7 SHIELDING EVALUATION

7.1 Conduct of Review

The objective of the staff's shielding review was to determine if the shielding design features of the proposed Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) meet U.S. Nuclear Regulatory Commission (NRC) criteria for radiation protection to workers and to the public. These criteria establish the limits for direct radiation resulting from all ISFSI operations. The shielding evaluation includes a review of the information in Chapter 7, "Radiation Protection," of the Diablo Canyon ISFSI Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2002) and relevant sections of Chapter 3, "Principal Design Criteria;" Chapter 4, "ISFSI Design;" and Chapter 8, "Accident Analyses," of the SAR.

Pacific Gas and Electric Company (PG&E) proposes to use the HI-STORM 100 System, which consists of interchangeable Multi-Purpose Canisters (MPCs) that contain the fuel, a storage overpack, and a transfer cask. The HI-TRAC transfer cask contains the MPCs during loading, unloading, and transfer operations. The system has been reviewed and approved by NRC for use under the general license provisions of 10 CFR Part 72. Amendment 1 of the HI-STORM 100 System Certificate of Compliance (CoC) became effective on July 15, 2002 (U.S. Nuclear Regulatory Commission, 2002). Cask-specific information reported in the HI-STORM 100 System Final Safety Analysis Report (FSAR), Revision 1 (Holtec International, 2002) was reviewed as part of the shielding evaluation of the Diablo Canyon ISFSI.

The shielding review considered how the information in the SAR addressed the following regulatory requirements:

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

- 10 CFR §72.24(c)(3) requires that the SAR describe all information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin for safety.

- 10 CFR §72.24(e) requires that the SAR describe the means of controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20, and for meeting the objective of maintaining exposures as low as reasonably achievable.

- 10 CFR §72.104(a) requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area be limited to 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid or 0.25 mSv (25 mrem) to any other critical organ.

- 10 CFR §72.106(b) requires that any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or any extremity may not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel, waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

- 10 CFR §72.126(a)(6) requires that radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposures must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. This design must include means to shield personnel from radiation exposure.

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

- 10 CFR §20.1201(a) requires that the licensee shall control the occupational dose to individual adults, except for planned special exposures under §20.1206, to the following dose limits. (1) An annual limit, which is the more limiting of (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or the lens of the eye being equal to 50 rems (0.5 Sv). (2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, (i) A lens dose equivalent of 15 rems (0.15 Sv), and (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

- 10 CFR §20.1301(a) requires that each licensee shall conduct operations so that (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under §35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with §20.2003, and (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with §35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

7.1.1 Contained Radiation Source

The gamma and neutron source specifications are presented in Section 7.2 of the Diablo Canyon ISFSI SAR. The sources of gamma and neutron radiation are the intact spent nuclear fuel assemblies, damaged fuel assemblies, fuel debris and nonfuel hardware to be stored in the HI-STORM 100 System. Only fuel and associated hardware irradiated at Diablo Canyon Power Plant (DCPP) Units 1 and 2 will be stored at the ISFSI. The spent nuclear fuel assemblies to be stored at the proposed ISFSI consist of two zircaloy-clad pressurized water reactor (PWR) designs. The two designs are the Westinghouse LOPAR 17 × 17, and the Westinghouse VANTAGE 5, 17 × 17, assemblies. Their characteristics are described in the Diablo Canyon ISFSI SAR Tables 3.1-1 and 3.1-2. Initial U-235 enrichment of the fuel to be stored at the proposed ISFSI will be limited to 5 percent. Average burnup will be limited to a maximum of 58,000 MWd/MTU and cooling time must exceed 5 years (Note: by letter dated January 16, 2004, PG&E further restricted the burnup of spent fuel assemblies to be stored at the ISFSI to 45,000 MWd/MtU; however, the staff's shielding evaluation assumed a higher burnup, where it was more limiting). Burnable poison rod assemblies, thimble plug devices, and rod cluster assemblies that will also contribute to the radiation source are described in Section 7.2 of the SAR. The gamma source term is composed of three distinct components. The first gamma source term is decay of fission products in the active fuel region. The second gamma source term is Co-60 activity of the steel structural material in the fuel element above and below the active fuel region. The third gamma source term is from (neutron, gamma) reactions.

For the ISFSI storage pads, PG&E conducted a shielding analysis that estimated bounding dose rates from direct radiation by assuming that storage overpacks containing identical MPC-32s are completely loaded with fuel assemblies having burnup of 32,500 MWd/MTU and 5-year cooling times. A Babcock and Wilcox 15 × 15 fuel assembly, a design-basis assembly for the HI-STORM 100 System, was used in the ISFSI shielding analyses. Additional credit was taken in the dose analyses for longer cooling times because the casks are to be placed at the ISFSI at the rate of only eight per year.

The PG&E shielding analysis of the transfer cask was performed for the MPC–24 using burnup and cooling time of 55,000 MWd/MTU and 12 years. These dose rates bound those for other MPC, burnup, and cooling time combinations. Gamma and neutron source terms were generated using the SAS2H (Hermann and Parks, 1995) and ORIGEN-S sequences from the SCALE 4.3 system (Hermann and Westfall, 1995).

The Co-60 gamma source for nonfuel hardware was evaluated. The Co-59 impurity level was assumed to be 0.8 g/kg [8×10^{-4} lb/lb] or 800 ppm in stainless steel and 4.7 g/kg [4.7×10^{-3} lb/lb] or 4,700 ppm in inconel. Burnup of 40,000 MWd/MTU and cooling time of 12 years were assumed for a design-basis burnable poison rod assembly. It was assumed that all overpacks were filled with the design-basis burnable poison rod assemblies.

For the shielding analysis, it was assumed that overpacks are placed on flat ground, so shielding design features included only the overpack and the transfer cask and no credit was taken for any additional radiation shielding provided by the hilly terrain surrounding the ISFSI storage pads. The dose-versus-distance analysis was conducted for a single overpack containing an MPC-32 loaded with design-basis burnup spent fuel and cooling times using the MCNP-4A Code (Briesmeister, 1993). The dose rates of direct radiation emanating from 140 casks were calculated at distances perpendicular to the long side of the ISFSI and assuming that eight overpacks are loaded per year. Radiation from 28 casks located along the long side of the ISFSI was accounted for as unshielded, and radiation from the remaining casks was accounted for as partly shielded by the casks located between those casks and dose locations.

The staff evaluated the applicant's analyses of the bounding radiation source terms for the Diablo Canyon ISFSI. The staff found the description of radiation sources and calculation methods of the shielding analysis to be consistent with the information provided in HI-STORM 100 System FSAR, Revision 1. The HI-STORM 100 System has been approved by NRC for use under the general license provisions of 10 CFR Part 72. The staff found that combinations of enrichment, burnup, and cooling time for DCPP spent nuclear fuel are conservative and were determined correctly.

7.1.2 Storage and Transfer Systems

7.1.2.1 Design Criteria

The design criteria for the proposed Diablo Canyon ISFSI are the regulatory dose limit requirements delineated in 10 CFR Part 20, §72.104(a), and §72.106(b). The Diablo Canyon ISFSI SAR specifies the shielding design criteria in Sections 3.3.1.1.2, 3.3.1.1.3, and 3.3.1.5.2. The SAR sections reference Chapters 1, 2, and 5 of the HI-STORM 100 System FSAR, Revision 1. The HI-STORM 100 System storage cask is designed to limit maximum average surface dose-rates radially, on top, and in areas adjacent to the inlet and exit vents to 600 µSv/hr [60 mrem/hr]. The transfer cask is designed to maintain personnel doses as low as reasonably achievable (ALARA). The staff finds the use of these design criteria to be appropriate. These design criteria provide reasonable assurance that the ISFSI will meet the dose limits delineated in 10 CFR Part 20, §72.104(a), and §72.106(b). The Diablo Canyon ISFSI will provide adequate radiological safety based on the use of suitable shielding for radiation protection in accordance with 10 CFR §72.128(a)(2).

7.1.2.2 Design Features

The Diablo Canyon ISFSI system is designed to provide both gamma and neutron shielding for all fuel loading, transfer, and storage conditions. The system shielding design features are described in Section 7.3 of the SAR. ISFSI design features that ensure that dose rates are ALARA include:

- There are no radioactive systems at the ISFSI storage pads other than overpacks containing MPCs.

- The MPC shielding is composed of a 1.3-cm-[0.5-in-] thick steel canister with a 6.4-cm-[2.5–in-] thick baseplate and a 24.1-cm-[9.5-in-] thick steel lid for the MPC-24 or a 25.4-cm [10-in] lid for the MPC–68.

- The MPCs are heavily shielded by the overpack that consists of 67.95-cm-[26.75-in-] thick concrete encased in inner and outer steel cases with a total thickness of 6.99 cm [2.75 in]. The top of the overpack is shielded by 26.7-cm-[10.5-in-] thick concrete encased in inner and outer steel cases with a total thickness of 13.3 cm [5.25 in]. The bottom of the overpack is shielded by the pedestal shield and baseplate with a total of 17.8 cm [7 in] of steel and 43.2 cm [17 in] of concrete.

- The HI-TRAC Transfer Cask radial shield is composed of a 4.45-cm- [1.75-in-] thick steel layer, an 11.4-cm-[4.5-in-] thick lead layer, and a 13.61-cm- [5.36-in-] thick water jacket. Its top lid consists of a 8.26-cm- [3.25-in-] thick Holtite-A layer encased in steel plates with a total thickness of 3.8 cm [1.5 in].

- The Cask Transfer Facility (CTF) positions the overpack below ground during loading operations. This position reduces dose rates during loading operations.

The staff evaluated the Diablo Canyon ISFSI shielding design features and found them acceptable. The information in the Diablo Canyon ISFSI SAR provides reasonable assurance that the shielding design features of the storage and transfer systems will meet the requirements in 10 CFR Part 20, §72.104(a), and §72.106(b). The staff evaluated the radiation protection design features of the Diablo Canyon ISFSI in Chapter 11 of this SER.

7.1.3 Shielding Composition and Details

7.1.3.1 Composition and Material Properties

The composition of the materials used in the shielding analysis is presented in the Diablo Canyon ISFSI SAR, Sections 3.3.1.5.2 and 7.3.2. These sections reference the HI-STORM 100 System FSAR, Revision 1. The primary shielding is provided by the concrete and steel in the HI-STORM 100 System storage cask during storage and by the steel, lead, water, and neutron shield in the HI-TRAC 125 Transfer Cask during transfer operations.

The staff found the description of the shielding composition to be sufficient to meet the requirements of 10 CFR §72.24(b) and §72.24(c)(3) by describing the design, the system shielding composition, and materials important to safety. This description is sufficiently detailed for evaluation of shielding effectiveness for maintaining the dose rates around the Diablo Canyon ISFSI within regulatory limits.

7.1.3.2 Shielding Details

The shielding details are described in Section 7.3.1 of the SAR. The HI-STORM 100 System storage casks will be stored on seven concrete pads in 4 × 5 arrays of casks. The total Diablo Canyon ISFSI storage capacity is 140 casks, which will be positioned on a 5.18 m [17-ft], center-to-center pitch.

The shielding models included streaming paths for the radial steel fins and pocket trunnions of the HI-TRAC 125 Transfer Cask, drain and vent ports in the MPC, annulus between the MPC and the concrete cask, annulus between the MPC and the transfer cask, and labyrinthine air inlet and outlet passages. Gamma and neutron doses were calculated using three-dimensional models with the MCNP–4A Code (Briesmeister, 1993).

The staff evaluated the shielding details and found that the description satisfies the requirements of 10 CFR §72.126(a)(6) and provides reasonable assurance that the radiation protection systems were adequately modeled in the shielding analysis.

7.1.4 Analysis of Shielding Effectiveness

7.1.4.1 Computational Methods and Data

The computational methods and data used to analyze shielding effectiveness in reducing the dose rates at the ISFSI are presented in Section 7.3.2 of the SAR and in the HI-STORM 100 System FSAR, Revision 1. Analyses were conducted to determine the surface and 1-m [3.28 ft] dose rates for storage and transfer casks, and dose rates at the restricted area fence, the makeup water facility, the power plant, the controlled-area boundary, and the nearest resident location.

The shielding analysis of the HI-STORM 100 System casks was performed using the MCNP-4A Code (Briesmeister, 1993). The MCNP Code is a general-purpose, continuous-energy, generalized geometry, time-dependent, coupled neutron-photon-electron Monte Carlo transport code system. The code system is able to model the complex surfaces associated with the storage casks. For neutrons, continuous cross-sectional data are accounted for in all reactions in any given cross-section evaluation such as ENDF/B-VI (Los Alamos National Laboratory, 1994). Thermal neutrons are described by both the free gas and S (alpha beta) models. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation and bremsstrahlung.

The gamma flux-to-dose conversion factors used in the Diablo Canyon ISFSI SAR were from American National Standards Institute/ American Nuclear Society (ANSI/ANS)–6.1.1 (American Nuclear Society Standards Committee Working Group, 1977). The computer code and the ANSI/ANS–6.1.1 flux-to-dose conversion factors used for shielding analysis are considered acceptable by the staff for use in the shielding evaluations.

7.1.4.2 Dose Rate Estimates

The estimates of dose rates and annual doses caused by direct neutron and gamma radiation at various on-site and off-site locations are presented in Sections 7.3.2.1, 7.3.2.2, 7.5.1, 7.5.3, and 7.5.4 of the SAR.

The HI-TRAC 125 Transfer Cask is designed to reduce dose rates from direct radiation emanated from a loaded MPC to levels that are ALARA. The design-basis MPC for the transfer cask analysis is the MPC-24 loaded with a fuel having a burnup of 55,000 MWd/MTU and 12-year cooling time period. The contact surface dose rate for the HI-STAR transfer cask was estimated to be approximately 3,893 µSv/hr [389.3 mrem/hr] outside the lid in its center. The transfer of MPC from the transfer cask to the overpack will take place outside the Fuel Handling Building/Auxiliary Building (FHB/AB) at the CTF. The impact on the offsite dose of the loaded MPC while it is transported inside the transfer cask from the plant to the CTF was considered in the analysis.

The design-basis MPC for the storage cask shielding analysis is the MPC–32 with a burnup of 32,500 MWd/MTU and 5-year cooling time period. The average contact surface dose rate for the HI-STORM 100 System storage cask (side-dose value) was estimated to be approximately 348 µSv/hr [34.8 mrem/hr] at the midplane of the overpack. To evaluate onsite and offsite dose rates from direct and scattered radiation emanating from the spent nuclear fuel stored at the ISFSI, the applicant employed an approach described in Section 5.4.3 of the HI-STORM 100 System FSAR, Revision 1. In this approach, the dose rates were estimated along the line perpendicular to the long side of the array of casks situated at the storage pad. In the first stage of analysis, the MCNP binary surface source file was generated for a single cask. This surface source was based on the real configuration of design-basis fuel in the cask, and those data were used in the second stage of the analysis. In the second stage of analysis, the contribution of the cask positioned in the second row of the array of casks and blocked from direct view line to the dose point was estimated. Then, the total direct and scattered dose rates were summed and dose rates at various distances were estimated for several small arrays of casks. The side-dose rate values, top-dose rate values, and in-air scattering of radiation (skyshine) dose rates from both side and top radiation components were taken into account. Results of the calculations employing this approach are presented in Section 5.4.3 of the HI-STORM 100 System FSAR, Revision 1, and are illustrative in nature and not site-specific. It is important to highlight that the contribution of the back row of casks to the total dose at a point 300 m [984 ft] away was estimated as 16 percent of the total dose. The dose fraction caused by neutrons from one cask at a point 200 m [656 ft] away from the cask was estimated at 7 percent of the total dose.

In addition to this approach, the applicant applied credit for the additional cooling times for the stored fuel based on its ISFSI loading plan. PG&E assumed in its dose rate analysis that eight overpacks per year will be loaded until all the ISFSI pads are filled to capacity. PG&E presented the reduction of gamma and neutron source intensity with time because of the longer cooling periods of design-basis fuel. Using this methodology instead of a full-scale 140-cask array modeling, the applicant calculated a site-boundary dose rate at a point located 427 m [1,400 ft] away from the ISFSI storage pad as 0.027 µSv/hr [0.0027 mrem/hr] from direct and scattered radiation exposure for 140 casks stored in a 5 × 28 array, as described in Chapter 7 of the SAR. This rate corresponds to a 56 µSv [5.6 mrem] annual dose for a 2,080-hr/yr occupancy rate for a hypothetical person located at the site boundary. The applicant estimated an annual direct and scattered radiation dose to the nearest resident located 2,414 m [1.5mi] away from the ISFSI as 0.0035 µSv [0.00035 mrem]. These dose values are less than the 250 µSv/yr [25 mrem/yr] whole-body dose limit specified in 10 CFR §72.104(a).

The staff evaluated the approach taken by the applicant in direct radiation dose analysis and found that the approach provides reasonable assurance that onsite and offsite dose rates at various distances from the array of casks will be estimated correctly. No off-normal events or accidents resulting in a loss of radiation shielding were identified in the SAR. Therefore, it was concluded in the applicant's analyses that dose rates at the controlled-area boundary and on site would not be affected by the minor damage to transfer or storage casks.

The staff evaluated the Diablo Canyon ISFSI SAR shielding calculations and found them to be acceptable. The dose rates at the on-site and off-site locations were found to be below the limits specified in 10 CFR §20.1201, §20.1301, and §72.104(a). The applicant's description, combined with a sample input file in the HI-STORM 100 System FSAR, Revision 1, provides reasonable assurance that the ISFSI shielding was adequately evaluated. The applicant's analysis demonstrates that no credible accident will cause a significant increase in public or personnel dose rates from direct radiation. This provides reasonable assurance that during accident conditions, dose rates from direct radiation will be below the limits specified in 10 CFR §72.106(b).

Chapter 11 of this SER discusses the overall offsite dose rates from the Diablo Canyon ISFSI estimated from the combined radiation exposure to the direct radiation, scattered radiation, and potential radioactive effluents. The staff has reasonable assurance that compliance with 10 CFR §20.1201, §20.1301, §20.1302, §72.104(a), and §72.106(b) will be achieved by the applicant by the means of its radiation protection design and radiological protection program described in the SAR.

7.1.5 Confirmatory Calculations

The staff examined the proposed contents listed in Tables 7.2.1 through 7.2.4 of the SAR. The staff performed independent calculations of the bounding source terms for the stored fuel at the proposed Diablo Canyon ISFSI. Neutron and gamma source terms, as well as the radionuclide inventory, were generated by the ORIGEN-ARP code (Bowman, 2000) for various cooling periods. These calculations provided reasonable assurance that design-basis gamma and neutron source terms for the MPC–32 are acceptable for the shielding analyses. The staff performed an independent calculation of the dose rates that could be expected around the storage casks and at the edge of the Diablo Canyon ISFSI controlled area. The staff used the MCNP-4C2 code, cross-sectional data supplied with the code distribution (Briesmeister, 2000), and gamma flux-to-dose conversion factors from ANSI/ANS 6.1.1 (American Nuclear Society Standards Committee Working Group, 1977).

The staff analyses were conducted in three stages. First, a single cask bounding surface source was determined. Second, a dose rate from the surface source was calculated at various distances. Third, the dose rates that would result from the maximum number of 140 casks planned to be stored at the Diablo Canyon ISFSI were calculated. A back-row shadow factor reported by PG&E was confirmed in separate calculations. Variable cask fuel cooling periods and back-row shade factors were taken into account in the total on-site and off-site dose calculations.

The annual doses were calculated for an individual located on the nearest boundary of the ISFSI controlled area and for the resident nearest to the ISFSI. These calculations confirmed the on-site dose rates calculated by the applicant and also confirmed that the off-site dose rates would be less than the 0.25-mSv/yr [25-mrem/yr] whole-body dose allowable to a member of the public, as required by 10 CFR §72.104. Based on these confirmatory calculations, the staff concluded that the applicant's shielding analysis is acceptable.

7.2 Evaluation Findings

The staff's shielding evaluation for the Diablo Canyon ISFSI assumed that only the HI-STORM 100 System will be used. The staff made the following findings regarding the shielding evaluation of the Diablo Canyon ISFSI:

- The design of the shielding system of the Diablo Canyon ISFSI satisfies the design criteria for radiological protection of 10 CFR §72.126(a)(6).

- The design of the Diablo Canyon ISFSI provides acceptable means for controlling occupation radiation exposures within the limits given in 10 CFR §20.1201 and for meeting the objective of maintaining exposures ALARA, in compliance with 10 CFR §72.24(e).

- The design of the Diablo Canyon ISFSI provides acceptable means for controlling exposures of the public to direct and scattered radiation within the limits given in 10 CFR §72.104(a) and 10 CFR §72.106(b)

- The design of the Diablo Canyon ISFSI provides suitable shielding for radioactive protection during normal and accident conditions in compliance with 10 CFR §72.128(a)(2).

7.3 References

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