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 control the performance of the operations, including inspection and testing. At a minimum, these procedures 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 *limits to establish and suspend supplemental cooling*. These sequences are controlled by the Diablo Canyon ISFSI TS and Section 10.2.

Several auxiliary components 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 work platform in the Unit 2 cask washdown area (CWA) assists in transfer cask and MPC preparation and closure operations. The work platform is part of the transfer cask seismic restraint system.

All handling of the transfer cask inside the FHB/AB will be made using a single failure proof crane to preclude a vertical cask drop event.

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.

An empty MPC-32 is also verified to have been cleaned, inspected, and is then raised, and inserted into the transfer cask. This insertion activity may take place either prior to entering the FHB/AB or once inside the FHB/AB. Upon completion of the insertion activity alignment marks are verified to ensure correct rotational alignment between the MPC and the transfer cask.

The transfer cask is brought into the FHB/AB through the Unit 2 roll-up door in a vertical orientation on a low-profile transporter (LPT). There is no LPT rail system for Unit 1, thus transfer casks designated for transporting spent fuel from both units enter through the Unit 2 roll-up door. If not previously installed, an empty MPC-32 will be installed when the transfer cask is in the CWA restraint. The LPT is equipped with heavy-duty rollers that engage with a set of temporary tracks that runs from inside the FHB/AB to the access road located outside the Unit 2 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.

After being transported into the FHB/AB, the transfer cask bolted to the LPT is positioned under the single failure proof FHB crane that is configured with a lift yoke. The lift yoke engages the transfer cask lifting trunnions, and the transfer cask is unbolted from the LPT. The transfer cask is then lifted above the LPT as it is moved into the Unit 2 CWA. There is no CWA seismic restraint for Unit 1, thus transfer casks designated for transporting spent fuel from both units are prepared in the Unit 2 CWA. Prior to moving the transfer cask into the CWA, the transfer cask is visually verified to have the bottom lid bolted to the cask. The transfer cask is placed within the CWA seismic restraint and secured. An empty MPC-32 is loaded into the transfer cask if not already loaded prior to entering the FHB/AB. To eliminate buoyancy effects the MPC is first filled with borated water and then the annulus between the transfer cask and the MPC is filled with demineralized water, in accordance with the ISFSI TS and Section 10.2. A seal is then installed in the top part of the annulus to minimize the risk of contaminating the external shell of the MPC.

When the transfer cask is ready for movement into the SFP, with the transfer cask engaged by the FHB crane, the transfer cask is released from the CWA seismic restraint and, along with its MPC, is raised approximately 12 inches above the floor of the FHB/AB (140 ft elevation). For Unit 1 spent fuel loading operations, the transfer cask is moved through the hot machine shop and into the FHB/AB bay area of Unit 1 and positioned adjacent to the Unit 1 SFP. For Unit 2 spent fuel loading operations, the transfer cask is positioned adjacent to the Unit 2 SFP.

The transfer cask annulus overpressure system is connected. The transfer cask is positioned over the cask recess area of the SFP and lowered using the FHB crane on to the SFP platform structure. The SFP cask restraint provides seismic restraint while the transfer cask is on the platform. The annulus over-pressure system 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 platform in the cask recess area of 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 loading of damaged fuel assemblies and fuel debris in the MPC-24E or -24EF, 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 attached, is placed in position in the MPC after the completion of fuel loading, while the transfer cask is in the SFP.

The FHB crane and the lift yoke are reattached, and the transfer cask is raised until the top of the MPC just breaks the SFP water surface. Rinsing of exterior surfaces and disconnecting the annulus pressurization system is performed as the transfer cask continues to emerge from the SFP. The transfer cask is raised completely out of the SFP to clear the SFP wall and lowered to about 12 inches above the floor of the FHB/AB (140 ft elevation). For Unit 1 fuel movement, Radiation Protection will prepare the transfer cask to preclude spreading contamination, prior to moving the transfer cask through the hot shop, to the cask restraint in the Unit 2 CWA. For Unit 2 fuel movement, the transfer cask is moved directly from the Unit 2 SFP to the Unit 2 CWA. The transfer cask is moved to the Unit 2 CWA restraint system and secured. Once the transfer cask is positioned in the CWA, the lift yoke is disconnected and removed from the area. Activities involving decontamination and placement of auxiliary equipment may occur in parallel or in a different sequence based on cask-loading experience at DCPP.

Procedural controls ensure that dilution of the MPC boron concentration will not occur from removal of the HI-TRAC from the spent fuel pool, until water is removed from the MPC in the blowdown process.

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 MPC boron concentration is verified in accordance with the Diablo Canyon ISFSI TS and 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, but maintaining the fuel covered with water to ensure the fuel is not exposed to an oxidizing environment. Oxidation of Boral or Metamic panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during, MPC lid welding operations. In addition, the space below the MPC lid shall be exhausted or purged with inert gas prior to, and during, MPC lid welding operations to provide additional assurance that explosive gas mixtures will not develop in this space. The automated welding system is installed. The MPC-lid welding, including nondestructive examinations, is completed. Once the MPC-lid welding is complete, the MPC is filled with borated water (in accordance with the Diablo Canyon ISFSI TS), vented, and hydrostatically tested.

After an acceptable hydrostatic test has been completed, 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. Using helium during MPC water displacement and moisture removal ensures that there will be no oxidization of the fuel cladding during loading operations (fuel is covered with water prior to blowdown).

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~~*is* accomplished using a ~~vacuum drying process (low burnup [that is, ≤45,000 MWD/MTU] fuel only) or the~~ forced helium dehydration (FHD) system~~ (low or high burnup fuel)~~. ~~During the vacuum drying process, the annular gap between the MPC and the HI-TRAC will be continuously flushed with water.~~ When using the FHD system the annular gap is verified to have no water present.

~~When vacuum drying is used, any water that has not drained from the MPC cavity evaporates as a result of the vacuum. This drying is aided by the temperature increase due to the decay heat of the fuel. To ensure adequate drying the vacuum drying pressure in the MPC must be verified to be at vacuum pressure criteria specified in the DC ISFSI TS. This low vacuum pressure is an indication that the cavity is dry and the moisture level in the MPC is acceptable. Following the successful completion of moisture removal from the MPC using the vacuum drying process, the MPC is backfilled with helium. When using the vacuum drying process for moisture removal, no additional preparation of the MPC cavity is necessary prior to helium backfill operations. The helium backfill system is attached, and the MPC is backfilled with helium to within the required pressure range in accordance with the DC ISFSI TS.~~

When the FHD system is used, any water that has not drained from the MPC cavity is removed through introducing dry gas into the MPC cavity that absorbs the residual moisture in the MPC. This humidified gas exits the MPC and the absorbed water is removed through condensation and/or mechanical drying. During use of the FHD system, the circulated helium is monitored until it meets the dryness criteria of the DC ISFSI TS. Once this is met, the helium pressure in the MPC cavity is adjusted to within the required pressure range in accordance with the DC 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, *if the MPC contains any high burnup (>45,000 MWD/MTU) fuel assemblies, the supplemental cooling system (SCS) is installed and the annulus between the transfer cask and MPC is filled with demineralized water within the time required by the Diablo Canyon ISFSI TS.* ~~t~~*T*he MPC vent and drain port cover plates are *then* installed, welded, inspected, examined, and helium leak tested in accordance with ANSI N14.5-1997. 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 that has references to the design drawings in the HI-STORM 100 System FSAR.

~~Any 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 transfer cask top lid is installed and secured with four bolts.

The lift yoke is re-attached to the transfer cask. The transfer cask is raised and 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 CWA seismic restraint is released and the FHB crane then moves the transfer cask laterally away from the CWA. The transfer cask is positioned on and bolted down to the LPT. If not performed earlier, the transfer cask and LPT are surveyed to ensure that any fixed contamination is within acceptable limits. The loaded transfer cask and LPT are then rolled out of the Unit 2 FHB/AB to an area outside of the FHB/AB where the cask transporter can access the transfer cask.

If necessary, the SCS may be removed from service during the movement of the transfer cask from the CWA restraint to the cask transporter, provided the time limits set forth in the Diablo Canyon ISFSI TS are met.

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 transfer cask. The transporter receives preoperational testing and maintenance and is 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 on the LPT is positioned under the lift beam of the cask transporter and the transfer cask lift links are rigged to the cask. The transporter lift system engages the transfer cask while the transfer cask is unbolted from the LPT. The transporter than raises the transfer cask*,* ~~and~~ it is secured within the transporter for the trip to the CTF*, and if required the SCS returned to service*. The LPT is than rolled out of the way and the transporter transports the transfer cask to the CTF along the approved transportation route as described in Section 4.3.3 and shown in Figure 2.1-2.

The overpack is prepared for loading, which involves general visual inspections and cleaning. Following the visual inspection and cleaning, the overpack is positioned in the CTF by the transporter. In preparation for receiving the loaded MPC, the overpack lid is removed (if previously installed). The mating device is secured to the overpack. To restrain the cask against seismically-induced impact loads on the main shell of the CTF, seismic restraints are installed to transmit the load from the overpack to the CTF shell (Section 3.3.4.2.3).

At the CTF, the transporter positions the transfer cask over the mating device*, the SCS is disconnected, if installed,* and ~~it~~ *the transfer cask* is then secured *to the mating device*. During this connection process, subsequent to MPC transfer, HI-TRAC removal, and HI-STORM closure operation, temporary shielding is provided around the mating device as needed to minimize occupational dose. Use of the temporary shielding during these processes will be administratively controlled. The cask transporter seismic anchor (TSA) restraints connect the cask transporter to the CTF TSA pads. The TSA restraints are described in Section 4.2.1.2 and depicted in plan view in Reference 39 of Section 4.2. The TSAs function to prevent the transporter from seismically interacting with the storage cask while in the CTF during MPC transfer operations. The transfer cask lift links are then disconnected and the MPC lift cleats are installed. 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 bottom lid. The bottom lid is supported by the mating device while the bottom lid bolts are removed. The bottom lid is removed from under the transfer cask.

The 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 lift links are reengaged on the transfer cask and the transporter lift system is engaged. The cask transporter TSA restraints are disconnected and the transfer cask is unfastened and lifted from the mating device and raised from the top of the overpack 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 bottom 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 lifted out of the CTF by the transporter and moved to the ISFSI pad, where it is placed in its designated storage location. During the transporter lifting of the HI-STORM, the probability of an earthquake occurring is so small as to make this event non-credible. Thus, the TSAs do not need to be attached to the transporter during the overpack lifting. Specific steps involved in these operations are described in the Diablo Canyon ISFSI approved procedures.

Prior to the loaded overpack arriving at the ISFSI pad, the designated storage location has been prepared for the cask to be placed on the pad. Specifically, a small number of alignment pins are installed in the anchor stud locations. These alignment pins ensure that the cask is properly located and the holes in the cask bottom flange match with the holes in the ISFSI pad embedment plate. When the cask is properly located and seated, the alignment pins are removed and the 16 anchor studs are threaded into the top of the embedded coupling (see Figure 4.2-2). The studs are pre-tensioned using a stud tensioner and the nuts tightened in a cross-pattern, roughly 180 degrees apart, to avoid uneven loads on the baseplate.

The preload on the cask anchor studs is applied without employing a torque wrench. Therefore, no torque is induced on the embedded anchor rods or compression couplings during the preload operation. A stud tensioner is used to apply preload on the anchor studs using hydraulic pressure to elastically "stretch" the bolt. The nuts are then tightened on the "stretched" stud to maintain the pre-load. This tension is transferred to the cask base/embedment plate interface as a compressive force via the stud nut and compression coupling. There is no significant torque applied on the nuts during tightening (i.e., hand-tightening is adequate).

The cask transporter is disconnected from the overpack and the lift brackets are removed and lid studs installed on the overpack. 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.

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 (Reference 1), is 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.

The Diablo Canyon ISFSI was initially licensed based on HI-STORM 100 CoC, Amendment 1, and used the applicable source terms based on fuel which providing limiting dose rates within the allowed loadable contents for the canister. Additionally, the original Diablo Canyon ISFSI License utilized canisters with an allowed leakage rate from the MPC, and hence a confinement dose analysis was performed to document potential effluent doses from the allowed MPC leakage.

When Diablo Canyon pursued License Amendment 2 (LA 2), to allow loading of high burnup fuel, the dose analyses were reperformed. Although the allowed loading of fuel was based on HI-STORM 100 CoC, Amendment 3, the revised dose analysis was performed using the HI-STORM 100 CoC, Amendment 5, source terms, which results in overstated doses since the Amendment 5 fuel was allowed to be loaded at a higher heat load, and hence higher dose rate. Additionally, as part of the system changes when pursing LA 2, the helium leak testing requirements for the MPC shell welds were revised to require them to meet the "leaktight" criteria of ANSI N14.5-1997. The vent and drain port cover plate welds helium leak testing requirements had been changed to the "leaktight" criteria of ANSI N14.5-1997 in LA 1. Since the lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore in accordance with ISG-18, leakage from this weld is considered non-credible. Since all the closure welds meet a leaktight criteria, the confinement boundary of the subsequently fabricated MPCs can be considered leak tight, and no dose contribution from confinement boundary leakage is required to be considered for the casks loaded to these requirements.

To preserve the previous licensing basis, where the previous analyses have not been superseded by the updated analyses, the new data is provided along side the previous data. When the information is contained in the Tables, the data supporting the current analyses is provided in the Table designated "(a)", and the previous data is in the Table designated "(b)".

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 ~~are~~ *have been* 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 are 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 (Reference 1). Therefore, no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean demineralized 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 *Diablo Canyon and* other nuclear power plants. ISFSI personnel 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 demineralized 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

- Maintaining the transfer cask water jacket filled with water during MPC processing

- Use of temporary portable shielding, as appropriate

- 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 1A, January 2003.

2. Deleted in Revision 2.

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.

4. *10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 3, May 29, 2007.*

5. *10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 5, July 14, 2008.*

6. *Holtec International Report No. HI-2002563, "Dose Evaluation for the ISFSI at Diablo Canyon Power Plant," Revision 8.*

7.2 RADIATION SOURCES

The source terms presented in this section were developed specifically for use in the Diablo Canyon ISFSI shielding analyses, *HI-2002563, Revision 8 (Reference 10)*. Other sections of this FSAR reference dose analyses from the HI-STORM 100 System FSAR (Reference 1) *and HI-STORM 100 System FSAR Revision 7 (Reference 11)*. The source terms used for the dose analyses referenced from the HI-STORM 100 System FSAR 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. *In* ~~T~~*t*he *original analysis,* burnup was assumed to be 32,500 MWD/MTU with an initial cooling time of 5 years. *To allow the HI-STORM 100 system at Diablo Canyon to be loaded with high burnup fuel, the shielding analysis was reperformed in support of License Amendment 2 (LA 2). The burnup assumed was increased to 69,000 MWD/MTU for assemblies of 4.8 wt% U-235 initial enrichment, with 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 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 *original* 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 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*ed* the dose rates from a transfer cask containing an MPC-32.

In the revised analysis for high burnup fuel, the transfer cask shielding analysis used the MPC-24 analysis from the HI-STORM 100 FSAR Revision 7 (Reference 11), which used a burnup of 75,000 MWD/MTU and cooling time of 5 years. This combination provides conservative doses as it exceeds the fuel allowed for loading in the system allowed by Section 10.2. To estimate the dose for an MPC-32, these doses are *multiplied by the ratio of assemblies contained, which provides conservative results since it does not take into consideration the increased self-shielding in the 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 *original* 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)
  - *Neutron Source Assemblies*

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 FSAR 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 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 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 *(a) &(b)* and 7.2-2 *(a) & (b)* 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 *(a) & (b)* and 7.2-4 *(a) & (b)* 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 *(a) & (b)* and 7.2-6 *(a) & (b)* 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 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 100 System FSAR describes the design-basis BPRA, RCCA, and TPD. The results 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 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 . An associated burnup of 40,000 MWD/MTU and a cooling time of 13 years were 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.

Neutron source assemblies (NSAs) are used in reactors for startup. During in-core operations, the stainless steel and inconel portions of the NSAs become activated, producing a significant amount of Co-60. Using the masses of steel and inconel for the NSAs it was determined that the total activation of a primary or secondary source is bound by the total activation of a BPRA (see Table 5.2.31 of Reference 11). Therefore, storage of NSAs is acceptable and a detailed dose rate analysis using the gamma source from activated NSAs is not performed.

Antimony-beryllium sources are used as secondary (regenerative) neutron sources in reactor cores. The Sb-Be source produces neutrons from a gamma-n reaction in the beryllium, where the gamma originates from the decay of neutron-activated antimony. The very short half-life of ^{124}Sb, 60.2 days, however results in a complete decay of the initial amount generated in the reactor within a few years after removal from the reactor. The production of neutrons by the Sb-Be source through regeneration in the MPC is *orders of magnitude lower than the design-basis fuel assemblies. Therefore Sb-Be sources do not contribute to the total neutron source in the MPC.*

Primary neutron sources (californium, americium-beryllium, plutonium-beryllium and polonium-beryllium) are usually placed in the reactor with a source-strength on the order of 5E+08 n/s. This source strength is similar to, but not greater than, the maximum design-basis fuel assembly source strength listed in Tables 5.2.15 and 5.2.16 of Reference 11.

By the time NSAs are stored in the MPC, the primary neutron sources will have been decaying for many years since they were first inserted into the reactor (typically greater than 10 years). For the ^{252}Cf source, with a half-life of 2.64 years, this means a significant reduction in the source intensity; while the ^{210}Po-Be source, with a half-life of 138 days, is virtually eliminated. The ^{238}Pu-Be and ^{241}Am-Be sources, however, have a significantly longer half-life, 87.4 years and 433 years, respectively. As a result, their source intensity does not decrease significantly before storage in the MPC. Since the ^{238}Pu-Be and ^{241}Am-Be sources may have a source intensity similar to a design-basis fuel assembly when they are stored in the MPC, only a single NSA is permitted for storage in the MPC. Since storage of a single NSA would not significantly increase the total neutron source in an MPC, storage of NSAs is acceptable and detailed dose rate analysis of the neutron source from NSAs is not performed.

For ease of implementation, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Section 10.2.

Instrument tube tie rods (ITTRs), which are installed after core discharge and do not contain radioactive materials, may also be stored in the assembly. ITTRs are authorized for unrestricted storage in an MPC.

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 dried within the FHB/AB allowing the liquid and gaseous waste released from the MPC during the draining and 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 the Section 10.2 limit for leakage of 5.0 x 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).

When the dose analysis was updated (Reference 10) to support loading of high-burnup fuel, the criteria for allowed leakage from the MPC was reduced to the leaktight criteria of ANSI N14.5-1997 and as such effluent releases do not need to be considered for casks tested to this criteria. The original effluent analysis is maintained for the 16 casks loaded to the original leakage criteria, however as noted in Reference 10, the off-site dose analysis for the original casks with the reduced source term and effluent release is bounded by the updated dose analysis with only direct dose. Therefore, the values for off-site dose assume all casks are loaded with the higher source term, and do not include a contribution from effluent release.

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 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 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

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1. *PG&E Calculation STA-140 (HI-2002513, Rev. 7), "Diablo Canyon ISFSI Site Boundary Confinement Analysis."*

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. 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 are positioned on a 17 ft, center-to-center pitch and the pads are 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 is 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 are 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 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 cask transporter places the overpack below ground prior to the MPC transfer. 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 MPC transfer 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 (Reference 1). 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 FSAR 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 is 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. *In the original analysis* ~~with~~ a burnup and cooling time of 32,500 MWD/MTU and 5 years, respectively, *was used* for all fuel assemblies in the MPC. *When the analysis was updated for high burnup fuel, a burnup and cooling time of 69,000 MWD/MTU and 5 years, respectively, was used 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 *in the original analysis, and an MPC-32 with a burnup and cooling time of 75,000 MWD/MTU and 5 years in the high burnup analysis*, 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*, as noted in the associated HI-STORM FSAR (References 1 and 6)*. 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. Tables 7.3-1 and 7.3-2*(a) & (b)* 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 7.3-1 was not revised for the new design basis fuel, since it is specifically used in the offsite dose analysis (Reference 5), and is not required for evaluation of on-site doses as discussed in Section 7.4.*~~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 Technical Specifications (TS) and 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 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*s* 7.3-4*(a) and (b)* show~~s~~ 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 are 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 takes 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 Section 10.2. Since the fuel in the casks on the ISFSI pads 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.

When the analysis was updated for implementation of storage of high burnup fuel, all casks were assumed loaded with the new higher source term. No credit was taken for lower design/actual source term from the initial 16 casks.

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, 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 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 are in place to alert personnel of high radiation conditions during fuel movement in the FHB/AB. In addition to the monitoring equipment, radiation protection coverage with hand-held radiation protection instruments and self-reading dosimetry for fuel movement evolutions is provided, 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 1A, January 2003.

2. Deleted in Revision 2.

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.

5. Holtec International Report No. HI-2002563, "Dose Evaluation for the ISFSI at Diablo Canyon Power Plant," Revision ~~6~~*8*.

6. *Final Safety Analysis Report for the HI-STORM 100 Cask System, Holtec International Report No. HI-2002444, Revision 7, August 9, 2008.*

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 *original analysis* 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 (Reference 1).

These values were not revised for the analysis performed in support of loading high burnup fuel (HBF). Since the actual burnup/cooling time allowed to be loaded by Section 10.2 is not appreciably different (burnup < 20% higher for same cooling time) from the values used in the original analysis, and the actual loading experience has shown that actual loading is done for less than 30% of the estimated dose, the following values will continue to be used to estimate the occupational exposure for loading and operations of the ISFSI at Diablo Canyon.

The estimated occupational exposure during overpack loading operations is approximately 2.1 rem. Refer to Holtec Report HI-2002563, Revision ~~6~~ *8* (Reference 1).

The estimated occupational exposure during overpack unloading operations is approximately 1.5 rem (Reference 1).

The list of operation steps is also provided in Reference 1, Appendix K. Numerous operations have been lumped together for ease of presentation. The duration of the operation and the time the personnel are 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 assure that the appropriate radiation monitoring is performed and that all operations are performed in a manner consistent with ALARA.

The occupational exposures during overpack loading and unloading operations are conservatively estimated. *ALARA practices take advantage of experience in loading* ~~By the time the Diablo Canyon ISFSI begins operation, other utilities will have loaded numerous overpacks using the~~ HI-STORM 100 System*s at Diablo Canyon as well as at other utilities*. 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*, strategically placed shielding* and shorter durations).

The estimated total annual per person occupational exposure as a result of daily ISFSI walkdowns, occasional maintenance repairs, and construction of additional ISFSI pads are 1.8 rem, 0.8 rem, and 2.9 rem, respectively (Reference 1). 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. A dose rate of 15 mrem/hr for the walk-downs is 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. A dose rate of 65 mrem/hr for repair operations is conservatively based on 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 and dose rate were assumed to be 15 and 6 mrem/hr, respectively.

The estimated dose rate at the assumed location for the restricted area fence, the makeup water facility (the nearest normally occupied location), and the power plant are 1.9 mrem/hr, 0.51 mrem/hr, and 0.022 mrem/hr, respectively. 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. The dose rate at the restricted area fence for the assumed location will be below 2 mrem/hr. Also, the dose rates in the normally-occupied locations, due to the ISFSI, are well below the 10 CFR 20 limits for monitored radiation workers. The 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 is assured via personnel dose monitoring in accordance with the DCPP Radiation Protection Program (Reference 1).

The dose rates for ISFSI walkdowns, occasional maintenance repairs, and construction of additional ISFSI pads and at the restricted area fence, the makeup water facility, and the power plant 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 above conservatively estimated values.

7.4.1 REFERENCES

1. Holtec International Report No. HI-2002563, “Dose Evaluation for the ISFSI at Diablo Canyon Power Plant,” Revision ~~6~~*8*.

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 (Reference 8). 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 these activities 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. Effluent doses for an accident condition are discussed in Section 8.2.7.

As noted in Section 7.2.2, when the dose analysis was updated to support the loading of high burnup fuel at the Diablo Canyon ISFSI, the need to consider effluent releases under conditions of normal storage was eliminated. Therefore, the remainder of this section is presented to document the historical licensing basis of the initial 16 casks only.

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, *for the original dose analysis,* 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 occurs outside the FHB/AB at the CTF. As a result, the impact of this operation on the offsite dose was considered. The~~re are~~ only ~~two~~ condition~~s~~ that need to be considered in this analysis~~.~~ ~~The first~~ is the condition of the MPC inside the transfer cask *when outside of the FHB/AB*. ~~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~~ *this* analys*i*~~e~~s.

7.5.4 TOTAL OFFSITE COLLECTIVE DOSE

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

8. Holtec International Report No. HI-2002563, "Dose Evaluation for the ISFSI at Diablo Canyon Power Plant," Revision ~~6~~*8*.

TABLE 7.2-1(a)

CALCULATED HI-STORM PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 69,000 MWD/MTU

Lower Energy	**Upper Energy**	**5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	3.26E+15	5.67E+15
7.0E-01	1.0	1.23E+15	1.44E+15
1.0	1.5	2.69E+14	2.15E+14
1.5	2.0	1.41E+13	8.08E+12
2.0	2.5	7.56E+12	3.36E+12
2.5	3.0	3.56E+11	1.29E+11
Totals		4.78E+15	7.34E+15

These values obtained from Reference 11, Table 5.2.4

TABLE 7.2-1*(b)*

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(a)

CALCULATED HI-TRAC PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 75,000 MWD/MTU

Lower Energy	**Upper Energy**	**5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	3.55E+15	6.17E+15
7.0E-01	1.0	1.36E+15	1.60E+15
1.0	1.5	2.94E+14	2.35E+14
1.5	2.0	1.50E+13	8.59E+12
2.0	2.5	7.63E+12	3.39E+12
2.5	3.0	3.72E+11	1.35E+11
Totals		5.23E+15	8.02E+15

These values obtained from Reference 11, Table 5.2.5

TABLE 7.2-2*(b)*

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(a)

CALCULATED HI-STORM PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 69,000 MWD/MTU

Lower Energy (MeV)	Upper Energy (MeV)	5-Year Cooling (Neutrons/s)
1.0E-01	4.0E-01	5.31E+07
4.0E-01	9.0E-01	2.71E+08
9.0E-01	1.4	2.48E+08
1.4	1.85	1.82E+08
1.85	3.0	3.21E+08
3.0	6.43	2.92E+08
6.43	20.0	2.60E+07
Total		1.39E+09

These values obtained from Reference 11, Table 5.2.15

TABLE 7.2-3*(b)*

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(a)

CALCULATED HI-TRAC PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 75,000 MWD/MTU

Lower Energy (MeV)	**Upper Energy (MeV)**	**5-Year Cooling (Neutrons/s)**
1.0E-01	4.0E-01	6.82E+07
4.0E-01	9.0E-01	3.48E+08
9.0E-01	1.4	3.18E+08
1.4	1.85	2.34E+08
1.85	3.0	4.11E+08
3.0	6.43	3.75E+08
6.43	20.0	3.34E+07
Total		1.79E+09

These values obtained from Reference 11, Table 5.2.16

TABLE 7.2-4*(b)*

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(a)

CALCULATED HI-STORM ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 69,000 MWD/MTU

Location	5-Year Cooling (curies)
Lower End Fitting	208.12
Gas Plenum Springs	15.88
Gas Plenum Spacer	9.11
Incore Grid Spacers	539.00
Upper End Fitting	102.08

These values obtained from Reference 11, Table 5.2.11

TABLE 7.2-5*(b)*

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(a)

CALCULATED HI-TRAC ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 75,000 MWD/MTU

Location	***5-Year Cooling (curies)***
Lower End Fitting	*219.47*
Gas Plenum Springs	*16.74*
Gas Plenum Spacer	*9.61*
Incore Grid Spacers	*568.40*
Upper End Fitting	*107.65*

These values obtained from Reference 11, Table 5.2.12

TABLE 7.2-6*(b)*

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 7.3-1

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.
- These values are taken from page A-7 of Holtec Report HI-2002563, Rev. ~~6~~8.

TABLE 7.3-2*(a)*

SURFACE AND 1 METER DOSE RATE~~S~~ *ESTIMATES* FOR THE TRANSFER CASK WITH THE MPC-~~24~~*32* ~~5~~*7*5,000 MWD/MTU AND ~~12~~*5*-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)	*Totals with BPRAs (mrem/hr)*
Surface Dose Rate						
1	*8.4*	*82.5*	*134.2*	*554.5*	*779.6*	*780.6*
2	*151.1*	*244.3*	*0.01*	*383.9*	*779.3*	*800.5*
3	*1.9*	*8.7*	*83.0*	*884.9*	*978.5*	*1004.8*
4	*55.4*	*11.2*	*454.2*	*1023.9*	*1544.8*	*1698.7*
4 (outer)	*6.5*	*8.0*	*56.4*	*21.5*	*92.3*	*111.3*
5 (pool)	*73.0*	*4.9*	*606.1*	*3844.7*	*4528.7*	*4539.0*
~~5 (pool with temp. shield)~~	~~7.2~~	~~11.5~~	~~110.0~~	~~58.5~~	~~187.2~~	
1 Meter Dose Rate						
1	*19.9*	*32.9*	*17.2*	*91.3*	*161.3*	*164.0*
2	*67.3*	*79.2*	*0.7*	*131.0*	*278.1*	*287.6*
3	*7.5*	*18.6*	*16.8*	*81.4*	*124.3*	*130.9*
4	*15.4*	*2.7*	*109.4*	*105.5*	*232.9*	*269.8*
5 (pool)(est)	*28.8*	*0.9*	*293.0*	*665.9*	*1603.9*	*1611.5*

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 bottom 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.
- These values are ~~taken from~~ *based on* Appendix ~~J~~ *P*of Holtec Report HI-2002563, Rev. ~~6~~*8*.. *As noted in the report to obtain values for an MPC-32, the MPC-24 values of HI-STORM*

FSAR Rev. 7 Table 5.1.8, are ratioed by the number of fuel assemblies (i.e. 32/24) to obtain the values in this Table.

- *1-meter dose rates for point 5 are estimated based on applying the ratio of dose rates, surface and 1-meter, for point 5(transfer) to the surface dose rate for 5 (pool), as only the pool lid is used.*
- Values in tables are nominal based on design basis fuel.
- Accident analysis values for complete loss of water in water jacket are provided in Section 8.2.11.3.

TABLE 7.3-2*(b)*

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	38.0	63.0	0.0	160.0	261.0
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 bottom 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.
- These values are taken from Appendix J of Holtec Report HI-2002563, Rev. 6.
- Values in tables are nominal based on design basis fuel.
- Accident analysis values for complete loss of water in water jacket are provided in Section 8.2.11.3.

TABLE 7.3-3

TOTAL SURFACE AND 1 METER DOSE RATES FOR THE TRANSFER CASK WITH VARIOUS MPCs

Dose Point Location	MPC-24 55,000 MWD/MTU 12-yr. Cooling (mrem/hr)	MPC-24 75,000 MWD/MTU 5-yr. Cooling (mrem/hr)	MPC-32 75,000 MWD/MTU 5-yr. Cooling (mrem/hr)
Surface Dose Rate			
1	195.3	585.4	780.6
2	261.0	600.4	800.5
3	291.1	753.6	1004.8
4	389.3	1274.0	1698.7
4 (outer)	23.1	83.5	111.3
5 (pool)	1151.7	3404.2	4539.0
1 Meter Dose Rate			
1	36.3	123.0	164.0
2	64.2	215.7	287.6
3	28.2	98.2	130.9
4	58.2	202.3	269.8

Notes:

- Refer to Figure 7.3-2 for the dose locations.
- 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 bottom 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.
- These values are taken from Holtec Report HI-2002563, Rev. 5.
- Values in tables are nominal based on design basis fuel.
- Accident analysis values for complete loss of water in water jacket are provided in Section 8.2.11.3.

~~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~~ [(a)]

~~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~~ [(b)]					
~~Surface dose rate (mrem/hr)~~	~~132.0~~	~~151.4~~	~~153.6~~	~~159.3~~	~~261.0~~
~~1 meter dose rate (mrem/hr)~~	~~54.9~~	~~57.8~~	~~61.5~~	~~59.5~~	~~64.1~~

[(a)] ~~Values in tables are nominal values.~~
[(b)] ~~Only the surface dose rate for the bounding MPC-24 (55,000 MWD/MTU) has been increased to account for a localized deformation in the water jacket.~~

TABLE 7.3-4(a)

DOSE RATE VERSUS DISTANCE FROM A SINGLE OVERPACK WITH THE MPC-32 69,000 MWD/MTU AND 5-YEAR COOLING

Distance		mrem/hr		
m	ft	Side-dose rate	Top-dose rate	Total dose rate
12.19	40.00	2.03E+00	1.42E-02	2.05E+00
18.29	60.00	1.00E+00	1.02E-02	1.01E+00
24.38	80.00	5.80E-01	7.48E-03	5.87E-01
30.48	100.00	3.73E-01	5.86E-03	3.79E-01
45.72	150.00	1.61E-01	3.23E-03	1.64E-01
50.00	164.04	1.35E-01	2.84E-03	1.38E-01
60.96	200.00	8.68E-02	2.03E-03	8.88E-02
91.44	300.00	3.45E-02	9.18E-04	3.54E-02
100.00	328.08	2.79E-02	7.51E-04	2.86E-02
121.92	400.00	1.68E-02	4.94E-04	1.73E-02
150.00	492.13	9.67E-03	2.83E-04	9.95E-03
200.00	656.17	4.33E-03	1.22E-04	4.45E-03
250.00	820.21	2.15E-03	5.70E-05	2.20E-03
300.00	984.25	1.18E-03	2.82E-05	1.20E-03
350.00	1148.29	6.59E-04	1.50E-05	6.74E-04
400.00	1312.34	3.81E-04	8.33E-06	3.90E-04
450.00	1476.38	2.29E-04	4.63E-06	2.34E-04
500.00	1640.42	1.44E-04	2.68E-06	1.47E-04
550.00	1804.46	9.35E-05	1.53E-06	9.50E-05
600.00	1968.50	6.42E-05	9.16E-07	6.51E-05
650.00	2132.55	4.13E-05	6.13E-07	4.19E-05
700.00	2296.59	2.80E-05	3.63E-07	2.83E-05
750.00	2460.63	1.84E-05	2.34E-07	1.86E-05
800.00	2624.67	1.26E-05	1.61E-07	1.27E-05
850.00	2788.71	9.10E-06	1.07E-07	9.20E-06
900.00	2952.76	6.16E-06	7.19E-08	6.24E-06

(These values are derived from the annual dose numbers at 5 years cooling from Reference 5, Appendix P, adjusted for the annual exposure of 8760 hours/yr.)

TABLE 7.3-4*(b)*

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 7.5-1

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~~8.5E-03	2,080	~~5~~17.6
Nearest Resident (1.5 mi / 7,920 ft / 2414 m)	3.4~~.0~~E-07~~8~~	8,760	3.0~~5~~E-03~~4~~

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

This Table is presented for historical information only. See discussion in Section 7.2.2.

(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)]	~~9~~*12*	8	~~1.44~~*3.22*E-01
~~MPC in overpack without a lid~~	~~9.0E-04~~	~~1.5~~	~~8~~	~~1.1E-02~~
~~Total~~				~~15.5E-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 bottom 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~~*HI-TRAC contribution at the site boundary is estimated by scaling the HI-TRAC dose at 1 meter (dose location 2 in Table 7.3-2(a) is used for this purpose*)*by the dose rates reduction obtained for HI-STORM between 1 and 400 meters (the site boundary is at 426.72 meters)*.

TABLE 7.5-4

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

	Normal Operations				**Off-Normal Operations**		
Organ	**Effluent Release[(c)]**	**Direct Radiation[(c)]**	**Overpack Loading Operations[(d)]**	**Other Uranium Fuel Cycle Operations[(a)]**	**Effluent Release[(~~d~~e)]**	**Total (normal + off-normal)**	**10 CFR 72.104 Regulatory Limit**
Site Boundary (1,400 ft / 427 m)							
Whole body ADE[(b)]	~~0.064~~0	~~5~~17.6	~~15.5E-02~~0.3	4.357E-02	1.27E-03	~~5.86~~17.9	25
Thyroid ADE	~~0.010~~0	~~5~~17.6	~~15.5E-02~~0.3	1.260E-01	1.02E-04	~~5.89~~18.0	75
Critical organ ADE (Max)	~~0.35~~0	~~5~~17.6	~~15.5E-02~~0.3	5.590E-02	9.31E-03	~~6.17~~18.0	25
Nearest Resident (1.5 miles / 7,920 ft / 2414 m)							
Whole body ADE	~~0.27~~0	3.0~~5~~E-03~~4~~	~~15.5E-02~~5.1E-05	4.357E-02	5.33E-03	0.047	25
Thyroid ADE	~~0.043~~0	3.0~~5~~E-03~~4~~	~~15.5E-02~~5.1E-05	1.260E-01	4.31E-04	0.~~32~~13	75
Critical organ ADE (Max)	~~1.46~~0	3.0~~5~~E-03~~4~~	~~15.5E-02~~5.1E-05	5.590E-02	3.92E-02	~~1.71~~0.098	25

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

(b) ADE is annual dose equivalent.

(c) 140 casks

(d) ~~Single cask~~*From Table 7.5-3. For nearest resident, the value is scaled by the ratio of direct radiation dose from the site boundary to the nearest resident.*

(e) From Table 8.1-1.

CHAPTER 8

ACCIDENT ANALYSES

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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 FSAR 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.

As discussed in Section 1.1 the licensed HI-STORM 100 System at the Diablo Canyon ISFSI has been modified to facilitate fuel-loading campaigns. These modifications were performed in accordance with 10 CFR 72.48 and involve the MPC-32 canister, HI-TRAC 125D transfer cask, HI-STORM 100SA overpack, CTF, low profile transporter, cask transporter, and auxiliary components used in the loading and transport to the ISFSI facility. The originally-licensed MPC-24s will likely not be used at the ISFSI and would require modifications, analyses and associated 10 CFR 72.48 evaluations similar to the MPC-32 prior to their use. Most of the accident and off-normal analyses and evaluations performed for the licensed HI-STORM 100 system remain bounding for the modified system. However, in cases where they do not and a re-analysis or site specific analysis was required, those analyses are identified and referenced in their related sections below.

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 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 used at Diablo Canyon 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 personnel 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 performed assuming normal ambient temperature (80°F), 10 percent of the fuel rods ruptured, peak insolation, maximum decay heat, maximum backfill pressure, IFBA fuel 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 the MPC-32 with 80°F ambient temperature is 79.~~0~~ *7 for the storage condition (Reference 13, Table B.5.14,* and 87.90 psig for the ~~storage and~~ transport condition~~s, respectively~~ (Reference 11, Table 9)*, respectively*. The added effect of increasing the ambient temperature from 80°F to the maximum off-normal temperature of 100°F *on the internal pressure* was ~~also~~ *included in the calculation*~~evaluated~~ in Reference ~~11~~ *13* for the storage condition. ~~The resulting pressure for this configuration is 81.1 psig (Reference 11, Table 9).~~ For the transport condition, 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 P2 for the transport condition is computed as follows:

$$P2 = P1 \times [(T1 + \triangle T)/T1]$$

Where,

P_1 = Absolute pressure at T_1 = 87.9 psig (102.6 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 89.8 psig for the transport condition. Pressure values for both the storage and transport conditions 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. 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 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. 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. Thermal analysis contained in the HI-STORM 100 System FSAR indicates that the MPC-32 has the highest design-basis decay heat load and always yields the highest cask system component and content temperatures. As such, only the MPC-32 is evaluated since the MPC-24 and MPC-24E thermal performance will be bounded by that of the MPC-32 under all conditions.

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 and the overpack) are shown in Table *B.5.*2 of Reference 1*6*~~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 *B.5.4* ~~3~~ of Reference 1*3*~~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 (Reference 7). The difference between the two evaluations is in the radioactive source term, the bounding temperature and pressure determined in the thermal analysis of Reference 11 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 FSAR 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.

This section only applies to the initial 16 casks loaded at the Diablo Canyon ISFSI. Following construction of the first 16 casks, the testing requirement for the MPC boundary welds was changed to the leaktight criteria of ANSI N14.5-1997. The vent and drain port cover plate welds helium leak testing requirements had been changed to the "leaktight" criteria of ANSI N.14.5-1997 in LA 1. Since the lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore in accordance with ISG-18, leakage from this weld is considered non-credible. Since all the closure welds meet a leaktight criteria, the confinement boundary of the subsequently fabricated MPCs can be considered leak tight.

8.1.3.1 Postulated Cause of Confinement Boundary Leakage

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

8.1.3.2 Detection of Confinement Boundary Leakage

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

8.1.3.3 Analysis of Effects and Consequences of Confinement Boundary Leakage

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

8.1.3.4 Corrective Action for Confinement Boundary Leakage

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

8.1.3.5 Radiological Impact of Confinement Boundary Leakage

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

8.1.4 PARTIAL BLOCKAGE OF AIR INLETS

The HI-STORM 100 System overpack is designed with inlet and outlet air ducts, four each at the top and bottom of the overpack structure with the lid installed. Each duct opening includes a stainless steel perforated plate (screen) across its outer face. These perforated plates (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. 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 perforated plates (screens) prevent foreign objects from entering into the ducts. The perforated plates (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 perforated plates (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. Computed component temperatures for two air inlet ducts blocked are less than the allowable component short-term temperature limits. ~~Computed component temperatures for three air inlet ducts blocked are less than the allowable component short-term temperature limits.~~ (Blocking of four ducts is treated as an accident in Section 8.2.15.) The results are shown in Table ~~4~~ *B.5.4* of Reference 1*3*~~1~~.

The MPC cavity pressure for ~~three~~ *two* blocked air ducts was also evaluated. An ~~MPC cavity gas bulk temperature rise of 89°F was evaluated and the resulting~~ MPC internal pressure was computed *using conservatively higher heat load and fill pressure,* to be

~~76.4~~*94.6* psig, which is less than the normal condition MPC design pressure of 100 psig (Reference 1*3*~~1~~, Table ~~9~~*B.5.12*).

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 perforated plates (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 H-STORM 100 System to safely store spent fuel for the long term.

Inspection of the overpack air duct perforated plates (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 perforated plates (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.

8.1.8 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 1A, January 2003.

4. Deleted in Revision 2.

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.

7. PG&E Calculation STA-140 (HI-2002513, Rev. 7), "Diablo Canyon ISFSI Site Boundary Confinement Analysis."

8. License Amendment Request 02-03, Spent Fuel Cask Handling, PG&E Letter DCL-02-044, April 15, 2002.

9. License Amendments 162 and 163, Spent Fuel Cask Handling, issued by the NRC, September 26, 2003.

10. Holtec International HI-STORM 100 Cask System Certificate of Compliance Number 1014, Amendment 1 dated 7/15/02

11. Holtec International Report No. HI-2053376, "Thermal-Hydraulic Analysis for Diablo Canyon Site-Specific HI-STORM System Design," Revision 7.

12. Holtec International Report No. HI-2053390, "Structural Evaluation of the Low Profile Transporter," Revision 4.

13. *Holtec International Document No. HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design", Revision 2.*

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

(6) Fire from mineral oil from the Unit 2 transformers

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 (Reference 41) 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 FSAR, 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, 2, 3, 4, and 6, 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 Technical Specifications (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 20 gallons. On the storage pad, the fuel is postulated to be burning in a pool surrounding the cask, therefore, the concrete short-term temperature limit will be exceeded and is an expected consequence of the event. Recovery from a fire event on the ISFSI pad will require a technical evaluation of the ability of the ISFSI pad, in the affected area, to perform its design function, and appropriate corrective actions taken as necessary

A potential fire in the CTF due to the release of the 50 gallons of fuel from the cask transporter has been addressed. The cask transporter will be designed with features (e.g., a limited fuel tank size and drip pan with drain) that ensure the fuel, if spilled, will not migrate into the CTF structure. The CTF opening will be located at a higher elevation than the local surrounding area such that any fuel spilled will flow away from the CTF by gravity. This ensures that any fire that may occur is bounded by the fire analysis described in Section 11.2.4 of the HI-STORM System FSAR.

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 any onsite, 20-gallon vehicle-fuel-tank fire (Event 2).

Portable generators and air compressors may be used during MPC transfer activities. If portable generators and air compressors are used, procedural controls will be established to ensure that they are bounded by the fire analysis described in Section 11.2.4 of the HI-STORM System FSAR.

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, and the transport route during transport operations to ensure the total energy received is less than the design-basis fire event. In addition, when the tanker truck is moving on the roadway past the ISFSI, the roadbed in all cases is below the level of the ISFSI pad, which ensures that even if there were a tank rupture, the fuel would not run toward the ISFSI. An analysis was performed for a ruptured 2,000-gallon gasoline tanker truck, which determined that at a distance of more than 4 meters it does not result in exceeding the design basis of the storage casks (Reference 34). There are fuel trucks on the DCPP site that carry up to 4,000 gallons of gasoline, however, those trucks are administratively maintained at least 1,100 ft from a cask being transported or the CTF/ISFSI facility. In addition, only trucks containing no more than 800 gallons of gasoline are allowed to pass the CTF/ISFSI facility at any time, and that movement is administratively controlled to ensure that the tanker is never at a distance that would not be bounded by the analysis performed for a ruptured 2,000-gallon gasoline tanker truck, which determined that at a distance of more than 4 meters, does not result in exceeding the design basis of the transfer cask. (Reference 34)

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). The wildfire evaluation uses predictive models called FARSITE and FLAMMAP (Reference 36) to determine the potential characteristics of wildfire in the Diablo Canyon. Both models utilize mapped data about the type of vegetation (fuel model), slope, aspect, elevation, wind, and moisture to predict wildfire characteristics such as flame length, rate of spread, heat per unit area, etc. The ISFSI site, located immediately southeast of the power plant's raw water reservoirs, is surrounded on the south, southeast, and north sides by a vegetation type of "annual grassland" (Reference 37). The main access road forms the northwest boundary of the proposed site. The annual grassland vegetation is grazed and has relatively low cover. Consequently, the fire risk of this fuel type is relatively low.

For Event 6, the physical properties of mineral oil limit the threat of a fire. 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 point of mineral oil is 275°F. 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 flash point of mineral oil. Therefore, under ambient or normal operating temperature, these materials do not represent a credible fire hazard. However, if an electrical fault were to occur in a transformer, the increase in heat within that transformer could cause it to rupture and its contents may support a local fire. The resulting fire is considered to be limited and bound by the design basis fire provided in Section 11.2.4 of the HI-STORM 100 System FSAR, and is further supported by an analysis performed for a ruptured 2000-gallon gasoline tanker truck, which determined that at a distance of more than 4 meters does not result in exceeding the design basis of the transfer cask. (Reference 34)

The probability of this event occurring while the transfer cask is in proximity and it affecting the transporter and transfer cask is extremely low. This is based on the properties of mineral oil, the minimum distance from the transformers to the transporter, the limited amount of exposure time, a dedicated transformer fire suppression system, and a significant difference in elevation between the transformers and the transporter route.

The transformers are approximately 240 ft from the transporter at its closest point during transport and the transporter is within a line of sight of the transformers for no more than 10 hours per year. Each of the transformers is surrounded by a dedicated fire suppression system that will act to control and minimize any fire that could potentially occur. There is also a 30-ft difference in elevation between the transporter route and the transformers that will not allow oil from a transformer to approach within approximately 120 ft of the transporter.

In addition, although a fire from a transformer is considered bounded by the design basis of the transfer cask and not an unacceptable hazard, in an effort to further minimize its probability, PG&E is taking prudent actions to minimize the transformer fire hazards during transport as follows:

For potential external hazards, administrative procedures will not allow any vehicle motion in the vicinity of the transformers during transport operations. In addition, administrative procedures are in place that will not allow transport of fuel when severe weather (which could result in lightning or other hazards) exists or is predicted to occur during the transport time in the vicinity of the DCPP plant site. To address the potential hazard for an internal short, PG&E administrative procedures consider offsite power conditions prior to transport operations in the vicinity of the Unit 2 transformers.

Based on the above discussion, the potential hazard from a transformer fire is considered credible; however, its potential effects are limited and considered bounded by the design basis fire analysis for the transfer cask.

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 *as demonstrated by the Diablo Canyon ISFSI specific thermal analyses (Reference 63)*.

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 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, 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~~ *shown* in ~~Section 8.2.11.2~~*Table C.3 of Reference 63*, the increase in fuel cladding and component material temperatures due to the *fire and* 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*, as shown in Reference 63, Table C.4*. Thus, there is no effect on the integrity of the MPC confinement boundary.

The ISFSI system is not 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.

8.2.7 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.

This section only applies to the initial 16 casks loaded at the Diablo Canyon ISFSI. Following construction of the first 16 casks, the testing requirement for the MPC boundary welds was changed to the leaktight criteria of ANSI N14.5-1997. The vent and drain port cover plate welds helium leak testing requirements had been changed to the "leaktight" criteria of ANSI N.14.5-1997 in LA 1. Since the lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore in accordance with ISG-18, leakage from this weld is considered non-credible. Since all the closure welds meet a leaktight criteria, the confinement boundary of the subsequently fabricated MPCs can be considered leak tight.

8.2.7.1 Cause of Accident

The analyses presented in Chapters 3 and 11 of the HI-STORM 100 System FSAR 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 (Reference 43). 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 leakage rate limit of $\leq 5.0 \times 10^{-6}$ 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).

8.2.10 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 Reference 6*3*~~1~~ and at -40°F in Section 4.4.3 of the HI-STORM 100 System FSAR. 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 Reference 6*3*~~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 site-specific 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 listed in ~~Table 5 of Reference 61~~*Table B.5.7 of Reference 63*. 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*, from Table B.5.12 of Reference 63, is calculated as 98.8 psig which is*~~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.

8.2.12 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 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 shows~~*Section B.5.4(e) of Reference 63 calculates* that the time to reach the short-term, fuel-cladding-temperature limit ~~varies from~~*is* approximately ~~45~~ *52* hours at a total cask heat load of ~~30~~ *36.9* 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.

8.2.14 100 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 ~~189.8 psia (175.1~~*172.9* psig~~)~~ (Reference 6*3*~~1~~, Table ~~9~~*B.5.14*), 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 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. 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 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 Section 10.2 and in the Diablo Canyon ISFSI TS. Incorporation of the MPC thermosiphon internal convection phenomenon, as described in Chapter 4 of the HI-STORM 100 System FSAR enables the maximum, design-basis, PWR-decay-heat load to rise to about ~~29~~ *37* 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~~*600*°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 *was* calculated at an ambient temperature of 80°F, ~~100 percent fuel rods ruptured,~~ design-basis insolation, and maximum decay heat ~~is 189.8 psia, as discussed in Section 8.2.14.2~~ *as part of the site specific thermal analysis (Reference 63). The analysis did not assume a simultaneous 100% rod rupture event, since the peak fuel cladding temperatures for the accident conditions never exceed the regulatory accident temperature limit, which ensures no significant cladding failures would occur. This is consistent with the latest NRC guidance on fuel cladding in dry storage casks (Reference 21), which states "In order to assure integrity of the cladding material … For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F)." The same result is confirmed for all accidents evaluated for the Diablo Canyon ISFSI. Therefore, no coincident 100% rod rupture postulations with an accident are evaluated. This is supported by the HI-STORM 100 CoC, Amendment 5.*~~This calculated pressure is for an MPC cavity bulk gas temperature of 509°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 71°F (39°K), and the Ideal Gas Law, t~~*T*he resultant MPC internal pressure is calculated to be ~~213.4~~*112.4* ~~psia (198.7~~ psig~~)~~ *(Reference 63, Table B.5.12)*, 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 .

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~~*32*-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 perforated plates (sheets) 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.

8.2.17 Supplemental Cooling System (SCS) Failure

The SCS system is a supplied fluid device used to provide supplemental HI-TRAC cooling. The SCS system maintains water in the MPC/HI-TRAC annulus to cool the MPC shell in order to maintain the fuel cladding below the ISG-11 Rev. 3 temperature limit. Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that renders it inoperable for an extended duration is postulated herein.

8.2.17.1 Cause of SCS Failure

Possible causes of SCS failure are: (a) Complete loss of annulus water from an uncontrolled leak or line break, or (b) other equipment problems after SCS has been secured for short term evolution, and SCS cannot be reestablished within the required restoration time due to equipment configuration.

8.2.17.2 Analysis of Effects and Consequences of SCS Failure

In the event of an SCS failure due to (a) or (b), a rapid water loss occurs and annulus is replaced with air. For the condition of a vertically oriented HI-TRAC with air in the annulus, the maximum steady state temperatures are below the accident temperature limits for fuel cladding and components (Reference 63).

Since none of the temperature or pressure limits are exceeded, shielding, criticality and confinement functions are unaffected. As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.

8.2.17.3 SCS Failure Dose Calculations

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

8.2.17.4 SCS Failure Corrective Action

In the vertical orientation the HI-TRAC is designed to withstand an SCS failure without an adverse effect on its safety functions. However, actions will be taken to either restore supplemental cooling or transfer the MPC into the HI-STORM in order to return the high burnup fuel cladding temperatures to below ISG-11 Rev. 3 limits.

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