7 SHIELDING EVALUATION

7.1 Conduct of Review

The objective of the staff's shielding review is to determine whether the shielding design features of the proposed Humboldt Bay ISFSI meet NRC criteria for protection against direct radiation from the material to be stored. Specifically, this evaluation establishes the validity of the dose rate estimates made in the Humboldt Bay ISFSI Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2004a). These estimates are used in the radiation protection review contained in Chapter 11 of this Safety Evaluation Report (SER) to determine compliance with regulatory limits for allowable dose rates and conformance with criteria for maintaining radiation exposures as low as is reasonably achievable (ALARA). The shielding evaluation includes a review of the information in Chapter 7, "Radiation Protection," and relevant sections of Chapter 3, "Principal Design Criteria;" Chapter 4, "ISFSI Design;" and Chapter 8, "Accident Analyses;" of the SAR, as well as supporting documentation.

The applicant proposes to use the HI-STAR HB cask system, which is comprised of the all-metal HI-STAR HB overpack and its integral multi-purpose canister (MPC-HB), that contains the fuel assemblies. Each HI-STAR HB cask is designed to store up to 80 Humboldt Bay Power Plant (HBPP) fuel assemblies. The HI-STAR HB system is a variation of the HI-STAR 100 system, which has been certified by NRC for use by 10 CFR Part 72 general licensees (U.S. Nuclear Regulatory Commission, 2001a). Holtec International developed the modified (shorter) MPC-HB for use at Humboldt Bay because of the smaller size (length and width) of the HBPP fuel assemblies. The modified HI-STAR HB system shielding analyses were performed in accordance with the methodologies documented in the Holtec HI-STAR 100 Final Safety Analysis Report (FSAR) (Holtec International, 2002a). There will be five HI-STAR HB casks stored at the ISFSI. The applicant also proposes to store an additional cask that contains reactor-related Greater than Class C (GTCC) waste at the ISFSI.

The review considered how the SAR and related documents address the regulatory requirements of 10 CFR §20.1201(a)(1), §20.1301(a), §20.1302(b), §72.24(b), §72.24(c)(3), §72.24(e), §72.104(a), §72.126(a)(6), and §72.128(a)(2). Complete citations of these regulations are provided in the Appendix of this SER.

7.1.1 Contained Radiation Sources

The gamma and neutron source specifications are presented in Section 7.2 of the SAR. The sources of gamma and neutron radiation are the intact spent nuclear fuel (SNF) assemblies, damaged fuel assemblies, fuel debris and nonfuel hardware to be stored in the HI-STAR HB system, as well as the reactor-related GTCC waste to be stored in a separate cask. HBPP Unit 3 was shut down in July 1976. For analysis purposes, the applicant used a cask loading date of July 2005, providing a minimum cooling and decay time of 29 years. The burnup of 23,000 megawatt-days per metric ton of uranium (MWd/MTU) is used as the bounding burnup value for all HBPP fuel assemblies (as this is the highest burnup of all HBPP fuel assemblies in the spent fuel pool inventory). Only fuel, associated hardware, and reactor-related GTCC waste irradiated at HBPP Unit 3 will be stored at the ISFSI. The SNF assemblies to be stored at the proposed ISFSI consist of four Zircaloy-2 clad boiling water reactor (BWR) designs. The four designs are the General Electric Type II, 7 × 7; General

Electric Type III, 6 × 6; Exxon Type III, 6 × 6; and Exxon Type IV, 6 × 6 assemblies. Their physical characteristics are described in Table 3.1-2 of the SAR. An enrichment of 2.09 wt % Uranium-235 was used for the shielding analysis as the lowest initial assembly planar average enrichment for all HBPP fuel. This adds conservatism to the analysis because lower enrichments for a given fuel burnup result in higher neutron source terms.

7.1.1.1 Gamma and Neutron Sources

The gamma source term is composed of three distinct components. The first gamma source term is decay of radioactive fission products. The second gamma source term is secondary photons from neutron capture (neutron, gamma) reactions in fissile and nonfissile radionuclides. The third gamma source term is from hardware activation products generated during power operations. Nonfuel portions of a fuel assembly, such as the steel and Inconel end fittings, become activated during in-core operations to produce a radiation source that is primarily Cobalt-60. Crud on the fuel assemblies is not explicitly accounted for in the source term because the crud source strength is negligible compared to the fuel source strength in the Cobalt-60 gamma energy range. The crud Cobalt-60 source strength calculations are based on the maximum amount of crud measured on HBPP fuel assemblies. The staff reviewed the information and calculations presented by the applicant and found the characterization of the source term due to crud to be acceptable.

The Humboldt Bay ISFSI vault is designed to house six storage casks; five HI-STAR HB casks containing HBPP spent nuclear fuel, and one cask containing HBPP GTCC waste. In order to analyze a bounding radiation source term, the model used for shielding analyses assumes that all six storage casks contain SNF. The applicant will characterize the GTCC waste as part of the loading procedure to ensure that the calculated radiation dose rate from the GTCC waste cask does not exceed the calculated dose rate from a SNF cask and therefore, will be within the bounds of the shielding analysis provided in the SAR. As described in SAR Section 7.2.1.1, the applicant has committed to dose rate measurements of the loaded GTCC waste cask to ensure the validity of the bounding source term used in the shielding analyses. The description of the potential GTCC waste components and procedures for the storage of GTCC waste at the ISFSI is contained in SAR Sections 3.1, 3.1.1.4, 4.2.3.1, and 10.2.1.3. These sections of the SAR were reviewed in accordance with the recommendations provided to the staff in Interim Staff Guidance (ISG)-17 (U.S. Nuclear Regulatory Commission, 2001b).

The neutron source term is composed of four distinct components. These are neutrons resulting from spontaneous fission; alpha particle-neutron reactions in fuel materials; secondary neutrons produced by fission from subcritical multiplication; and gamma-neutron reactions.

The gamma and neutron source terms described previously are grouped into three components: the fuel-gamma source, the fuel-neutron source, and the nonfuel activation source. Gamma and neutron source terms were generated using the SAS2H (Hermann and Parks, 1998) and ORIGEN-S (Hermann and Westfall, 1998) modules of the SCALE 4.4 system.

The physical characteristics of the fuel used at HBPP Unit 3 and to be stored at the ISFSI are summarized in Table 3.1-2 and Section 10.2 of the SAR. Section 10.2 of the SAR provides the proposed operating controls and limits with the intact and damaged fuel assembly limits specified in Tables 10.2-1 and 10.2-2. The design basis fuel assembly chosen for the shielding analysis is the General Electric Type III. This fuel assembly was chosen because it has the highest uranium mass loading and makes up the largest percentage of the HBPP Unit 3 SNF inventory. The shielding design basis fuel assembly is described in Section 7.2.1.1 and Tables 7.2-1, 7.2-2, and 7.2-3 of the SAR. The design basis fuel assembly is specified with the minimum initial enrichment (2.09 wt% Uranium-235), the maximum burnup (23,000 MWd/MTU), and the minimum cooling time (29 years) of the fuel assemblies to be stored at the Humboldt Bay ISFSI. These specifications provide bounding source term analyses for all of the fuel assemblies proposed for storage. Regarding damaged fuel, the applicant stated in Section 10.2.1.1 of the SAR that the amount of material contained in a damaged fuel container (DFC) is limited to the equivalent of a single intact fuel assembly. Based on this statement, the staff finds the application of the calculated source term to damaged fuel to be acceptable. However, as discussed in Section 7.1.4.2 of this SER, the applicant relies upon a conservative model of the damaged fuel source to demonstrate the effect on dose rates of loading damaged fuel into a cask. The fuel-gamma source, fuel-neutron source, and nonfuel activation source discussed above are summarized in Section 7.2 and Tables 7.2-4, 7.2-5, and 7.2-6 of the SAR.

The staff evaluated the analyses of the bounding radiation source terms for the Humboldt Bay ISFSI. The staff finds that the specified design basis enrichment, burnup, and cooling time for the HBPP SNF are conservative and were determined correctly to provide a bounding source term for all fuel assemblies to be loaded into the SNF casks. Based its review of the applicant's descriptions of the crud and GTCC source terms, as discussed in this SER section, the staff also finds the analyses regarding these source terms to be acceptable.

7.1.2 Storage and Transfer Systems

7.1.2.1 Design Criteria

The design criteria for the proposed Humboldt Bay ISFSI are the regulatory dose limit requirements delineated in 10 CFR Part 20.1201(a)(1), §20.1301(a), §20.1302(b), and §72.104(a). The SAR specifies the shielding design criteria in Section 3.3.1.5.2 and Table 3.4-2. The HI-STAR HB system is designed to minimize radiation dose to workers and the public using a combination of the steel MPC-HB, overpack steel, and Holtite-A neutron shielding material. Significant shielding is also provided by the below-grade vault design of the ISFSI through its use of concrete, steel, and the surrounding soil. 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 §20.1201(a)(1), §20.1301(a), §20.1302(b), and §72.104(a). The Humboldt Bay 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 Humboldt Bay ISFSI system is designed to provide both gamma and neutron shielding for all fuel loading, transfer, and storage conditions. The shielding design features are described in Section 7.3 of the SAR. The six casks are designed to be in a single row positioned vertically in a below-grade concrete vault. ISFSI design features that ensure that dose rates are ALARA include:

- There are no radioactive systems at the ISFSI other than the GTCC cask and the overpacks containing the MPC-HB canisters.

- The MPC-HBs are shielded by the heavy-walled steel overpack (SAR Figure 3.3-3). The large mass of steel used to provide shielding by the overpack includes 21.6 cm [8.5 in] in the radial direction for gamma shielding. Additionally, there is a radial neutron shield (Holtite-A) that is a minimum of 10 cm [4 in] thick. The top and bottom of the overpack are shielded by a 15-cm [6-in] thick steel lid and the bottom forging, respectively. The shielding at the top, where there is no soil above the vault, is also enhanced by the 24.1-cm [9.5-in] thick steel lid of the MPC-HB (SAR Figure 3.3-1).

- The MPC-HBs are heavily shielded by the below-grade concrete vault (SAR Figure 3.2-1) and the surrounding soil. The vault design includes a vault lid composed of a minimum of 38-cm [15-in] thick concrete encased in inner and outer steel lid plates with a total thickness of 3.18 cm [1.25 in] of steel.

- The MPC-HB is loaded for storage and decontaminated in the HBPP Refueling Building (RFB) prior to transfer to the ISFSI. The overpack is a bolted, sealed pressure vessel that is leak tested. The MPC-HB is designed such that leakage from the confinement barrier is not credible. Confinement is evaluated in Chapter 9 of this SER.

The staff finds the shielding design features described above acceptable. The information provided in the SAR meets the requirements of 10 CFR §72.24(b) and (c)(3) and provides reasonable assurance that the shielding design features will meet the requirements of 10 CFR §72.126(a)(6) and §72.128(a)(2). The staff evaluated the radiation protection design features 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 Sections 3.3.1.5.2 and 7.3.2 of the SAR. These sections reference the HI-STAR 100 system FSAR (Holtec International, 2002a), specifically Section 5.3, as it relates to the shielding evaluation. The staff finds 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 the evaluation of shielding effectiveness for maintaining the dose rates at and around the Humboldt Bay ISFSI within regulatory limits.

7.1.3.2 Shielding Details

The shielding details are described in Section 7.3 of the SAR. The MPC-HB is heavily shielded by the overpack, the vault, and the surrounding soil. The HI-STAR HB system storage casks will be stored in a below-grade concrete vault in a 1 × 6 array of casks. The overpack has a large mass of steel and a radial neutron shield to provide gamma and neutron radiation shielding. The neutron shield of the overpack has been specifically designed as a solid Holtite-A radial shield enclosed in steel to eliminate possible neutron streaming paths created by channel-based enclosure shell designs.

The staff finds the description of the shielding details to be sufficient based on evaluating the description and drawings provided to identify the geometric arrangement and physical dimensions of sources and shielding materials. This evaluation included the description of the design features used to minimize potential gamma and neutron streaming paths (SAR Section 7.3). The staff finds that the description satisfies the requirements of 10 CFR §72.126(a)(6) and provides reasonable assurance that the radiation protection systems are 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 supporting documents (Holtec International, 2004, HI-2033047), including a reference to the methods and approach of Section 5.4 of the HI-STAR 100 System FSAR (Holtec International, 2002a). Analyses were conducted to determine the surface and 1-m [3.28-ft] dose rates for the casks, as well as dose rates at the ISFSI site boundary {16 m [53 ft]}, the 100-m [328-ft]-controlled-area boundary during cask transfer and vault loading operations, and the location of the nearest resident {247 m [811 ft]}. Also,   Section 8.2.5.3 of the SAR presents analyses conducted to calculate dose rates for accident conditions for a cask that has undergone a fire during transfer operations. The complete loss of Holtite-A radial neutron shielding in the overpack is assumed in this calculation, which represents the worst case condition for all cask accidents analyzed by the applicant.

The shielding analysis of the HI-STAR HB system casks was performed using the MCNP-4A code (Briesmeister, 1993). The MCNP code is a general-purpose, continuous-energy, coupled neutron-photon-electron Monte Carlo transport code system. The code system is able to model the complex surfaces associated with the storage casks. The individual cross-section libraries are data contained in the MCNP-4A code system and are based on the cross-section data recommended in the MCNP manual. The staff finds the use of MCNP acceptable, as discussed in NUREG-1567 (U.S. Nuclear Regulatory Commission, 2000), and agrees that the code and cross-section data used in the applicant's shielding analyses are appropriate for this application.

The flux-to-dose-rate conversion factors used in the MCNP-4A shielding calculations were from American National Standards Institute/American Nuclear Society (ANSI/ANS)-6.1.1 (American Nuclear Society Standards Committee Working Group, 1977), as verified by reviewing the MCNP shielding model input data provided by the applicant (Pacific Gas and Electric Company, 2004b). The computer code and the ANSI/ANS-6.1.1 flux-to-dose conversion factors used for shielding analyses are considered acceptable by the staff for use in shielding calculations.

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, 7.4, and 7.5 of the SAR.

The HI-STAR HB system is designed to reduce dose rates from direct radiation emanated from a loaded MPC-HB to levels that are ALARA. The design basis MPC for the shielding analysis is the MPC-HB loaded with fuel assemblies having a burnup of 23,000 MWd/MTU and a 29-year cooling time period. The contact surface dose rate for the HI-STAR HB cask (SAR Table 7.3-1) was estimated to be approximately 99 μSv/hr [9.9 mrem/hr] outside the overpack lid in its center and 83 μSv/hr [8.3 mrem/hr] at the midplane of the overpack. The dose rate at all locations adjacent to a single storage cell with the ISFSI vault lid installed is estimated to be less than 1.5 μSv/hr [0.15 mrem/hr].

To assess onsite and offsite doses from direct radiation emanating from the SNF stored at the ISFSI, the applicant employed an approach described in Sections 7.4 and 7.5 of the SAR. The onsite assessment applies the calculated dose rates for the MPC-HB locations shown in Figure 7.3-1 of the SAR. The dose rate versus distance calculations used for onsite and offsite dose assessments for the storage phase of ISFSI operations were conducted using a simplified model, in which a single storage cell in the ISFSI vault contains a fully loaded HI-STAR HB cask. The single vault cell is modeled with reflective boundary conditions on both sides at a distance halfway between storage cells such that an infinite single line of cells is modeled as a conservative calculation of the 1 × 6 array of storage cells in the actual ISFSI vault. This approach models the single GTCC storage cask as a SNF storage cask to provide a bounding analysis, as discussed in Section 7.1.1.1 of this SER.

Table 7.4-1 of the SAR provides the estimated occupational exposures to the HBPP personnel during the operational phases of ISFSI operation including (i) loading SNF into the MPC-HB contained in the overpack, (ii) decontaminating the MPC-HB and overpack, (iii) transferring the HI-STAR HB cask from the RFB to the ISFSI vault, (iv) transferring the HI-STAR HB cask into a storage cell of the vault, and (v) closing the storage cell lid. Tables 7.4-1 and 7.4-2 of the SAR provide a list of the operational steps involved in loading and unloading an overpack and MPC-HB. These tables include the estimated number of personnel, dose rates, and time for each operational task. The number of personnel and operation duration estimates were based on industry experience with the Holtec HI-STAR and HI-STORM cask systems. The estimated dose from loading, transfer, and emplacement into the ISFSI vault of a single HI-STAR HB cask was 5.68 person-mSV [567.98 person-mrem]. When compared to similar systems, these estimates are low because of the long cooling time of the HBPP fuel.

Section 7.4 of the SAR discusses the dose estimates for routine maintenance operations with a summary presented in Table 7.4-3 of the SAR. The annual occupational exposure from ISFSI walkdowns was estimated to result in a dose of 92 person-μSv [9.2 person-mrem]. The estimated annual exposure for overpack repair activities was estimated at 36 person-μSv [3.6 person-mrem]. The staff reviewed the occupational dose estimates and found them acceptable. Based on these estimates, there is reasonable assurance that personnel exposures will be below the annual occupational dose limit of 0.05 Sv [5 rem] specified in 10 CFR §20.1201(a)(1).

The preceding analyses are limited to a HI-STAR HB cask loaded with intact spent fuel. However, damaged fuel may be loaded into the cask in the two different configurations shown in Figures 10.2-1 and 10.2-2 of the SAR. The applicant referred to an analysis performed for the HI-STORM 100 (Holtec International, 2002b) to demonstrate the effects of damaged fuel on cask dose rates. The results of that HI-STORM 100 analysis showed the dose rate above the cask to change negligibly and the dose rate at the side of the cask to increase by less than 20%.

The staff reviewed the HI-STORM 100 analysis, including the modeling of the damaged fuel source and the analyzed loading pattern. Based on this review and the applicant's statement in Section 10.2.1.1 of the SAR restricting the amount of damaged fuel material in a DFC to that equivalent to an intact fuel assembly, the staff finds the modeling technique for the damaged fuel source to be applicable to the HI-STAR HB evaluation. The staff also finds the results of the HI-STORM 100 analysis to be applicable to a HI-STAR HB cask loaded according to Figure 10.2-1 of the SAR. However, the HI-STORM 100 analysis is not bounding for a HI-STAR HB cask loaded according to Figure 10.2-2 of the SAR. For this case, the staff estimated that dose rates above the cask would have a non-negligible increase. Using this estimate, and the estimate for the increase in cask side dose rates from the HI-STORM 100 analysis, the staff determined that the dose rates from a HI-STAR HB cask loaded with damaged fuel would be greater than those estimated for a cask loaded with intact fuel only. However, the resulting increase in doses would still be within regulatory limits. The staff considers this analysis, particularly the model of the damaged fuel source term, to be conservative and to provide a bounding estimate of the dose rates resulting from a cask loaded with damaged fuel.

The staff evaluated the radiation dose analyses and the SAR shielding calculations and found them to be acceptable. The staff finds that there is reasonable assurance that the dose rates at the onsite and offsite locations will be below the limits specified in 10 CFR §20.1201(a)(1), §20.1301(a), and §72.104(a). The description in the SAR, combined with the review of input and output files (Pacific Gas and Electric Company, 2004b), provides reasonable assurance that the HI-STAR HB shielding was adequately evaluated. The MCNP input and output files were reviewed an found to be consistent with the description of the shielding model and the results provided in the SAR. Chapter 11 of this SER discusses the overall onsite and offsite dose rates from the Humboldt Bay ISFSI estimated from the combined radiation exposure to direct radiation and potential radioactive effluents. The staff has reasonable assurance that compliance with 10 CFR §20.1201(a)(1), §20.1301(a), §20.1302(b), and §72.104(a) will be achieved by means of the radiation protection design and radiological protection program described in the SAR and evaluated in Chapter 11 of this SER. Based on this finding, the staff has reasonable assurance that ALARA objectives will be met.

7.1.5 Confirmatory Calculations

The staff independently calculated the bounding source terms for the stored fuel at the proposed Humboldt Bay ISFSI. Neutron and gamma source terms, as well as the hardware Cobalt-60 radionuclide inventory, were generated using the ORIGEN-ARP module (Gauld, et al., 2004) of the SCALE Version 5 code. ORIGEN-ARP does not provide a default library for a BWR 6 × 6 fuel assembly geometry; therefore, the confirmatory source terms were generated using the default BWR 7 × 7 fuel assembly geometry library, and the design-basis fuel assembly parameter values provided in Table 7.2-1 of the SAR. This was done for ease of calculation using the ORIGEN-ARP module, as well as to provide confirmatory analysis of the General Electric Type II 7 × 7 fuel assembly also present in the SNF inventory at the HBPP. The staff independently confirmed the results provided in Tables 7.2-4 through 7.2-6 of the SAR. The confirmatory calculations provide reasonable assurance that design basis neutron and gamma source terms for the HI-STAR HB system are accurate and acceptable for the shielding analyses.

The staff independently calculated the dose rates that could be expected around the storage casks and annual doses at the boundary of the Humboldt Bay ISFSI controlled area. A distance to the public trail of 15 m [50 ft] and an occupancy time of 2,080 hours per year was assumed for comparison to the applicant's analyses. The staff used the SCALE Version 5 code system SAS4 module (Tang and Emmett, 2004), the 27N-18COUPLE cross-section library supplied with the code, and neutron and gamma flux-to-dose conversion factors from ANSI/ANS-6.1.1 (American Nuclear Society Standards Committee Working Group, 1977). The SAS4 control module performs a three-dimensional Monte Carlo shielding analysis of a nuclear fuel transport or storage container using an automated biasing procedure. The coupled 27 neutron group, 18 gamma group (27N-18COUPLE) library based on ENDF/B-IV data is widely used in light water reactor SNF shielding calculations and has been validated against experimental data (Jordan, et al., 2004). The fuel assemblies were modeled within the MPC-HB basket cell as a homogenized fuel pellet and cladding material assembly. The SNF cask and vault geometry and associated material properties were modeled explicitly and developed using the dimensions and properties provided in Tables 7.2-1 and 7.2-2 of the SAR and the features and specifications that were discussed previously in Sections 7.1.2.2 and 7.1.3 of this SER.

The staff's analysis was conducted to confirm the applicant's calculations for normal conditions and the transporter fire accident scenario. A single cask geometry was considered, and the dose rates at various distances were computed, including a calculation assuming that no overpack Holtite-A neutron shielding was present for the fire accident scenario during cask transfer. Annual doses were computed from the dose rates for the ISFSI vault geometry. The annual dose was calculated for an individual located slightly inside {15 m [50 ft]} the nearest boundary of the ISFSI controlled area for a conservative occupancy time on the public trail of 2,080 hours per year. The dose rates and annual doses presented in Tables 7.3-1 and 7.5-1 through 7.5-3 of the SAR were independently confirmed by the staff through shielding calculations performed using the SCALE Version 5 code. The dose rates for the fire accident scenario discussed in Section 8.2.5.3 of the SAR were also confirmed.

The staff's calculations confirmed the onsite dose rates estimated by the applicant, providing reasonable assurance that the requirements of 10 CFR §20.1201(a)(1), §20.1301(a), and §20.1302(b) will be met. The applicant has an established radiation protection program, as required by 10 CFR Part 20, Subpart B. This program will be used to meet ALARA objectives and demonstrate compliance with dose limits to members of the public by evaluations and measurements. The staff calculations also confirmed that the offsite dose rates will be less than the 0.25-mSv/yr [25-mrem/yr] whole-body dose allowable to any real individual located beyond the controlled area, as required by 10 CFR §72.104(a). Based on these confirmatory calculations, the staff finds that the applicant's shielding analysis is acceptable.

7.2 Evaluation Findings

The staff made the following findings regarding the shielding evaluation of the Humboldt Bay ISFSI:

- The design and description of the shielding system in the SAR satisfy the criteria for radiological protection of 10 CFR §72.24(b), §72.24(c)(3), §72.126(a)(6), and §72.128(a)(2).

- The design of the Humboldt Bay ISFSI provides acceptable means for controlling occupational radiation exposures within the limits given in 10 CFR §20.1201(a)(1) and for meeting the objective of maintaining exposures ALARA.

- The design of the Humboldt Bay ISFSI provides acceptable means for controlling exposures of the public to direct radiation within the limits given in 10 CFR §72.104(a), §20.1301(a), and §20.1302(b).

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

7.3 References

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