM 100 System FSAR, the shielding capabilities of the HI-STORM 100 System are unchanged because the section average concrete temperature does not exceed its short-term-condition design temperature limit for the duration of the accident. The Diablo Canyon ISFSI TS require the blockage to be cleared within 8 hours of declaring the heat removal system inoperable. Assuming the blockage occurs just after the last 24-hour surveillance is performed, the 8-hour completion time provides a total of 32 hours in this condition, which is less than the 72-hour analyzed duration of the event. The concrete, fuel cladding and MPC shell do not reach their short-term-temperature limits over the entire analyzed 72-hour duration of the event. In addition, the emergency procedures will require an inspection of the ISFSI following a tornado, which will shorten the time to complete clearing the blockage. The elevated temperatures will not cause a breach of the confinement system and the short-term, fuel-cladding-temperature limit is not exceeded. Therefore, there are no direct or airborne radiation consequences of this accident.

For complete blockage of air inlet ducts it is estimated that the removal, cleaning, and replacement of the affected mesh screens will take two people approximately 2 hours. The radiation doses to workers who remove debris blocking the inlet ducts are estimated to be double those conservatively estimated for the analysis of the partial inlet blockage in Section 8.1.4. The dose rate at this location is estimated to be 58 mrem/hour. The total exposure for two people taking 2 hours to perform these corrective actions is 0.232 man-rem.

8.2.16 TRANSMISSION TOWER COLLAPSE

Two 500-kV transmission towers are located in the vicinity of the ISFSI storage pads and CTF. This section addresses the impact of a fallen transmission tower on a loaded overpack. During transportation to the CTF and all handling and lifting activities at the CTF, a loaded transfer cask is protected from the impact of a falling transmission tower at all times by the structure of the cask transporter. Therefore, an analysis of the transfer cask for tower collapse impact loads is not required and has not been performed. A postulated transmission tower collapse at both the ISFSI storage site and CTF was analyzed to demonstrate that there is no loss of confinement from damage to an MPC during both transfer operations or while stored at the ISFSI pad in an overpack. The collapse of a transmission tower is classified as Design Event IV, as defined by ANSI/ANS-57.9.

8.2.16.1 Cause of Transmission Tower Collapse

The transmission tower collapse is postulated as a consequence of extreme wind speeds (above 84 mph) creating greater than design loads on the tower structure.

8.2.16.2 Analysis of the Transmission Tower Collapse

The location of the transmission towers with respect to the CTF and ISFSI storage pads is shown in Figure 2.1-2. A transmission tower is postulated to collapse by hinging of the legs and failure of braces without incident of leg or pile foundation pullout or lateral failure due to wind- or tornado-wind-generated loads. The transmission tower is a four-legged structure with a "T" shape at the top. Based on the location of the transmission corridor with respect to the CTF and the ISFSI storage pad and the conduct of loading operations, in the unlikely event of a collapse, a tower could impact the loaded overpack in different orientations at the CTF and the storage pad. At the CTF, the tower collapse is modeled with the pointed section of the "T" cross-bar impacting the MPC lid directly because the overpack may not have its top lid installed at the time of the event. At the ISFSI, the flat side of the "T" cross-bar impacts the overpack top lid.

A commercial computer code developed by the Livermore Software Technology Corporation and QA validated by Holtec International, LS-DYNA (Reference 26), was used to numerically model the problem and develop the impact forces of the tower structure on the target. LS-DYNA is a general purpose, explicit finite element program used to analyze the nonlinear dynamic response of two- and three-dimensional inelastic structures.

There are two towers that are close enough in proximity to the CTF and ISFSI storage site to impact a cask if a tower collapse were to occur. The applicable physical characteristics for the two transmission towers are:

(1) One tower has a height of approximately 125 ft, measured from the ground to the highest point. It is located, at its nearest foundation, approximately 100 ft west of the ISFSI pads and 60 ft south of the CTF. It has a total structural weight of approximately 25 kips.

(2) The other tower has a height of approximately 135 ft, measured from the ground to the highest point. It is located, at its nearest foundation, approximately 60 ft east of the ISFSI pads. It has a total structural weight of approximately 31 kips.

The analysis evaluates the impact forces generated by collapse of the second tower as the governing case since it is a taller and heavier tower.

8.2.16.2.1 Tower Collapse at the CTF

The LS-DYNA computer simulation of the tower collapse at the CTF models the pointed portion of the "T" bar impacting the MPC lid. The force of the tower impact on the MPC lid is 427 kips. This force is much smaller than the allowable impact force for the weld (2,789 kips) determined in the tornado-missile analysis, and thus will not cause a breach of the MPC confinement boundary. The maximum local stress of the MPC lid due to the impact is 14.6 ksi, which is smaller than the yield stress of the lid material (18.8 ksi). The potential for MPC-lid puncture due to this event is bounded by the intermediate-missile evaluation described in Section 8.2.2. The design-basis intermediate missile (a 760-lb insulator string traveling at 157 mph) is shown not to penetrate the 9-1/2-inch-thick MPC lid.

8.2.16.2.2 Tower Collapse at the ISFSI Storage Pad

The LS-DYNA computer simulation of the tower collapse at the ISFSI storage pad models the flat side of the "T" bar impacting the overpack top lid. The unfiltered impact force was computed to be 534 kips. To convert this to an equivalent g-load on the overpack, the 534 kips is divided by the weight of the loaded overpack:

$$534 / 360 = 1.48 \text{ g}$$

The overpack structure is designed to withstand a 45-g deceleration. Therefore, the impact of the force due to the transmission tower collapse is bounded with margin. The horizontal component of the impact force is less than 93 kips, which is bounded by the large tornado missile load of 122 kips described in Section 8.2.2. The overturning moments are also bounded for the effects on the anchorage to the ISFSI pad. MPC confinement boundary integrity related to tower impact discussed in Section 8.2.16.2.1 is applicable at the pad.

8.2.16.3 Dose Calculation for Transmission Tower Collapse

There are no offsite dose consequences as a result of this accident because the MPC confinement boundary remains intact. Potential damage to the overpack structure as a result of this event will vary based on the actual location and severity of the impact on the overpack. Based on the loads described above, no significant damage to the shielding effectiveness of the overpack is expected. If necessary, corrective actions will be implemented based on the nature of the damage in a time frame commensurate with safety significance.

8.2.17 NONSTRUCTURAL FAILURE OF A CTF LIFT JACK

This section addresses the nonstructural failure of one CTF lift jack on a loaded overpack requiring convective cooling. Three lift jacks are used simultaneously to raise and lower the CTF lifting platform on which the overpack rests. A postulated failure of one lift jack at the CTF was evaluated as a hypothetical accident. The nonstructural failure of a lift jack at the CTF is classified as Design Event IV, as defined by ANSI/ANS-57.9.

The lift jacks and platform are designed using the applicable guidelines of NUREG-0612 and seismically analyzed to ensure that structural failure is not a credible event. The CTF design criteria, facility description, and operations and maintenance activities are presented in Sections 3.3.4, 4.4.5, and 5.1, respectively.

8.2.17.1 Cause of Nonstructural Failure of a CTF Lift Jack

The nonstructural failure of a lift jack is postulated as a consequence of an electrical or mechanical malfunction of a lift jack component causing all lift jacks to stop.

8.2.17.2 Analysis of the Nonstructural Failure of a CTF Lift Jack

The CTF is designed to position an overpack sufficiently below grade where the transfer cask can be mated to the overpack using the cask transporter. In this position, the top approximately 3 ft of the overpack remains above grade while the base of the overpack is in a confined air space. The CTF lift platform, suspended by each jack screw, raises and lowers the overpack. Three lift jacks provide the lifting force for the lifting platform. The jacks are located on the circumference of the main shell in the extensions, 120 degrees apart. The jacks are supported at the top end and use a traveling-nut design. The captured nut travels along the rotating threaded jack screw shaft to provide the lifting and lowering motion for the lifting platform. All jacks operate in unison to keep the platform level through the entire travel range (approximately 150 inches).

The CTF lifting platform provides the support of the overpack and transmits the lifting jack force to the overpack. The platform provides a level base on which the overpack rests. To interface with the lifting jacks, the platform has extensions that enter into each main shell extension. The location and controlled movement of the jacks afford uniform loading of the lifting platform. The main shell provides radial guidance of the lifting platform.

It is postulated that if one lift jack fails, the platform and potentially a loaded overpack requiring convective cooling would be unable to be raised out of the confined air space for an extended period of time while corrective actions are performed. The design of the jack control system incorporates protective features whereby all jacks are stopped when a mismatch in the performance between operating jacks is detected. Thus, there is no mechanical damage to the overpack, and the only concern in this event is keeping the MPC and overpack sufficiently cooled and removing the overpack from the CTF.

By conservative analysis, the overpack can withstand a loss of normal ventilation cooling for up to 22 hours before the short-term temperature limit of the fuel cladding is reached. The conservative limit of 22 hours is based on the observation that the HI-STORM 100 System FSAR Section 4.5.2 case of a transfer cask in an underground silo envelopes the overpack in the CTF vault due to the overpack's larger thermal mass, greater opportunity for convective cooling, and lower initial temperature. If it is determined that the 22 hours may be exceeded during an actual event, the overpack is capable of being removed using the cask transporter with the HI-STORM lift links and lifting brackets.

It is concluded that the postulated nonstructural failure of a lift jack accident will not result in the breach of MPC confinement, fuel cladding damage, or prevent MPC retrievability.

8.2.17.3 Dose Calculation for Nonstructural Failure of a CTF Lift Jack

Because the confinement boundary is not breached, there are no releases and no corresponding offsite dose consequences as a result of this accident.

The dose consequences to personnel implementing corrective actions for this accident are estimated using the dose rate for the removal of blockage from the air inlet ducts (Section 8.1.4). Using the blockage removal dose rate of 58 mrem/hour for these corrective actions is conservative because it includes contribution from the affected cask, as well as adjacent casks on the ISFSI storage pad. This accident involves only one cask at the CTF. Assuming it takes a crew of 5 a total of one, 8-hour shift spent in close proximity to the cask, the total accumulated dose to mitigate this event would be:

$$58 \text{ mrem/hr} \times 8 \text{ hr} \times 5 \text{ people} = 2.32 \text{ man-rem}$$

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TABLE 8.1-1

OFF-NORMAL OPERATION ANNUAL DOSES AT THE SITE BOUNDARY AND FOR THE NEAREST RESIDENT DUE TO EFFLUENT RELEASE FROM A SINGLE HI-STORM CASK

	Site Boundary Dose[a] (mrem)	Nearest Resident Dose[b] (mrem)
Whole body ADE[c]	1.27E-03	5.33E-03
Thyroid ADE	1.02E-04	4.31E-04
Critical Organ ADE (Max)	9.31E-03	3.92E-02

Note:
[a] Occupancy at the site boundary is assumed to be 2,080 hrs/yr.
[b] Occupancy for the nearest resident is assumed to be 8,760 hrs/yr. Also, the site boundary χ/Q is used for the nearest resident; this is conservative because the nearest resident is located farther away from the release point than the site boundary.
[c] ADE is annual dose equivalent.

CHAPTER 9

CONDUCT OF OPERATIONS

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CHAPTER 9

CONDUCT OF OPERATIONS

This chapter discusses the PG&E organization for the design, fabrication, construction, testing, operation, modification, and decommissioning of the Diablo Canyon ISFSI. Included are descriptions of organizational structure, personnel responsibilities and qualifications, and PG&E interface with contractors and other outside organizations.

Programs under 10 CFR 50 for DCPP, such as radiation protection, environmental monitoring, emergency preparedness, quality assurance, and training will be adopted as necessary to ensure the safe operation and maintenance of the Diablo Canyon ISFSI under 10 CFR 72. PG&E has included in the ISFSI license application the following proposed plans that support the conduct of ISFSI operations: an Appendix to the DCPP Physical Security Plan, a Safeguards Contingency Plan, a Security Training and Qualification Plan, an Emergency Plan, a Quality Assurance (QA) Program, and a Training Program.

As appropriate, 10 CFR 50 license requirements will be removed from ISFSI procedures upon termination of the 10 CFR 50 licenses. During this transition period, appropriate 10 CFR 72.48 reviews will be conducted to ensure continued compliance with ISFSI 10 CFR 72 license requirements. This process will result in stand-alone ISFSI programs that implement the 10 CFR 72 license. PG&E will maintain the appropriate administrative and managerial controls at the ISFSI until the DOE takes title to and assumes responsibility for the spent fuel.

9.1 ORGANIZATIONAL STRUCTURE

9.1.1 CORPORATE ORGANIZATION

The organization charts shown in Figures 9.1-1 and 9.1-2 represent the organizational relationships throughout the life of the ISFSI while DCPP units are operating. Relationships between corporate personnel and Diablo Canyon ISFSI onsite personnel are depicted in the figures. While DCPP units are operating, the costs for construction and operation of the Diablo Canyon ISFSI will be funded from revenues generated from operation of the units. Upon shutdown of the operating units, the costs for construction, operation, and decommissioning of the Diablo Canyon ISFSI will be funded from the DCPP Decommissioning Trust, which has been approved by the California Public Utilities Commission (CPUC). All costs are monitored and controlled by the ISFSI Program Manager during the ISFSI preoperations phase, and by the Station Director during the ISFSI operations phase.

Following decommissioning of both operating units and termination of the 10 CFR 50 operating licenses, the Diablo Canyon ISFSI organization will change. The revised ISFSI organization will be dependent on the new PG&E organization that will result following the decommissioning of the operating units. PG&E will notify the NRC of the new Diablo Canyon ISFSI organization at that time. (The operating licenses for DCPP Units 1 and 2 expire in 2021 and 2025, respectively.)

9.1.2 CORPORATE FUNCTIONS, RESPONSIBILITIES, AND AUTHORITIES

The Senior Vice President, Generation and Chief Nuclear Officer is the corporate executive responsible for overall ISFSI safety and is responsible for taking measures needed to ensure acceptable performance of the staff in designing, fabricating, constructing, testing, operating, modifying, decommissioning, and providing technical support to the ISFSI. The Senior Vice President, Generation and Chief Nuclear Officer, reports to the President and Chief Executive Officer of PG&E.

The Vice President, Nuclear Services, is responsible for providing engineering and design services, safety assessments, and licensing services for the ISFSI. He is the corporate interface with the CPUC for all ISFSI cost matters. The Vice President, Nuclear Services, reports to the Senior Vice President, Generation and Chief Nuclear Officer.

The Vice President, Diablo Canyon Operations, will be responsible for ISFSI operations. The Vice President, Diablo Canyon Operations, reports to the Senior Vice President, Generation and Chief Nuclear Officer.

The Nuclear Safety Oversight Committee (NSOC) is a corporate committee that reports to the Senior Vice President, Generation and Chief Nuclear Officer, and is chaired by the Vice President, Nuclear Services. NSOC membership, functions, meeting requirements and responsibilities are described in Sections 17.1 and 17.2 of the DCPP Final Safety Analysis Report (FSAR) Update (Reference 1).

The Diablo Canyon ISFSI will be operated under the same corporate management organization responsible for the operation of DCPP. Throughout the ISFSI lifetime, legal support will be available from PG&E corporate headquarters; technical and operational support will be available from DCPP personnel and outside consultants. This support will be provided, when needed, for licensing, QA, engineering, radiation protection, maintenance, testing, emergency planning, security, and decommissioning.

As shown in Figures 9.1-1 and 9.1-2, the QA and quality control functions will be performed by personnel independent of the ISFSI line organization. The results of QA audits and recommendations for improvement will be provided directly to the ISFSI Program Manager (during the preoperations phase), the Station Director (during ISFSI operations phase), and the Senior Vice President, Generation and Chief Nuclear Officer (during both phases). The frequency and scope of QA audits is described in Section 17.18 of the QA Program that is included as Attachment E to the ISFSI license application.

9.1.3 IN-HOUSE ORGANIZATION

The Diablo Canyon ISFSI will be designed, constructed, tested, and operated under the same organization responsible for the design, testing, and operation of the DCPP. The only difference is that during the Diablo Canyon ISFSI preoperations phase, the ISFSI Program Manager will be responsible for day-to-day management of ISFSI activities; whereas during the Diablo Canyon ISFSI operations phase, the Station Director will be responsible for the day-to-day management of the ISFSI.

Figure 9.1-1 shows the organization that will be in place during the ISFSI preoperations phase, including design, fabrication, construction, fuel loading, testing, and initial operation of the first cask. During the preoperations phase, the Diablo Canyon ISFSI Program Manager is responsible for day-to-day management of ISFSI activities and ensuring that the design, fabrication, construction, fuel loading, testing, and initial operation of the first cask are safely conducted. Cost control for all of these activities is the responsibility of the Diablo Canyon ISFSI Program Manager. The ISFSI Program Manager is responsible also for the development of the ISFSI license application and associated coordination with appropriate federal and state agencies leading to obtaining the 10 CFR 72 license. The Diablo Canyon ISFSI Program Manager reports to the Director, Stragetic Projects and Assistant to the Vice President, Nuclear Services. The Director Strategic Projects and Assistant to the Vice President, Nuclear Services reports to the Vice President, Nuclear Services. The Vice President, Nuclear Services, is responsible for overall safety of ISFSI activities, and the industrial safety program, during the ISFSI preoperations phase.

Figure 9.1-2 shows the organization that will be in place during the ISFSI operations phase, including design, fabrication, construction, fuel loading, and testing of all casks subsequent to the initial cask. During ISFSI operations, the Station Director will be responsible for the overall safety of ISFSI activities, including fuel loading, testing, maintenance, and operation of all subsequent casks. The Station Director reports directly to the Vice President, Diablo Canyon Operations. The Manager, Operations and the Director, Maintenance Services report directly to the Station Director. The Manager, Operations is responsible for administering, coordinating, planning, and scheduling all ISFSI operating activities. He is responsible for ensuring that appropriate operating procedures are available and that operating personnel are familiar with the procedures. The Director, Maintenance Services exercises direct supervision over ISFSI maintenance and work planning. ISFSI Specialist will report to either the Manager, Operations or the Director, Maintenance Services according to their discipline. The Engineering Director will be responsible for the design, fabrication, and modification of all subsequent casks. The Engineering Director reports to the Vice President, Nuclear Services.

Throughout both phases, functions such as engineering, design, construction, QA, radiation protection, testing, operations, and security will be performed by DCPP personnel. The existing DCPP Plant Staff Review Committee (PSRC) reviews matters affecting the safe storage of spent nuclear fuel. The PSRC is chaired by the Station Director. PSRC membership, functions, meeting requirements and responsibilities are described in Sections 17.1 and 17.2 of the DCPP FSAR Update.

9.1.4 RELATIONSHIPS WITH CONTRACTORS AND SUPPLIERS

All activities associated with the ISFSI are managed and approved by PG&E. The cask vendor is responsible for providing the HI-STORM 100 System. Consulting firms may be used to support the design and engineering efforts for the ISFSI project, and for the construction of associated structures and components, including the ISFSI storage pad. Qualified vendors may be selected to provide other services and/or equipment as needed.

During the preoperations phase, the Diablo Canyon ISFSI Program Manager is responsible for providing oversight of work activities performed by contractors. Fewer contractors will be involved during the ISFSI operations phase, and their activities will be managed by the Station Director.

9.1.5 TECHNICAL STAFF

The PG&E staff that supports DCPP Units 1 and 2 operations is described in Section 13.1 of the DCPP FSAR Update. This staff will also support the Diablo Canyon ISFSI. The functions, responsibilities and authorities of the Diablo Canyon ISFSI personnel identified in Figures 9.1-1 and 9.1-2 are described in Section 13.1 of the DCPP FSAR Update. Not identified in Section 13.1 of the DCPP FSAR Update is the ISFSI Program Manager during the preoperations phase, whose responsibilities are as described in Section 9.1.3. Also not identified in Section 13.1 of the DCPP FSAR Update is the ISFSI Specialist during the operations phase, whose responsibilities are as described in Section 9.1.6. The qualifications of the PG&E technical staff meet or exceed the requirements specified in Section 9.1.7.

The design for the ISFSI storage system will be primarily performed by the cask vendor. Designs, calculations, and analyses performed by the cask vendor and any other vendors will be reviewed and approved by Diablo Canyon personnel prior to construction.

9.1.6 OPERATING ORGANIZATION, MANAGEMENT, AND ADMINISTRATIVE CONTROL SYSTEM

9.1.6.1 Onsite Organization

This section describes the ISFSI operations organization that will be in place during long-term storage of spent nuclear fuel. The ISFSI operations organization is shown in Figure 9.1-2 and is the same organization currently responsible for the operation of DCPP. Approximately 11 full-time equivalent personnel will be used from the existing DCPP organization to perform the functions of ISFSI specialists and security. Lines of authority, responsibility, and communication will be defined and established for all ISFSI organization positions. These relationships will be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions.

9.1.6.2 Personnel Functions, Responsibilities and Authorities

The Station Director will report directly to the Vice President, Diablo Canyon Operations, and will be responsible for the safe operation of the ISFSI, maintaining personnel trained and qualified in accordance with the Diablo Canyon ISFSI operations training program (as described in Attachment D to the ISFSI license application), and operation of ISFSI equipment that is important to safety. The Station Director will provide direction for the safe operation, maintenance, radiation protection, training and qualification, and security of the ISFSI and personnel.

ISFSI specialists and security staff will be responsible for the day-to-day operation of the ISFSI. They will perform their activities in accordance with the requirements of the Diablo Canyon ISFSI license, TS, physical security plan, plant procedures, and applicable state and federal regulations. Security staff personnel will be responsible for ISFSI site security during routine, emergency, and contingency operations.

In order to ensure continuity of operation and organizational responsiveness to off-normal situations, a formal order of succession and delegation of authority will be established. The Station Director will designate in writing personnel who are qualified to act as the Station Director in his absence.

9.1.6.3 Administrative Control

Planned and scheduled internal and external quality assurance audits in accordance with the DCPP Quality Assurance Program will be performed to evaluate the application and effectiveness of management controls, procedures, and other activities affecting safety. The audit program will describe audit frequency, methods for documenting and communicating audit findings, resolution of issues, and implementation of corrective actions.

The existing DCPP change control program will be revised to incorporate 10 CFR 72.48 and other ISFSI regulatory requirements. The DCPP change control program will be used to manage Diablo Canyon ISFSI change control.

9.1.7 PERSONNEL QUALIFICATION REQUIREMENTS

Each member of the DCPP staff performing work on the Diablo Canyon ISFSI will meet or exceed the qualifications of Regulatory Guide 1.8 (Reference 2), with the exceptions as noted in the License Application, Attachment E, "Quality Assurance Program," Table 17.1-1. In addition, the Station Director and the ISFSI specialists and security staff are qualified as described below:

The Station Director, at the time of assuming the responsibilities for ISFSI operations, shall have a minimum of 8 years of power plant experience, of which a minimum of 3 years shall be nuclear power plant experience. A maximum of 2 years of the remaining 5 years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one basis. The Station Director will be trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program.

The ISFSI specialists and security staff, at the time of appointment to their positions, shall have a high school diploma or successfully completed the General Education Development test. ISFSI specialists shall have 2 years of power plant experience of which a minimum of 1 year shall be nuclear power plant experience. Consistent with the assigned duties, ISFSI specialists will be trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program training and qualification requirements. Security staff that support the ISFSI will be trained and qualified in accordance with the DCPP Security Training and Qualifications Plan requirements.

During loading of the ISFSI, fuel handling operations will either be performed by, or supervised by, DCPP personnel trained and qualified by the Diablo Canyon ISFSI Operations Training Program. During ISFSI operations, operation of equipment and controls that are identified as important to safety for the ISFSI will be limited to personnel who are trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program, or personnel who are under the direct visual supervision of a person who is trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program.

9.1.8 LIAISON WITH OUTSIDE ORGANIZATIONS

All activities associated with ISFSI operations are managed and approved by PG&E. These activities will be performed in accordance with approved procedures. The cask vendor provides engineering, technical support, and other services for the ISFSI project relating primarily to the design and construction of structures and components. Other qualified vendors may be selected to provide specialty services and/or equipment. Interface with DOE, cask vendor, and other outside organizations is performed in accordance with contractual agreements.

9.1.9 REFERENCES

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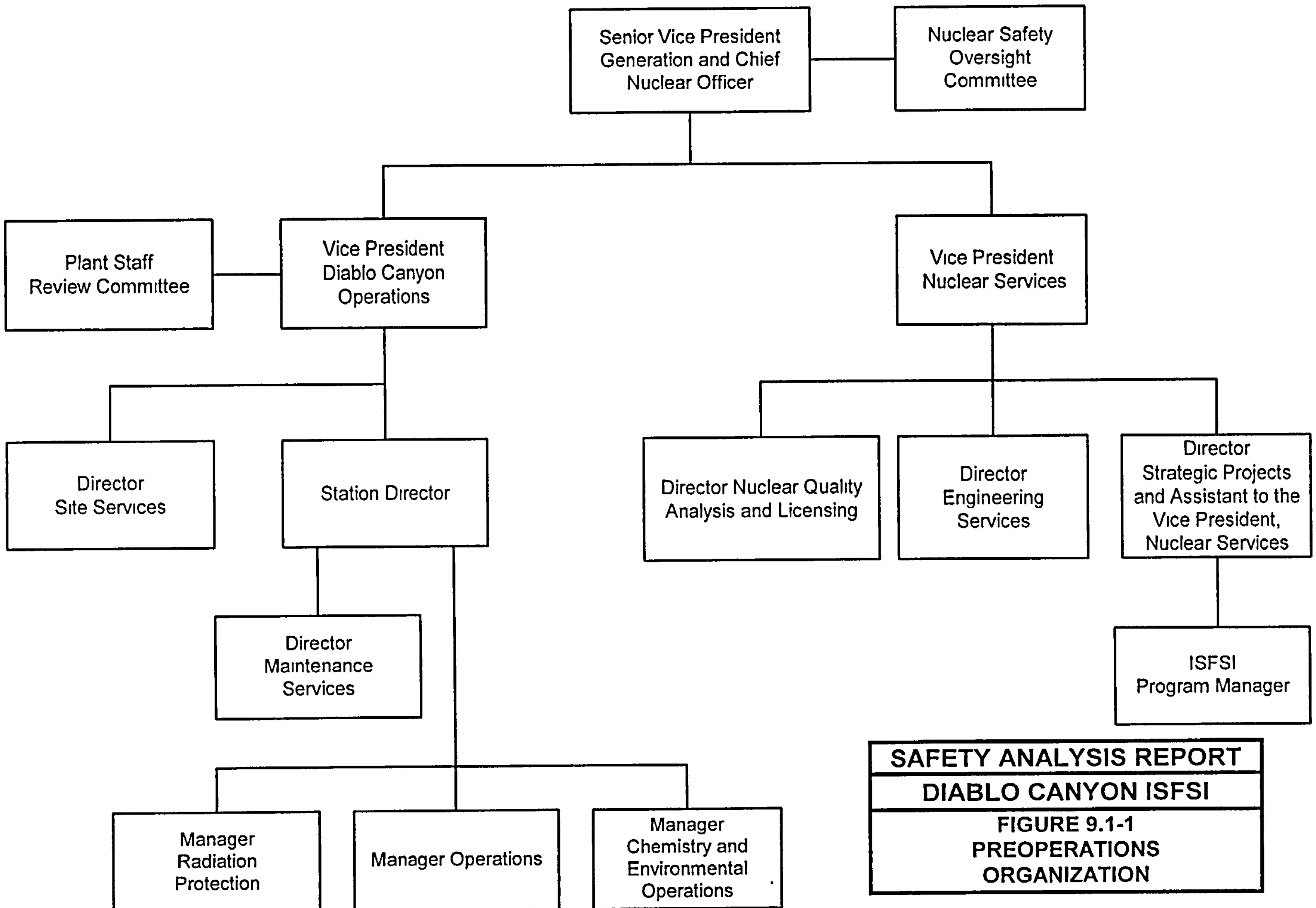

SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

FIGURE 9.1-1
PREOPERATIONS ORGANIZATION

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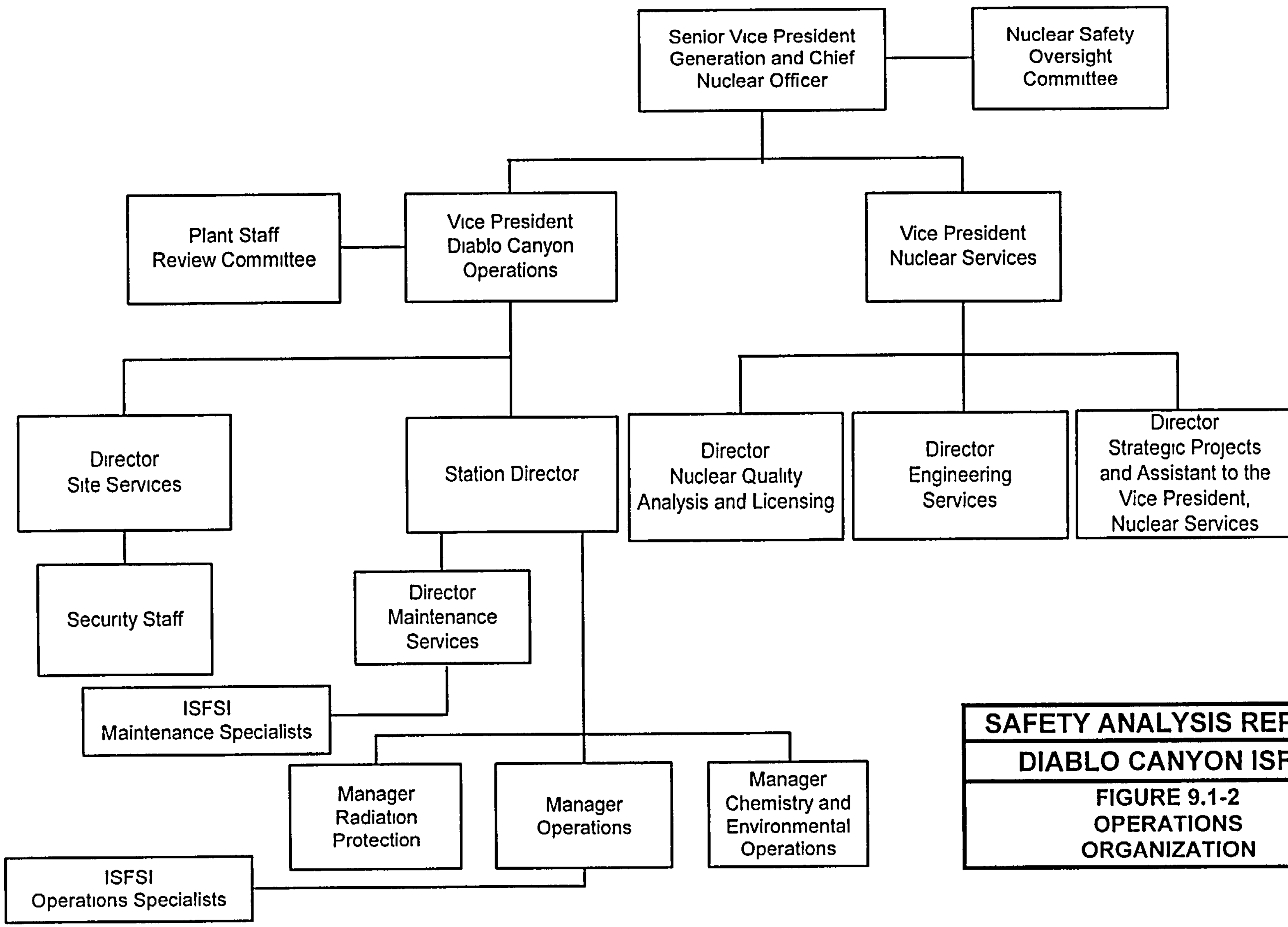

SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

FIGURE 9.1-2
OPERATIONS ORGANIZATION

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CHAPTER 10

OPERATING CONTROLS AND LIMITS

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10.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides an overview of, and the general bases for, operating controls and limits specified for the Diablo Canyon ISFSI.

10.2.1 FUNCTIONAL AND OPERATING LIMITS, MONITORING INSTRUMENTS, AND LIMITING CONTROL SETTINGS

This section provides requirements for the controls or limits that apply to operating variables classified as important to safety and are observable and measurable. The operating variables required for the safe operation of the Diablo Canyon ISFSI are:

- Spent fuel characteristics
- Spent fuel storage cask (SFSC) heat removal capability
- Multi-purpose canister (MPC) dissolved boron concentration level
- Annulus gap water requirement during moisture removal for loading and reflooding for unloading
- Water temperature of a flooded MPC
- MPC vacuum pressures
- MPC recirculation gas exit temperature
- Helium purity
- MPC helium backfill pressures
- Gas exit temperature of a MPC prior to reflooding
- SFSC time limitation while seated in the cask transfer facility (CTF)
- Fuel cladding oxide thickness

Each of the specifications for these characteristics is provided below with the exception of the MPC dissolved boron concentration, SFSC time limitation in the CTF, and heat removal parameters, which are provided in the Diablo Canyon ISFSI Technical Specifications (TS) and their bases. Although provided in the SAR sections below, the TS and bases also provide Limiting Conditions for Operation and bases for maintaining the integrity of the MPC during loading and unloading. These include vacuum pressure, recirculation gas temperature, backfill pressure, and leak rate during loading, and exit gas temperature during unloading.

10.2.1.1 Fuel Characteristics

The Diablo Canyon ISFSI is designed to provide interim storage for up to 4,400 fuel assemblies, which accommodates the number of assemblies predicted to be used during the licensed operating life of the plant. The Diablo Canyon ISFSI storage system will use four MPC types for the storage of fuel assemblies, fuel debris and associated nonfuel hardware. The DCPP fuel will normally be stored as nonconsolidated fuel assemblies both with and without control components. The intact fuel assemblies will be stored in either the MPC-24, MPC-24E, MPC-24EF, or MPC-32 canisters. The damaged fuel assemblies can only be stored in MPC-24E or MPC-24EF canisters, and the fuel debris can only be stored in MPC-24EF canisters. Damaged fuel or fuel debris will be placed in a damaged fuel container before loading into an MPC. The fuel debris can be consolidated, however, the amount of debris is limited to the equivalent of a single intact fuel assembly.

Fuel qualification is based on the requirements for criticality safety, decay heat removal, radiological protection, and structural integrity. The analysis presented in Chapters 4, 7 and 8 of this SAR documents the qualification of DCPP inventory of spent fuel assemblies and associated nonfuel hardware for storage in the Diablo Canyon ISFSI storage system design.

During the operation of DCPP, fuel integrity has been, and continues to be, monitored. Through the detection of radiochemistry changes in the reactor coolant system, most fuel damage is assessed. When damaged rods are suspected, assemblies are inspected as they are removed from the core. All assemblies with positive indication of damage are again inspected in the spent fuel pool (SFP) to determine numbers and location of rods in the assembly that have failed cladding. If the fuel assembly is to be placed back in the reactor core, any failed rods are removed and replaced with nonfuel rods of equivalent dimensional properties. If the suspected damaged fuel assemblies are at the end of their cycle, the assemblies may be stored in the SFP without repair. During this process, all known rod failures are noted and their assemblies are tracked. If the failure is visible from the exterior of the assembly, the damage may be video taped. For assemblies that are removed from the reactor core and were not inspected at that time, similar inspections will be performed prior to loading these assemblies into an MPC for storage. This will ensure that there are no undetected failed rods in any assembly that is placed in an MPC.

Under this failure detection process, inspections to date have found limited failures. Where single failed rods have been identified and removed, they are being stored in the SFP and will ultimately be stored in an MPC that can contain fuel debris. This detection process, along with the past history of plant operations and SFP fuel storage, provide a high level of confidence that the current spent fuel and associated nonfuel hardware will meet the criteria for storage in the appropriate MPC. In addition, based on the condition of the current spent fuel, the continued maintenance of the reactor coolant and SFP water chemistry requirements, and proper handling of the fuel, there is a high level of confidence that future spent fuel assemblies will meet the criteria for storage in the appropriate MPC.

DCPP will develop a cask-loading plan to ensure that no damaged fuel assemblies are loaded into an MPC-24 or MPC-32 canister. Damaged fuel will only be allowed to be stored in either an MPC-24E or MPC-24EF canister. Fuel debris will only be allowed to be stored in an MPC-24EF canister. If the structural integrity criterion is met, then approval for dry storage for a given assembly is made. This qualification will be documented and subsequently referenced in Diablo Canyon ISFSI operating procedures prior to loading spent fuel assemblies into the MPC.

The cask-loading plan will provide a loading sequence based on the various characteristics of the fuel assemblies being loaded. There are two main fuel-loading strategies that are used: uniform fuel loading and regionalized fuel loading. In addition, there is a fuel loading sub-strategy called preferential fuel loading. All of these loading strategies are designed to ensure that the design bases of the fuel, MPCs, and overpacks are maintained.

Uniform fuel loading is used when the fuel assemblies being loaded are all of similar burnup rates, decay heat levels, and post-irradiation cooling times. In this case the actual location of each assembly is less critical and assemblies can be placed at any location in the MPC. However, if the post-irradiation cooling times for any of the assemblies are different by ≥ 1-year, preferential fuel loading is required to be considered.

Preferential fuel loading requires that the fuel assemblies with the longest post-irradiation cooling times be located at the periphery of the MPC basket. Fuel assemblies with shorter post-irradiation cooling times are placed toward the center of the basket. Preferential fuel loading is a requirement in addition to other MPC loading restrictions such as those for nonfuel hardware and damaged fuel containers.

Regionalized fuel loading is used when high heat emitting fuel assemblies are to be stored in an MPC. This loading strategy allows these specific assemblies to be stored in locations in the center of the MPC basket provided lower heat emitting fuel assemblies are stored in the peripheral storage locations. Use of regionalized fuel loading must consider other restrictions on loading such as those for nonfuel hardware and damaged fuel containers. Regionalized fuel loading meets the intent of preferential fuel loading.

The following controls will ensure that each fuel assembly is loaded into a known cell location within a qualified MPC:

- A cask-loading plan will be independently verified and approved.
- A fuel movement sequence will be based upon the written loading plan. All fuel movements from any rack location will be performed under controls that will ensure strict, verbatim compliance with the fuel movement sequence.

- Prior to placement of the MPC lid, all fuel assemblies and associated nonfuel hardware, if included, will be either video taped or visually documented by other means, and independently verified, by ID number, to match the fuel movement sequence.

A cognizant engineer is responsible for performing a third independent verification to ensure that the fuel in the MPCs is placed in accordance with the original cask-loading plan.

Based on the qualification process of the spent fuel and the administrative controls used to ensure that each fuel assembly is loaded into the correct location within an MPC, incorrect loading of an MPC is not considered to be a credible event.

10.2.1.2 Fuel Characteristics (Allowable Content)

The characteristics of the fuel that are allowable for storage in the MPCs are as follows:

- Intact fuel assemblies, damaged fuel assemblies, fuel debris, and nonfuel hardware meeting the limits specified in Tables 10.2-1, 10.2-2, 10.2-3, and 10.2-4 and other referenced tables may be stored in the SFSC system. These SAR tables and specifications are duplicated in Tables 2.2-1 through 2.2-10 of the Diablo Canyon ISFSI TS.

- For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining intact fuel assemblies in the MPC shall meet the decay heat generation limits for the damaged fuel assemblies. This requirement applies only to uniform fuel loading.

10.2.1.3 Uniform and Preferential Fuel Loading

Fuel assemblies used in uniform or preferential fuel loading shall meet all applicable limits specified in Tables 10.2-1, 10.2-2, 10.2-3, 10.2-4, and 10.2-5. Fuel assembly burnup, decay heat, and cooling time limits for uniform loading are specified in Tables 10.2-6 and 10.2-7. Preferential fuel loading shall be used during uniform loading (that is, any authorized fuel assembly in any fuel storage location) whenever fuel assemblies with significantly different post-irradiation cooling times ($\geq$ 1 year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling times shall be loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times shall be placed toward the center of the basket. Regionalized fuel loading as described in 10.2.1.4 below meets the intent of preferential fuel loading.

10.2.1.4 Regionalized Fuel Loading

Fuel may be stored using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Figures 10.2-1 through 10.2-3 (these figures are duplicated in the Diablo Canyon

ISFSI TS as Figures 2.1-1 through 2.1-3), define the regions for the MPC-24; MPC-24E/MPC-24EF; and MPC-32 models, respectively. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in Tables 10.2-8 and 10.2-9. In addition, fuel assemblies used in regionalized loading shall meet all other applicable limits specified in Tables 10.2-1, 10.2-2, 10.2-3, 10.2-4, and 10.2-5. Limitations on nonfuel hardware to be stored with their associated fuel assemblies are provided in Table 10.2-10.

10.2.1.5 For Allowable Content - Functional and Operating Limits Violations

If any fuel specifications or loading conditions above are violated, the following Diablo Canyon ISFSI TS actions shall be completed:

- The affected fuel assemblies shall be placed in a safe condition.
- Within 24 hours, notify the NRC Operations Center.
- Within 30 days, submit a special report that describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.

10.2.2 MPC LOADING CHARACTERISTICS

The confinement of radioactivity during the storage of spent fuel and associated nonfuel hardware in the MPC is ensured by the structural integrity of the strength-welded MPC. However, long-term integrity of the fuel and cladding depends on storage in an inert heat removal environment inside the MPC. This environment is established by removing water from the MPC and backfilling the cavity with an inert gas.

The loading process of an MPC involves placing a transfer cask with an empty MPC in the SFP and loading it with fuel assemblies (intact or damaged that meet the specifications for allowable content discussed above), fuel debris, and/or nonfuel hardware allowed per the type of MPC. Once this is complete a lid is then placed on the MPC. An MPC lid retention device is placed over the MPC lid and attached to the transfer cask. The transfer cask and MPC are raised to the SFP surface. The transfer cask and MPC are then moved into the cask washdown area where dose rates are measured and the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and moisture removal is performed. The MPC cavity is backfilled with helium. Additional dose rates are measured and the MPC vent and drain cover plates and closure ring are installed and welded. Nondestructive examination (NDE) inspections are performed on the welds.

As a part of the loading process there are several characteristics that must be maintained to ensure that the allowable contents placed in any MPC remains stable and intact. These characteristics involve maintaining the MPC cavity temperature. During the loading process there are times when the loaded MPC is water filled and times when it is empty of water. As a result, there are characteristics that must address each of these two conditions. One of these characteristics is MPC water temperature and the other is maintaining the borated water level and recirculation in the annular gap between the transfer cask and the MPC.

Also during the loading process there are several characteristics vital to ensuring that the resulting MPC internal environment is conducive to long-term heat removal and maintaining the integrity of the fuel cladding. These characteristics are; limiting the moisture in the MPC; backfilling the MPC with high quality inert gas; and limiting the leakage of this inert environment over time. The dry, inert and sealed MPC atmosphere is required to be in place during loading, transport and storage operations after an acceptable final NDE on the first weld of the MPC lid to its outer shell.

10.2.2.1 Annulus Gap Water Requirement

During the loading and unloading processes there are time periods when there is no water in the MPC, or it is being removed, or the inert environment in the MPC cavity has not been completely established or maintained at levels that will continue to provide adequate cooling and maintain fuel cladding integrity. During these time periods maintaining the water level in the annular gap and continuous recirculation for high heat fuel (> 22 kw) between the loaded MPC and the transfer cask ensures that the cooling capability is adequate to maintain the fuel cladding integrity. As long as the annular gap water level is maintained with borated water and the temperature of the water in the gap is maintained below boiling through recirculation, there is no time limitation for refilling the MPC with borated water or establishing an acceptable inert environment in the MPC for moderate burnup fuel (≤ 45,000 MWD/MTU). However, without recirculation there is a limit of 2 hours to establish this process or establish an inert environment. For higher burnup fuel (>45,000 MWD/MTU), which requires the use of a forced helium dehydration (FHD) system for drying, once the drying process is completed and if residual helium is not removed from the MPC, there is a limit of 2 hours to re-establish an inert environment in the MPC. This is discussed further in Section 10.2.2.3.

During the loading process, prior to start of the removal of water from the MPC through the drying process, the annular gap shall be filled and maintained full throughout the drying and backfill process. This water level shall be maintained until the MPC inert environment is established at an acceptable level to support long-term storage or the MPC is refilled with water. In addition, during an unloading process the annular gap shall be filled with water prior to removal of the inert environment in the MPC cavity.

10.2.2.2 MPC Water Temperature

During the loading and unloading processes, maintaining the integrity of the fuel in the MPC is the critical activity. As a result of decay heat produced by the spent fuel assemblies, providing a coolant source is imperative to maintaining control of cladding temperature and the fuel integrity. During these processes when there is water in the MPC, the water is considered the coolant source. As long as there is water in the MPC it will continue to perform the coolant function. This water should continue to perform its function as long as it does not reach the boiling temperature. As a result, the parameter that will best indicate the potential reduction of water would be the temperature of the water in the MPC. However, since monitoring the water temperature in the MPC directly may not always be possible, an analysis of the potential for the water to reach the boil-off temperature is performed to ensure that the boil-off temperature cannot be reached. This analysis will be based on the decay heat levels of the contents and the various volumes of water in the MPC as it is loaded. The results of this analysis will provide any time limitation or any requirement for compensatory measures.

While there is water in the MPC, there will be adequate assurance through analysis that the temperature of that water in the MPC will not reach the boil-off level and that the volume of water in the MPC is not allowed to decrease significantly. If the water temperature is shown to potentially reach the boiling level, action will be taken to limit the time of the activity to less than the time to boil off or, as a minimum, continue to replace the volume of water that is boiling off. If no action is possible to correct this condition, then the content loaded in the MPC shall be removed and placed back in the SFP.

10.2.2.3 MPC Drying Characteristics

Dependent on the allowable content of a specific MPC, cavity moisture removal can be performed by using either vacuum drying or a Forced Helium Dehydration (FHD) system after the MPC has been drained of water. See Figure 10.2-4 for a schematic diagram of the FHD system. The Standard Review Plan (SRP) acceptance criterion for dryness is ≤ 1 gram-mole per cask of oxidizing gases. This has been translated by the industry to be 3 torr for vacuum drying. For the recirculation drying process using the FHD system, measuring the temperature of the gas exiting the demoisturizer of the FHD system provides an indication of the amount of water vapor entrained in the helium gas in the MPC. Maintaining a demoisturizer exit temperature of less than or equal to 21°F for 30 minutes or more during the recirculation drying process ensures that the partial pressure of the entrained water vapor in the MPC is less than 3 torr.

If the MPC contains only moderate burnup fuel (≤ 45,000 MWD/MTU) vacuum drying can be used. In this process any water that has not drained from the MPC cavity evaporates from the MPC cavity due to the vacuum. This drying is aided by the temperature increase due to the decay heat of the fuel. To ensure adequate drying the vacuum drying pressure in the MPC must be verified to be at ≤ 3 torr for ≥ 30 minutes. This low vacuum pressure is an indication that the cavity is dry and the moisture level in the MPC is acceptable.

For any MPC that contains fuel assemblies of any authorized burnup, the FHD system can be used to remove the remaining moisture in the MPC cavity after all of the water that can practically be removed through the drain line using a hydraulic pump has been expelled in the water blowdown operation. The FHD system is required to be used for any MPCs containing at least one high burnup fuel assembly (>45,000 MWD/MTU).

The recirculation process using the FHD involves introducing dry gas into the MPC cavity that absorbs the residual moisture in the MPC. This humidified gas exits the MPC and the absorbed water is removed through condensation and/or mechanical drying. The dried gas is then forced back through the MPC until the gas exit temperature from the FHD demoisturizer is ≤ 21°F for at least 30 minutes. Meeting these temperature and time criteria ensures that the cavity is dry and the moisture level in the MPC is acceptable. The FHD system shall be designed to ensure that during normal operation (that is, excluding startup and shutdown ramps) the following criteria are met:

(1) The temperature of helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.

(2) The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).

(3) The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulated flow regime in the MPC cavity.

(4) The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr if the helium temperature at the demoisturizer outlet is ≤ 21°F for a period of 30 minutes.

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

(1) The condensing module shall be designed to devaporize the recirculating helium gas to a dew point of 120°F or less.

(2) The demoisturizer module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to ensure that the bulk moisture vaporization in the MPC has been completed.

(3) The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.

(4) The preheater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.

The design of the FHD system is subject to the confirmatory analyses listed below to ensure that the system will accomplish the performance objectives set forth in this SAR.

(1) System thermal analysis in Phase 1: Characterize the rate of condensation in the condensing module and helium temperature variation under Phase 1 operation (i.e., the scenario where there is some unevaporated water in the MPC) using a classical thermal-hydraulic model wherein the incoming helium is assumed to fully mix with the moist helium inside the MPC.

(2) System thermal analysis in Phase 2: Characterize the thermal performance of the closed loop system in Phase 2 (no unvaporized moisture in the MPC) to predict the rate of condensation and temperature of the helium gas exiting the condensing and the demoisturizer modules. Establish that the system design is capable to ensure that partial pressure of water vapor in the MPC will reach less than or equal to 3 torr if the temperature of the helium gas exiting the demoisturizer is predicted to be at a maximum of 21°F for 30 minutes.

(3) Fuel Cladding Temperature Analysis: A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4.1.1.1 through 4.4.1.1.4 with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation (design maximum heat load, no moisture, and maximum helium inlet temperature), is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

If Diablo Canyon is the first user of the FHD system designed and built for the MPC drying function, the system will be subject to confirmatory testing as follows:

(1) A representative quantity of water will be placed in a manufactured MPC (or equivalent mock-up) and the closure lid and RVOAs installed and secured to create a hermetically sealed container.

(2) The MPC cavity drying test will be conducted for the worst case scenario (no heat generation within the MPC available to vaporize water).

(3) The drain and vent line RVOAs on the MPC lid will be connected to the terminals located in the preheater and condensing modules of the FHD system, respectively.

(4) The FHD system will be operated through the moisture vaporization (Phase 1) and subsequent dehydration (Phase 2). The FHD system operation will be stopped after the temperature of helium exiting the demoisturizer module has been at or below 21°F for 30 minutes (nominal). Thereafter, a sample of the helium gas from the MPC will be extracted and tested to determine the partial pressure of the residual water vapor in it. The FHD system will be deemed to have passed the acceptance testing if the partial pressure in the extracted helium sample is less than or equal to 3 torr.

At completion of the drying operation using the FHD system, the partial pressure of the helium/water vapor will be at 3 torr or less, however, the total pressure in the MPC will be approximately 2000 torr or 3 atm. This is the result of the MPC still containing helium and the approved contents continuing to heat that helium. To complete the backfill and loading process when the FHD system is used, the contained helium/water vapor mixture in the MPC must be withdrawn down to an MPC total pressure of 10 torr. This ensures the helium backfill process can be properly completed. Once the residual helium/water vapor mixture is drawn down to 10 torr the cooling capability of the MPC is reduced. As a result, there is a 2-hour limitation during which either the backfill gas must be introduced into the MPC; or as a minimum the MPC must be refilled with helium. Either of these actions will re-establish adequate cooling capability in the MPC and ensure that the fuel cladding short-term temperature limit is not exceeded.

If the cavity moisture removal limits are not met, an engineering evaluation will be necessary to determine the potential quantity of moisture left within the MPC cavity. Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. As the quantity of moisture estimated can range over a broad scale, different recovery strategies may be necessary.

Since moisture remaining in the cavity may represent a potential long-term degradation concern, immediate action is not necessary. The actions to develop and initiate the corrective actions should be undertaken as soon as possible commensurate with the safety significance of the condition. Completion times for the determined corrective actions will be controlled by the DCPP corrective actions program and will be determined and controlled based on the safety significance of the condition.

10.2.2.4 MPC Helium Backfill Characteristics and Purity

Having the proper helium backfill pressure or density ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC. During the loading operation, once the dryness limits are met, the MPC cavity is backfilled with helium to provide the inert environment required for long-term storage. To ensure the proper environment is established the helium used in the backfill process shall have a purity of ≥ 99.995 percent. In addition, the helium backfill pressure shall be verified during loading for all MPCs to be ≥ 29.3 psig and ≤ 33.3 psig.

If it has been determined that the helium backfill pressure limit has not been met, an engineering evaluation shall be undertaken to determine the actual helium pressure within the MPC cavity. Since too much or too little helium in the MPC cavity represents a potential overpressure or heat removal degradation concern, the engineering evaluation shall be performed in a timely manner commensurate with the safety significance of the condition (that is, if it is not addressed there is a possibility of a failure to adequately cool the contained fuel resulting in cladding damage).

Once the helium pressure in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the helium pressure estimated can range over a broad scale, different recovery strategies may be necessary. Completion times for the determined corrective actions will be controlled by the DCPP corrective actions program and will be determined and controlled based on the safety significance of the condition.

10.2.2.5 MPC Leakage Characteristics

The MPC helium leak rate limit ensures there is adequate helium in the MPC for long-term storage and proper heat removal. The leak rate acceptance limit of ≤ 5.0E-6 atm cc/sec (He) is assumed in the confinement analyses and is bounding for offsite dose. This is a mass-like leakage rate as specified in ANSI N 14.5 (1997). This is defined as the rate of change of the pressure-volume product of the leaking fluid at test conditions. This allows the leakage rate as measured by a mass spectrometer leak detector (MSLD) to be compared directly to the acceptance limit without the need for unit conversion from test conditions to standard, or reference conditions.

During transport operations or storage operations if the helium leak rate limit is determined not to be met, an engineering evaluation shall be performed to determine the impact of increased helium leak rate on heat removal and offsite dose. Since the SFSC is a ventilated system, any leakage from the MPC is transported directly to the environment. An increased helium leak rate represents a potential challenge to MPC heat removal and the offsite doses calculated in the Diablo Canyon ISFSI SAR confinement analyses, reasonably rapid action is warranted.

Once the cause and consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed, different recovery strategies may be necessary. An elevated helium leak rate represents a challenge to heat removal rates and offsite doses, reasonably rapid action and completion of the corrective actions shall be commensurate with the safety significance of the condition. Completion times for the determined corrective actions are controlled by the DCPP corrective actions program and will be determined based on the safety significance of the condition

10.2.2.6 Returning MPC to Safe Condition

If for a loaded MPC the fuel cavity dryness, backfill pressure, or helium leakage rate cannot be successfully met or maintained for any reason, the MPC must be returned to a safe analyzed condition, which may ultimately require the fuel to be placed back in the SFP. The completion time for this effort shall be based on the safety significance of the condition. The completion time shall consider the time required to perform fuel cool-down operations, reflood the MPC, cut the MPC lid welds, move the transfer cask into the SFP, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

10.2.3 MPC UNLOADING CHARACTERISTICS

In the event that an MPC must be unloaded, the transfer cask with its enclosed MPC is returned to the auxiliary building/fuel handling building to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are then removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The MPC is attached to the cool-down system. The cool-down system is a closed-loop forced ventilation gas cooling system that cools the fuel assemblies by cooling the surrounding helium gas inside the MPC.

During fuel cool-down, the MPC/transfer cask annular gap is reflooded with borated water to ensure adequate cooling capability is maintained. Once the fuel cool-down process is complete the MPC is reflooded with borated water and the MPC lid weld is removed leaving the MPC lid in place. The transfer cask and MPC are placed in the SFP and the MPC lid is removed. The contents are removed from the MPC and the MPC and transfer cask are removed from the SFP and decontaminated.

10.2.3.1 Gas Exit Temperature Of An MPC Prior To Reflooding

The integrity of the MPC depends on maintaining the internal cavity pressures within design limits. During the unloading process, reducing the fuel cladding temperatures significantly reduces the temperature gradients across the cladding, thus minimizing thermally-induced stresses on the cladding during MPC reflooding. In addition, reducing the MPC internal temperatures eliminates the risk of high MPC pressure due to sudden generation of steam during reflooding. This is accomplished by using the cool-down system that reduces the MPC internal temperatures such that there is no sudden formation of steam during MPC reflooding. Monitoring the circulating MPC gas exit temperature from the cool-down system ensures that there will be no large thermal gradient across the fuel assembly cladding during reflooding, which could be potentially harmful to the cladding. The exit gas temperature limit of ≤ 200°F ensures that the MPC gas exit temperature will closely match the desired fuel cladding temperature prior to reflooding the MPC. This temperature was selected to be lower than the boiling temperature of water with additional margin to eliminate the possibility of flashing to steam during reflooding.

During the fuel cool-down process, if the MPC helium gas exit temperature limit is not met, proceeding with reflooding shall be prohibited and actions must be taken to restore the parameters to within the limits before reflooding. In addition, while this parameter is being restored within limits, the proper conditions must be verified to exist for the transfer of heat from the MPC to the surrounding environs to ensure the fuel cladding remains below the short-term temperature limit. Maintaining the annular gap water level between the MPC and the transfer cask will ensure that adequate cooling capability exits.

10.2.4 OTHER OPERATING CONTROLS AND LIMITS

10.2.4.1 Fuel Cladding Oxide Thickness

In determining whether fuel assemblies are considered intact or damaged, several parameters are considered as is discussed in Section 10.2.1. Most of these parameters concern known or suspected cladding failures. However, for high burnup fuel (> 45,000 MWD/MTU), fuel-cladding oxidation is also a concern and shall be evaluated prior to a specific fuel assembly being identified as an intact assembly. A very high oxidation level can mean that a fuel assemble is not structurally sound and may fail in storage causing a change in the conditions inside the affected MPC. The evaluation of fuel cladding oxidation can be performed by actual physical measurement or an appropriate predictive methodology. For a high burnup spent fuel assembly to be classified as an intact fuel assembly, the computed or measured average oxidation layer thickness shall not exceed the applicable maximum allowable average fuel cladding oxidation layer thickness provided in the Diablo Canyon ISFSI Technical Specifications.

For a high burnup fuel assembly, if the fuel cladding oxidation layer thickness that is computed or measured on any fuel rod exceeds the limit, that fuel assembly will be considered a damaged fuel assembly. As such it will require storage in a damaged fuel container and limited to what MPC type it may be stored in.

10.2.5 LIMITING CONDITIONS FOR OPERATION

10.2.5.1 Equipment

All Diablo Canyon ISFSI equipment important to safety is passive in nature, therefore, there are no limiting conditions regarding minimum available equipment or operating characteristics. The MPC, transfer cask, CTF, and overpack have been analyzed for all credible equipment failure modes and extreme environmental conditions. No credible postulated event results in damage to fuel, release of radioactivity above acceptable limits, or danger to the public health and safety. All operational equipment is to be maintained, tested, and operated according to the implementing procedures developed for the ISFSI. The failure or unavailability of any operational equipment can delay the transfer of an MPC to the transfer cask or to the SFSC, but would not result in an unsafe condition.

10.2.5.2 Technical Conditions and Characteristics

The following technical conditions and characteristics are required for the Diablo Canyon ISFSI:

- Spent fuel characteristics
- SFSC heat removal capability

- MPC dissolved boron concentration level
- Annulus gap water requirement during moisture removal for loading and reflooding for unloading
- Water temperature of a flooded MPC
- MPC vacuum pressures
- MPC recirculation gas exit temperature
- Helium purity
- MPC helium backfill pressures
- Gas exit temperature of an MPC prior to reflooding
- SFSC time limitation while seated in the CTF
- Fuel cladding oxide thickness

The spent fuel specifications for allowable content for storage in the ISFSI and their bases are detailed in Section 10.2.1. In addition, the spent fuel specifications are also contained in Diablo Canyon ISFSI TS Section 2.0. A description of bases for selecting the above remaining conditions and characteristics are detailed in Sections 10.2.2 through 10.2.4, with the exception of the heat removal capability, SFSC time limitation in the CTF, and dissolved boron concentration. These are provided in the Diablo Canyon ISFSI TS bases. Although provided in the above SAR sections, the Diablo Canyon ISFSI TS and TS Bases also provide Limiting Conditions for Operations and bases for maintaining the integrity of the MPC during loading and unloading. These include vacuum pressure, recirculation gas temperature, backfill pressure, and leak rate during loading, and exit gas temperature during unloading.

The technical and operational considerations are to:

- Ensure proper internal MPC atmosphere to promote heat transfer, minimize oxidation, and prevent an uncontrolled release of radioactive material.
- Ensure that dose rates in areas where operators must work are ALARA and that all relevant dose limits are met.
- Ensure that the fuel cladding is maintained at a temperature sufficiently low to preclude cladding degradation during normal storage conditions.

Through the analyses and evaluations provided in Chapters 4, 7, and 8, this SAR demonstrates that the above technical conditions and characteristics are adequate and that no significant public or occupational health and safety hazards exist.

10.2.6 SURVEILLANCE REQUIREMENTS

The analyses provided in this SAR show that the Diablo Canyon ISFSI and the storage system fulfill its safety functions during all accident conditions as described in Chapter 8. Surveillance requirements are provided in the Diablo Canyon ISFSI TS. No continuous surveillance of the MPC is required during long-term storage. Surveillance of the SFSC duct screens is in the Diablo Canyon ISFSI TS and ensures freedom of air movement and adequate heat dissipation during long-term storage.

10.2.7 DESIGN FEATURES

The following storage system design features are important to the safe operation of the Diablo Canyon ISFSI and require design controls and limits:

- Material mechanical properties for structural integrity confinement and shielding
- Material composition and dimensional control for subcriticality
- Decay heat removal

Component dimensions are not specified here since the combination of materials, dose rates, criticality safety, and component fit-up define the operable limits for dimensions (that is, thickness of shielding materials, thickness of concrete, MPC plate thicknesses, etc.) The values for these design parameters are specified in the HI-STORM 100 System FSAR and LAR 1014-1 (References 1 and 2 respectively). Changes to any of these design features will be implemented only after conducting a safety evaluation in accordance with 10 CFR 72.48.

The combination of the above controls and limits and those discussed previously in Section 10.2 define requirements for the Diablo Canyon ISFSI storage system components that provide radiological protection and structural integrity during normal storage and postulated accident conditions.

10.2.8 ADMINISTRATIVE CONTROLS

Use of the existing DCPP organizational and administrative systems and procedures, record keeping, review, audit, and reporting requirements coupled with the requirements of this SAR ensure that the operations involved in the storage of spent fuel at the ISFSI are performed in a safe manner. This includes both the selection of assemblies qualified for ISFSI storage and the verification of assembly identification numbers prior to and after placement into individual MPCs. The spent fuel qualification, identification, and control are discussed in

Sections 10.2.1 through 10.2.4 above. Other administrative programs will control revisions to the Diablo Canyon ISFSI TS Bases; radioactive effluents; fuel-cladding-oxide thickness; MPC loading and unloading processes; ISFSI operations, and transportation route conditions. These other programs are defined in the Diablo Canyon ISFSI TS.

10.2.9 OPERATING CONTROL AND LIMIT SPECIFICATIONS

The operating controls and limits applicable to the Diablo Canyon ISFSI, as documented in this SAR, are delineated in the Diablo Canyon ISFSI TS and the TS Bases. These include:

- MPC dryness, backfill pressure and leak rate limitations
- SFSC heat removal capability
- Fuel Cool-Down exit gas temperature limitation
- SFSC time limitation in the CTF
- Dissolved boron concentration

10.2.10 REFERENCES

Detailed information describing the HI-STORM 100 System is provided in the following two references, which must be used together:

1. Final Safety Analysis Report for HI-STORM 100 System, Revision 0, July 2000.

2. License Amendment Request 1014-1, Holtec International, Revision 2, July 2001, including Supplements 1 through 4 dated August 17, 2001; October 5, 2001; October 12, 2001; and October 19, 2001; respectively.

Reference 2 contains information related to MPC-32, MPC-24, MPC-24E, MPC-24EF, and the HI-STORM 100SA. General references to these documents are made in Chapter 10 as needed to supplement SAR information.

TABLE 10.2-1

MPC-24 FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

B. Quantity per MPC: Up to 24 fuel assemblies.

C. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

NOTE 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel cell location. Fuel assemblies containing RCCAs may only be loaded in fuel storage locations 9, 10, 15, and/or 16 of Figure 10.2-1. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

NOTE 2: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO. Use of ZIRLO clad fuel is limited to a maximum burnup of 45,000 MWD/MTU.

TABLE 10.2-2

MPC-24E FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

2. Uranium oxide, damaged fuel assemblies, with or without nonfuel hardware, placed in damaged fuel containers. Uranium oxide damaged fuel assemblies shall meet the criteria specified in Table 10.2-5 and meet the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	≤ 4.0 wt% ^{235}U.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware and DFC)

TABLE 10.2-2

B. Quantity per MPC: Up to four (4) damaged fuel assemblies in damaged fuel containers, stored in fuel storage locations 3, 6, 19 and/or 22 of Figure 10.2-2. The remaining MPC-24E fuel storage locations may be filled with intact fuel assemblies meeting the applicable specifications.

C. Fuel debris is not authorized for loading in the MPC-24E.

NOTE 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel storage location. Fuel assemblies containing RCCAs must be loaded in fuel storage locations 9, 10, 15 and/or 16 of Figure 10.2-2. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

NOTE 2: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO. Use of ZIRLO clad fuel is limited to a maximum burnup of 45,000 MWD/MTU.

TABLE 10.2-3 Sheet 1 of 2

MPC-24EF FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 3)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

2. Uranium oxide, damaged fuel assemblies and fuel debris, with or without nonfuel hardware, placed in damaged fuel containers. Uranium oxide damaged fuel assemblies shall meet the criteria specified in Table 10.2-5 and meet the following specifications (Note 1 and 2):

Cladding type	Zr (Note 3)
Initial enrichment	≤ 4.0 wt% ^{235}U.
Post-irradiation cooling time and average burnup per assembly	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware and DFC)

TABLE 10.2-3 Sheet 2 of 2

B. Quantity per MPC: Up to four (4) damaged fuel assemblies and/or fuel debris in damaged fuel containers, stored in fuel storage locations 3, 6, 19 and/or 22 of Figure 10.2-2. The remaining MPC-24EF fuel storage locations may be filled with intact fuel assemblies meeting the applicable specifications.

NOTE 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel storage location. Fuel assemblies containing RCCAs must be loaded in fuel storage locations 9, 10, 15 and/or 16 of Figure 10.2-2. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

NOTE 2: The total quantity of fuel debris permitted in a single damaged fuel container is limited to the equivalent weight and special nuclear material quantity of one intact fuel assembly.

NOTE 3: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4, and ZIRLO. Use of ZIRLO clad fuel is limited to a maximum burnup of 45,000 MWD/MTU.

TABLE 10.2-4

MPC-32 FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

B. Quantity per MPC: Up to 32 intact fuel assemblies.

C. Damaged fuel assemblies and fuel debris are not authorized for loading in the MPC-32.

NOTE 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel storage location. Fuel assemblies containing RCCAs must be loaded in fuel storage locations 13, 14, 19 and/or 20 of Figure 10.2-3. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

NOTE 2: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO. Use of ZIRLO clad fuel is limited to a maximum burnup of 45,000 MWD/MTU.

TABLE 10.2-5

FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Type	Vantage 5	Standard or LOPAR
Cladding Material	Zr (Note 5)	Zr (Note 5)
Design Initial U (kg/assy.) (Note 2)	≤ 467	≤ 467
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt% ^{235}U) (Note 4)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit) (wt% ^{235}U) (Notes 3 and 4)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	264	264
Fuel Rod Cladding O.D. (in.)	≥ 0.360	≥ 0.372
Fuel Rod Cladding I.D. (in.)	≤ 0.3150	≤ 0.3310
Fuel Pellet Dia. (in.)	≤ 0.3088	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.496	≤ 0.496
Active Fuel Length (in.)	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.016	≥ 0.014

NOTE 1: All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies.

NOTE 2: Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or DCPP. For each fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with DCPP fuel records to account for manufacturers tolerances.

NOTE 3: Soluble boron concentration per Technical Specification LCO 3.2.1.

NOTE 4: For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum initial enrichment of the intact fuel assemblies is limited to the maximum initial enrichment of the damaged fuel assemblies and fuel debris (i.e., 4.0 wt.% ^{235}U).

NOTE 5: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO. Use of ZIRLO clad fuel is limited to a maximum burnup of 45,000 MWD/MTU.

TABLE 10.2-6

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(UNIFORM FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Burnup (Intact Fuel Assemblies) (MWD/MTU)	MPC-24E/24EF Assembly Burnup (Intact Fuel Assemblies) (MWD/MTU)	MPC-24E/24EF Assembly Burnup (Damaged Fuel Assemblies and Fuel Debris) (MWD/MTU)	MPC-32 Assembly Burnup (Intact Fuel Assemblies) (MWD/MTU)
≥ 5	40,600	41,100	39,200	32,200
≥ 6	45,000	45,000	43,700	36,500
≥ 7	45,900	46,300	44,500	37,500
≥ 8	48,300	48,900	46,900	39,900
≥ 9	50,300	50,700	48,700	41,500
≥ 10	51,600	52,100	50,100	42,900
≥ 11	53,100	53,700	51,500	44,100
≥ 12	54,500	55,100	52,600	45,000
≥ 13	55,600	56,100	53,800	45,700
≥ 14	56,500	57,100	54,900	46,500
≥ 15	57,400	58,000	55,800	47,200

NOTE 1: Linear interpolation between points is permitted.
NOTE 2: Burnup for fuel assemblies with cladding made of ZIRLO is limited to 45,000 MWD/MTU or the value in this table, whichever is less.

TABLE 10.2-7

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(UNIFORM FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Decay Heat (Intact Fuel Assemblies) (Watts)	MPC-24E/24EF Assembly Decay Heat (Intact Fuel Assemblies) (Watts)	MPC-24E/24EF Assembly Decay Heat (Damaged Fuel Assemblies and Fuel Debris) (Watts)	MPC-32 Assembly Decay Heat (Intact Fuel Assemblies) (Watts)
≥ 5	1157	1173	1115	898
≥ 6	1123	1138	1081	873
≥ 7	1030	1043	991	805
≥ 8	1020	1033	981	800
≥ 9	1010	1023	972	794
≥ 10	1000	1012	962	789
≥ 11	996	1008	958	785
≥ 12	992	1004	954	782
≥ 13	987	999	949	773
≥ 14	983	995	945	769
≥ 15	979	991	941	766

NOTE 1: Linear interpolation between points is permitted.

NOTE 2: Includes all sources of heat (i.e., fuel and nonfuel hardware).

TABLE 10.2-8

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(REGIONALIZED FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Burnup for Region 1 (MWD/MTU)	MPC-24 Assembly Burnup for Region 2 (MWD/MTU)	MPC-24E/24EF Assembly Burnup for Region 1 (MWD/MTU)	MPC-24E/24EF Assembly Burnup for Region 2 (MWD/MTU)	MPC-32 Assembly Burnup for Region 1 (MWD/MTU)	MPC-32 Assembly Burnup for Region 2 (MWD/MTU)
≥ 5	49,800	32,200	51,600	32,200	39,800	22,100
≥ 6	56,100	37,400	58,400	37,400	43,400	26,200
≥ 7	56,400	41,100	58,500	41,100	44,500	29,100
≥ 8	58,800	43,800	60,900	43,800	46,700	31,200
≥ 9	60,400	45,800	62,300	45,800	48,400	32,700
≥ 10	61,200	47,500	63,300	47,500	49,600	34,100
≥ 11	62,400	49,000	64,900	49,000	50,900	35,200
≥ 12	63,700	50,400	65,900	50,400	51,900	36,200
≥ 13	64,800	51,500	66,800	51,500	52,900	37,000
≥ 14	65,500	52,500	67,500	52,500	53,800	37,800
≥ 15	66,200	53,700	68,200	53,700	54,700	38,600
≥ 16	-	55,000	-	55,000	-	39,400
≥ 17	-	55,900	-	55,900	-	40,200
≥ 18	-	56,800	-	56,800	-	40,800
≥ 19	-	57,800	-	57,800	-	41,500
≥ 20	-	58,800	-	58,800	-	42,200

NOTE 1: Linear interpolation between points is permitted.

NOTE 2: These limits apply to intact fuel assemblies, damaged fuel assemblies, and fuel debris.

NOTE 3: Burnup for fuel assemblies with cladding made of ZIRLO is limited to 45,000 MWD/MTU or the value in this table, whichever is less.

TABLE 10.2-10

NONFUEL HARDWARE COOLING AND AVERAGE ACTIVATION

Post-Irradiation Cooling Time (years)	BPRA and WABA Burnup (MWD/MTU)	TPD Burnup (MWD/MTU)	RCCA Burnup (MWD/MTU)
≥3	≤20,000	NA	NA
≥4	≤25,000	≤20,000	NA
≥5	≤30,000NA	≤25,000	≤630,000
≥6	≤40,000	≤30,000	
≥7	≤45,000	≤40,000	
≥8	≤50,000	≤45,000	
≥9	≤60,000	≤50,000	
≥10		≤60,000	
≥11		≤75,000	
≥12		≤90,000	
≥13		≤180,000	
≥14		≤630,000	

<u>NOTE 1</u>: Linear interpolation between points is permitted, except that TPD burnups > 180,000 MWD/MTU and ≤630,000 MWD/MTU must be cooled ≥ 14 years.

<u>NOTE 2</u>: Applicable to uniform loading and regionalized loading.

<u>NOTE 3</u>: NA means not authorized for loading.

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CHAPTER 1

PROPOSED ACTIVITIES

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1.2 NEED FOR THE FACILITY

The Nuclear Waste Policy Act (NWPA) of 1982 mandated that the Department of Energy (DOE) assume responsibility for the permanent disposal of spent nuclear fuel from the nation's commercial nuclear power plants. Pending the availability of a permanent DOE repository, nuclear power plant operators such as PG&E have been given the responsibility under the NWPA to provide for the interim onsite storage of spent fuel until it is accepted by DOE. DOE has not complied with its NWPA mandate to have a repository in operation commencing in 1998, and no interim spent fuel storage facility has been established. Moreover, no such DOE facility is expected to become operational in a timeframe to meet the spent fuel storage needs of DCPP. Thus, spent fuel generated by DCPP will need to remain at DCPP until a DOE or other facility is available. Consequently, additional spent fuel storage capacity is needed at DCPP no later than 2006.

The additional capacity to accommodate discharged spent fuel as proposed herein will allow DCPP to continue to generate electricity. Any interruption in the availability of this capacity would almost certainly cause a negative impact on the domestic sector power supply in California. Considering the power supplies in California and in the western United States, as well as uncertainties about future power supplies, any loss of power from DCPP could have significant adverse impacts on the population, the infrastructure, and the economy. Expansion of the onsite spent fuel storage capacity at DCPP as planned by PG&E is necessary to avoid these potential significant negative impacts.

PG&E has considered several alternative means for accommodating the additional spent fuel that will be generated at DCPP through the licensed operating life of each unit. The onsite alternatives include a second reracking of the spent fuel pools to replace the existing high-density racks with racks of higher-density design, and building an onsite ISFSI using dry cask storage technology. PG&E has also considered the possibility of participating in the Private Fuel Storage venture, which has an application pending before the NRC for a license to independently store spent fuel from nuclear power plants.

Based on an overall assessment of operational and safety considerations, the amount of spent fuel to be generated, the transportation requirements associated with the alternatives, resources needed, and scheduling restraints, PG&E has concluded that dry cask storage of spent fuel at DCPP is the best available method at this time for providing the necessary storage capacity. However, as discussed below, increasing the spent fuel pool storage capacity through a second reracking with higher density racks remains a viable option if it appears that the Diablo Canyon ISFSI cannot be licensed on a schedule that meets PG&E storage requirements.

The expanded storage capacity provided by the use of dry casks at the ISFSI will be used to store aged spent fuel that has been stored for 5 years or longer in the DCPP spent fuel pools. The storage spaces in the respective spent fuel pools that become available following this transfer of the aged spent fuel into dry cask storage then can be used to store future discharged spent fuel from the reactor core. Storage casks will be acquired as needed to accommodate the spent fuel generated until shipment offsite occurs.

The Diablo Canyon ISFSI will consist of: the storage pads, a cask transfer facility (CTF), an onsite cask transporter, and the dry cask storage system. The dry cask storage system that has been selected by PG&E for the Diablo Canyon ISFSI is the Holtec International (Holtec) HI-STORM 100 System (Reference 2). The HI-STORM 100 System is comprised of a multi-purpose canister (MPC), the storage overpack, and the HI-TRAC transfer cask. The HI-STORM 100 System is certified by the NRC for use by general licensees as well as site-specific licensees, presently with a 24 PWR fuel assembly MPC and storage overpack (NRC 10 CFR Part 72 Certificate of Compliance [CoC] No. 1014) (Reference 3).

Holtec has proposed a number of changes to the certified HI-STORM 100 System in License Amendment Request (LAR) 1014-1, submitted to the NRC on August 31, 2000 (Reference 4). These proposed changes include a HI-STORM 100SA storage overpack, a higher-capacity MPC-32 design (for storage of 32 PWR spent fuel assemblies), and MPC designs with different fuel storage capabilities (for example, high burnup fuel and certain damaged fuel). As discussed below, several of these proposed changes are desirable for the Diablo Canyon ISFSI. PG&E understands, however, that several of the proposed changes in LAR 1014-1, such as the designs to accommodate high burnup fuel, may involve extensive NRC review. As discussed below, issuance of a revised CoC (1014-1) may not necessarily be required to support the plant-specific Diablo Canyon ISFSI license.

The Diablo Canyon ISFSI is designed to hold up to 140 storage casks (138 casks plus 2 spare locations). Based on the current fuel strategy and use of the MPC-32, the ISFSI with a storage pad capacity of 140 casks will be capable of storing the spent fuel generated by DCPP Units 1 and 2 over the term of the current operating licenses (2021 and 2025, respectively). The ISFSI is sized to accommodate a single 40-year operating license term for both units and to support subsequent decommissioning of the units and termination of the 10 CFR 50 licenses. Because of its higher capacity, the principal MPC to be used will be the MPC-32. In addition, to accommodate spent fuel generated during the licensed period, as well as any damaged fuel assemblies, debris, and nonfuel hardware, PG&E may use three other MPC designs from the HI-STORM 100 System: the MPC-24, MPC-24E, and MPC-24EF. All four MPC designs use the same storage overpack and are either licensed by current CoC No. 1014 or will be licensed by future revisions to CoC No. 1014. These cask designs will accommodate most of the DCPP-specific fuel characteristics.

The PG&E license application incorporates these designs in a preferred cask system licensing approach as follows:

(1) The initial Diablo Canyon ISFSI site-specific license would incorporate the MPC capabilities as specified in CoC No. 1014, as proposed to be amended in Holtec LAR 1014-1. The NRC is anticipated to issue a final technical review on LAR 1014-1 and a preliminary Safety Evaluation Report in late December 2001 or early 2002. Rulemaking is expected to be completed in mid-2002. While the MPC capabilities covered by the Holtec CoC No. 1014 and LAR 1014-1 will not completely envelope all of the spent fuel

characteristics eventually needed for storage of DCPP fuel, they will cover most of the current spent fuel pool inventory and will permit the storage of nearly all spent fuel and associated nonfuel hardware generated through the license term.

(2) MPC designs that may be needed to store the balance of the DCPP spent fuel will be addressed in future revisions to the CoC. As these changes are submitted by Holtec and approved by the NRC, PG&E will amend the Diablo Canyon ISFSI site-specific license to incorporate these changes. The resulting capability will provide PG&E with the flexibility to store onsite all of the spent fuel and nonfuel hardware from DCPP Units 1 and 2 generated during the term of its operating licenses.

(3) In a Federal Register Notice dated October 11, 2001 (66 FR 51823), NRC issued the final rule change regarding greater than class C (GTCC) waste (for example, split pins and thimble tubes). The rule change applies only to the interim storage of GTCC waste generated or used by commercial nuclear power plants. The rule change allows interim storage of reactor-related GTCC wastes under a 10 CFR 72 site-specific license. In accordance with the guidance of ISG 17, PG&E plans to request a modification to its proposed site-specific license at a future date to allow interim storage of GTCC wastes at the Diablo Canyon ISFSI.

Licensing of the Diablo Canyon ISFSI also involves NRC review of a number of site-specific issues. They include the site-specific environmental review, geotechnical issues related to the site, site-specific environmental conditions and natural phenomena, and other site-specific matters. Although the Holtec LAR 1014-1 includes a high-seismic capability for the storage overpack (the HI-STORM 100SA), it does not incorporate some Diablo Canyon specific information (for example, the pad design, the overpack seismic anchorage design, the cask transporter seismic design, and the CTF design). PG&E is submitting information on these items as part of this site-specific application and intends that these issues be reviewed and licensed as part of the PG&E site-specific 10 CFR 72 license.

In addition to the approval from the NRC under 10 CFR Part 72, other state and local permits and licenses will be required to support the construction and operation of the Diablo Canyon ISFSI, as discussed in ER Chapter 9. With respect to the State of California, PG&E will apply for a Coastal Development Permit (CDP). The CDP application will require an environmental review in accordance with State law. The County of San Luis Obispo acts as the lead agency. PG&E initiated the necessary state environmental review process in November 2001 and encourages NRC coordination with the County. This Environmental Report is being written to address the requirements of the National Environmental Policy Act and the California Environmental Quality Act.

Separate and apart from the present 10 CFR 72 application, PG&E intends to submit a 10 CFR 50 LAR for DCPP Units 1 and 2 in early 2002 related to cask handling activities in the DCPP fuel handling building/auxiliary building (FHB/AB). PG&E also submitted a 10 CFR 50 LAR on September 13, 2001, to allow credit for soluble boron in the DCPP spent fuel pools and thus provide additional storage in the existing high-density racks. Credit for soluble boron will extend full core offload capability in the spent fuel pools from 2003 to 2006.

PG&E has evaluated the above proposed actions and has determined that the proposed actions and mitigating measures do not involve: (a) a significant hazards consideration, (b) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (c) a significant increase in individual or cumulative occupational radiation exposure. This document should allow federal and state agencies to conclude that PG&E proposed actions to implement a used fuel storage program consisting of a 10 CFR 72 license application and modification to the DCPP 10 CFR 50 operating license do not involve any significant adverse environmental impacts.

In its Waste Confidence Decision, the NRC examined the environmental impacts of the operation of ISFSIs built at operating nuclear power plant sites. The Commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored without significant environmental impacts for at least 30 years beyond the licensed life for operation of that reactor at onsite or offsite ISFSIs (10 CFR 51.23: 49 Fed. Reg. 34688, August 31, 1984). The NRC has reviewed the Waste Confidence decision twice since it was first issued (in 1990 [55 Fed. Reg. 38474, September 18, 1990] and in 1999 [64 Fed. Reg. 68005, December 6, 1999]), and in both cases, the Commission basically reaffirmed the findings of the original decision. On July 18, 1990, the NRC published a final rule on "Storage of Spent Nuclear Fuel in NRC-Approved Storage Casks at Nuclear Power Plant Sites" (55 Fed. Reg. 29181-29190, July 18, 1990), and issued a general license for storage of spent nuclear fuel at reactor sites (10 CFR 72.210). The environmental impacts of spent nuclear fuel storage at reactor sites were also addressed in an environmental assessment and its accompanying "finding of no significant impact" (NRC 1989). The finding of no significant impact states that the Commission concludes that the proposed rulemaking, entitled "Storage of Spent Nuclear fuel in NRC-Approved Storage Casks at Nuclear Power Reactor Sites," will not have a significant incremental impact on the quality of the human environment. In addition, the NRC has issued seven site-specific licenses for at-reactor ISFSIs located in various parts of the country. For these seven ISFSIs, environmental assessments were completed and findings of no significant impact were reached.

DIABLO CANYON ISFSI
ENVIRONMENTAL REPORT

CHAPTER 2

THE SITE AND ENVIRONMENTAL INTERFACES

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2.9 REGIONAL HISTORIC, SCENIC, CULTURAL, AND NATURAL FEATURES

A cultural resources (historic and prehistoric archaeological, historic built environment, sites or areas of interest to local Native American community) investigation was conducted for the ISFSI project. This investigation consisted of the following steps:

- Archival Database search
- On-the-ground archaeological "field verification" of proposed impact areas
- Native American consultation

The results of the cultural resources investigation along with a description of scenic and natural resources are provided in this section.

2.9.1 ETHNOGRAPHIC OVERVIEW

The ISFSI is located in the ethnographic territory of the Obispeño, the northernmost group of the Chumash Indians of Southern California. The Obispeño language (Hokan language family) is considered the most divergent from other Chumash groups (Reference 1). The term Obispeño refers to the group associated with Mission San Luis Obispo and does not reflect the native term that the group used for themselves.

The Obispeño Chumash, although having a rich material culture, differed from other Southern Chumash groups. The plank canoe, a Southern Chumash trait, did not appear to be used or built this far north in Chumash territory (Reference 1). Mission record research suggests that the population in the north was also more mobile than in the south, as Mission San Luis Obispo recruitment was smaller than other missions (Reference 2). Population density also appears to be smaller with an estimated density of 25 to 45 individuals per square mile for the Obispeño (Reference 3). Explorers' accounts of northern Indian settlements substantiate the proposition that there was a different settlement pattern in the north than in the south. There appeared to be smaller more dispersed settlements in the north and larger aggregated settlements in the south.

The focus of Obispeño subsistence practices was based on maritime resources; the rocky and chaotic coastline may have precluded the use of the canoe. However, many mollusks and fish species present along the shoreline and the tidal pools were used. The Obispeño also used acorns, hard seeds, and terrestrial game. Trade relationships of the Obispeño were mainly to the north, with various items exchanged with the Southern San Joaquin Valley Yokuts and coastal Salinans.

The founding of the missions in Southern California had a devastating effect on Native Americans. With the founding of the Mission San Luis Obispo de Tolosa in 1772, Native Americans from the surrounding area were recruited to build, farm, and work on the Mission. Poor living conditions and the introduction of Euro-American diseases led to the decimation of the native population. At Mission San Luis in 1803, just over 900 Native Americans were recorded as living there; in 1938, only 170 remained. In 1914 only one known Obispeño speaker was alive (Reference 1). Consequently, little has been preserved or recorded of Native American culture for the area. The establishment of the Santa Ynez reservation, 40 miles northeast of Santa Barbara, is the only land given to the Northern Chumash by the United States government.

2.9.2 HISTORIC OVERVIEW

The ISFSI is part of the Cañada de Los Osos y Pecho y Islay Mexican land grant patented to John Wilson in 1869 by the United States government. The area was once two separate land grants, the Cañada de Los Osos and the Pecho y Islay. The Cañada de Los Osos was granted to Victor Linares in December of 1842 by Governor Alvarado (References 4 and 5). The area was also granted to Modesta Castro in 1844 but her claim was rejected (Reference 4). Governor Michelorena granted Pecho y Islay to Francisco Padilla in April of 1843.

In 1845, captains James Scott and John Wilson became the grantees of the combined Ranchos, Cañada de Los Osos y Islay, which included 32,430 acres. These two men also purchased the San Luis Obispo Mission (Reference 6). The land was patented by the United States government to John Wilson in September of 1869. Wilson later married the widow of Don Romualdo Pacheco, whose son of the same name became governor of California in 1876. In 1851, Wilson is reported to have had land holdings in excess of 53,000 acres (Reference 7). Portions of the land grant were subsequently obtained by W. H. Patterson, H.M. Warden, Ramona Hilliard, and L. Marre (Reference 7).

The land to the north of Diablo Creek was then owned by the Spooner family from 1892 until 1942, and after that by Oscar Field (Reference 2). The land to the south of Diablo Creek was leased and/or owned by the Marre family since 1879 until recently. Luigi Marre leased the Pecho holdings beginning in 1879; he later bought the land (Reference 2). Luigi Marre leased the Pecho holdings beginning in 1879; he later bought this land (Reference 2). The area, which remained isolated for years, has been used primarily for cattle grazing and agricultural purposes.

2.9.3 ARCHEOLOGICAL OVERVIEW

During the Mid-19th century, a number of collectors excavated sites along the coastline from Morro Bay to Avila Beach. In 1872, one collector, Charles H. Jackson, reported on the finding of a cemetery near San Luis Obispo Bay (Reference 7).. Other early "excavations" along the coastline near the Project area include Leon de Cessac in 1878, Summers and Freer in 1894-95, and the Los Angeles County Museum in 1929 (Reference 2). On an 1874 map prepared by Schumacher for the Smithsonian Institute, the Terrace area to the north of Avila Beach is marked as the location of "Indian kitchen middens," although specific locales were not noted.

Over a period of some 30 years there have been a number of cultural resource investigations at prehistoric archeological sites located in the ISFSI area. The first study to record sites in the area was that of Arnold R. Piling in 1955 (Reference 8). His survey, which was focused along the Marine terraces from Avila Beach to Morro Bay, recorded sites CA-SL0-2 (SLO-2) and SL0-3 in 1947 and SLO-61 in 1948 at the mouth of Diablo Creek. In addition he noted two other sites, SL0-7 and SL0-8, located northwest of Diablo Creek.

In 1966, Francis Riddell conducted a survey for PG&E of approximately 250 acres to be used as the site for DCPP. This reconnaissance (Reference 9) resulted in the description of five sites located within the ISFSI area. These are known as Riddell Nos. 1, 2, 3, 4, and 5. Very little descriptive information concerning the sites, area surveyed, and method of survey is contained in Riddell's report. Although it is not stated in the report, SL0-2 is the same as Riddell's No. 1 and SL0-61 is Riddell's No. 2. Thus, as a result of Riddell's survey, two previously recorded sites were relocated and three new sites (Riddell Nos. 3, 4, and 5) were recorded. One of the new sites, Riddell No. 4 was assigned the designation CA-SL0-584 in 1966.

In 1968, Greenwood and associates undertook subsurface investigations at six sites within the construction areas for the DCPP facilities and a proposed access road from the plant locale to Avila Beach. The excavations in the area included SL0-2, SL0-61, and SL0-584. Further, Greenwood (Reference 2) conducted minimal work at Riddell No. 3. In addition, the report contains a summary of ethnographic research concerning the immediate ISFSI area (Reference 2) and an analysis of fish remains from SL0-2 (Reference 10). Excavation appears to been restricted to the direct impact areas of the form of proposed facilities or remaining portions of the sites which had not been disturbed by grading or construction activities.

The subsurface excavation of thirty-two 1 x 2 meter units resulted in the inspection of about 190 cubic meters of soil from SL0-2 at two different locations (Reference 2). Thirty units were excavated at CA-SL0-2, site one, revealing a midden soil that ranged in depth from 260 to 340 cm. Two other units placed in an area northeast of Site 1, designated Site 1a, had a cultural deposit ranging in depth from 70 to100 cm. This latter area was also the location of SL0-3. Based on this investigation, Greenwood suggests that sites SL0-2 and SL0-3 should be considered as one site. Subsequent investigations have validated this proposition.

The excavations at SL0-2 resulted in the exposure of a cemetery complex containing 54 inhumations, 24 that were identified as female, and 15 male (Reference 2).. Due to grading for road construction, an additional six inhumations were recovered from the site in November of 1968 and six fragmentary inhumations collected in June 1969. A total of 66 burials were exposed. Grave goods were associated with some of the burials. The burials recovered from these excavations were turned over to a local Native American group and were reported to have been reburied.

The artifact inventory from the site is quite impressive. A total of 2,885 stone, bone, wood, and shell artifacts comprise the catalog material from the site. This includes stone projectile points, blades, knives, shoppers, scrapers, boring or drilling implements, and cores. Groundstone items include bowls, bull borders, manos, milling stones, pestles, pitted stones, and charmstones. A number of mammal, shell, and bird bone artifacts were recovered in addition to 1,607 shell beads. The be inventory also includes Olivella, Mytilus, and clam beads. A few sherds of pottery, similar to Owens Valley brown where were also collected. A large quantity of faunal material was also recovered.

Temporal affiliations ascribed to the materials recovered from CA-SL0-2 cover a time span of some 9,000 years. Based on results of three radiocarbon dates, 930 ± 50 (UCLA 1686B: human bone from burial 44), 8960 ± 190 (GAK-2044: Haliotis shell adjacent to burial 5), 9320 ± 140 (UCLA-1686A: human bone from burial 20 [dates expressed in years before present]), a suite of 23 hydration rim readings from 21 samples, and cross- dating of artifact types, Greenwood (1972: 85-95) postulated three temporal components at the site. These are, from earliest to latest periods, the Early Milling Stone, Hunting, and Canaliño horizons specifically related to the Diablo Canyon area (Reference 2). Greenwood was also able to identify assemblages and present evidence for different settlement/subsistence patterns for each period. It was suggested that SL0-2 was a major village that figured prominently in the social, economic, and political life of the indigenous occupants of the area. The 1968 excavations of the site resulted in the identification one of the oldest cultural list ratified sites identified to date in San Luis Obispo County.

Excavations were also undertaken at CA-SLO-61 along the bluff overlooking the Diablo Coast. Five 1 x 2 meter excavation units were completed at the site. Six cubic meters of soil were examined and resulted in the recovery of 40 artifacts that included a bowl mortar, pitted stones, a cobble pestle, a drill, and 21 scrapers. The cultural deposit ranged in depth from 20 to 100 cm. Based on a comparison of materials recovered from the limited excavation to those recovered from the upper levels SL0-2, Greenwood (Reference 2) assigned the site to the Canaliño.

Another site within the ISFSI area, but not located on the coast (SL0-584), was also excavated by Greenwood (Reference 2). The site was located on a small flat on the South Bank of Diablo Creek. It is now the site of the DCPP switchyard. Seven units were excavated at the site that ranged in depth from 50 to 100 cm. A total of 76 catalog artifacts were recovered from the site. Materials collected included 10 projectile points, leaf-shaped blades, scrapers, three bowl fragments, a hopper mortar fragment, a pestle, pitted cobbles, brownware sherds, Olivella disks, and Mytilus and Tivela beads. Historic materials included five glass trade beads and one brass ring. In addition, three cupule boulders were located within the site boundaries. Based on cross-dating of artifact types similar to the upper levels of SL0-2 and the occurrence of historic period artifacts, the site is considered to be a late (Canaliño) site. It is also suggested that it was a task-specific site with a short-term for seasonal occupation (Reference 2).

The last site within the ISFSI area examined during the 1968-69 investigations was Riddell Site 3. The site is located at the southern tip of Diablo Coast. Greenwood (Reference 2) provides the following description of the work completed at the site:

> "It should be noted that one additional locale was described in the contract agreement but not excavated during the fieldwork described in this report. A light scatter of shell which appeared fresh and recent was on the surface of 1968, but test pits test dug by shovel disclosed only very shallow soil covering on the volcanic outcrop and no shell, chipping waste, or artifacts below the surface. In view of the total priorities, no systematic excavation was attempted."

In September of 1974, Greenwood removed human remains in the area around two potholes and reported upon natural erosion occurring along the coastline strip of the site (Reference 11). She further recommended some measures for controlling vandalism at the site and midden sluffing along the bluff area.

In 1978, Greenwood and Associates completed the survey of 90 acres of land thought to be the aerial extent of site SL0-2. She concludes:

> "Based on visible indications, the extent of the site is revealed as a minimum of 350 meters east-west by a minimum of 427 meters north-south, for a known area of 130,235 square meters...that the locality recorded by Pilling as SL0-3 is actually within the boundaries of SL0-2, and a part of the larger site."

Also contain within this report was information regarding the nomination of SL0-2 to the National Register of Historic Places as part of the existing Rancho Canada de Los Osos y Pechos Historic District, located some 1.4 miles to the south of the ISFSI area (Reference 1). Another document prepared by Greenwood and Associates (1978) that supplemented the survey report, addressed background research regarding Diablo Canyon and SLO-2 (Reference 12). The main purpose of this research was to investigate the reported location of the Chumash village location Tsuhanu that was asserted to be within the DCPP site. It was determined, based on a record search and interviews, that SL0-2 was not the location of this village.

Since the 1968 investigations, PG&E has instigated various procedures for the protection and management of SL0-2. In 1980, an Archeological Resources Management Plan (ARMP) was incorporated into the operating license for DCPP (Reference 13). Surface alterations addressed in this management plan include provisions for fire protection, storage of materials confined to areas protected by fill, restrictions on traffic flow, and limiting maintenance of roads and existing utility lines to areas have been previously disturbed. The site area has beenfenced and warning signs are posted at entry points of road access to the site. Since November of 1983, photographs have been taken at regular intervals from 23 stations within the site in order to monitor any physical changes to the site caused by natural or other processes.

In 1986, Holson reported on the survey of the unsurveyed portions the NRC license regulated area for DCPP (Reference 13). A total of six prehistoric sites were reported. Three new sites, SLO-1161, SLO-1162, and SLO-1163, two of Riddell's sites, SLO-1159 (Riddell 3) and SLO-1160 (Riddell 5), and a new site form were prepared for SLO-61.

From 1979 to the present, other archeological investigations conducted within the ISFSI area have been associated with ongoing maintenance and construction of facilities for DCPP. In general, the majority of the work has focused on monitoring surface modifications in the area of SL0-2.

2.9.4 ARCHIVAL DATABASE SEARCH

A cultural resource record database search was conducted at PG&E's cultural resource library in San Francisco. The research was performed to identify previously recorded or otherwise known cultural resources and previous cultural resource studies within or adjacent to the proposed ISFSI, and archaeologically sensitive portions of the study area, as determined by the locations of previously recorded archaeological sites nearby and their relationship to environmental factors and topography. As the project area is an area controlled by PG&E for the past 30 plus years, a record search of the California Historic Resource Information System at U.C. Santa Barbara was not deemed necessary.

The database search revealed that the project area (areas of direct and indirect impacts including construction and operation of the ISFSI) had been previously examined (Reference 13). No sites listed in or eligible for inclusion in the National Register of Historic Places (NRHP) were identified within the area of the proposed ISFSI. One archaeological site that is listed in the NRHP (CA-SLO-2) is located within 150 meters of the proposed ISFSI site. Seven other sites (CA-SLO-61, -584, -1159, -1160, -1161, -1162, and -1163) are located within the 750-acre exclusion zone surrounding DCPP. Several other prehistoric and historic archaeological sites are located within the coastal terrace between Diablo Creek to the south and Coon Creek to the north. Over 70 prehistoric archaeological sites are located south of Diablo Creek, including a large NRHP District.

A record search for significant natural features that are listed in the National Registry of Natural Landmarks was conducted by contacting the National Park Service Land Management Division in San Francisco. This research indicated that no natural landmarks are located near the ISFSI.

Early low-level aerial photos of DCPP were also examined. The purpose of this review was to verify areas of previous disturbance in the ISFSI site area. The aerial photos clearly revealed that the ISFSI site has been disturbed during construction of the switchyards. The entire ISFSI site was cut for the fill to provide switchyards.

2.9.5 FIELD VERIFICATION

An on-the-ground field survey of the area near the ISFSI was conducted in May and September 2000 and January 2001. The primary focus of this field study was to verify the previous ground disturbance to the ISFSI site and the absence of any archaeological sites as reported in Reference 13. These field verifications determined that the ISFSI site has had major soil removal and no archaeological sites were noted. Other project components are located on landfill; no archaeological sites were noted in the fill.

2.9.6 NATIVE AMERICAN CONSULTATION

Four local Native American individuals of Chumash descent and the federally recognized Santa Ynez Band of Chumash were contacted by letter on April 13, 2000. The letter requested concerns and comments from the local Native American community and extended a meeting invitation to discuss the proposed ISFSI. A meeting was eventually held with one individual and another individual responded by mail.

The concerns expressed by the two Chumash individuals ranged from general concern for activities at Diablo Canyon to more specific concerns on potential harm to the environment from the ISFSI. While the individual who responded by letter noted no specific concerns, the area is of great spiritual importance to that individual.

On August 18, 2000, a second letter was sent to the Santa Ynez Band of Chumash. The letter was sent to the Tribal Elders, the Tribal subgroup that is responsible for commenting on proposed projects. No comments have been received to date in response to this letter.

2.9.7 SCENIC AND NATURAL RESOURCES

The proposed ISFSI is situated along a 12-mile stretch of California coast located between Montana de Oro State Park to the north and Avila Bay to the south. This stretch of coast is characterized by a relatively narrow and flat coastal plain or marine terrace that abuts the base of the Irish Hills. The seaward cliff edge of the marine terrace is typically rugged and rocky. Numerous offshore stacks, rock-lined coves, extensive tide pools, and near shore rocky crevices are prominent features of this landscape. Lion Rock, Diablo Rock, Pecho Rock and the Point San Luis Lighthouse are prominent scenic resources along this stretch of coast. Other than the DCPP development, the area has only a few rural structures associated with cattle grazing and farming and a 12-kV distribution line that parallels a farm road to the north of Diablo Creek.

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CHAPTER 3

THE FACILITY

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THE FACILITY

3.1 EXTERNAL APPEARANCE

The major features of the Diablo Canyon ISFSI are the storage casks and pads, the cask transfer facility (CTF), security light poles, and fences surrounding the storage site and related staging areas. The ISFSI storage site, CTF, and transport route from the DCPP fuel handling building/auxiliary building (FHB/AB) are within the owner-controlled area as described in ER Section 2.1. Travel distance from the FHB/AB to the CTF and the ISFSI storage site is approximately 1.2 miles.

The storage casks are placed on a series of reinforced concrete pads, each built as needed, within a protected area separate from that of DCPP. Each pad is designed to accommodate up to 20 storage casks in a 4 by 5 array. Ultimately, seven such pads (containing up to 138 casks plus 2 spare locations) could be built to accommodate a full offload of DCPP Units 1 and 2 reactor cores and their spent fuel pools at the end of their existing operating licenses (2021 and 2025, respectively). Initially, two pads will be constructed followed by the remaining five pads on a schedule to meet DCPP operational requirements. Prior to construction of the remaining five pads, the area to be occupied by future pads will be filled with sand or aggregate. PG&E will continue to evaluate the availability of the alternative methods of meeting its National Waste Policy Act of 1982 obligations to store spent fuel as discussed in Section 8.2. PG&E may decide to not utilize the full capacity if other alternative options of transferring spent fuel to a high level offsite storage facility become a viable option to meet the plant spent fuel storage needs. Figure 3.1-1 shows the fully developed storage pad with the seven 4 by 5 arrays end to end. Each loaded storage cask is approximately 11 ft in diameter, 20 ft high, and weighs about 360,000 pounds. There is approximately 6 ft surface-to-surface distance between casks. The series of seven storage pads will cover an area approximately 500 ft by 105 ft. An asphalt concrete paved corridor, 40 ft wide on 3 sides and 50 ft wide on the north side provides access around the concrete pads.

A security fence, with a locked gate, serves as the protected area boundary and surrounds the storage pads. There is approximately 50 ft between the storage casks and this security fence. There is a second fence, termed the restricted area fence, around the protected area, which forms the restricted area boundary. The restricted area fence is 100 ft from the nearest cask to ensure the dose rate at this boundary will be less than 2 mrem/hr in compliance with 10 CFR 20 requirements.

The CTF is located about 100 ft off the northwest corner of the western-most storage pad.

A detailed description of the storage site, the CTF, and the HI-STORM 100 System is provided in Chapter 4 of the ISFSI SAR.

CHAPTER 5

ENVIRONMENTAL EFFECTS OF ACCIDENTS

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CHAPTER 5

ENVIRONMENTAL EFFECTS OF ACCIDENTS

TABLES

TABLE 5.1-1
TOTAL ANNUAL OFFSITE COLLECTIVE DOSE (MREM) AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM THE DIABLO CANYON ISFSI [(a)]

	Normal Operations				Off-Nornal Operations		
Organ	**Effluent Release[(d)]**	**Direct Radiation[(d)]**	**Overpack Loading Operations[(e)]**	**Other Uraniun Fuel Cycle Operations[(a)]**	**Effluent Release(e)**	**Total (normal + off-normal)**	**10 CFR 72.104 Regulatory Limit**
Site Boundary (1,400 ft / 427 m)							
Whole body ADE[(b)]	0.064	5.6	13.1E-02	4.357E-02	1.27E-03	5.84	25
Thyroid ADE	0.010	5.6	13.1E-02	1.260E-01	1.02E-04	5.87	75
Critical organ ADE (Max)	0.35	5.6	13.1E-02	5.590E-02	9.31E-03	6.15	25
Nearest Resident (1.5 miles / 7,920 ft / 2414 m)							
Whole body ADE	0.27	3.5E-04	13.1E-02	4.357E-02	5.33E-03	0.45	25
Thyroid ADE	0.043	3.5E-04	13.1E-02	1.260E-01	4.31E-04	0.30	75
Critical organ ADE (Max)	1.46	3.5E-04	13.1E-02	5.590E-02	3.92E-02	1.69	25

(a) This table was taken from ISFSI SAR Table 7.5-4.

(b) Data for uranium fuel cycle operations were obtained from the DCPP FSAR Update, Rev. 11, Table 11.3-32. Table 11.3-32 was selected based on the highest dose values in the sectors at the site boundary (0.5 miles). These dose values for the site boundary were conservatively applied to the nearest resident. The critical organ dose listed was based on the total liver dose in Table 11.3-32. The values listed in Table 11.3-32 should bound the results calculated from effective dose equivalent (EDE) methodology.

(c) ADE is the annual dose equivalent.

(d) 140 casks

(e) Single cask

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SITING AND DESIGN ALTERNATIVES

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SITING AND DESIGN ALTERNATIVES

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8.2 DESIGN ALTERNATIVES

8.2.1 ALTERNATIVE ACTIONS

8.2.1.1 No Action

The purpose of the Diablo Canyon ISFSI is to provide spent fuel storage such that spent fuel can be removed from the DCPP Units 1 and 2 spent fuel pools and stored until it can be moved to a permanent federal repository. If the present inventory of fuel remains in the DCPP spent fuel pools, the ability to discharge a full core from either unit would be lost by 2006 with the presently anticipated refueling schedules. Ultimately, the lack of onsite spent fuel storage could force a premature shutdown of the DCPP units and would pose significant electrical supply problems. Short and long term replacement power facilities would generate power at a much higher cost and result in higher electricity rates for PG&E customers. Also, these replacement power sources would likely involve the use of fossil fuels, thereby causing greater environmental impacts.

In the Nuclear Waste Policy Act of 1982 (NWPA) [42 U.S.C. Section 10101 et seq], Congress determined that the operators of civilian nuclear power plants have "primary responsibility" for interim storage of spent nuclear fuel pending federal development of a permanent disposal repository. The NWPA further specified that operators should meet their responsibility "by maximizing, to the extent practical, the effective use of existing storage facilities at the site of each civilian nuclear power reactor, and by adding new onsite storage capacity in a timely manner where practical" [42 U.S.C. Section 10151(a)(1)]. Congress also declared that the purpose of the NWPA was to promote the "addition of new spent nuclear fuel storage capacity" at civilian reactor sites [Id. at Section 10151(b)(1)], and directed federal agencies to "take such actions. . . necessary to encourage and expedite the effective use of available storage, and necessary additional storage" at reactor sites [Id. at Section 10152]. The Diablo Canyon ISFSI is in accord with the mandate of the NWPA. Accordingly, the "no action" alternative is not a viable approach.

8.2.1.2 Increasing Capacity of Existing Pools

As discussed below, increasing the capacity of the existing spent fuel pools through reracking or fuel rod consolidation are not preferred or viable, respectively, for long-term solutions to spent fuel storage at DCPP. However, implementing one of these options could provide short-term relief. Both options would require more operational, maintenance, and surveillance activities than a dry cask storage system, which relies upon passive features to maintain cooling and radiation shielding. The additional operational, maintenance, and surveillance activities would result in a loss of efficiency and increased costs.

8.2.1.2.1 Reracking

The existing Diablo Canyon spent fuel storage racks are a high-density design that could be replaced with an even more densely configured rack system to expand the number of storage cells per unit from 1324 to 2102. For Unit 1, this increase would provide storage for all spent fuel assemblies to the end of the operating license in 2021. However, for Unit 2, this expansion would be insufficient to allow operation to the end of the operating license in 2025. In addition, the increased heat load from the higher density racks could require modifications to increase the cooling capacity of the spent fuel pool cooling systems.

Since additional storage capacity from an increased density design would not allow operation of both DCPP units until the end of their initial operating licenses, this alternative is not the preferred alternative.

8.2.1.2.2 Spent Fuel Rod Consolidation

Spent fuel rod consolidation involves removing all the fuel rods from two spent fuel assemblies, reconfiguring them into a closely packed array, and then placing them into a canister of the same outside dimensions as a fuel assembly. The canister is then stored in a rack location formerly occupied by a single fuel assembly. The remaining fuel assembly skeletons are then compacted and placed in another canister designed to hold approximately ten such skeletons. In this way ten fuel assemblies can be consolidated into approximately five canisters of fuel rods and one canister of skeletons.

Due to the high seismic design requirements at DCPP, the fuel racks are not designed to accommodate the higher mass of consolidated fuel. Additionally, consolidation requires extensive operational resources that could interfere with normal plant operations. For these reasons, this alternative was not considered viable.

8.2.1.3 Construction of a New Storage Pool

This alternative involves the construction of a new storage pool and support facilities separate from the existing DCPP spent fuel pool. A new storage pool would require the same support facilities and maintenance as the existing pool (e.g., fuel handling equipment, large capacity cask crane, building ventilation, and water quality systems). This option would require the spent fuel to be transferred via a dry cask storage system to the new storage pool. This option would require more operational, maintenance, and surveillance activities to maintain its safety than a dry cask storage system, which relies upon passive features to maintain cooling and radiation shielding. A wet storage pool would also involve additional handling of the fuel, which would likely result in higher radiation doses to workers, as well as an increase in the risk of a fuel handling accident. This alternative was not selected because of its high cost, the potential for higher occupational radiation exposures, and the time needed for design, licensing, and construction.

8.2.1.4 Ship Fuel to a Permanent Federal Repository

This is the PG&E preferred solution to storage of spent fuel from Diablo Canyon. However, a permanent Federal repository will not be available in a timeframe consistent with the spent fuel storage needs of PG&E. DOE is currently working to develop a repository as required by the NWPA of 1982, as amended in 1987. DOE is evaluating a site in Yucca Mountain, Nevada, to determine if it is a suitable location for a high-level radioactive waste repository. Currently, DOE does not anticipate having a licensed repository ready to receive spent fuel until 2010. Although DOE recommended that a Monitored Retrievable Storage (MRS) facility be constructed and in operation by 1988, the NWPA prohibits siting an MRS before obtaining a construction permit for the permanent repository. Given the uncertainties of schedules for either a repository or an MRS, this alternative does not meet the needs of PG&E to store spent fuel.

8.2.1.5 Ship Fuel to a Reprocessing Facility

There are no operating commercial reprocessing facilities in the United States nor is there the prospect for one in the foreseeable future. Reprocessing facilities are in operation in other countries (e.g., United Kingdom, France, Germany, and Japan); however, the shipment of domestic spent fuel to a foreign country for storage or disposal involves a number of political, legal, and logistical uncertainties. This is not considered a viable alternative.

8.2.1.6 Ship Fuel to a Private Spent Fuel Storage Facility

There are no private licensed storage facilities available at this time to provide for interim storage of spent fuel and other radioactive materials from DCPP. However, several utilities have formed the Private Fuel Storage L. L. C. (PFSLLC) to construct a privately owned ISFSI that will store spent fuel from several nuclear plants at a central site. This ISFSI, called the Private Fuel Storage Facility (PFSF), will be located on the Skull Valley Indian Reservation in Utah. The PFSLLC has entered into a lease agreement with the Skull Valley Band of Goshute Indians for the site.

The PFSF would incorporate the dry cask storage technology that is currently in use or proposed for use at several operating nuclear power plants, including the HI-STORM 100 System. The construction and operation of the PFSF is therefore a potential substitute for building individual onsite ISFSIs at various nuclear power plant sites. Presently, there is no assurance the project will be successfully licensed and built. Moreover, based on current licensing and construction schedules, the PFSF would not be available until 2003 at the earliest, and there is no assurance as to when DCPP spent fuel could be accepted. Therefore, efforts must begin now for the design, licensing, and construction of the Diablo Canyon ISFSI to meet the requirements for DCPP fuel storage by 2006. Moreover, even if the PFSF were available, this alternative would involve an extra offsite shipment of the spent fuel for ultimate disposal at a DOE repository. This is not considered a viable alternative at this time.

8.2.1.7 Ship Fuel to Another Nuclear Power Plant

This alternative would involve shipping the DCPP spent fuel to another nuclear power plant with sufficient storage capacity. The other utility would have to be licensed for and agree to accept the DCPP spent fuel. Since all the power reactor operators are expected to face spent fuel pool storage shortfalls, they are not expected to be willing to reduce their own storage capacity. No reactor licensees have requested approval from the NRC to accept spent fuel from a reactor site owned by another company, and no proposals for such requests have been identified to date. Therefore, this is not considered to be a viable alternative.

8.2.1.8 Conclusions

PG&E will continue to evaluate the availability of the alternative methods of meeting its National Waste Policy Act of 1982 obligations to store spent fuel as discussed in Section 8.2. PG&E may decide to not utilize the full capacity of the ISFSI if other alternative options of transferring spent fuel to a high level offsite storage facility become a viable option to meet the plant spent fuel storage needs.

8.2.2 ALTERNATIVE DRY STORAGE DESIGNS

PG&E evaluated proposals from four different vendors for the Diablo Canyon ISFSI. The fuel storage systems evaluated were all dry storage systems and included:

- Canister-based dual purpose systems suitable for both storage and eventual offsite shipment
- Horizontal and vertical concrete vault systems suitable only for fuel storage

PG&E's evaluation process compared the various designs based on a number of factors including:

- Compatibility with the proposed site
- Potential radiation exposure
- Effects of postulated off-normal events
- Regulatory compliance and licensing issues
- Cost and other commercial considerations
- Engineering/licensing capability of the vendor

Based on this evaluation, PG&E selected the HI-STORM 100SA System designed by Holtec International. The HI-STORM 100SA System is a high-seismic storage system that is anchored to the storage pad. Chapter 4 of the Diablo Canyon ISFSI Safety Analysis Report contains a detailed description of the HI-STORM 100SA System.