7 SHIELDING EVALUATION

7.1 Conduct of Review

This review was performed to determine if the shielding design features of the proposed Idaho Spent Fuel (ISF) Facility meet U.S. Nuclear Regulatory Commission criteria for radiation protection of workers and the public against direct radiation emanating from the materials to be stored during all operations at the facility. The shielding evaluation includes a review of the information in Chapter 7, "Radiation Protection," of the Safety Analysis Report (SAR) (Foster Wheeler Environmental Corporation, 2003a) and relevant sections of Chapter 3, "Principal Design Criteria;" Chapter 4, "Facility Design;" and Appendix A, "Safety Evaluation of DOE–ID Provided Transfer Cask," of the SAR and Foster Wheeler Environmental Corporation Responses to U.S. Nuclear Regulatory Commission Requests for Additional Information (Foster Wheeler Environmental Corporation, 2003b,c). Also considered was "TRIGA Spent Nuclear Fuel Data for Transfer from IFSF to the Spent Nuclear Fuel Dry Storage Project" (Foster Wheeler Environmental Corporation, 2002).

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 Independent Spent Fuel Storage Installation (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 is reasonably achievable (ALARA).

- 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 m [328 ft].

- 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. This design must provide 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 0.05 Sv [5 rem] 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 0.5 Sv [50 rem]. (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 0.15 Sv [15 rem] and (ii) a shallow-dose equivalent of 0.5 Sv [50 rem] 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 1 mSv [0.1 rem] 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 0.02 mSv [0.002 rem] in any one hour.

7.1.1 Contained Radiation Source

The sources of gamma and neutron radiation are the spent nuclear fuel (SNF) from Training, Research, and Isotope reactors built by General Atomics (TRIGA); and Peach Bottom Unit 1, Cores 1 and 2; as well as Shippingport reflector modules and rods arriving at the ISF Facility. These contents will be transferred within the ISF Facility, reloaded into sealed storage canisters, and stored in the ISF Facility vault storage. These materials are described in Section 3.1.1, "Material To Be Stored" of the SAR. The Peach Bottom Unit 1 reactor Cores 1 and 2 fuel compacts are similar to each other and consist of carbides of thorium and uranium initially enriched to 93.2 weight percent of U-235. The Peach Bottom Core 1 operation was terminated in October 1969 with the equivalent burnup level of 30,795 MWD/MTHM. The Peach Bottom Core 2 operated until October 1974 to a burnup of 72,717 MWD/MTHM. The Shippingport Light Water Breeder Reactor operated more than 29,000 effective full-power hours until it was shut down in 1983. The limits on TRIGA fuel initial enrichment and burnup are specified as 20 weight percent of U-235 and 34,103 MWD/MTU. A detailed assessment of the radiation sources, specific radiological source terms, and calculational methods for these fuels is provided in Section 7.2 of the SAR.

The ISF Facility is proposed to provide storage for 1,601.5 Peach Bottom reactor SNF elements; 2,971 SNF rods from the Shippingport reactor; and approximately 1,600 TRIGA reactor SNF elements.

The applicant analyzed radionuclide inventory data for SNF and reflector modules and rods obtained from the U.S. Department of Energy - Idaho Operations Office (DOE-ID) via the ISF contract (DOE-ID, 2000) to determine the burnups and cooling times for site-specific dose estimates at the ISF Facility. To generate a TRIGA source term (Sterbentz, 1997), the U.S. Department of Energy used the Monte Carlo N-Particle (Los Alamos National Laboratory, 1994), MCNP–ORIGEN2 Coupled Utility Program (Babcock, et al., 1994) and the ORIGEN2 Version 1 (Radiation Shielding Information Center, 1991) codes. For its analyses, the applicant adjusted the date of the U.S. Department of Energy inventory for radiation decay to July 1, 2004, the projected ISF Facility operational start date. The applicant used the ORIGEN2 code (Croff, 1980) to perform this decay date adjustment and did not conduct its own independent depletion and decay analyses of the fresh fuels.

The TRIGA stainless steel-clad SNF was selected by the applicant for shielding analyses because it provides the bounding photon flux for all fuel types and its inventory in Section C, Attachment C-A-A, Table 3.2 (U.S. Department of Energy–Idaho Operations Office, 2000). The applicant used 34,103 MWD/MTU burnup for a 4-year long reactor campaign at steady power and 6.5-year long cooling period to determine radionuclide inventory and radiation source terms. The axial burnup profile of the TRIGA fuel was accounted for by multiplication of the total photon intensity per assembly by a peaking (scaling) factor of 2.62. For the TRIGA fuel source term description, the applicant used the highest, most conservative value (2.62) in its analyses. This value, however, was not used in the applicant's updated shielding analyses (Foster Wheeler Environmental Corporation, 2003c). Further discussion about the consideration of the SNF rods peaking (scaling) factor follows in Section 7.1.4.2 of this SER. Because the actual TRIGA fuel loading will include individual fuel elements that have varying cooling times, uranium enrichment/loading, and burnup, a bounding decay heat rate corresponding to the TRIGA fuel source term was determined, since it is proportional to the level of radioactive decay and is a function of reactor power history, initial enrichment, and cooling time. For TRIGA fuel, 36 watts decay heat production per ISF canister corresponds to the design basis source term. This limit for the TRIGA fuel will be included in ISF Facility Technical Specification 2.1.

The staff finds acceptable the description of radiation sources, calculational methods, the bounding burnups and cooling times used for subsequent calculations of average onsite occupational exposures and bounding offsite dose rates. The staff conducted independent, confirmatory calculations of the ISF Facility photon source term using a stainless steel-clad TRIGA fuel element containing 800 g [1.76 lb] of SS-304 per element with a 2,200 ppm of Co-59 impurity level. The SCALE4.4A/SAS2H (Hermann and Parks, 2000) and ORIGEN-ARP Version 2.0 (Bowman and Leal, 2000) computer codes and the 44-group ENDF/B-V neutron cross-sectional library, which contains data collapsed from the fine group 238-group ENDF/B-V library (Greene, et al., 1994), were used for the calculations. The photon source term obtained from the confirmatory calculations is in close agreement with that of the applicant. The neutron dose rate as estimated by the applicant is expected to be lower by several orders of magnitude than the gamma dose rate at the same receptor locations.

7.1.2 Storage and Transfer Systems

The two existing Peach Bottom transfer casks provided by the DOE-ID will be used to transfer the SNF transfer from the Idaho Nuclear Technology and Engineering Center to the ISF Facility. These two Peach Bottom casks were previously certified for use as transportation casks in accordance with 10 CFR Part 71 by the U.S. Atomic Energy Commission (1974) in September 1974; the Certificate of Compliance expired in March 1982. The casks are described in Appendix A of the SAR.

7.1.2.1 Design Criteria

The design criteria for the proposed ISF Facility are the regulatory dose limit requirements delineated in 10 CFR Part 20, §72.104(a), and §72.106(b). The ISF Facility SAR specifies the shielding design criteria in Sections 3.3.5, 7.1.2, 7.3.2, and 7.4.2 and in Tables 7.4-1 and 7.4-2. Maintaining doses as low as is reasonably achievable (ALARA) is an ISF Facility design constraint. The staff finds the use of these design criteria to be appropriate. These design criteria provide reasonable assurance that the ISF Facility will meet the dose limits delineated in 10 CFR Part 20, §72.104(a), and §72.106(b). Additionally, these design criteria provide reasonable assurance that the ISF Facility 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 ISF Facility is designed to provide gamma and neutron shielding for all fuel loading, transfer, and storage conditions. The storage system and transfer system shielding design features are described in Sections 7.3 and 7.4.2 of the SAR. Facility design features that ensure that dose rates are ALARA and within regulatory limits include:

- The Peach Bottom transfer cask is heavily shielded to minimize external dose rate. The transfer cask consists of the 15.88-cm [6.25-in] thick layer of lead in the cask central section, the 13.33-cm [5.25-in] thick layer of lead at the cask ends, the 3.8-cm [1.5-in] thick mild steel clad of outer shell, and the 0.64-cm [0.25-in] thick stainless steel inner shell. The top and bottom lids consist of 10.2 cm [4 in] of lead sandwiched between two 3.8-cm [1.5-in] thick stainless steel plates.

- The concrete walls of the Fuel Packaging Area (FPA) cell where fuel will be reloaded from transfer casks into storage canisters, are 1.22 m [4 ft] thick.

- The concrete wall separating the storage area and Transfer Tunnel is 0.91 m [3 ft] thick.

- The ISF Facility site layout provides substantial distance between the fuel handling and storage vault areas and the controlled area boundary {at least 12.9 km [8 mi]}, thereby minimizing radiation exposures to members of the public located beyond the restricted area boundary.

The staff finds that the description of the design of the shielding components important to safety and the means for controlling and limiting occupational radiation exposures and for meeting ALARA goals for the ISF Facility storage and transfer systems is sufficient and satisfies the requirements of 10 CFR §72.24(b), §72.24(c)(3), and §72.24(e). The staff also finds that the ISF Facility shielding design features satisfy the requirements of 10 CFR §72.126(a)(6) because they include heavy shielding to minimize personnel radiation exposure. Furthermore, the SAR provides sufficient information to demonstrate that 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 for normal, off-normal, and accident conditions.

The effectiveness of the shielding design features in limiting dose rates around the ISF Facility below the values specified in 10 CFR Parts 20 and 72 is evaluated 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 Section 7.3 and Table 7.3-2 of the SAR. The primary shielding during transfer operations will be from the steel and lead shield of the Peach Bottom transfer cask, from the steel and neutron absorbing material, Jabroc-N, used in the Canister Handling Machine (CHM), and from the concrete of the walls surrounding the FPA, Canister Closure Area (CCA) and Transfer Tunnel. The primary shielding during fuel repackaging operations will be from the FPA concrete walls. The primary shielding during storage will be from the concrete of the storage area walls. Concrete modeled in the ISF Facility shielding analysis has a density of 2.4 g/cm^3 [149.8 lb/ft^3] and is used for radiation shielding of nuclear power plants (American National Standards Institute, 1997). No temperature effect on shielding properties of the materials is expected by the applicant.

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 §72.24(c)(3). The staff further determines that the description of material composition, density, and geometry is in sufficient detail for evaluating the effectiveness of the shielding in maintaining the dose rates around the ISF Facility to within regulatory limits.

7.1.3.2 Shielding Details

The details of the shielding are described in Section 7.3.2 of the SAR. The ISF Facility storage area will house 246 metal storage tubes sealed and filled with inert gas with a single ISF canister in each tube. The storage area will consist of two adjacent storage vaults where tubes will be arranged in one 6 × 17 array and one 8 × 18 array along the Fuel Transfer Tunnel wall. Radially, the storage area provides at least 0.91 m [3 ft] of concrete shielding, and the FPA provides 1.22 m [4 ft] of concrete shielding. The SAR identified three potential gaps in shielding: the gap between the canister and canister collet, the gap between the shield plug inside the canister and the inner wall of the canister, and the gap between the Transfer Cask and the FPA cask port. The CHM shielding consists of 30.48 cm [12 in] steel and has an outer layer of 10.2 cm [4 in] of special radial neutron shielding manufactured of the proprietary Jabroc-N material.

The Peach Bottom transfer cask provides 15.88 cm [6.25 in] of lead in the cask central section {279 cm [110 in] long}, 13.34 cm [5.25 in] of lead at the cask ends {76.2 cm [30 in] long each}, 3.8-cm [1.5-in] thick mild steel clad outer shell, and 0.64-cm [0.25-in] thick stainless steel inner shell. The top and bottom lids provide 10.2 cm [4 in] of lead sandwiched between two 3.8-cm [1.5-in] thick stainless steel plates. The Peach Bottom cask has no special neutron shielding.

The staff evaluated the description of the shielding composition and details and finds that it satisfies the requirements of 10 CFR §72.126(a)(6). The staff also finds that 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 ISF Facility are presented in Section 7.3.2 and Tables 7.4.1 and 7.4.2 of the SAR and in Foster Wheeler Environmental Corporation's response to the staff's request for additional information (2003c). The analyses were conducted to determine the dose rates in close proximity to the Peach Bottom transfer cask, behind the FPA wall, in the CCA, in close proximity to the ISF Facility steel cask of the canister trolley, in close proximity to the CHM, and behind a concrete wall of storage vault 2 containing an 8 × 18 array of storage tubes. The TRIGA SNF was modeled in these analyses. The storage and transfer fuel configurations were considered in the analyses. The storage configuration consisted of a total of 108 TRIGA fuel elements packaged vertically in 2 baskets stacked on top of each other and with rods distributed relatively uniformly within the baskets (2 × 54 configuration). The transfer configuration consisted of a total of 90 TRIGA fuel elements packaged vertically in 3 buckets with 6 fuel cans filled with 5 rods each (5 × 6 × 3 configuration).

The shielding analysis was performed using the computer codes MCNP 4B2 (Los Alamos National Laboratory, 1997) and MCNP 4C2 (Los Alamos National Laboratory, 2001a). MCNP 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. The gamma flux-to-dose conversion factors used in the ISF Facility 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 employed by the applicant is widely used for shielding analyses and is considered acceptable by the staff for use in modeling the shielding configurations and materials at the ISF Facility. 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 at various locations on the site and beyond the edge of the restricted area are described in Sections 7.3.2, 7.4.1, and 7.4.2 and in Tables 7.4.1 and 7.4.2 of the SAR; in Sections 7.3.2, 7.4, and 7.6 of Appendix A of the SAR; and in Foster Wheeler

Environmental Corporation's response to the staff's requests for additional information (2003b,c).

The applicant calculated onsite dose rates for the ISF Facility based on stainless steel TRIGA fuel source terms with initial enrichment of 20 weight percent of U-235, burnup of 34,103 MWD/MTU, and cooling time of 6.5 years as discussed in Section 7.1.1 of this SER. The applicant calculated side surface and 0.3-m [1-ft] radial Peach Bottom cask surface doses to be approximately 0.33 and 0.12 mSv/hr [33 and 12 mrem/hr]. The calculated bottom surface and 0.3-m [1-ft] axial Peach Bottom cask dose rates are approximately 0.15 and 0.11 mSv/hr [15 and 11 mrem/hr]. The estimated dose rate drops to approximately 0.038 mSv/hr [3.8 mrem/hr] at 0.9 m [3 ft] from the cask (Foster Wheeler Environmental Corporation, 2003c). The dose rate behind the 1.22-m [4-ft] concrete FPA wall at the operator location was estimated as 4.4×10^{-4} mSv/hr [0.044 mrem/hr]. The applicant estimated an external dose rate of 0.15 mSv/hr [15 mrem/hr] at a distance of 0.3 m [1 ft] from the 0.9-m [3-ft] thick concrete wall between the storage area and Transfer Tunnel and an external dose rate of 0.033 mSv/hr [3.3 mrem/hr] at a distance of 0.3 m [1 ft] from the 0.9-m [3-ft] thick concrete wall outside vault 2 containing 144 canisters filled with TRIGA fuel in storage configuration. The 0.3-m [1-ft] radial dose rate from the CHM surface was estimated as 0.024 mSv/hr [2.4 mrem/hr]. The applicant did not apply a peaking (scaling) factor of 2.62 for the bounding source term in the shielding effectiveness calculations. Even if the factor was applied to the source term, the resulting personnel annual doses still would meet the 10 CFR §20.1201(a) annual occupational dose limits. The SAR presented streaming analyses that evaluated dose rates in the CCA during canister closure and inspection operations. The estimated dose rate for the minimum expected shield plug/canister gap of 2.54 mm [0.1 in] is 0.5 mSv/hr [50 mrem/hr]. The estimated dose rate for the maximum expected shield plug/canister gap of 5.08 mm [0.2 in] is 1.43 mSv/hr [143 mrem/hr].

The applicant provided a qualitative discussion of the negligible contribution of direct and scattered radiation to the total dose at the controlled area boundary. No accidents were identified in the SAR that would cause a significant loss of shielding. The credible losses of shielding were described and analyzed. The radiological consequences of these events will include localized elevated radiation fields, however, no increased exposure rates are expected.

Based on the results of the applicant's shielding analysis, the staff finds that the predicted dose rates resulting from normal operation of the ISF Facility are below the limits specified in 10 CFR §72.104(a) at locations offsite.

The applicant's shielding analysis provided sufficient information to satisfy the requirements of 10 CFR §20.1201 and §72.24, by demonstrating that radiation exposures to workers will be limited to acceptable levels through the use of shielding at the ISF Facility.

Chapter 11 of this SER discusses the overall offsite dose rates from the ISF Facility estimated from the combined radiation exposure from 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) can be achieved by the applicant by means of its radiation protection design and radiological protection program described in the SAR.

7.1.5 Confirmatory Calculations

The staff performed confirmatory calculations to determine the TRIGA photon source term that is a bounding source term for the SAR shielding analysis. In performing the confirmatory calculations, the staff used the SCALE4.4A/SAS2H (Hermann and Parks, 2000), ORIGEN-ARP Version 2.0 (Bowman and Leal, 2000) computer codes and 44-group ENDF/B-V neutron cross-sectional library (Greene, et al., 1994). The stainless steel-clad TRIGA fuel element containing 800 g [1.76 lb] of SS-304 per element with 2,200 ppm of Co-59 impurity level was used for the analyses. The photon source term determined from the staff's confirmatory calculations is in close agreement with that obtained by the applicant.

The staff also performed independent calculations of the dose rates that could be expected behind 0.91- and 1.22-m [3- and 4-ft] thick walls of the FPA at the potential operator location. The staff used the MCNP 4C2 code (Los Alamos National Laboratory, 2001a), ENDF/B-VI cross-sectional data (Los Alamos National Laboratory, 2001b), and gamma flux-to-dose conversion factors from ANSI/ANS 6.1.1 (American Nuclear Society Standards Committee Working Group, 1977) for these calculations. These calculations confirmed that neutron dose rates are negligible in comparison with photon dose rates, and that the onsite dose rates for an operator located behind the FPA concrete walls exposed to an ISF storage canister loaded with 108 TRIGA SNF elements are below 0.07 and 0.002 mSv/hr [7.0 and 0.2 mrem/hr]. Therefore, the estimated FPA operator annual dose would meet the 10 CFR §20.1201(a) annual dose limit for workers. The staff performed independent confirmatory calculations for dose rate estimates to personnel from a loaded Peach Bottom transfer cask in the Cask Receipt Area (CRA), conservatively taking into account a peaking (scaling) factor of 2.62 for bounding the TRIGA SNF source. These calculations confirmed the surface 0.304-m [1-ft] and 1-m [3.29-ft] radiation dose rates at and around a Peach Bottom transfer cask loaded with 90 TRIGA SNF elements in three tiers are below 1.12, 0.93, and 0.42 mSv/hr [112, 93, and 42 mrem/hr] for the side surface; 0.38, 0.22, and 0.13 mSv/hr [38, 22, and 13 mrem/hr] for the bottom surface; and 0.016, 0.015, and 0.003 mSv/hr [1.6, 1.5, and 0.3 mrem/hr] for the top surface; so the estimated personnel annual dose in the CRA would meet the 10 CFR §20.1201(a) annual dose limit.

Based on the staff's confirmatory calculations, the staff concludes that the applicant's shielding analysis is acceptable.

7.2 Evaluation Findings

Based on its review of the information provided by the applicant regarding its shielding evaluation for the proposed ISF Facility, the staff made the following findings:

- The design of the ISF Facility provides acceptable means for limiting exposure of the public to direct and scattered radiation within the limits given in 10 CFR §72.104.

- The design of the ISF Facility provides acceptable means for limiting occupational radiation exposures within the limits given in 10 CFR §20.1201 and for maintaining exposures ALARA, in compliance with 10 CFR §72.24(e).

- The design of the shielding system(s) of the ISF Facility satisfies the criteria for radiological protection of 10 CFR §72.126(a)(6).

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

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