CHAPTER 5

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. 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. 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 required periodic visual surveillance of the overpack inlet and outlet air ducts to ensure they remain free of blockage and intact. 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.

5.2 CONTROL ROOM AND CONTROL AREAS

Due to the welded closure of the MPC, the passively-cooled storage cask design, and the Diablo Canyon ISFSI TS requirement for periodic checks of the casks, the Diablo Canyon ISFSI does not require continuous surveillance and monitoring or operator actions to ensure that its safety functions are performed during normal, off-normal, or postulated accident conditions. Therefore, a control room or control area is not considered necessary, as allowed by 10 CFR 72.122(j).

Normal loading and unloading operations will take place in the DCPP fuel handling building/auxiliary building under local control and in coordination with the DCPP control room staff and subject to the controls established under the DCPP 10 CFR 50 license.

Operation during the transport phase will be under local control by DCPP personnel.

5.3 SPENT FUEL ACCOUNTABILITY PROGRAM

Accountability and control of spent fuel will be maintained at all times during loading, transfer, and storage operations. Loading, transfer, and inventory records for spent fuel moved from the DCPP spent fuel storage pools to the Diablo Canyon ISFSI storage site will be maintained in accordance with existing DCPP procedures. The Diablo Canyon ISFSI storage site will be treated as a separate material balance area from DCPP.

As required by 10 CFR 72.72, records will be maintained showing the receipt, inventory (including location), disposal, acquisition, and transfer of all spent fuel and radioactive waste in storage. In addition, accountability records for all fuel assemblies transferred to, stored at, or removed from the Diablo Canyon ISFSI will be maintained for as long as fuel assemblies are stored at the ISFSI and retained for a period of 5 years after the fuel is transferred out of the ISFSI. Section 9.4.2 of this SAR provides the justification for an exemption from the method of storage requirements of 10 CFR 72.72(d), which will allow records of spent fuel storage to be maintained in the same manner as the DCPP QA records.

All nonfuel hardware associated with the DCPP spent fuel assemblies is identified by a unique serial number permanently stamped or engraved on the hardware. Verification of the nonfuel serial numbers will be made to ensure that only appropriate nonfuel hardware is stored with the spent fuel assemblies. The verification will include verifying in which fuel assembly the nonfuel hardware is stored.

Material status reports will be completed and submitted to the NRC as specified in 10 CFR 72.76. Nuclear material stored at the ISFSI is not expected to be transferred from PG&E until eventual transfer to DOE for transportation to a DOE storage facility. Therefore, Nuclear Transaction reports (DOE/NRC Form-741) required by 10 CFR 72.78 will not be needed until that time.

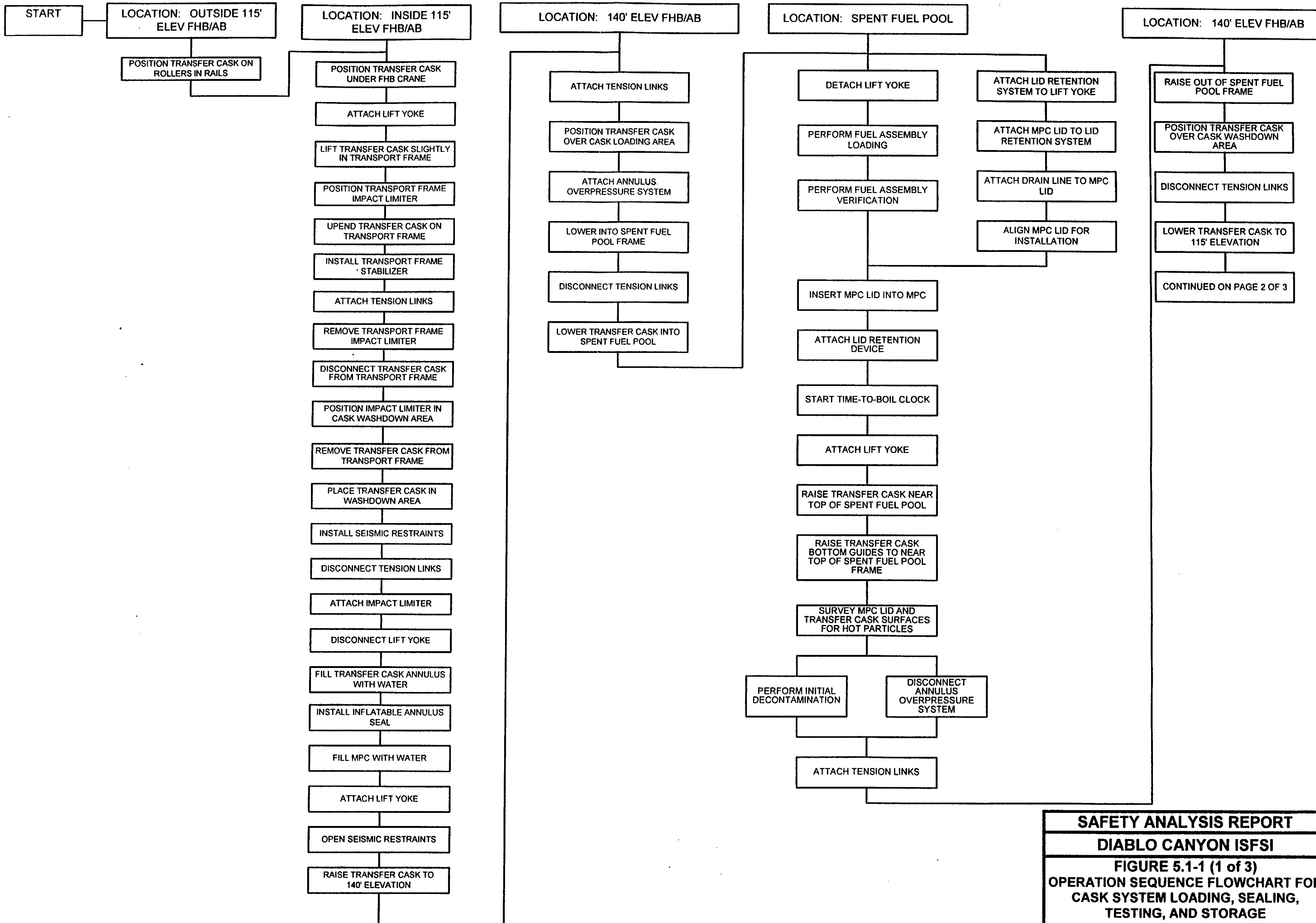
5.4 SPENT FUEL TRANSPORT

Spent fuel transport from the fuel handling building/auxiliary building to the CTF and, subsequently, to the ISFSI storage pads, is accomplished using a specifically designed transporter. Design criteria for the transporter are presented in Sections 3.2 and 3.3.3. A description of the transporter is provided in Section 4.3. Operation of the transporter is described in Sections 4.4.1.2.4 and 5.1.1.3. The location and construction features of the transport route are described in Section 4.3.3.

TABLE 5.1-1

MEASURING AND TEST EQUIPMENT

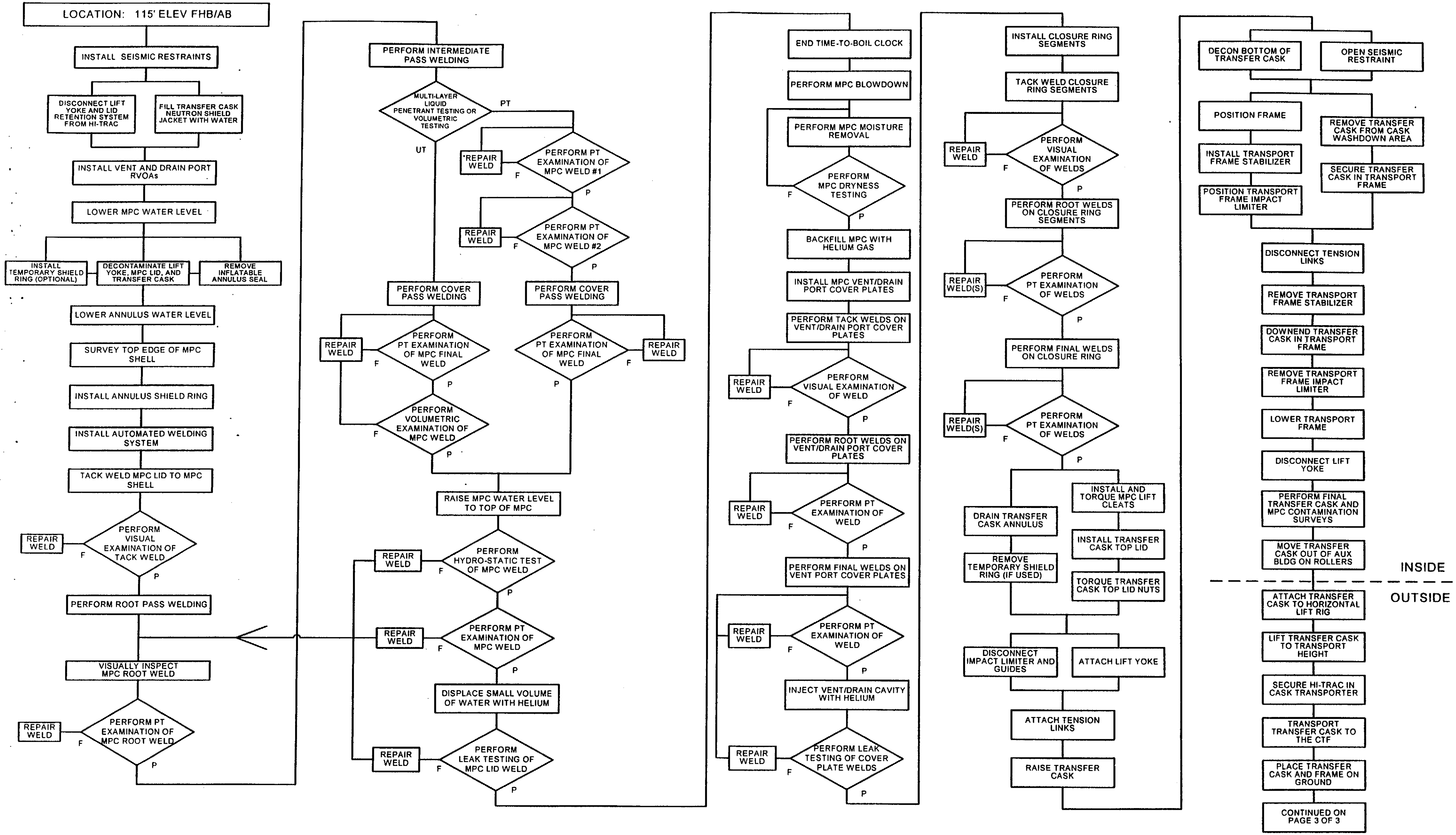
Instrument	Function
Contamination Survey, Radiation Monitoring Instruments	Measures contamination levels and dose rate levels on HI-STORM 100SA overpack, MPC lid, HI-TRAC transfer cask and ancillaries.
Flow Rate Monitor	Monitors gas flow rate during assembly cool-down.
Helium Mass Spectrometer Leak Detector	Ensures leakage rates are within acceptance criteria.
Pressure and Vacuum Gauges	Ensures correct helium backfill and MPC dryness during loading operations.
Temperature Gauge	Monitors the state of fuel cooldown prior to MPC flooding and ensures MPC dryness during loading operations when FHD system is used..
Water Totalizer	Used for water pumpdown prior to lid welding operations.



SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

FIGURE 5.1-1 (1 of 3)
OPERATION SEQUENCE FLOWCHART FOR CASK SYSTEM LOADING, SEALING, TESTING, AND STORAGE

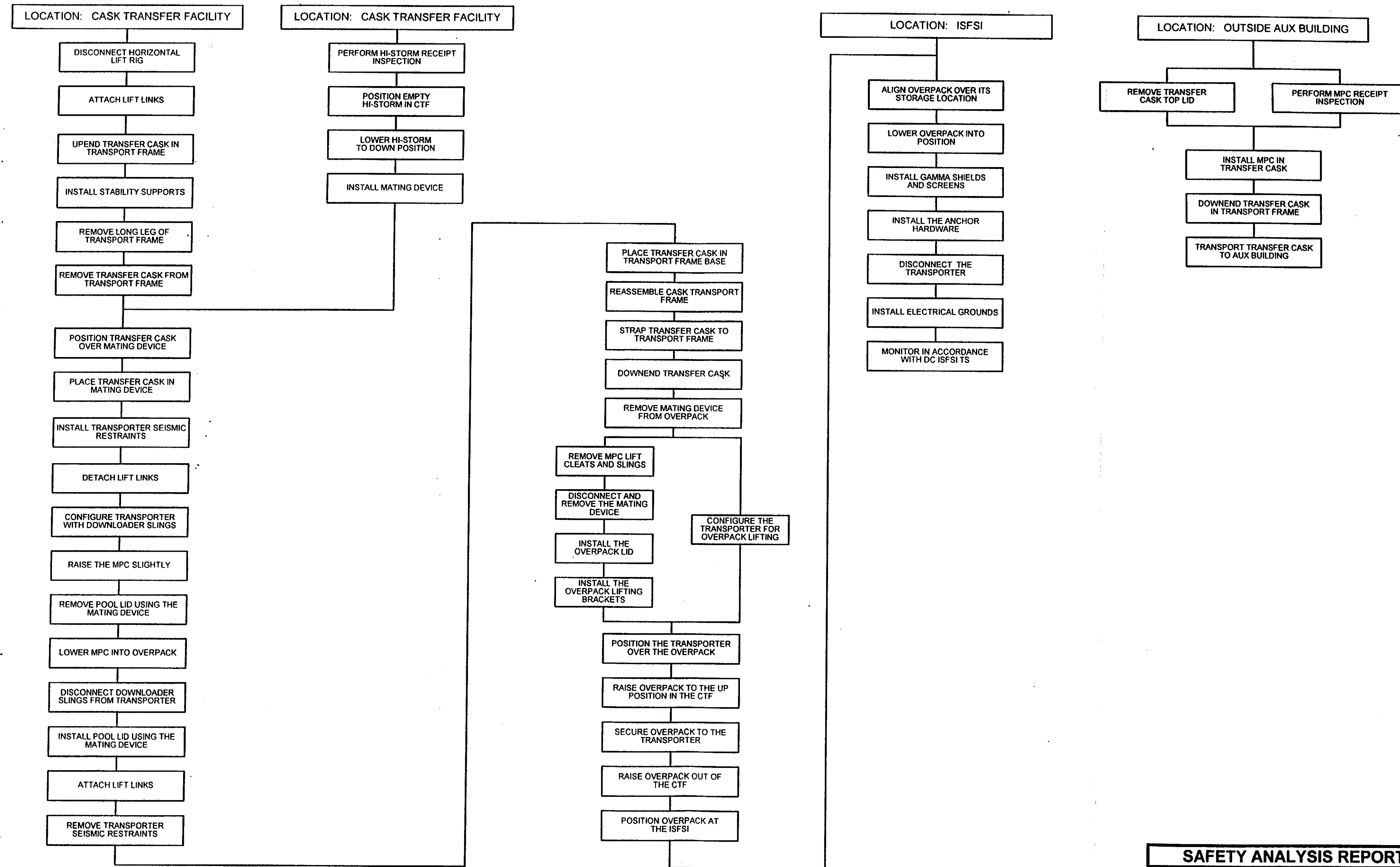


Note: F = Fail
P = Pass

SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

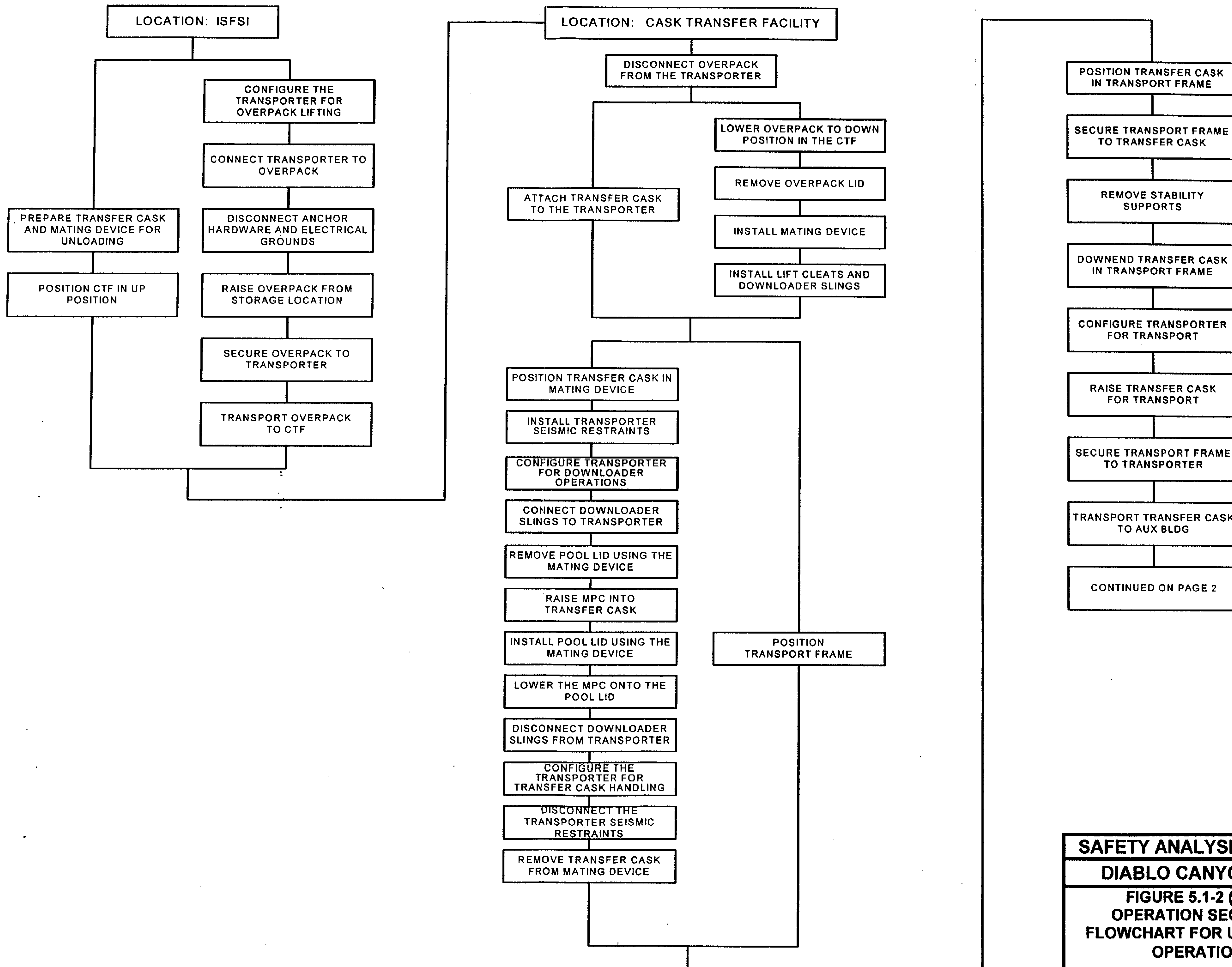
FIGURE 5.1-1 (2 of 3)
OPERATION SEQUENCE FLOWCHART FOR CASK SYSTEM LOADING, SEALING, TESTING, AND STORAGE



SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

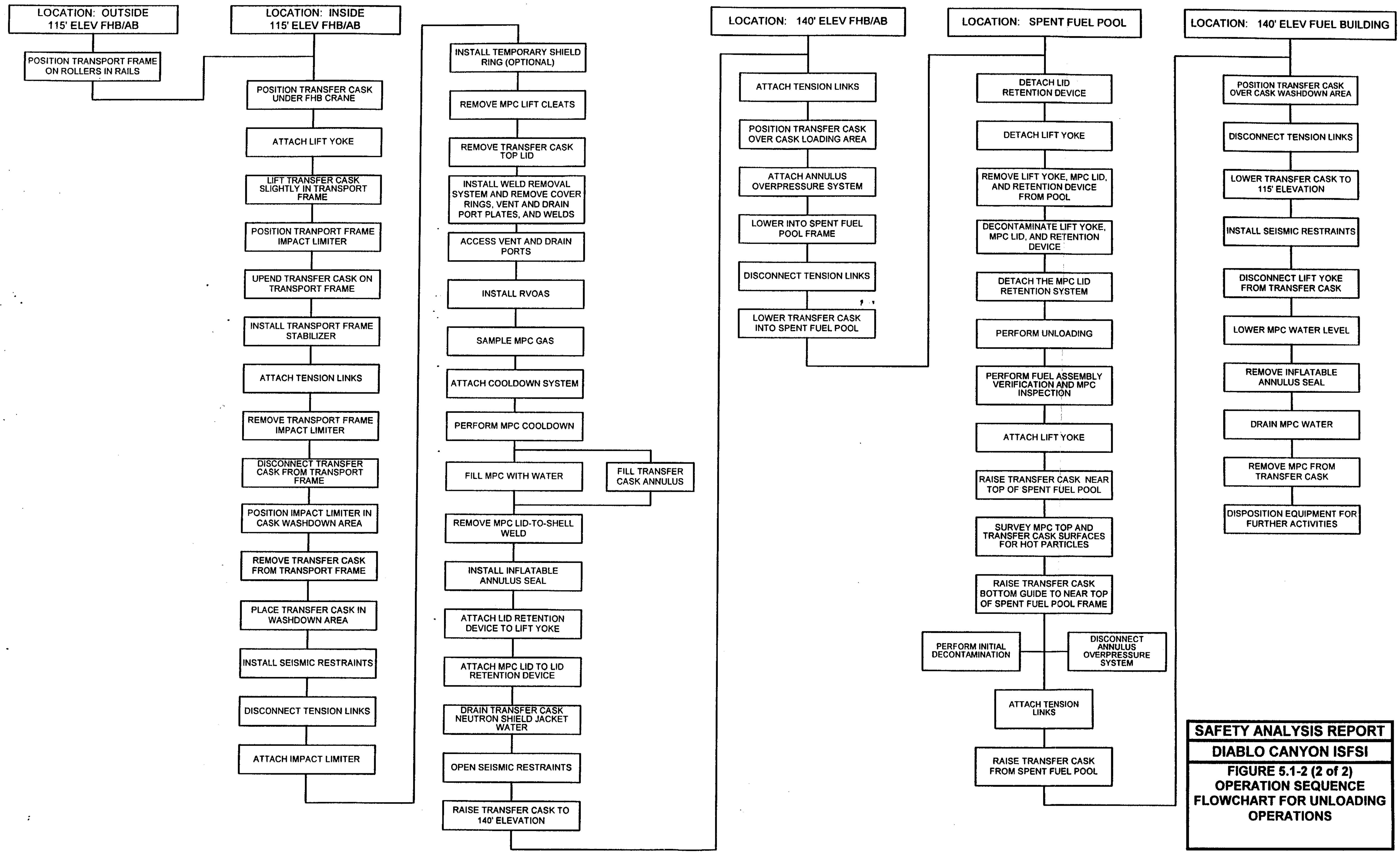
FIGURE 5.1-1 (3 of 3)
OPERATION SEQUENCE FLOWCHART FOR CASK SYSTEM LOADING, SEALING, TESTING, AND STORAGE



SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

**FIGURE 5.1-2 (1 of 2)
OPERATION SEQUENCE
FLOWCHART FOR UNLOADING
OPERATIONS**



SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

**FIGURE 5.1-2 (2 of 2)
OPERATION SEQUENCE FLOWCHART FOR UNLOADING OPERATIONS**

DIABLO CANYON ISFSI
SAFETY ANALYSIS REPORT

CHAPTER 6

WASTE MANAGEMENT

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CHAPTER 6

WASTE MANAGEMENT

6.1 MPC CONFINEMENT BOUNDARY DESIGN

The MPC is designed to endure normal, off-normal, and accident conditions of storage with maximum decay heat loads without loss of confinement. The MPC confinement boundary ensures that there will be no release of radioactive materials from the cask storage system under all postulated loading conditions. Refer to Chapter 3 for additional detail regarding confinement barriers and systems.

6.2 RADIOACTIVE WASTES

No radioactive wastes will be generated due to transport or storage of the loaded MPC at the ISFSI. Radioactive wastes generated during MPC loading operations in the fuel handling building/auxiliary building (FHB/AB) will be treated using existing DCPP radioactive waste control systems as described in the DCPP Final Safety Analysis Report (FSAR) Update, Chapter 11, "Radioactive Waste Management" (Reference 1).

Contaminated water from loaded MPCs will normally be drained back into the spent fuel pool with no additional processing. A small amount of liquid waste will result from transfer cask and MPC decontamination. The decontamination procedure may result in a small amount of detergent/demineralized mixture being collected in the FHB/AB. Liquid wastes in this area are directed to the liquid radwaste disposal system.

If necessary, potentially contaminated air and helium from the MPC during loading and unloading operations will be connected to the gaseous radwaste system. A small quantity of low-level solid waste may be generated during MPC loading operations. The solid waste may include disposable anti-contamination garments, paper, rags, tools, etc., and will be processed as described in the DCPP FSAR Update, Section 11.5, "Solid Waste System."

Any water collected in the CTF sump will be sampled before it is discharged. If the water is found to be contaminated, it will be disposed of in accordance with the DCPP radioactive waste management program.

6.3 REFERENCES

1. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update, Revision 14, November 2001.

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RADIATION PROTECTION

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CHAPTER 7

RADIATION PROTECTION

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CHAPTER 7

RADIATION PROTECTION

This chapter provides information regarding the radiation protection design features of the ISFSI and the estimated onsite and offsite doses expected due to operation of the Diablo Canyon ISFSI. The generic HI-STORM 100 System, described in the HI-STORM 100 System FSAR, as amended by Holtec License Amendment Request (LAR) 1014-1 (References 1 and 2, respectively) will be deployed at the Diablo Canyon ISFSI. The generic shielding analyses, including methodology, computer codes, and modeling were performed and licensed in accordance with NUREG-1536. These same, previously-licensed techniques, were used in performing the site-specific analyses described in this chapter.

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

7.1.1 POLICY CONSIDERATION AND ORGANIZATION

It is the policy of Pacific Gas and Electric Company (PG&E), through Nuclear Power Generation (NPG), to design, operate, and maintain the Diablo Canyon ISFSI in a manner that maintains personnel radiation doses as low as is reasonably achievable (ALARA).

DCPP's ALARA program, which complies with the requirements of 10 CFR 20 and 10 CFR 50, is considered sufficient for ISFSI operations under 10 CFR 72. The ALARA program is implemented through NPG program directives, administrative procedures, and working level procedures. These documents will be revised as needed to address ISFSI operations prior to operation of the ISFSI.

The Health Physics Program used for operating the Diablo Canyon ISFSI is described in Section 7.6 and implements the requirements of 10 CFR 20, 10 CFR 72, and the NPG policy for implementation of the ALARA philosophy for all site activities involving potential radiation exposure. The Radiation Protection Manager is responsible for administering, coordinating, planning, and scheduling all radiation protection activities involving the ISFSI.

The primary objective of the Health Physics Program is to maintain radiation exposures to workers, visitors, and the general public below regulatory limits and otherwise ALARA.

The Holtec HI-STORM 100 System, chosen for use at the Diablo Canyon ISFSI, has been designed with the principles of ALARA considered for the operation, inspection, maintenance, and repair of the cask system. PG&E provides the facilities, equipment, and the trained and qualified staff to ensure that any radiation exposures due to ISFSI operations are ALARA. The ISFSI storage pad will be monitored and evaluated on a routine basis to ensure that radiation exposures from the ISFSI storage pad to unrestricted areas are ALARA.

Specific design- and operations-oriented ALARA considerations are described in the following sections.

7.1.2 DESIGN CONSIDERATIONS

The Diablo Canyon ISFSI storage pad site is located in an area adjacent to the raw water reservoir. The location was chosen based on two ALARA considerations as follows:

- The ISFSI is centrally located within the DCPP site boundary, thus maintaining offsite doses ALARA.

- The ISFSI is sufficiently distant from buildings and occupied spaces so that the doses to onsite personnel are maintained ALARA.

The layout of the ISFSI storage pads is designed to minimize personnel exposures during routine surveillance, maintenance, and repair activities. The overpacks will be sufficiently spaced to allow adequate personnel access between the casks.

Regulatory Position 2 of NRC Regulatory Guide 8.8 (Reference 3) provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the Diablo Canyon ISFSI and the HI-STORM 100 System as described below:

- Regulatory Position 2a, regarding access control, is met by the use of a restricted area fence for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials and a security perimeter fence with a locked gate that surrounds the ISFSI storage pad and prevents unauthorized access.

- Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure to the extent practicable. Fundamental design considerations that directly influence occupational exposures and which have been incorporated into the HI-STORM 100 System design include:

  - Minimization of the number of handling and transfer operations for each spent fuel assembly

  - Minimization of the number of handling and transfer operations for each MPC loading

  - Maximization of fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be stored at the ISFSI

  - Minimization of planned maintenance requirements

- Minimization of decontamination requirements at ISFSI decommissioning
- Optimization of the placement of shielding with respect to anticipated worker locations and fuel placement during loading and transfer operations
- A thick-walled overpack that provides gamma and neutron shielding
- A single, thick MPC lid (rather than separate structural and shield lids) that provides effective shielding for operators during MPC loading and transfer operations
- Multiple welded barriers to confine radionuclides
- Smooth surfaces to reduce decontamination times
- MPC penetrations located and configured to reduce streaming paths
- Overpack and transfer cask designed to reduce streaming paths
- MPC vent and drain ports, with remotely operated valves, to prevent the release of radionuclides during loading and unloading operations and to facilitate draining, drying, and backfill operations
- Use of an annulus overpressure system to minimize contamination of the MPC shell outer surfaces during loading operations
- Minimization of maintenance to reduce doses during storage operation
- Use of a dry environment inside the MPC cavity to preclude the possibility of release of contaminated liquids.

• Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at the ISFSI.

• Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM 100 System is designed to withstand all normal, off-normal, and accident design-basis conditions without loss of confinement function, as described in Chapter 7 of the HI-STORM 100 System FSAR. Therefore, no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean borated water in the transfer cask MPC annulus and by using an inflatable annulus seal to preclude spent fuel pool (SFP) water contacting the exterior surface of the MPC.

- Regulatory Position 2e, regarding crud control, is not applicable to the Diablo Canyon ISFSI since there are no radioactive systems at the ISFSI that could transport crud.

- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the DCPP fuel handling building/auxiliary building (FHB/AB). The exterior surface of the transfer cask is designed with a minimal number of crud traps and a smooth, painted surface for ease of decontamination. In addition, an inflatable annulus seal and annulus overpressure system are used to prevent SFP water from contacting and contaminating the exterior surface of the MPC.

- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at the ISFSI.

- Regulatory Position 2h, regarding resin treatment systems, is not applicable to the Diablo Canyon ISFSI since there are no treatment systems containing radioactive resins.

- Regulatory Position 2i, regarding other miscellaneous features, is met because the ISFSI storage pad is located in a cut into an existing hill and located away from normally-occupied power plant areas. The hill provides natural shielding on one side and partial shielding on two sides, and the ISFSI pads are set back a sufficient distance from the controlled area boundary to ensure low dose rates in the uncontrolled area. In addition, the MPC is constructed from stainless steel. This material is resistant to corrosion and the damaging effects of radiation, and is well proven in spent nuclear fuel storage cask service.

7.1.3 OPERATIONAL CONSIDERATIONS

Operating procedures for the Diablo Canyon ISFSI, including cask loading, unloading, transfer to the cask transfer facility (CTF), MPC transfer, and movement to the ISFSI storage pad are detailed in Chapter 5. The operating procedures were developed with an underlying ALARA philosophy and have been modified, as appropriate, to incorporate lessons learned from actual loading campaigns conducted at other nuclear power plants. ISFSI personnel will follow site-specific implementing procedures consistent with the philosophy of Regulatory Guides 8.8 and 8.10. Personnel radiation exposure during ISFSI operations is minimized through the incorporation of the following concepts:

- Fuel loading procedures that follow accepted practice and build on lessons learned from operating experience

- Preparation of the loaded MPC and transfer cask inside the FHB/AB using existing plant equipment and procedures, where possible
- Use of an optional regionalized loading strategy, where feasible, to take advantage of shielding provided by placing lower burnup and longer cooled fuel assemblies on the periphery of the MPC basket
- Filling of the annulus between the MPC and the transfer cask with clean borated water and using the inflatable annulus seal and annulus overpressure system to minimize contamination of the outer surface of the MPC
- Performance of as many MPC preparation activities as possible with water in the MPC cavity
- Filling of the transfer cask water jacket with water before draining the water out of the MPC cavity
- Use of temporary portable shielding, as appropriate, including a bottom shield for the transfer cask when oriented horizontally
- Use of power-operated tools, when possible, to install and remove bolts on the transfer cask and overpack
- Consideration of the ALARA philosophy in job briefings prior to fuel movement, cask loading, and MPC preparation
- Use of classroom training, mock-ups and dry-run training to verify equipment operability and procedure adequacy and efficiency.

7.1.4 REFERENCES

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

2. License Amendment Request No. 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. Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable, USNRC, June 1978.

7.2 RADIATION SOURCES

The source terms presented in this section of the SAR were developed specifically for use in the Diablo Canyon ISFSI shielding analyses. Other sections of this SAR reference dose analyses from the HI-STORM 100 System FSAR (Reference 1) and the HI-STORM 100 System FSAR, as amended by LAR 1014-1 (Reference 2). The source terms used for the dose analyses referenced from the HI-STORM 100 System FSAR, as amended by LAR 1014-1, are contained in those documents and, therefore, are not repeated in this section.

7.2.1 CHARACTERIZATION OF SOURCES

Shielding analyses for dose rates from direct radiation were performed assuming that the overpacks contain MPC-32s completely loaded with fuel assemblies having identical burnup and cooling times. The burnup was assumed to be 32,500 MWD/MTU with an initial cooling time of 5 years. In the estimation of the doses presented in Sections 7.4 and 7.5, credit was taken for additional cooling time from 5 years to 20 years as the casks are placed at the ISFSI over time. An annual loading campaign of eight casks each year was assumed. This initial burnup and cooling time value is based on SAR Section 10.2 for uniform fuel loading. It is demonstrated in Section 7.3 that the dose rates on the surface of the overpack calculated using this burnup and cooling time bound the dose rates calculated using the other allowable burnup and cooling times. In addition, it is demonstrated that the dose rates calculated for an overpack containing an MPC-32 bound the dose rates calculated for an overpack containing an MPC-24, MPC-24E, or MPC-24EF.

The shielding analysis for the transfer cask that is presented in this chapter was performed for the MPC-24 using a burnup and cooling time of 55,000 MWD/MTU and 12 years, respectively, based on SAR Section 10.2 for uniform loading. It is demonstrated in Section 7.3 that the dose rates on the surface of the transfer cask using this burnup and cooling time bound the dose rates using other allowable burnup and cooling times. It is also demonstrated that the dose rates from a transfer cask containing an MPC-24 bound the dose rates from a transfer cask containing an MPC-32.

A review of the fuel inventory, as of November 2000, indicates that fuel assemblies with burnups between 30,000 and 35,000 MWD/MTU have an average initial enrichment of 3.01 wt percent ^{235}U and that assemblies with burnups between 50,000 and 55,000 MWD/MTU have an average initial enrichment of 4.2 wt percent ^{235}U. Since lower enrichments result in slightly higher neutron source terms, enrichments of 2.9 and 4.0 wt percent ^{235}U were conservatively used for the analysis of the 32,500 and 55,000 MWD/MTU burnups, respectively.

The principal sources of direct radiation in the HI-STORM 100 System are:

- Gamma radiation originating from the following sources
  - Decay of radioactive fission products

  - Secondary photons from neutron capture in fissile and nonfissile nuclides
  - Hardware activation products generated during power operations

- Neutron radiation originating from the following sources
  - Spontaneous fission
  - Alpha, neutron (α, n) reactions in fuel materials
  - Secondary neutrons produced by fission from subcritical multiplication
  - Gamma, n (γ, n) reactions (this source is negligible)

The foregoing can be grouped into three distinct sources, each of which is discussed below: fuel-gamma source, fuel-neutron source, and nonfuel-hardware-activation source. The source terms for the analyses presented in this SAR were calculated using the same methods described in the HI-STORM 100 System FSAR. The neutron and gamma source terms, along with the quantities of radionuclides available for release, were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system (References 3 and 4, respectively).

7.2.1.1 Design-Basis Fuel Assembly

The physical characteristics of the fuel used at DCPP are summarized in Table 3.1-1 and SAR Section 10.2.

Section 5.2 of the HI-STORM 100 System FSAR describes the design basis pressurized water reactor (PWR) fuel assembly based on a comparison of source terms from the PWR fuel assembly classes permitted for storage under the HI-STORM 100 System general certification. It was determined that the B&W 15-by-15 fuel assembly, which has the highest uranium mass of the allowable fuel assemblies, was the assembly with the highest radiation source and therefore was the design-basis fuel assembly. Since the fuel assemblies used for DCPP are permitted for storage under the HI-STORM 100 general certification, they are bounded by the determination of the design-basis fuel assembly in the HI-STORM 100 System FSAR. Therefore, for conservatism, the B&W 15-by-15 design basis PWR fuel assembly described in Table 5.2.1 of the HI-STORM 100 System FSAR was used for the analysis presented in this chapter. Tables 5.3.1 and 5.3.2 of the HI-STORM 100 System FSAR describe the axial location of the sources in the fuel assembly and the material composition of the assembly. The axial burnup profile used in these analyses and the position of the assembly within the MPC were identical to those described in Chapter 2 of the HI-STORM 100 System FSAR.

The HI-STORM 100 System FSAR, as amended by LAR 1014-1, describes the shielding analysis to qualify generic damaged fuel assemblies. The discussion in Section 5.4.2 of the HI-STORM 100 System FSAR describes the effect of damaged fuel assemblies on the external dose rates. This discussion indicates that the change in dose rate associated with the storage of damaged fuel assemblies is not significant. Based on that analysis and the reasonable expectation that there will be few damaged fuel assemblies stored in the Diablo Canyon ISFSI, a specific evaluation of damaged fuel assemblies was not performed. Rather, all assemblies in all casks were assumed to be intact at the design basis burnup and cooling times.

7.2.1.2 Fuel-Gamma Source

Tables 7.2-1 and 7.2-2 present the gamma source terms that were used for the active fuel portion of the design basis assemblies for the overpack and transfer cask analyses, respectively. The source is presented in both MeV/sec and photons/sec for an energy range of 0.45 MeV to 3.0 MeV. Section 5.2.1 of the HI-STORM 100 System FSAR provides the justification that only photons in this energy range need to be considered in the dose evaluation. The HI-STORM 100 System FSAR states: "Photons with energies below 0.45 MeV are too weak to penetrate the overpack or transfer cask, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose."

As mentioned above, the cooling time was varied from 5 to 20 years for the HI-STORM analysis to account for residency time on the ISFSI storage pad as the casks are assumed to be deployed in annual, 8-cask increments. In order to minimize the volume of data presented, Table 7.2-1 only presents the source term for the odd-year cooling times beginning at 5 years and ending at 15 years. This approach is also used in presenting the other source terms described below.

7.2.1.3 Fuel-Neutron Source

Table 7.2-3 and 7.2-4 present the neutron source term used for the active fuel portion of the design-basis fuel assemblies for the overpack and transfer cask analyses, respectively. The neutron source is presented in neutrons/sec. Section 5.2.2 of the HI-STORM 100 System FSAR provides additional discussion on the calculation of the neutron source.

The neutron source term increases as the ^{235}U enrichment decreases for the same burnup and cooling time. Therefore, as discussed earlier in this section, a bounding low enrichment was chosen for the source term calculations. The neutron source strength also varies with burnup, by the power of 4.2 (Reference 1). Since this relationship is nonlinear and since burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 of the HI-STORM 100 System FSAR was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnup listed in Table 2.1.11 of the HI-STORM 100 System FSAR for the PWR fuel is 1.105. Using the power of 4.2 relationship results in a 37.6 percent ($1.105^{4.2}/1.105$) increase in the neutron source strength in the peak nodes and the total neutron source strength listed in Tables 7.2-3 and 7.2-4 increases by 15.6 percent. This increase in neutron source term is not reflected in the data presented in Tables 7.2-3 and 7.2-4, but is accounted for in the shielding analysis.

7.2.1.4 Nonfuel-Hardware Source

As mentioned above, the nonfuel hardware of a fuel assembly (for example, steel and inconel in the end fittings) activate during in-core operations to produce a radiation source. The primary radiation from these portions of the fuel assembly is ^{60}Co activity. Radiation from other isotopes within the steel and inconel has a negligible impact on the radiation dose rate compared with the ^{60}Co activity. Therefore, ^{60}Co was the only isotope considered in the analysis. The method used to calculate the activity in the nonfueled regions of the assembly is fully described in Section 5.2.1 of the HI-STORM 100 System FSAR. The ^{59}Co impurity level assumed in the steel and inconel of the fuel assembly was 1.0 g/kg or 1000 ppm. It was also assumed for this analysis that the fuel assemblies contained nonzircaloy grid spacers with a ^{59}Co impurity level of 1.0 g/kg. This assumption also conservatively bounds nonzircaloy fuel clips, which are present on a limited number of fuel assemblies. The HI-STORM 100 System FSAR (Chapter 8) discusses how this ^{59}Co impurity level value is conservative relative to fuel manufactured since the late 1980s.

Tables 7.2-5 and 7.2-6 list the ^{60}Co source that was used in the nonfuel portions of the fuel assemblies for the overpack and transfer cask analyses, respectively. Tables 5.2.1 and 5.3.1 of the HI-STORM 100 System FSAR describe the mass and dimensions of these nonfuel portions of the fuel assembly.

The HI-STORM 100 System FSAR, as amended by LAR 1014-1, includes burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), and rod cluster control assemblies (RCCAs) in the authorized contents of the HI-STORM 100 System. Since the DCPP fuel inventory includes assemblies containing all of these devices in some quantity, they were considered in the analysis. The HI-STORM LAR 1014-1 describes the design-basis BPRA, RCCA, and TPD. Results presented in the HI-STORM LAR 1014-1 demonstrate that the design-basis BPRA results in the highest dose rates compared to the TPD and RCCA. This is because the BPRA and TPD are very similar with the exception that the BPRA has an activated portion within the active fuel region. Since the RCCAs are limited to a quantity of four per cask in the center four locations, their contribution to the external dose rate is negligible compared to that of the BPRAs, which can be stored in any position. Therefore, only the BPRAs were considered in this analysis. As described above, the only isotope of concern in the activation of the BPRA is ^{60}Co. Consistent with the analysis in the HI-STORM 100 System FSAR, as amended by LAR 1014-1, the ^{59}Co impurity level was assumed to be 0.8 g/kg or 800 ppm in stainless steel and 4.7 g/kg or 4700 ppm in inconel. Table 7.2-7 provides the source term that was calculated for the BPRAs. This source was calculated using the design basis BPRA from the HI-STORM 100 System FSAR, as amended by LAR 1014-1. An associated burnup of 40,000 MWD/MTU and a cooling time of 13 years was used for the BPRA. This burnup and cooling time bounds the current inventory of BPRAs at DCPP. DCPP has stopped using BPRAs and TPDs. Therefore, the number of these devices in the SFP is not increasing. However, for conservatism, it was assumed that all overpacks were filled with design-basis BPRAs. In the calculation of the dose rate from the ISFSI storage pads, the source shown in Table 7.2-7 was decayed (similar to the neutron and gamma source) to credit the additional cooling time arising from the assumption of eight casks per year being loaded and deployed at the ISFSI storage pads.

7.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Loading of spent fuel into the MPC in the transfer cask is performed under water in the SFP cask loading pit, which prevents the spread of effluent radioactivity during fuel loading. The MPC is sealed and vacuum dried within the FHB/AB allowing the liquid and gaseous waste released from the MPC during the draining and vacuum drying to be processed by the appropriate DCPP systems. Therefore, no airborne releases to the environment from the spent nuclear fuel assemblies are expected to occur during loading and handling operations.

The MPC, which provides the confinement boundary for the HI-STORM 100 System, is a welded pressure vessel and has no bolted closure or mechanical seals. Chapter 3 of the HI-STORM 100 System FSAR demonstrates that all confinement boundary components are maintained within Code-allowable stress limits under all design-basis normal, off-normal, and accident conditions. The all-welded construction of the MPC in conjunction with the extensive inspections and testing performed during closing operations ensures that no release of radioactive effluents will occur from the HI-STORM 100 System. The above discussion notwithstanding, an analysis has been performed to calculate the dose to an individual at the Diablo Canyon site boundary due to an effluent release based on SAR Section 10.2 limit for leakage of 5.0×10^{-6} atm-cm^3/sec under the conditions of the helium leak rate test. This calculation is based on the guidance of NUREG-1536 (Reference 5), ISG-5 (Reference 6) and ISG-11 (Reference 7), as applicable, and is discussed in Sections 7.5 (for normal conditions), Section 8.1.3 (for off-normal conditions), and Section 8.2.7 (for accident conditions).

7.2.2.1 External Contamination Control

The external surface of the MPC is protected from contamination by preventing it from coming into contact with the SFP water. Prior to submergence in the SFP, an inflatable seal is installed at the top of the annulus formed between the MPC shell and the transfer cask cavity. This annulus is filled with clean, demineralized water (borated as required by the Diablo Canyon ISFSI TS), and the seal is inflated. An annulus water overpressurization system is used to maintain the water behind the inflated seal at a slight positive pressure. This system, in the unlikely event of a leak in the inflated seal, will preclude the entry of contaminated water into the annulus. These steps ensure that the MPC surface is free of contamination that could become airborne during storage. Additionally, following fuel-loading operations and removal from the SFP, the MPC lid, the upper end of the MPC shell, and the exterior surfaces of the transfer cask are decontaminated, to the extent practicable, and then surveyed for any remaining, loose surface contamination.

7.2.2.2 Confinement Vessel Releasable Source Term

The inventory for isotopes other than ^{60}Co is calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system, as described in Chapter 5 of the HI-STORM 100 System FSAR. The isotopic inventory for PWR fuel in the MPC-32 was based on the design-basis fuel assembly with a burnup of 55,000 MWD/MTU, 5-years cooling time, and an enrichment of 4.0 wt percent ^{235}U. These assumed burnup and cooling times were chosen to conservatively bound the actual burnup and cooling times for all spent fuel currently at the DCPP site. This burnup is different from that used for the direct radiation source, because unlike the direct radiation source, where the dose rate decreases as the burnup and cooling time increase, the dose rate from effluent release is primarily driven by burnup and is not significantly affected by cooling time.

The enrichment chosen for the confinement evaluation, 4.0 wt percent ^{235}U, is a conservatively low enrichment for the burnup of 55,000 MWD/MTU. The dose to all organs, with the exception of the lung, and the whole body either increases or remains constant with decreasing enrichment. Therefore, a lower enrichment is generally conservative. The dose rate to the lung increases less than 5 percent for a 1 percent increase in enrichment. Section 7.5 presents the offsite dose due to a non-mechanistic normal effluent release. In that section, the dose rate to the lung is bounded by the dose rate to the bone and therefore the slight increase in dose rate for the lung that would be expected from a higher enrichment is not considered.

The 55,000 MWD/MTU burnup bounds the allowable burnups for the MPC-32 as specified in the Diablo Canyon ISFSI TS and SAR Section 10.2. This burnup, though, does not bound all the allowable burnups for the MPC-24 or MPC-24E. However, the reduced fuel contained in an MPC-24 versus an MPC-32 offsets the slight increase in isotopic inventory associated with the slightly higher allowable burnups in the MPC-24. Therefore, the confinement analysis in Section 7.5 of an MPC-32 with a burnup of 55,000 MWD/MTU and a cooling time of 5 years is conservative.

All isotopes that contribute greater than 0.1 percent to the total curie inventory for the fuel assembly are considered in the evaluation as fines. This analysis also includes those actinides that contribute greater than 0.01 percent to the total curie inventory as the dose conversion factors for these isotopes are in general, greater than other isotopes (for example, isotopes of plutonium, americium, curium, and neptunium). A summary of the isotopes available for release is provided in Table 7.2-8.

7.2.2.3 Crud Radionuclides

The majority of the activity associated with crud is due to ^{60}Co (Reference 8). The inventory for ^{60}Co was determined by using the crud surface activity for PWR rods (140×10^{-6} Ci/cm^2) provided in NUREG/CR-6487, multiplied by the surface area per assembly (3×10^5 cm^2 for

PWR fuel, also provided in NUREG/CR-6487). The source terms were then decay corrected 5 years using the basic radioactive decay equation:

$$A(t) = A_0 e^{-\lambda t}$$

where:

$A(t)$	=	activity at time t (Ci)
A_0	=	the initial activity (Ci)
λ	=	the $\ln 2/t_{1/2}$ (where $t_{1/2}$ = 5.272 years for ^{60}Co (Reference 9))
t	=	the time in years (5 years)

A summary of the ^{60}Co inventory available for release is provided in Table 7.2-8.

7.2.3 REFERENCES

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

2. License Amendment Request No. 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. O.W. Hermann, C.V. Parks, SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module, NUREG/CR-0200, Revision 5, (ORNL/NUREG/CSD-2/V2/R5), Oak Ridge National Laboratory, September 1995.

4. O.W. Hermann, R.M. Westfall, ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms, NUREG/CR-0200, Revision 5, (ORNL/NUREG/CSD-2/V2/R5), Oak Ridge National Laboratory, September 1995.

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

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

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

8. B.L. Anderson, B.L. et al., Containment Analysis for Type B Packages Used to Transport Various Contents, NUREG/CR-6487, UCRL-ID-124822, Lawrence Livermore National Laboratory, November 1996.

9. B. Shleien, The Health Physics and Radiological Health Handbook, Scinta Inc., Silver Spring, MD, 1992.

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 STORAGE SYSTEM DESIGN FEATURES

The Diablo Canyon ISFSI is described in Chapters 1, 2, and 4 of this SAR. The HI-STORM 100 System dry storage casks will be stored on up to seven concrete pads. Each pad contains a 4-by-5 array of casks. Therefore, the ISFSI has a total capacity of 140 casks (138 plus 2 spare locations). Figures 2.1-2 and 4.1-1 illustrate the ISFSI location and pad layout. The casks will be positioned on a 17 ft, center-to-center pitch and the pads will be positioned such that the pitch between casks on adjacent pads is also 17 ft. As discussed in Section 4.1, the restricted area fence surrounding the ISFSI will be positioned to ensure that the dose rate at the fence is below 2 mrem/hr, in accordance with the requirements of 10 CFR 20 for unrestricted areas.

The ISFSI and dry storage system has a number of design and administrative control features that ensure that radiation exposures are ALARA.

- There are no radioactive systems at the ISFSI storage pads other than the overpacks containing MPCs.
- The fuel is stored dry inside the MPC, so that no radioactive liquid is available for leakage.
- The MPCs will be loaded, welded, and the upper lid decontaminated in the DCPP FHB/AB prior to being moved to the CTF located near the ISFSI storage pads.
- The overpacks are loaded and the lids installed prior to movement from the CTF to the ISFSI pads.
- Fuel is not removed from the MPCs at either the ISFSI storage pads or the CTF. Unloading of the fuel from the MPC, if necessary, would only occur in the SFP in the FHB/AB.
- The MPCs are heavily shielded by the overpack.
- A locked restricted area fence surrounds the ISFSI storage pads to prevent unauthorized access.
- The ISFSI storage area is typically not occupied.
- Lastly, the MPC design includes a 9.5-inch thick steel lid for shielding of workers.

The HI-STORM 100 System FSAR (Reference 1) describes the transfer cask and overpack in detail. The design features of the HI-STORM 100 System and CTF that ensure radiation exposures are ALARA follow:

- The overpack has a large concrete body encased in steel. The concrete is over 2-ft thick and the steel on the inside and outside of the concrete is each more than 0.5-inch thick. The concrete provides both neutron and gamma radiation shielding while the steel provides predominantly gamma radiation shielding.

- The transfer cask uses a heavy bottom shield made of steel and Holtite-A, which decreases the dose rates during transfer operations from the FHB/AB to the CTF.

- The use of the short overpack eliminates the need for the upper vent duct shield inserts during MPC loading operations. This is accomplished by incorporating the upper vent ducts into the lid.

- The CTF positions the overpack below ground during the loading operations. This minimizes the time involved in loading the overpack by significantly reducing the lift height of the transfer cask above the overpack. This contributes to reduced dose rates during loading operations.

- The HI-STORM 100 System and the CTF have been designed for ease of operation to minimize the duration of the operational sequences.

- In order to minimize dose to personnel consistent with the ALARA philosophy, procedures will be reviewed and dry runs will be performed prior to loading the first cask.

7.3.2 SHIELDING

The design of the HI-STORM 100 System, including the transfer cask, as it relates to the shielding evaluation, is described in Section 5.3 of the HI-STORM 100 System FSAR, as amended by LAR 1014-1 (Reference 2). Summary design targets are given in Table 3.4-2. Besides the overpack and transfer cask, no other radiation shielding features are required for the Diablo Canyon ISFSI. However, due to the choice of the ISFSI storage pad location, which is excavated into the side of a hill, there is a partial natural earth berm located around three sides of the ISFSI storage pads. The terrain around the Diablo Canyon ISFSI storage pads is naturally hilly, which will also provide additional radiation shielding. Conservatively, the analysis documented in this SAR does not take credit for any additional radiation shielding, which would be provided by the surrounding terrain. Rather, the calculations conservatively assume that the ISFSI storage pads are located on flat ground. The details of the calculations are described in Sections 7.4 and 7.5.

The HI-STORM 100SA overpack design will be used at the Diablo Canyon ISFSI. The overpack anchorage hardware has no significant impact on the shielding evaluation. Therefore, the shielding analyses and models emulate the HI-STORM 100S overpack and are applicable to the HI-STORM 100SA overpacks used at the Diablo Canyon ISFSI.

7.3.2.1 Surface and One Meter Dose Rates

As described in Section 7.2, the design-basis MPC for the HI-STORM analysis is the MPC-32 with a burnup and cooling time of 32,500 MWD/MTU and 5 years, respectively, for all fuel assemblies in the MPC. The design-basis MPC for the transfer cask analysis is the MPC-24 with a burnup and cooling time of 55,000 MWD/MTU and 12 years, respectively, for all fuel assemblies in the MPC. These MPCs and burnup/cooling time combinations were chosen to bound all models of MPC in each case. Figures 7.3-1 and 7.3-2 show the overpack and the transfer cask with dose rate locations marked. These are the same dose locations for which values were reported in the HI-STORM 100 System FSAR, as amended by LAR 1014-1. Tables 7.3-1 and 7.3-2 present the surface and 1-meter dose rates for the overpack and the transfer cask loaded with the MPC-32 and MPC-24, respectively, and design basis fuel, including BPRAs. The dose from the individual source components (neutron, photon, and cobalt) is explicitly listed. Table 7.3-3 shows the dose rates at the surface and 1 meter from the overpack and transfer cask as a function of different burnup and cooling times. These burnup and cooling times were chosen based on the allowable contents in the Diablo Canyon ISFSI TS and SAR Section 10.2. The results in this table indicate that the dose rates for the design basis burnup and cooling time are bounding.

7.3.2.2 Dose Versus Distance

The dose rate versus distance from both an overpack and the Diablo Canyon ISFSI were calculated using the Monte Carlo N-Particle (MCNP) transport code (Reference 3). Figure 7.3-3 provides a pictorial representation of the ISFSI with all seven storage pads completely filled with loaded overpacks. The cooling time of the fuel assemblies assumed in the shielding analysis is superimposed on the cask locations in Figure 7.3-3. Based on the storage capacity of the ISFSI (138 plus 2 spare locations), it is not practical to try to model the entire ISFSI in MCNP or any other computer code. Therefore, a methodology similar to that described in Section 5.4 of the HI-STORM 100 System FSAR and LAR 1014-1 was used in the calculation of the dose rate versus distance from the ISFSI. The dose rate versus distance was calculated first for a single overpack. Then numerous MCNP calculations, using relatively small models, were performed to develop ratios for the dose rate contribution from casks situated behind other casks. These ratios were used in conjunction with the dose rate versus distance from a single overpack to estimate the dose rate from the entire ISFSI storage area.

The dose rate from the radiation source was separated into two components. For the purposes of this discussion, the first is referred to as the top-dose. This is the dose rate from radiation that leaves the top of the overpacks. The second component is referred to as side-dose. This is the dose rate from radiation that leaves the sides of the overpacks. In both cases, top-dose and side-dose, in-air scattering of radiation (skyshine) were accounted for in the dose calculations.

In calculating the dose rate from the entire ISFSI storage area, the cask array geometry impacted each of the dose components (top and side) in a different fashion. The total top-dose rate was a summation of the top-dose rates from all 140 casks where the actual distance from the dose location to the individual cask was accounted for.

The total side-dose rate was a summation of the side-dose rates from all 140 casks where the distances within the facility and the self-shielding of one row of casks to another row were accounted for. Since the side-dose rate is from particles leaving the side of the overpack, this dose contribution is greatly reduced if the cask is situated behind another cask. The front cask blocks some, but not all of the radiation from the back cask from reaching the site-boundary. The fraction of radiation blocked was therefore calculated with MCNP, as mentioned above, and used in the determination of the total side-dose.

Dose locations along the long side of the cask array are facing 28 casks directly, that is, without being shielded by other casks. Dose locations along the short side of the array only face five casks directly. Dose rates at dose points along the long side of the array will, therefore, always be higher than dose rates at dose points along the short side of the array. As a bounding approach, all dose rates from the ISFSI storage area reported in this chapter are calculated perpendicular to the long side of the array, regardless of the actual orientation of the dose location relative to the cask array. The results of the dose rate calculations are discussed in Sections 7.4 and 7.5.

As mentioned earlier, the models assumed a flat terrain surrounding the overpack and the ISFSI storage area. The MCNP models consisted of the overpack surrounded by 1,050 meters of air in the radial direction and 700 meters of air in altitude. The cask was assumed to be sitting on an infinite slab of soil. The dose rate versus distance from a single overpack was calculated for the top and side of the overpack separately. Table 7.3-4 shows the dose rate versus distance from a single overpack for the design basis burnup and cooling time. The dose rate due to radiation exiting the top and radiation exiting the side of the overpack are explicitly listed in addition to the total dose rate. Figure 7.3-4 shows the total dose rate versus distance from a single overpack for the design basis burnup of 32,500 MWD/MTU and cooling times of 5 and 20 years.

7.3.2.3 ISFSI Loading Plan

As mentioned in Section 7.2, it was assumed for the purpose of the dose rate analysis that eight overpacks will be loaded per year every year until the ISFSI storage pads are completely filled. Credit for source-strength reduction was taken for the additional cooling time that occurs as a result of this loading plan. At a rate of 8 casks per year, it will take 17.5 years to fill the ISFSI to capacity for a total minimum cooling time after core discharge of 22.5 years for the first casks deployed. However, the oldest fuel in the casks in the ISFSI was conservatively assumed to be 20 years old. No credit was taken for additional cooling from 20 to 22.5 years. Note that this approach also conservatively assumes that all fuel is loaded in the HI-STORM 100 System casks at 5-years cooling time, which is the shortest cooling time allowed by the Diablo Canyon ISFSI TS and SAR Section 10.2. Since the fuel in the casks on the ISFSI pads will have different cooling times after the ISFSI is filled, the position of the casks relative to the dose locations is important.

Section 4.1 states that up to 7 ISFSI pads will be constructed and each pad will contain a 4-by-5 array of casks. The pads will be constructed beginning at the east end of the ISFSI and progressing west, as needed. This loading plan was credited in the shielding analysis. However, it was conservatively assumed that the casks with the "youngest" fuel were positioned on the pads closest to the dose locations. Figure 7.3-3 shows the ISFSI in its final configuration after all seven storage pads have been filled. The age of the fuel in the casks assumed for the analysis is shown in the center of the circle representing a cask. Since it is assumed that 8 casks are loaded per year and credit is taken for the additional cooling time up to 20 years, the age of the fuel in the casks on Pad 1 (the first pad to be used) is assumed to be 20 years. The age of the fuel in the casks on the last pad loaded, Pad 7, is assumed to be 5 to 7 years. Since the highest dose rate from the ISFSI will occur after the ISFSI is completely loaded, this was the only configuration analyzed. As discussed earlier, the dose rate was conservatively calculated perpendicular to the long side of the ISFSI. However, because of the loading pattern of the casks, the location of highest dose rate is not in the center of the ISFSI. Calculations determined that the highest dose rate occurs at approximately the center of Pad 6. Therefore, the dose versus distance calculations from the ISFSI were conservatively performed for distances perpendicular to the center of Pad 6.

7.3.3 VENTILATION

10 CFR 72.122(h)(3) requires that ventilation systems and offgas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal and off-normal conditions. However, as discussed in Section 3.3.1.2 of this SAR, the HI-STORM 100 System is designed to prevent the release of radioactive materials and gases during normal and off-normal conditions. Thus, there are no offgas systems required once the spent fuel is enclosed in the welded MPCs.

Nonetheless, Section 7.5 provides an evaluation of the offsite dose consequences from the hypothetical leakage of all loaded MPC-32s in the ISFSI under normal and off-normal conditions. 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 Section 8.2.7.

7.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Permanent area radiation and airborne radioactivity monitors are not needed at the Diablo Canyon ISFSI since the storage system is passive. Temporary, hand-held radiation protection instruments and self-reading dosimeters will be used during transfer operations at the CTF and routine maintenance at the ISFSI storage area. Thermoluminescent dosimeters will be used to monitor, record, and trend area doses at appropriate intervals in all four directions around the ISFSI restricted area fence. Neutron radiation detection devices may also be used if deemed necessary by the DCPP radiation protection organization.

During fuel loading, existing SFP monitors will monitor for any releases of airborne radioactivity. These monitors are designed to automatically change the building ventilation exhaust system from normal to emergency mode upon detection of radiation levels above preset alarm levels. An area radiation monitoring system is provided for personnel protection and general surveillance of the SFP area (Reference 4, Section 11.4.2.3). Continuous monitoring, recorded readouts, and high radiation level alarms are available in the control room, plus local audible and visual indicators will be in place to alert personnel of high radiation conditions during fuel movement in the FHB/AB. In addition to the monitoring equipment, PG&E will provide radiation protection coverage with hand-held radiation protection instruments and self-reading dosimetry for fuel movement evolutions, which is standard practice for these activities.

7.3.5 REFERENCES

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

2. License Amendment Request No. 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. J.F. Briesmeister, Ed., MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A., Los Alamos National Laboratory, LA-12625-M (1993).

4. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update, Revision 14, November 2001.

7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENTS

The results presented in this section are based on the analysis of the overpack and the transfer cask using design basis fuel, including BPRAs (bounding nonfuel hardware). The discussion in Section 7.2 states that the transfer cask was analyzed with the MPC-24 and the overpack was analyzed with the MPC-32 because these were the bounding MPCs for those overpacks. Consistent with that approach, the analysis presented in this section assumed the transfer cask was loaded with an MPC-24 with a design basis burnup and cooling time of 55,000 MWD/MTU and 12 years, respectively. This analysis also conservatively assumed that the overpack was loaded with an MPC-32 with a design basis burnup and cooling time of 32,500 MWD/MTU and 5 years, respectively.

Table 7.4-1 provides the estimated occupational exposures to DCPP personnel during the following phases of ISFSI operation:

(1) Loading of fuel into the MPC in the transfer cask.

(2) Decontamination of the transfer cask and MPC preparation for storage.

(3) Transport of the transfer cask from the FHB/AB to the CTF adjacent to the ISFSI storage area.

(4) Transfer of the MPC from the transfer cask to the overpack at the CTF.

(5) Closing of the overpack and emplacement on the ISFSI pad.

Table 7.4-2 provides the estimated occupational exposures during the unloading of an overpack (the reversal of the steps listed above). In Tables 7.4-1 and 7.4-2, the total duration of the operation is shown, as well as the time the personnel will be located in the higher dose rate areas. Therefore, total dose for each operation is a product of the number of personnel, dose at location, and time in dose field.

The list of operation steps is provided in Tables 7.4-1 and 7.4-2. Numerous operations have been lumped together for ease of presentation. The duration of the operation and the time the personnel will be located in the higher dose rate areas are based on industry experience with the Holtec HI-STAR and HI-STORM casks and casks from other vendors. The dose rates used for this analysis are conservatively estimated using design-basis fuel. Diablo Canyon radiation protection personnel will assure that the appropriate radiation monitoring is performed and that all operations are performed in a manner consistent with ALARA.

The results presented in Tables 7.4-1 and 7.4-2 are conservatively estimated. By the time the Diablo Canyon ISFSI begins operation, other utilities will have loaded numerous overpacks using the transfer cask and a CTF. Based on the experience to be gained and the lessons to be learned, it is expected that the dose rates from loading an overpack will be somewhat less than those listed here (that is, fewer activities and shorter durations).

Table 7.4-3 provides the estimated annual occupational exposure as a result of daily ISFSI walkdowns, occasional maintenance repairs, and construction of additional ISFSI pads. The dose associated with the clearing of debris from a blocked ventilation duct is presented in Sections 8.1.4 and 8.2.15.

The daily walkdown of the ISFSI requires a person to walk the full length of the ISFSI outside each pad of casks and between each row of casks. This walkdown is to look for obstructions that may be blocking the air vents of the overpack. It was assumed, based on a walking speed of 2 miles/hour, that it would take a person 20 minutes to perform the walk-down at the completion of the ISFSI when all pads are filled with overpacks. This results in a total occupancy time of 122 hours per year. The dose rates shown in Table 7.4-3 for the walk-downs are conservatively based on the 1-meter dose rates, times 4 casks.

The doses for the repair operations assume 1 repair operation per month of 1-hour duration with 2 people performing the operation. The dose rates were conservatively calculated inside an infinite array of casks.

The dose during construction of additional storage pads was calculated for the construction of Pad 7. It was assumed that the previous six pads were completely filled. Doses estimated for the construction of Pad 7 bound the construction of any other pad. The dose rate was conservatively estimated at the center of Pad 7 with no credit for temporary shielding. It was assumed that construction would take 3 months at 40 hours per week in the dose field. The number of personnel was assumed to be 15.

Table 7.4-4 presents the dose rate at the assumed location for the restricted area fence, the makeup water facility (the nearest normally occupied location), and the power plant. The occupancy time was assumed to be 2,080 hours, which is the equivalent of a 40-hour workweek for 52 weeks per year. Also, the dose rates at these locations were conservatively calculated perpendicular to the long side of the storage array. This table demonstrates that the dose rate at the restricted area fence for the assumed location will be below 2 mrem/hr. This table also demonstrates that the dose rates in the normally-occupied locations, due to the ISFSI, are well below the 10 CFR 20 limits for monitored radiation workers. Table 7.4-4 indicates that workers at the makeup water facility may have to become monitored workers as the storage pad approaches the full capacity. Compliance with 10 CFR 20 for these and other workers will be assured via personnel dose monitoring in accordance with the DCPP Radiation Protection Program.

The dose rates presented in Tables 7.4-3 and 7.4-4 demonstrate that the estimated occupational exposures from the Diablo Canyon ISFSI meet the regulatory requirements of 10 CFR 20. The actual doses from the ISFSI are expected to be considerably less than the conservatively estimated values in Tables 7.4-3 and 7.4-4.

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. 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.

7.6 HEALTH PHYSICS PROGRAM

7.6.1 ORGANIZATION

The health physics program, which is described in the DCPP FSAR Update, Section 12.3, is considered sufficient for ISFSI activities. The Manager, Radiation Protection, is responsible for health physics activities related to ISFSI operations for the life of the facility, including all decontamination and decommissioning activities. The radiation protection manager is independent of the operations manager.

7.6.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

DCPP health physics program objectives, equipment, instrumentation, and facilities are described in the DCPP FSAR Update, Section 12.3.2. Except for access control to the ISFSI, many of the same facilities identified in the DCPP FSAR Update will be used for ISFSI operations and surveys. Once the storage site is operational, entrance to and work within the ISFSI protected area will be controlled by radiation protection and security personnel. Radiation work permits will be required in accordance with applicable DCPP procedures.

Available equipment and instrumentation includes personal monitoring equipment, portable radiation measuring instruments, portable air sampling equipment, facilities for internal radiation monitoring, count room equipment, personnel protective equipment, and decontamination equipment and facilities.

7.6.3 POLICIES AND PROCEDURES

The health physics program is carried out in accordance with PG&E program directives, administrative procedures, and working level procedures, which will be revised as needed to address ISFSI operations prior to operation of the ISFSI. The revised procedures will help to maintain exposure ALARA to personnel consistent with operating the ISFSI in a safe, reliable, and efficient manner and will ensure compliance with all applicable regulations and PG&E policies pertaining to radiation protection and release of radioactive materials.

The operation and use of radiation monitoring instrumentation at the Diablo Canyon ISFSI, including personnel monitoring equipment and measurement and sampling techniques, will be described in written procedures.

7.7 ENVIRONMENTAL MONITORING PROGRAM

The DCPP radiological environmental monitoring program will also be used for the Diablo Canyon ISFSI. This program will be augmented to include additional thermoluminescent dosimeters. Since there are no effluents from the ISFSI, there will be no additional radiological effluent monitoring.

TABLE 7.2-1

CALCULATED HI-STORM PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 32,500 MWD/MTU

Lower Energy	Upper Energy	5-Year Cooling		7-Year Cooling		9-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	1.47E+15	2.56E+15	1.17E+15	2.04E+15	1.02E+15	1.77E+15
7.0E-01	1.0	4.49E+14	5.28E+14	2.40E+14	2.83E+14	1.35E+14	1.59E+14
1.0	1.5	1.07E+14	8.53E+13	6.85E+13	5.48E+13	4.96E+13	3.97E+13
1.5	2.0	7.51E+12	4.29E+12	3.63E+12	2.07E+12	2.48E+12	1.42E+12
2.0	2.5	6.42E+12	2.86E+12	1.23E+12	5.46E+11	2.49E+11	1.11E+11
2.5	3.0	2.38E+11	8.67E+10	6.08E+10	2.21E+10	1.58E+10	5.73E+09
Totals		2.04E+15	3.18E+15	1.49E+15	2.38E+15	1.20E+15	1.97E+15
Lower Energy	**Upper Energy**	**11-Year Cooling**		**13-Year Cooling**		**15-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	9.20E+14	1.60E+15	8.53E+14	1.48E+15	8.02E+14	1.39E+15
7.0E-01	1.0	7.99E+13	9.40E+13	5.06E+13	5.95E+13	3.44E+13	4.05E+13
1.0	1.5	3.86E+13	3.08E+13	3.12E+13	2.50E+13	2.59E+13	2.07E+13
1.5	2.0	1.99E+12	1.14E+12	1.69E+12	9.67E+11	1.46E+12	8.36E+11
2.0	2.5	5.75E+10	2.55E+10	1.81E+10	8.05E+09	9.47E+09	4.21E+09
2.5	3.0	4.29E+09	1.56E+09	1.37E+09	4.99E+08	6.27E+08	2.28E+08
Totals		1.04E+15	1.73E+15	9.36E+14	1.57E+15	8.63E+14	1.46E+15

TABLE 7.2-2

CALCULATED HI-TRAC PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 55,000 MWD/MTU

Lower Energy	Upper Energy	12-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	1.48E+15	2.58E+15
7.0E-01	1.0	1.30E+14	1.52E+14
1.0	1.5	7.07E+13	5.65E+13
1.5	2.0	3.64E+12	2.08E+12
2.0	2.5	4.08E+10	1.81E+10
2.5	3.0	4.01E+09	1.46E+09
Totals		1.69E+15	2.79E+15

TABLE 7.2-3

CALCULATED HI-STORM PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 32,500 MWD/MTU

Lower Energy (MeV)	Upper Energy (MeV)	5-Year Cooling (Neutrons/s)	7-Year Cooling (Neutrons/s)	9-Year Cooling (Neutrons/s)	11-Year Cooling (Neutrons/s)	13-Year Cooling (Neutrons/s)	15-Year Cooling (Neutrons/s)
1.0E-01	4.0E-01	6.35E+06	5.89E+06	5.46E+06	5.07E+06	4.70E+06	4.36E+06
4.0E-01	9.0E-01	3.24E+07	3.01E+07	2.79E+07	2.59E+07	2.40E+07	2.23E+07
9.0E-01	1.4	2.98E+07	2.76E+07	2.56E+07	2.38E+07	2.21E+07	2.05E+07
1.4	1.85	2.20E+07	2.04E+07	1.90E+07	1.76E+07	1.64E+07	1.53E+07
1.85	3.0	3.90E+07	3.63E+07	3.38E+07	3.15E+07	2.94E+07	2.74E+07
3.0	6.43	3.52E+07	3.27E+07	3.04E+07	2.83E+07	2.63E+07	2.44E+07
6.43	20.0	3.11E+06	2.88E+06	2.67E+06	2.48E+06	2.30E+06	2.13E+06
Total		1.68E+08	1.56E+08	1.45E+08	1.35E+08	1.25E+08	1.16E+08

TABLE 7.2-4

CALCULATED HI-TRAC PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 55,000 MWD/MTU

Lower Energy (MeV)	**Upper Energy (MeV)**	**12-Year Cooling (Neutrons/s)**
1.0E-01	4.0E-01	2.31E+07
4.0E-01	9.0E-01	1.18E+08
9.0E-01	1.4	1.08E+08
1.4	1.85	7.97E+07
1.85	3.0	1.41E+08
3.0	6.43	1.28E+08
6.43	20.0	1.13E+07
Total		6.09E+08

TABLE 7.2-5

CALCULATED HI-STORM ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 32,500 MWD/MTU

Location	5-Year Cooling (curies)	7-Year Cooling (curies)	9-Year Cooling (curies)	11-Year Cooling (curies)	13-Year Cooling (curies)	15-Year Cooling (curies)
Lower End Fitting	139.25	106.90	82.30	63.19	48.62	37.27
Gas Plenum Springs	10.62	8.16	6.28	4.82	3.71	2.84
Gas Plenum Spacer	6.10	4.68	3.60	2.77	2.13	1.63
Incore Grid Spacers	360.64	276.85	213.15	163.66	125.93	96.53
Upper End Fitting	68.30	52.43	40.37	31.00	23.85	18.28

TABLE 7.2-6

CALCULATED HI-TRAC ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 55,000 MWD/MTU

Location	**12-Year Cooling (curies)**
Lower End Fitting	75.11
Gas Plenum Springs	5.73
Gas Plenum Spacer	3.29
Incore Grid Spacers	194.53
Upper End Fitting	36.84

TABLE 7.2-7

CALCULATED ^{60}Co SOURCE PER BPRA PER ASSEMBLY
FOR A BURNUP OF 40,000 MWD/MTU AND A COOLING TIME OF 13 YEARS

Region	**Curies Co-60**
Upper End Fitting	12.1
Gas Plenum Spacer	1.8
Gas Plenum Springs	3.3
Incore	313.8

TABLE 7.2-8

ISOTOPE INVENTORY AND RELEASE FRACTION
Ci/ASSEMBLY

Nuclide	PWR Fuel Ci/Assembly	Release Fraction[a]
Gases		
^{3}H	2.97E+02	0.30
^{129}I	2.64E-02	0.30
^{85}Kr	4.82E+03	0.30
Crud		
^{60}Co	2.18E+01	0.15 normal/offnormal 1.0 accident
Volatiles		
^{90}Sr	5.10E+04	2.0E-04
^{106}Ru	1.44E+04	2.0E-04
^{134}Cs	3.01E+04	2.0E-04
^{137}Cs	7.82E+04	2.0E-04
Fines		
^{241}Pu	7.75E+04	3.0E-05
^{90}Y	5.10E+04	3.0E-05
^{147}Pm	2.57E+04	3.0E-05
^{154}Eu	4.51E+03	3.0E-05
^{244}Cm	5.57E+03	3.0E-05
^{238}Pu	3.76E+03	3.0E-05
^{125}Sb	1.99E+03	3.0E-05
^{155}Eu	1.28E+03	3.0E-05
^{241}Am	8.06E+02	3.0E-05
^{240}Pu	3.65E+02	3.0E-05
^{239}Pu	1.99E+02	3.0E-05
^{137m}Ba	7.38E+04	3.0E-05
^{106}Rh	1.44E+04	3.0E-05

TABLE 7.2-8

Nuclide	PWR Fuel Ci/Assembly	Release Fraction[a]
^{144}Ce	8.14E+03	3.0E-05
^{144}Pr	8.14E+03	3.0E-05
^{125m}Te	4.86E+02	3.0E-05

[a] B.L. Anderson, et al., Containment Analysis for Type B Packages Used to Transport Various Contents, NUREG/CR-6487, UCRL-ID-124822, Lawrence Livermore National Laboratory, November 1996.

Note: The isotopes, which contribute greater than 0.1 percent to the total curie inventory for the fuel assembly, are considered in the evaluation as fines. The analysis also includes actinides, which contribute greater than 0.01 percent to the total curie inventory for the fuel assembly. This is in accordance with ISG-5.

TABLE 7.3-1

SURFACE AND 1 METER DOSE RATES FOR THE OVERPACK WITH AN MPC-32 32,500 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Surface Dose Rate				
1	6.8	17.0	2.9	26.7
2	33.9	0.1	0.8	34.8
3	9.9	18.3	2.6	30.8
4	1.6	1.5	0.9	3.9
4a	2.5	13.4	13.0	28.9
1 Meter Dose Rate				
1	4.9	5.3	0.3	10.5
2	17.0	0.6	0.4	18.0
3	4.6	5.0	0.4	10.0
4	0.4	0.5	0.4	1.3

Notes:

- Refer to Figure 7.3-1 for the dose locations.
- Gammas generated by neutron capture are included with fuel gammas.
- Gammas from BPRAs are included in the fuel gammas for the portion of the BPRA in the active fuel zone and included in the ^{60}Co gammas for the portion of the BPRA above the active fuel zone.
- Dose location 4a is located directly above the top duct. This is a very localized area of increased dose. Dose location 4a was only calculated at the surface of the lid.

TABLE 7.3-2

SURFACE AND 1 METER DOSE RATES FOR THE TRANSFER CASK WITH THE MPC-24 55,000 MWD/MTU AND 12-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Surface Dose Rate					
1	1.9	23.3	36.5	133.6	195.3
2	26.2	62.6	0.0	88.3	177.1
3	0.3	4.2	25.2	261.4	291.1
4	9.9	2.9	115.1	261.4	389.3
4 (outer)	1.1	2.0	14.5	5.5	23.1
5 (pool)	12.9	1.2	155.6	982.0	1151.7
5 (pool with temp. shield)	7.2	11.5	110.0	58.5	187.2
1 Meter Dose Rate					
1	3.6	8.5	4.1	20.1	36.3
2	12.1	20.1	0.3	31.7	64.2
3	2.0	5.4	4.0	16.8	28.2
4	2.7	0.7	27.9	26.9	58.2

Notes:
- Refer to Figure 7.3-2 for the dose locations.
- Gammas from BPRAs are included in the fuel gammas for the portion of the BPRA in the active fuel zone and included in the ^{60}Co gammas for the portion of the BPRA above the active fuel zone.
- Dose location 4 (outer) is the radial segment at dose location 4, which is 18-24 inches from the center of the overpack.
- Dose rates are based on no water within the MPC. During the MPC lid welding the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- The dose rate below the pool lid is calculated in the center of the lid. The HI-STORM 100 System FSAR demonstrates that this dose rate will be greatly reduced at the outer edge of the overpack. In addition, during transfer operations from the FHB/AB to the CTF, additional shielding will be installed on the bottom of the transfer cask. The dose rates on the pool lid with temporary shielding were calculated assuming the temporary shielding was only 2.5 inches of Holtite-A.

TABLE 7.3-3

SURFACE AND 1 METER DOSE RATE AT THE MIDPLANE OF THE OVERPACK AND THE TRANSFER CASK AS A FUNCTION OF DIFFERENT BURNUP AND COOLING TIMES

MPC	32	32	24	24	24
Burnup (MWD/MTU)	32,500	45,000	41,500	50,000	55,000
Cooling time (years)	5	8	5	8	12
Initial enrichment (wt.% ^{235}U)	2.9	4.0	3.4	4.0	4.0
Overpack					
Surface dose rate (mrem/hr)	34.8	22.4	32.8	19.8	15.5
1 meter dose rate (mrem/hr)	18.0	11.5	16.8	10.1	7.7
Transfer Cask					
Surface dose rate (mrem/hr)	132.0	151.4	153.6	159.3	177.0
1 meter dose rate (mrem/hr)	54.9	57.8	61.5	59.5	64.1

TABLE 7.3-4

DOSE RATE VERSUS DISTANCE FROM A SINGLE OVERPACK
WITH THE MPC-32
32,500 MWD/MTU AND 5-YEAR COOLING

Distance		mrem/hr		
m	ft	Side-dose rate	Top-dose rate	Total dose rate
12.19	40.00	1.02E+00	1.97E-03	1.02E+00
18.29	60.00	5.00E-01	1.42E-03	5.01E-01
24.38	80.00	2.88E-01	1.05E-03	2.89E-01
30.48	100.00	1.86E-01	8.31E-04	1.87E-01
45.72	150.00	8.04E-02	4.68E-04	8.09E-02
50.00	164.04	6.64E-02	4.14E-04	6.68E-02
60.96	200.00	4.27E-02	2.99E-04	4.30E-02
91.44	300.00	1.69E-02	1.40E-04	1.71E-02
100.00	328.08	1.37E-02	1.16E-04	1.38E-02
121.92	400.00	8.33E-03	7.71E-05	8.41E-03
150.00	492.13	4.76E-03	4.56E-05	4.81E-03
200.00	656.17	2.12E-03	2.02E-05	2.14E-03
250.00	820.21	1.05E-03	9.63E-06	1.06E-03
300.00	984.25	5.80E-04	4.88E-06	5.85E-04
350.00	1148.29	3.19E-04	2.61E-06	3.22E-04
400.00	1312.34	1.84E-04	1.45E-06	1.86E-04
450.00	1476.38	1.11E-04	8.08E-07	1.11E-04
500.00	1640.42	6.88E-05	4.61E-07	6.93E-05
550.00	1804.46	4.45E-05	2.63E-07	4.47E-05
600.00	1968.50	2.99E-05	1.58E-07	3.01E-05
650.00	2132.55	1.90E-05	1.01E-07	1.91E-05
700.00	2296.59	1.25E-05	5.96E-08	1.26E-05
750.00	2460.63	8.24E-06	3.71E-08	8.28E-06
800.00	2624.67	5.42E-06	2.48E-08	5.45E-06
850.00	2788.71	3.90E-06	1.61E-08	3.91E-06
900.00	2952.76	2.66E-06	1.06E-08	2.67E-06

TABLE 7.4-1

OCCUPATIONAL EXPOSURE DURING OVERPACK LOADING OPERATIONS

	Operation	Duration of Operation (hours)	Time in Dose Field (minutes)	Dose Rate at Location (mrem/hr)	Number of Personnel	Total Dose (mrem)
1	Insert MPC into transfer cask	2	120	0	3	0
2	Place transfer cask in cask washdown area	2	120	0	4	0
3	Attach impact limiter	0.5	10	0	2	0
4	Fill annulus	1	60	0	2	0
5	Fill MPC with water	2	20	0	2	0
6	Move transfer cask over the spent fuel pool	2	60	2	3	6
7	Place transfer cask in the spent fuel pool	1.5	60	2	3	6
8	Load fuel assemblies into MPC	8	480	2	2	32
9	Perform fuel assembly identification check	2	60	2	2	4
10	Install MPC lid and lid retention system	2	120	2	2	8
11	Remove transfer cask from spent fuel pool and washdown external portion	1.5	45	23	2	34.5
12	Place transfer cask in the cask washdown area	1	5	46	2	7.7
13	Install the seismic restraints	0.5	30	46	2	46
14	Disconnect lid retention system and lift yoke	1	60	24	2	48
15	Fill water jacket with water	1	5	38	2	6.3
16	Perform initial decontamination	2	120	23	3	138
17	Install temporary shield ring	0.5	15	39	2	19.5
18	Lower MPC water level	0.5	15	29	2	14.5
19	Install automated welding system	1.5	30	29	2	29
20	Perform MPC lid welding and NDE	12	210	29	1	101.5
21	Hydro test MPC	1	30	29	2	29
22	Perform leakage testing	1	30	29	1	14.5
23	Blowdown MPC	3	15	29	2	14.5
24	Perform MPC moisture removal	20	60	29	2	58
25	Perform helium backfill	3	30	29	2	29
26	Install/weld vent and drain cover plates	2	120	29	2	116

TABLE 7.4-1

	Operation	Duration of Operation (hours)	Time in Dose Field (minutes)	Dose Rate at Location (mrem/hr)	Number of Personnel	Total Dose (mrem)
27	NDE vent and drain cover plate welds	0.5	30	29	1	14.5
28	Leak test cover plates	0.5	30	29	1	14.5
29	Install/weld closure ring	2	60	29	2	58
30	NDE closure ring welds	1	60	29	2	58
31	Remove automated welding system	1	15	29	2	14.5
32	Drain annulus	1	5	36	1	3
33	Remove temporary shield ring	0.5	15	39	2	19.5
34	Decontaminate MPC lid and transfer cask	2	120	23	3	138
35	Install transfer cask top lid	0.5	30	33	2	33
36	Install lift cleats	0.5	30	45	2	45
37	Remove impact limiter	0.5	5	49	2	8.2
38	Raise transfer cask and decontaminate bottom	2	90	36	2	108
39	Remove seismic restraints	0.5	30	46	2	46
40	Remove transfer cask from cask washdown area	1	60	18	2	36
41	Install bottom shield	1	60	49	2	98
42	Place transfer cask in transport frame	1	45	23	2	34.5
43	Downend transfer cask in transport frame	1	10	23	2	7.7
44	Perform transfer cask surveys	1	30	61	2	61
45	Transport transfer cask to cask transporter	1	60	23	2	46
46	Attach transfer cask to the cask transporter	0.5	30	23	2	23
47	Transport transfer cask to the cask transfer facility	1.5	90	12	2	36
48	Prep overpack for receiving MPC	3	180	0	3	0
49	Upend transfer cask in the transport frame	1.5	30	23	2	23
50	Remove transfer cask from the transport frame	0.5	30	23	2	23
51	Remove bottom shield	0.5	30	49	2	49
52	Mate transfer cask with overpack	0.5	30	36	2	36
53	Secure transfer cask to cask transporter	1	60	23	2	46
54	Attach MPC downloader	0.5	15	45	2	22.5
55	Remove transfer cask pool lid	0.5	30	49	2	49

Table 7.4-1 Sheet 3 of 3

	Operation	Duration of Operation (hours)	Time in Dose Field (minutes)	Dose Rate at Location (mrem/hr)	Number of Personnel	Total Dose (mrem)
56	Transfer MPC into overpack	1	60	23	2	46
57	Disconnect MPC downloader slings	0.5	30	2	2	2
58	Remove transfer cask from mating device	0.5	30	2	2	2
59	Remove MPC downloader slings/cleats	0.5	15	45	2	22.5
60	Remove mating device	0.5	30	6	2	6
61	Install overpack lid	1	15	6	2	3.0
62	Raise overpack to full up position	0.5	30	18	2	18
63	Attach overpack lift bracket	0.5	30	5	2	5
64	Transport overpack to ISFSI	1	60	18	2	36
65	Attach overpack to ISFSI pad	1	60	16	2	32
66	Perform post loading testing	3	30	18	2	18
Total dose during loading operations						2.1 rem

TABLE 7.4-2

OCCUPATIONAL EXPOSURE DURING OVERPACK UNLOADING OPERATIONS

	Operation	Duration of Operation (hours)	Time in Dose Field (minutes)	Dose Rate at Location (mrem/hr)	Number of Personnel	Total Dose (mrem)
1	Recover overpack from ISFSI pad	6	360	16	2	192
2	Attach overpack lift bracket	0.5	30	5	2	5
3	Transport overpack to CTF	1	60	18	2	36
4	Lower overpack to full down position	0.5	30	18	2	18
5	Remove overpack lid	1	15	6	2	3
6	Attach mating device	0.5	30	6	2	6
7	Install MPC downloader slings/cleats	0.5	30	45	2	45
8	Install transfer cask into mating device	0.5	30	2	2	2
9	Attach MPC downloader	0.5	30	2	2	2
10	Secure transfer cask to cask transporter	1	60	2	2	4
11	Transfer MPC into transfer cask	1	60	23	2	46
12	Install transfer cask pool lid	0.5	30	49	2	49
13	Disconnect MPC downloader	0.5	15	45	2	22.5
14	Disconnect transfer cask from overpack	0.5	30	36	2	36
15	Install bottom shield	1	60	49	2	98
16	Place transfer cask in the transport frame	1	45	23	2	34.5
17	Downend transfer cask in the transport frame	1	10	23	2	7.7
18	Attach transfer cask to the cask transporter	0.5	30	23	2	23
19	Transport transfer cask to the fuel building	1.5	90	12	2	36
20	Transport transfer cask into the fuel building	1	60	23	2	46
21	Upend transfer cask in transport frame	1.5	30	23	2	23
22	Remove transfer cask from transport frame	1	60	23	2	46
23	Remove bottom shield	0.5	30	49	2	49
24	Place transfer cask in cask washdown area	0.5	5	46	2	7.7
25	Install the seismic restraints	0.5	30	46	2	46
26	Remove lift cleats	0.5	30	45	2	45

TABLE 7.4-2

	Operation	Duration of Operation (hours)	Time in Dose Field (minutes)	Dose Rate at Location (mrem/hr)	Number of Personnel	Total Dose (mrem)
27	Remove transfer cask top lid	0.5	30	37	2	37
28	Attach impact limiter	0.5	10	49	2	16.3
29	Install temporary shield ring	0.5	15	39	2	19.5
30	Fill annulus	1	6	53	2	10.6
31	Install weld removal system	1.5	30	29	2	29
32	Core drill vent and drain cover plates	1	60	29	2	58
33	Perform MPC gas sampling	2	10	29	2	9.7
34	Perform MPC helium cooldown	20	60	29	2	58
35	Flood MPC	3	15	29	2	14.5
36	Remove MPC lid weld	12	210	29	1	101.5
37	Remove weld removal system	1	15	29	2	14.5
38	Remove temporary shield ring	0.5	15	39	2	19.5
39	Drain water jacket	1	5	49	2	8.2
40	Install lid retention system and lift yoke	0.5	20	65	2	43.3
41	Remove the seismic restraints	0.5	30	65	2	65
42	Raise transfer cask to spent fuel pool level	1	5	61	2	10.2
43	Place transfer cask over spent fuel pool	1.5	60	2	2	4
44	Place transfer cask in spent fuel pool	1.5	45	2	2	3
45	Remove lid retention system bolts	2	120	2	2	8
46	Remove MPC lid	0.5	30	2	2	2
47	Unload fuel assemblies from transfer cask	8	480	2	2	32
48	Remove transfer cask from spent fuel pool	1.5	45	2	3	4.5
49	Place transfer cask in cask washdown area	1	6	2	2	0.4
50	Drain MPC and transfer cask annulus	6	30	2	2	2
51	Remove MPC from transfer cask	2	20	2	2	1.3
52	Decontaminate MPC and transfer cask	8	240	2	2	16
Total dose during unloading operations						1.5 rem

TABLE 7.4-3

OCCUPATIONAL EXPOSURES ASSOCIATED WITH ISFSI ACTIVITIES

Activity	Dose Rate (mrem/hr)	Duration (hours/year)	Number of Personnel	Total Annual Dose (rem)
Completion of ISFSI (140 casks)				
ISFSI walk-downs	15.0	122	1	1.8
Overpack repairs	65.0	12	2	1.6
Construction of last storage pad	6.0	480	15	43.2

TABLE 7.4-4

OCCUPATIONAL EXPOSURES AT ONSITE LOCATIONS

Location	Dose Rate (mrem/hr)	Occupancy (hours/year)	Annual Dose (rem)
Completion of ISFSI (140 casks)			
Restricted area fence (100 ft / 30.5 m)	1.9	N/A	N/A
Make-up water facility (223 ft/ 68.0 m)	5.1E-01	2,080	1.1
Auxiliary building wall (798 ft/ 243.2 m)	2.2E-02	2,080	0.05

TABLE 7.5-1

NORMAL OPERATION DOSE RATES AND ANNUAL DOSES AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM DIRECT RADIATION FROM THE 140 CASKS AT THE DIABLO CANYON ISFSI

Location	Dose Rate (mrem/hr)	Occupancy (hours/year)	Annual Dose (mrem)
Site Boundary (1,400 ft / 427 m)	2.7E-03	2,080	5.6
Nearest Resident (1.5 mi / 7,920 ft / 2414 m)	4.0E-08	8,760	3.5E-04

TABLE 7.5-2

NORMAL OPERATION ANNUAL DOSES AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM AN ASSUMED EFFLUENT RELEASE FROM THE 140 CASKS AT THE DIABLO CANYON ISFSI

	Annual Dose[(a)] (mrem)
Site Boundary (1,400 ft / 427 m)	
Whole body ADE[(b)]	0.064
Thyroid ADE	0.010
Critical Organ ADE (Max)	0.35
Nearest Resident (1.5 mi / 7,920 ft / 2,414 m)	
Whole body ADE	0.27
Thyroid ADE	0.043
Critical Organ ADE (Max)	1.46

[(a)] The effluent release dose for the nearest resident is conservatively chosen to be the site boundary dose, adjusted for full-time occupancy (8,760/2,080). This is conservative since the χ/Q for the nearest resident would be less than that used for the site boundary. The occupancy time for the site boundary is 2,080 hours and the occupancy time for the nearest resident is 8,760 hours.

[(b)] ADE is annual dose equivalent.

TABLE 7.5-3

DOSE RATES AT THE SITE BOUNDARY
FROM OVERPACK LOADING OPERATIONS

Condition	Dose Rate (mrem/hr)	Event Duration (hours)	Loadings per year	Annual Dose (mrem)
MPC in transfer cask	2.0E-03[a]	7.5	8	1.20E-01
MPC in overpack without a lid	9.0E-04	1.5	8	1.1E-02
Total				13.1E-02

[a] The dose rate for the transfer cask was calculated by scaling the highest dose rate on the surface of the transfer cask (not including the pool lid) by the ratio of the highest contact dose rate to distance dose rate calculated for the overpack. Specifically, 389.3 mrem/hr (Table 7.3-2) was multiplied by 1.8E-04 (Table 7.3-4 (400 m))/34.8 (Table 7.3-1).

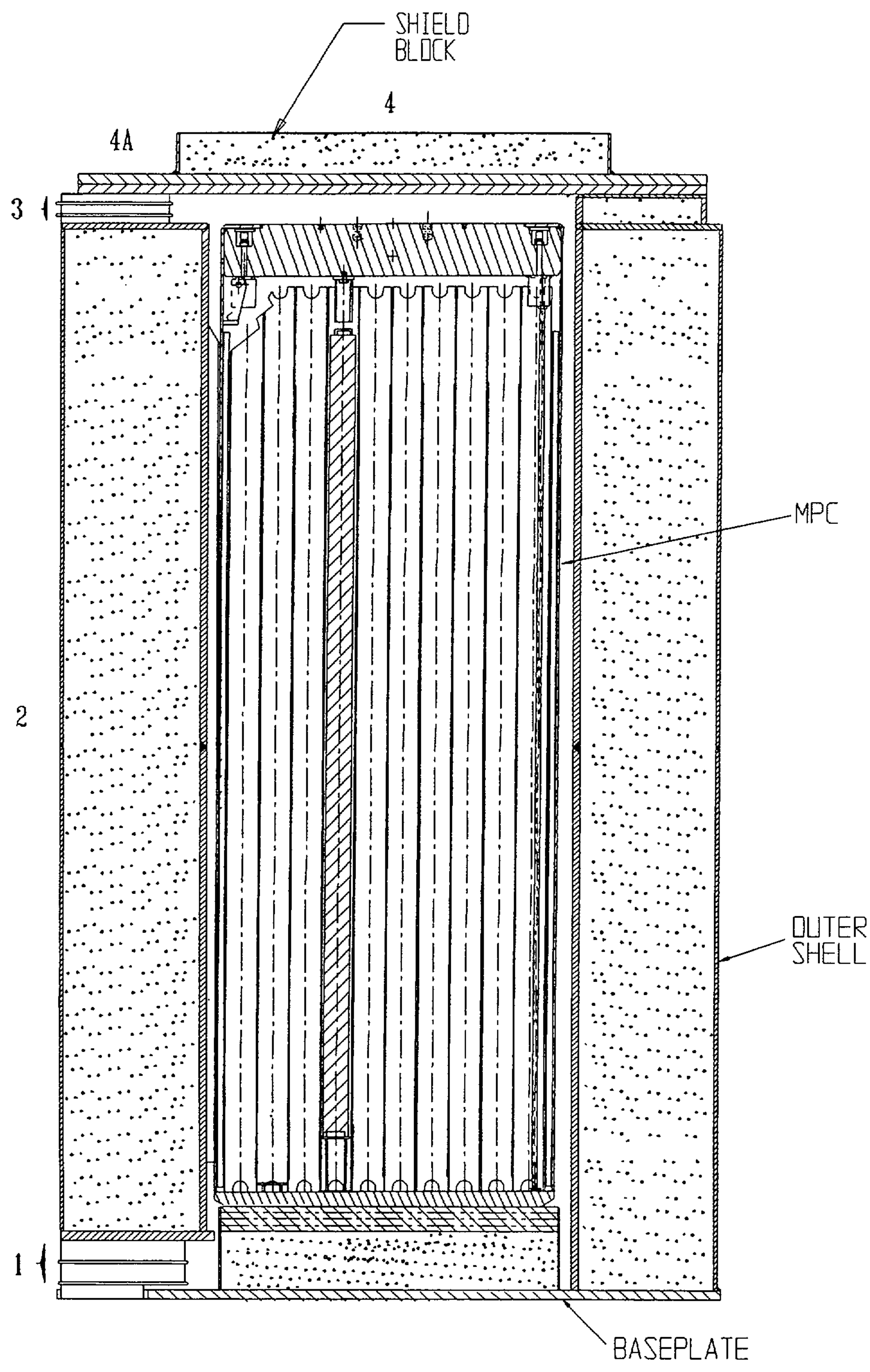
TABLE 7.5-4

TOTAL ANNUAL OFFSITE COLLECTIVE DOSE (MREM) AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM THE DIABLO CANYON ISFSI CONTAINING 140 CASKS FOR NORMAL OPERATION

Organ	Effluent Release	Direct Radiation	Overpack Loading Operations	Other Uranium Fuel Cycle Operations[a]	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	25
Thyroid ADE	0.010	5.6	13.1E-02	1.260E-01	75
Critical organ ADE (Max)	0.35	5.6	13.1E-02	5.590E-02	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	25
Thyroid ADE	0.043	3.5E-04	13.1E-02	1.260E-01	75
Critical organ ADE (Max)	1.46	3.5E-04	13.1E-02	5.590E-02	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.

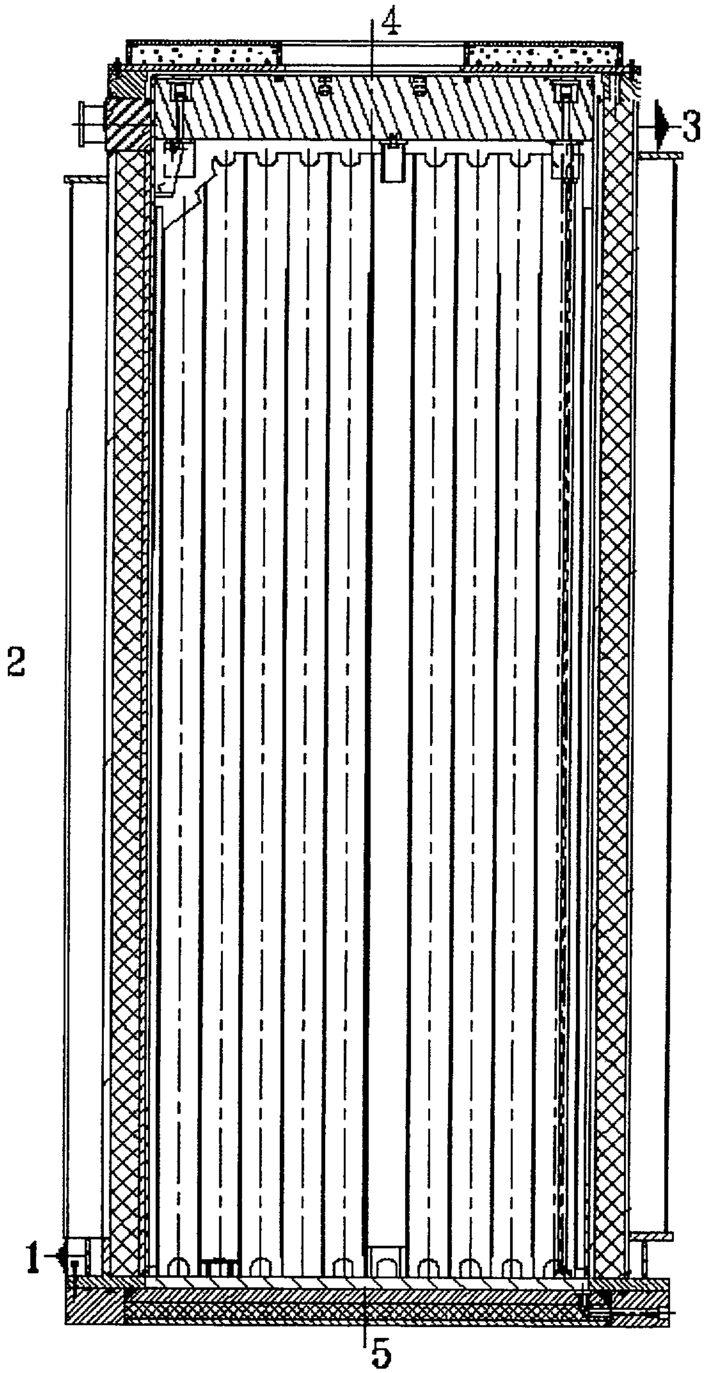
[b] ADE is annual dose equivalent.



SAFETY ANALYSIS REPORT

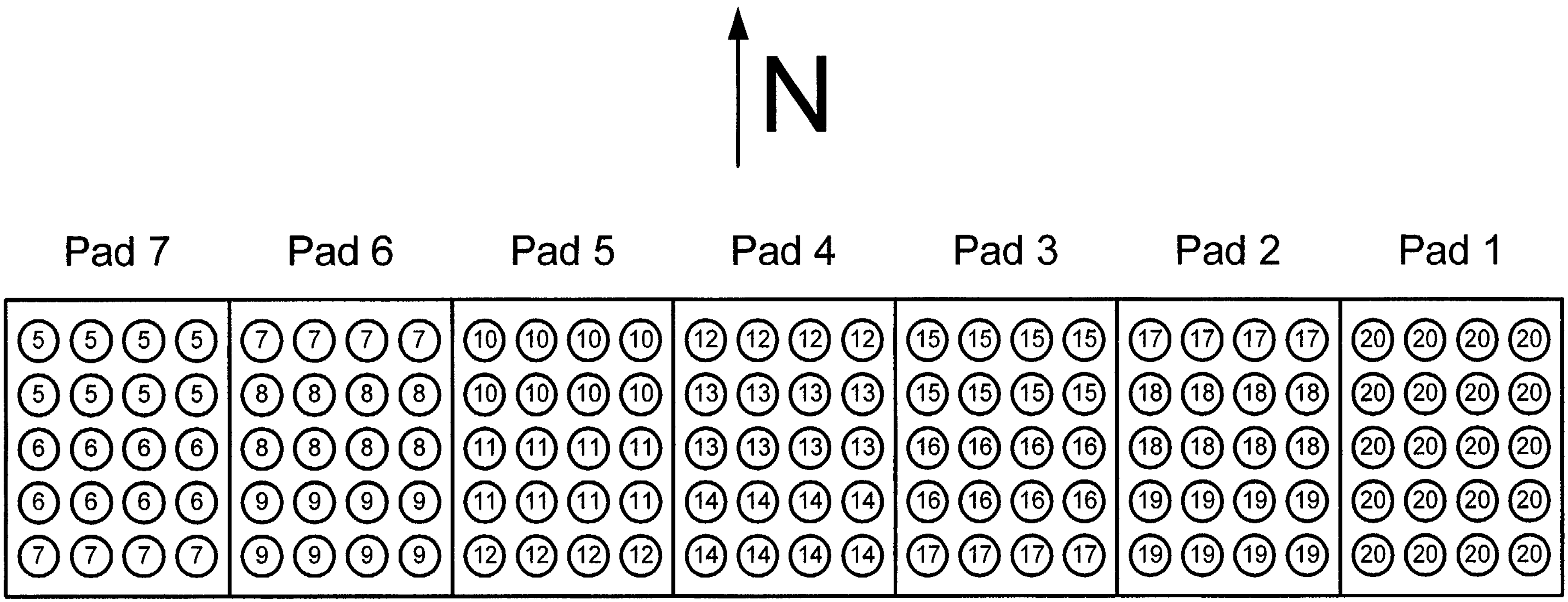
DIABLO CANYON ISFSI

FIGURE 7.3-1
CROSS SECTION ELEVATION VIEW OF THE HI-STORM 100S OVERPACK WITH DOSE POINT LOCATIONS



Note:
Numbers refer to SAR Table 7.3-2

SAFETY ANALYSIS REPORT
DIABLO CANYON ISFSI
FIGURE 7.3-2 CROSS SECTION ELEVATION VIEW OF TYPICAL HI-TRAC TRANSFER CASK WITH DOSE POINT LOCATIONS



○ = Cask showing cooling time, in years, of the fuel assemblies

SAFETY ANALYSIS REPORT

DIABLO CANYON ISFSI

FIGURE 7.3-3
A PLAN VIEW OF THE ISFSI AT THE COMPLETION OF LOADING OPERATIONS

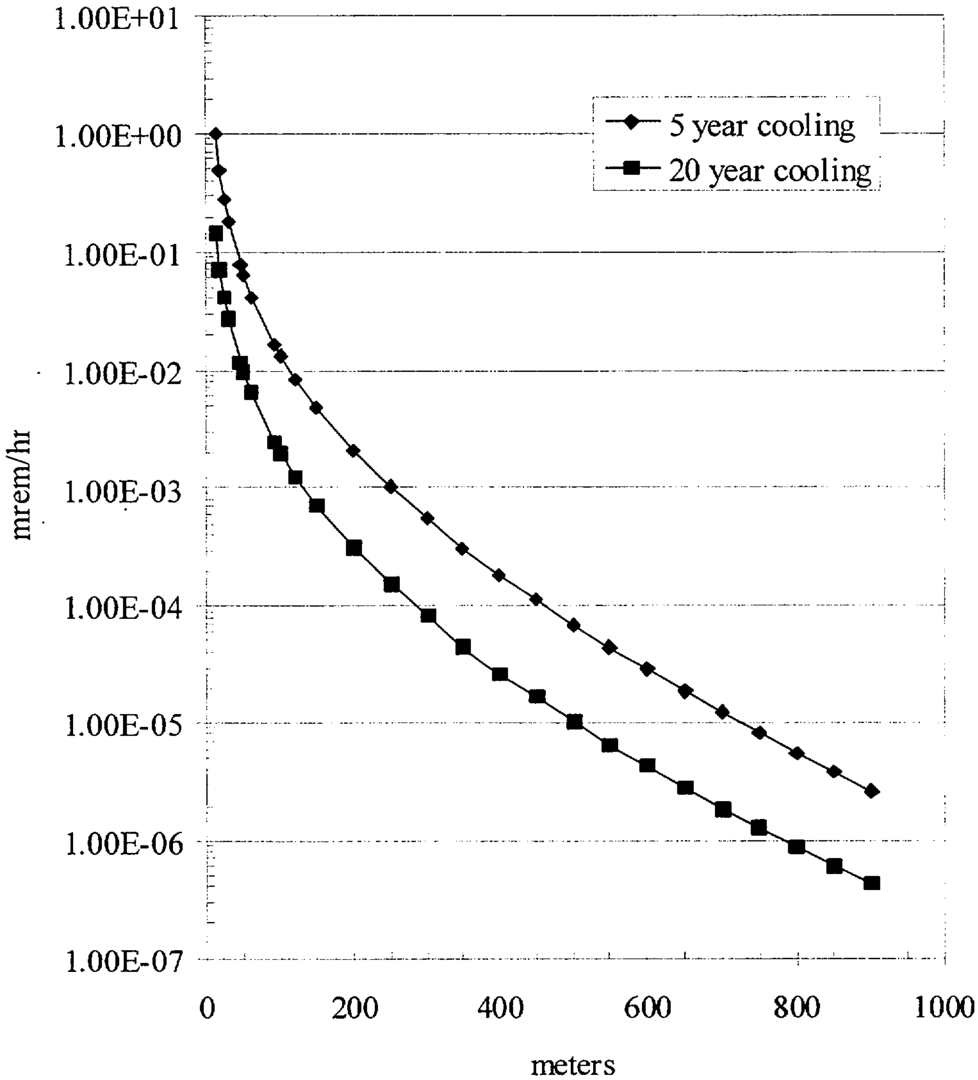


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FIGURE 7.3-4
DOSE RATE VERSUS DISTANCE FROM A SINGLE HI-STORM 100SA OVERPACK CONTAINING AN MPC-32 LOADED WITH FUEL OF BURNUP 32,500 MWD/MTU

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CHAPTER 8

ACCIDENT ANALYSES

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ACCIDENT ANALYSES

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FIGURES

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CHAPTER 8

ACCIDENT ANALYSES

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:

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

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)

$\triangle T$ = 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. 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.

8.1.8 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, 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 2, 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 while the transporter is on bedrock or bedrock overlain by surficial soil (where the seismic event has been judged as not being credible), 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.

Methodology- Structural Analysis

The analysis 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.

<u>Acceptance Criteria</u>

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 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.

<u>Methodology</u>

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.

Acceptance Criteria:

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.

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 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 concrète 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 0.2×10^{6} psi to 2.0×10^{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. 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.

Acceptance Criteria

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

Assumptions

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

Key Input Data

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

Results

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.

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 1814 kg automobile, the 344.7 kg 500-kV insulator string, and the 4 kg, 1-inch-diameter steel rod were determined to be the bounding large, intermediate, and small missiles, respectively.

The bounding large and intermediate 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. 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) 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 intermediate missile entering the transfer cask through the hole in the top lid and impacting the MPC lid. If the intermediate missile directly impacts the MPC, it will not penetrate the 9-1/2-inch-thick, stainless-steel lid. The safety factor against failure of the peripheral MPC lid-to-shell weld is 7.1.

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 and was evaluated for its effect after impacting the outer shell (including penetration) and the top lid of the overpack at design-basis velocity. 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 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 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 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 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 energy received is less than the design basis for the event, or (b) a risk assessment will be performed using Regulatory Guide 1.91 risk acceptance criteria. 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:

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

8.2.18 REFERENCES

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8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

The Diablo Canyon ISFSI storage site is located as shown in Figure 2.1-2 of this SAR. The protected area for the storage site is shown in Figure 4.1-1 and described in Section 4.1. The nearest road to the ISFSI site is a DCPP access road that is used to access various onsite facilities. Use of this road is controlled by PG&E. This access road will be relocated to the north side of the raw water storage reservoir. As concluded in Section 2.2, there are no credible accident scenarios involving any offsite industrial, transportation, or military facilities in the area around the DCPP site that will have any significant adverse impact on the ISFSI. In addition, there are no potential onsite fires, explosions, or chemical hazards that would have a significant or unacceptable impact on the ISFSI. Site characteristics that affect the safety analysis, and how they have been considered in developing suitable margins of safety for the storage of DCPP's spent fuel, are summarized in Table 8.3-1.