7 SHIELDING EVALUATION

7.1 Conduct of Review

The shielding evaluation includes a review of the information in Chapter 7 (Private Fuel Storage Limited Liability Company, 2000) of the SAR. Chapter 7 of the SAR describes the radiation protection features of the Facility that ensure that radiation exposures to workers and to the public meet NRC regulatory criteria and are maintained ALARA. This chapter also evaluates radiation doses to the public and to workers from the operation of the Facility. Relevant information in Chapter 3, Principal Design Criteria, and Chapter 4, Facility Design, of the SAR was also considered.

The shielding evaluation of the PFS Facility is based on the use of the HI-STORM 100 Cask System, which has been approved by the NRC for use under the general license provisions of 10 CFR Part 72 (Nuclear Regulatory Commission, 2000a). Cask-specific information presented in the HI-STORM 100 FSAR (Holtec International, 2000) was also reviewed as it pertained to the shielding and radiation protection aspects of the ISFSI. Cask-specific information already documented in the staff's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b) will not be repeated in this SER.

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

- 10 CFR 72.24(b) requires that the SAR describe the ISFSI structures with special attention to design and operating characteristics.
- 10 CFR 72.24(c)(3) requires that the SAR describe all structures, systems, and components important to safety.
- 10 CFR 72.24(e) requires that SAR describe the means of controlling and limiting occupational radiation exposures, within the limits given in 10 CFR Part 20, to ALARA.
- 10 CFR 72.104(a) requires that the annual dose equivalent to any real individual located beyond the controlled area be limited to 25 mrem/yr to the whole body, 75 mrem/yr to the thyroid, or 25 mrem/yr to any other organ during normal operations and anticipated occurrences.
- 10 CFR 72.106(b) requires that the dose to any individual located beyond the controlled area be no greater than 5 rem to the whole body or any organ from any design basis accident.
- 10 CFR 72.126(a)(6) requires that structures, systems, and components be designed, fabricated, located, shielded, controlled, and tested to control external and internal radiation exposures to personnel.

- 10 CFR 72.128(a) requires that spent fuel storage systems be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with suitable shielding under normal and accident conditions.

- 10 CFR 20.1101 requires that doses to workers and members of the public be reduced to levels that are ALARA

- 10 CFR 20.1201 requires licensees to control occupational dose such that radiation workers at the site receive a total effective dose equivalent of less than 5 rem/yr, a sum of the deep dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye of less than 50 rem/yr, a lens dose equivalent of less than 15 rem, and a shallow dose equivalent to skin or any extremity of 50 rem.

- 10 CFR 20.1301(a) requires that each licensee conduct operations so that (1) the total effective dose equivalent to individual members of the public from licensed operations does not exceed 0.1 rem in a year, and (2) the dose in any unrestricted area from external sources does not exceed 0.002 rem in any one hour.

- 10 CFR 20.1302(b) contains requirements for how a licensee may demonstrate that the dose limit in 10 CFR 72.1301 will be met.

7.1.1 Contained Radiation Sources

The source of gamma and neutron radiation is the spent fuel stored in the HI-STORM 100 Cask System. As specified in Technical Specification 2.1.1, the spent fuel that will be stored in the storage casks is limited to the approved contents specified in Section 2.0 of Appendix B to NRC Certificate of Compliance No. 72-1014 for the HI-STORM 100 Cask System. The burnup and cooling time limits specified for intact PWR fuel in the Certificate of Compliance range from 33,300 MWD/MTU for 5 years to 44,700 MWD/MTU for 15 years. The burnup and cooling time limits specified for intact BWR in the Certificate of Compliance range from 29,900 MWD/MTU for 5 years to 41,000 MWD/MTU for 15 years. A detailed assessment of the radiation sources, specific radiological source terms, and calculation methods for this fuel are provided in Chapter 5 of the HI-STORM 100 FSAR, which the staff has previously reviewed and found acceptable in the HI-STORM 100 SER. The PFS Facility SAR references the HI-STORM 100 FSAR and provides a description of the radiation sources and calculation methods used in the site-specific dose calculations for the Facility.

The applicant evaluated discharged spent fuel inventory data gathered by the Department of Energy (U.S. Department of Energy, 1996) to determine appropriate cask-average burnup and cooling times for site-specific dose estimates at the Facility. The applicant evaluated the data and calculated weighted-average burnups of 32,400 MWD/MTU for PWR spent fuel and 23,800 MWD/MTU for BWR spent fuel which will be stored at the Facility. The applicant calculated a weighted-average cooling time of 23 years for all spent fuel that is stored at the Facility, assuming that 200 casks are loaded each year at the Facility. Based on this information, the applicant used PWR fuel (in the MPC-24 canister configuration) with a cask-average burnup and cooling time of 35,000 MWD/MTU for 20 years to calculate average on-site occupational exposure estimates from the average fuel that is expected at the Facility. The applicant used a bounding, cask-average burnup and cooling time of 40,000 MWD/MTU for 10 years to calculate bounding off-site dose estimates to members of the public.

The staff finds the description of radiation sources and calculation methods to be consistent with the information approved in the HI-STORM 100 FSAR. The use of the PWR design basis fuel in the MPC-24 canister configuration for generic dose calculations is acceptable because external dose rates from the PWR and BWR fuel in their respective canister designs are similar, as shown in the HI-STORM 100 FSAR. Based on the fuel inventory analysis presented by the applicant, the average and bounding burnup and cooling times used for subsequent calculations of average on-site occupational exposures and bounding off-site dose rates are acceptable. Actual dose rates during operation of the Facility will be measured by active and passive radiation monitoring in order to verify compliance with the radiological limits in 10 CFR Parts 20 and 72. The applicant will also operate the Facility under a Radiation Protection Program as required in Technical Specification 5.5.3 to assure that radiation fields are continually monitored and radiation doses to workers and members of the public are maintained ALARA, as actual dose information is gathered during operations. Radiation monitoring at the Facility and the Radiation Protection Program are evaluated in Sections 11.1.2 and 11.1.4 of this SER.

7.1.2 Storage and Transfer Systems

7.1.2.1 Design Criteria

The shielding design criteria for the Facility are described in Sections 3.3.5.2, 4.2.1.5.3, and 7.3 of the SAR; these sections reference the HI-STORM 100 FSAR. The HI-STORM 100 storage cask is designed to limit the average external contact dose rates (gamma and neutron) to 40 mrem/hr on the sides, 10 mrem/hr on top, and 60 mrem/hr at the air inlets and outlets, based on design basis fuel. The transfer cask is designed to reduce dose rates from a loaded canister to ALARA levels. The staff finds the use of these design criteria for the Facility to be appropriate. These design criteria provide reasonable assurance that the Facility will meet the dose limits specified in 10 CFR 72.104(a) and 10 CFR 72.106(b). Additionally, these design criteria provide reasonable assurance that the Facility will provide adequate safety based on the use of sufficient shielding in accordance with 10 CFR 72.128(a)(2).

7.1.2.2 Design Features

The storage and transfer system shielding design features are described in Section 7.3 of the SAR. Facility design features that ensure that dose rates are ALARA and within regulatory limits include:

- The only source of radiation, that can lead to significant exposure to workers or members of the public at the Facility, are the sealed canisters containing spent fuel assemblies, which will always be shielded by a shipping, storage, or transfer cask.

- The shipping, transfer, and storage casks are heavily shielded to minimize external dose rates. The storage cask consists of thick layers of steel and concrete. The transfer cask is composed of layers of steel, lead, and neutron-shielding materials.

- The PFS Facility site layout provides substantial distance (at least 2,130 ft) between the cask storage area and the restricted area boundary, thereby minimizing radiation exposures to members of the public located beyond the restricted area boundary.

- The Canister Transfer Building is located inside the restricted area, which minimizes the route between the handling facility and storage pad and maintains substantial distance (500 m or 1,650 ft) from the controlled area boundary.

The description of the Facility storage and transfer systems satisfies the requirements of 10 CFR 72.24(b), (c)(3), and (e) because the design of the shielding components important to safety and the means for controlling and limiting occupational radiation exposures and for meeting ALARA goals are sufficiently described. The Facility shielding design features satisfy the requirements of 10 CFR 72.126(a)(6) because they include heavy shielding to minimize personnel radiation exposure. The SAR provides sufficient information to determine whether the requirements of 10 CFR 72.128(a)(2) are met; this information includes a description of the shielding materials that will provide radiation protection under normal and accident conditions.

The staff finds the description of the storage and transfer system shielding design features to be sufficient. Based on this description, the effectiveness of the shielding design features in limiting dose rates around the Facility to the values specified in 10 CFR Parts 20 and 72 can be evaluated. This evaluation is provided in Section 7.1.4.2 and Chapter 11 of this SER.

7.1.3 Shielding Composition and Details

7.1.3.1 Composition and Material Properties

The shielding composition and material properties are described in Sections 4.2.1.5.3 and 7.3.3 of the SAR. These sections reference the HI-STORM 100 FSAR. The primary shielding during storage will be from the concrete and steel in the HI-STORM 100 storage cask. The primary shielding during transfer operations will be from the steel, lead, and neutron shield in the HI-TRAC transfer cask. The detailed properties of the materials included in the shielding model are given in the HI-STORM 100 FSAR.

The staff finds that the description of the shielding composition and details are sufficient to meet the requirements of 10 CFR 72.24(b) and 10 CFR 72.24(c)(3) by describing the shielding components important to safety. The description of material composition, density, and geometry are described in sufficient detail to evaluate the effectiveness of the shielding in reducing the dose rates around the Facility to within regulatory limits.

7.1.3.2 Shielding Details

The details of the shielding are described in Section 7.3.3 of the SAR. The ISFSI will consist of approximately 4,000 storage casks containing a total of up to 40,000 MTU of spent nuclear fuel. The storage casks will be placed on a concrete storage pad in a 2 × 4 array (up to eight storage casks per pad). The concrete pads will be arranged in a 20 × 25 array. This arrangement maximizes the amount of shielding that the outermost casks provide to the casks in the middle of the array.

The HI-STORM 100 FSAR provides a detailed description of the HI-STORM 100 storage cask and HI-TRAC transfer cask. Radially, the HI-STORM 100 storage cask provides 26-3/4 inches of concrete shielding and 2-3/4 inches of steel shielding. Axially, the storage cask provides 10-1/2 inches of concrete shielding and 5-1/4 inches of steel shielding in the storage cask lid. The storage cask vent ports are designed with sharp bends to avoid radiation streaming out of these ports. In calculating the dose rates around the Facility, the vent ports have been explicitly modeled. The HI-TRAC transfer cask is equipped with heavy neutron and gamma shielding to reduce the dose rates from a loaded canister to ALARA levels.

The description of the shielding composition and details satisfy the requirements of 10 CFR 72.126(a)(6). The radiation protection systems that will shield onsite personnel from radiation exposure have been sufficiently described.

7.1.4 Analysis of Shielding Effectiveness

7.1.4.1 Computational Methods and Data

The computational methods and data used to analyze the effectiveness of the shielding at the Facility are described in Section 7.3.3.2 of the SAR and in the HI-STORM 100 FSAR. Analyses were conducted to determine the dose rates close to the transfer cask and the dose rates both close to and far from the storage casks.

The shielding analysis of the HI-STORM casks was performed using the computer code MCNP 4A (Los Alamos National Laboratory, 1995). MCNP 4A is a three-dimensional transport code that uses Monte Carlo techniques with a combinatorial geometry modeling capability able to model the complex surfaces associated with the storage casks. Continuous energy cross-sectional data from ENDF/B-V (Los Alamos National Laboratory, 1994) were used by the computer code to determine gamma and neutron cross sections. The gamma flux-to-dose conversion factors used in the PFS Facility SAR were from ANSI/ANS-6.1.1 (American Nuclear Society Standards Committee Working Group, 1977).

The computer codes employed by the applicant and cask vendor (Holtec International) are widely used for shielding analyses and are considered acceptable by the staff for use in modeling the shielding configurations and materials at the Facility. The ANSI/ANS-6.1.1 flux-to-dose conversion factors are acceptable values for use in the shielding evaluations.

7.1.4.2 Dose Rate Estimates

The estimates of dose rates at various locations on the site and beyond the edge of the restricted area site are described in Sections 7.3.3.3, 7.3.3.4, and 7.3.3.5 of the SAR.

The HI-TRAC transfer cask is designed to reduce dose rates from a loaded canister to ALARA levels. All transfers of spent fuel canisters will occur remotely within the heavily shielded

Canister Transfer Building, which will cause offsite doses from these operations to be negligible.

The applicant calculated off-site dose rates for the HI-STORM 100 based on PWR design basis fuel source terms with a burnup and cooling time of 40,000 MWD/MTU for 10 years as discussed in Section 7.1.1 of this SER. The applicant calculated average contact surface dose rates for the HI-STORM 100 storage cask to be approximately 10 mrem/hr at the sides, 3 mrem/hr on top, and 6 mrem/hour at the vents. Based on these values, the applicant calculated a site boundary dose rate of 0.0029 mrem/hr for 4,000 casks from direct and scattered radiation exposure. As discussed in Chapter 9 of this SER, no release of radioactive material in effluent is expected during normal operations; therefore, the dose due to effluents is not considered. The applicant extrapolated the site boundary dose rate out to a distance of two miles and calculated an annual dose of 0.0356 mrem to the nearest resident, assuming the resident is continually present for 8,760 hr/yr. The applicant also calculated an annual dose of 5.85 mrem for a hypothetical person at the site boundary (e.g., non-Facility worker), assuming the person is at the site boundary for 2,000 hr/yr which is approximately equal to 40 hr/week. These dose rates are less than the 10 CFR 72.104(a) dose limit of 25 mrem/yr to the whole body to a member of the public.

No accidents were identified in the SAR that could cause significant loss of shielding or complete penetration of the storage casks. The worst accident identified was the tornado missile impact. This design basis accident could cause localized thinning of the storage cask shielding. However, only a small number of casks would be damaged and much of the shielding would remain intact. Therefore, the offsite dose due to direct and scattered radiation would not increase significantly as a result of this accident. Further, in Chapter 9 of this SER, the dose due to a hypothetical release from a single canister is estimated to be 2.68 mrem (to an individual continuously present at the owner controlled area boundary for 30 days). Thus, there is reasonable assurance that the total dose (i.e., dose due to direct and scattered radiation and to a hypothetical release) from any design basis accident will be less than the 10 CFR 72.106(b) limit of 5 rem.

Based on the results of the applicant's shielding analysis, the staff finds that the dose rates, at locations onsite and offsite, are below the limits specified in 10 CFR 20.1201, 20.1301, 20.1302, and 72.104(a). As discussed in Section 7.1.1 of this SER, the burnup and cooling parameters are acceptable for demonstrating the shielding capability of the Facility. The applicant's shielding analysis satisfies the requirements of 10 CFR 72.24, 72.126, and 72.128 by demonstrating that radiation exposures to workers and members of the public will be adequately limited through the use of shielding at the Facility. The shielding analysis also demonstrates that no credible accident will significantly increase the dose rates. Therefore, there is reasonable assurance that under accident conditions, dose rates will remain below the limits specified in 10 CFR 72.106(b).

Chapter 11 of the SER evaluates the combined radiological exposure from the direct and scattered radiation doses and any potential radioactive materials in effluents. The purpose of the evaluation is to verify the radiation protection design of the Facility and that its radiological protection program satisfies the occupational exposure and public dose requirements in 10 CFR 20.1201, 20.1301, 20.1302, 72.104(a), and 72.106(b).

7.1.5 Confirmatory Calculations

The staff performed independent calculations of the dose rates that could be expected around the storage casks and at the edge of the PFS Facility controlled area. The staff used the MCNP 4A code (Los Alamos National Laboratory, 1995), ENDF/B-VI cross section data (Los Alamos National Laboratory, 1994), and gamma flux-to-dose conversion factors from ANSI/ANS 6.1.1 (American Nuclear Society Standards Committee Working Group, 1977). These calculations confirmed the onsite dose rates calculated by the applicant and also confirmed that the offsite dose rate would be less than the 25 mrem/yr whole body dose allowable to a member of the public as required by 10 CFR 72.104. Based on its confirmatory calculations, the staff finds the applicant's shielding analysis to be acceptable.

7.2 Evaluation Findings

Evaluation of shielding at the Facility assumed that only the HI-STORM 100 Cask System will be used. Based on the staff's review of the SAR, the staff finds that the requirements of 10 CFR 72.104(a) and 72.106(b) are met with respect to dose rates due to direct and scattered radiation. The staff also finds that the other applicable requirements of 10 CFR Parts 20 and 72, as identified in Section 7.1 of this SER, have been satisfied. The staff found that: (1) the description of the contained radiation sources of the Facility is sufficient to determine that the Facility will satisfy all radiological protection criteria; (2) the design of the storage and transfer systems at the Facility is adequate to maintain exposures to ALARA and within applicable regulatory dose limits; and (3) the shielding features of the Facility are adequate to ensure that radiation exposures to workers and to the public are within the applicable regulatory limits under normal and accident conditions.

7.3 References

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