9 CONFINEMENT EVALUATION

9.1 Conduct of Review

The staff reviewed the confinement evaluation presented in the Idaho Spent Fuel (ISF) Facility Safety Analysis Report (SAR) (Foster Wheeler Environmental Corporation, 2003).

The ISF Facility will use a combination of helium-filled spent nuclear fuel (SNF) canisters and storage tubes as primary and secondary confinement. Additional protection is afforded by radiological monitoring, a concrete storage vault, and ventilation and off-gas systems that use high-efficiency particulate air (HEPA) filtration and negative pressure to filter and control air flow. The confinement approach follows the overall guidance provided in Interim Staff Guidance Number 5, Revision 1, “Confinement Evaluation” (U.S. Nuclear Regulatory Commission, 1999).

Foster Wheeler Environmental Corporation performed analyses of hypothetical radiological releases to demonstrate compliance with 10 CFR Part 72 and included a discussion of radiological release calculations. Information about chemical composition and mechanical properties of materials for construction of critical cask components is also provided in the ISF Facility SAR (Foster Wheeler Environmental Corporation, 2003).

This review was conducted in accordance with the guidance presented in Chapter 9 of NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000) and, as appropriate, NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997). The review focused on analyses and results presented and referenced by the applicant in the ISF Facility SAR (Foster Wheeler Environmental Corporation, 2003).

9.1.1 Radionuclide Confinement Analysis

The application was reviewed to identify the quantity of radionuclides that hypothetically could be released during normal, off-normal, and accident conditions including design basis accidents. The staff reviewed Sections 3.3.2.1, “Confinement Barriers and Systems;” 3.3.2.2, “Ventilation and Off-Gas Systems;” 3.3.5.3, “Radiological Alarm Systems;” 4.2.1.2, “ISF Storage Tube Assembly;” 4.2.1.3, “ISF Canister;” 4.2.2.3, “Confinement Features;” 4.2.3.2, “Components;” 4.3.1, “Ventilation and Off-Gas Systems;” “4.3.2.1.2, Instrumentation and Controls;” 4.7.2.3, “Confinement Features;” 4.7.3.1.3, “Fuel Packaging Area;” 7.2.1, “Characterization of Sources;” 7.2.2, “Airborne Radioactive Material Sources;” 7.4.2, “Site Dose Assessment;” 7.6.2, “Analysis of Multiple Contribution;” 7.6.3, “Estimated Dose Equivalents;” 7.6.4, “Liquid Release;” 8.1, “Off-Normal Events;” 8.2, “Accidents;” and Chapter 6, “Generated Waste Confinement and Management,” of the SAR (Foster Wheeler Environmental Corporation, 2003). The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR §72.24(l)(1) requires a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences. The description must identify the design objectives and the means to be used for keeping levels of radioactive material in effluents released to the environment as low as is reasonably achievable and

within the exposure limits stated in §72.104. The description must include an estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal Independent Spent Fuel Storage Installation (ISFSI) operations.

- 10 CFR §72.44(c)(1)(i) requires that each license issued under this part include technical specifications for functional and operating limits and monitoring instruments and limiting control settings. The functional and operating limits for an ISFSI are limits on fuel or waste handling and storage conditions that are found to be necessary to protect the integrity of the stored fuel or waste container, to protect employees against occupational exposures and to guard against the uncontrolled release of radioactive materials.

- 10 CFR §72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual beyond the controlled area must not exceed 0.25 mSv [25 mrem] to the whole body, 0.75 mSv [75 mrem] to the thyroid, and 0.25 mSv [25 mrem] to any other organ as a result of exposure to planned discharges of radioactive material, direct radiation, and radiation from other nearby operations.

- 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, high-level radioactive waste, or reactor-related Greater Than Class C waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 m [328 ft].

- 10 CFR §72.122(b)(4) requires that if the ISFSI is located over an aquifer, which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

- 10 CFR §72.122(h)(3) requires that ventilation systems and off-gas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions.

- 10 CFR §72.126(d) requires that the ISFSI be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in §72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in §72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

- 10 CFR §72.128(a)(3) requires that spent fuel storage systems be designed with confinement structures and systems.

The ISF Facility confinement system is designed for long-term confinement and dry storage of the Peach Bottom Unit 1 fuel elements, TRIGA research reactor fuel elements, and the Shippingport reactor reflector modules and fuel rods. The Peach Bottom Core 1 fuel elements are known to be damaged. To account for the damaged fuel elements, Foster Wheeler Environmental Corporation conservatively assumed that no credit would be taken for the existing cladding for any of the fuel types. The design of the confinement system is discussed in detail in Section 4.2.3 of the ISF Facility SAR. The structures, systems, and components of the confinement system classified as important to safety, are introduced in Section 3.4 and discussed in more detail throughout Chapter 4 of the ISF Facility SAR. There are structures, systems, and components classified as important to safety contained in the cask receipt area, transfer area, and storage area. The major components of the confinement system that are classified as important to safety include the sealed SNF canisters, canister baskets, canister impact plate, canister shield plug, pressure boundary components of sealed storage tubes, and charge face cover plate. The confinement system is designed to maintain a confinement barrier under all normal, off-normal, and accident conditions.

The confinement boundary for the ISF Facility includes helium-filled sealed SNF canisters and storage tubes. These systems are described in detail in Section 4.2 of the ISF Facility SAR. The confinement system is designed, fabricated, and tested in accordance with the applicable requirements of the ASME code, Section III, Subsections NB and NC, to the maximum extent practicable (ASME International, 2001). The SNF canister double-seal welds and storage tube double metallic seal rings are designed to maintain confinement during normal and design basis accident conditions.

Although no releases are anticipated from the confinement system, Section 7.6.2 of the SAR reports that releases from the Fuel Packaging Area (FPA) during normal operations are estimated to add 3×10^{-4} μSv/yr [3×10^{-5} mrem/yr] total effective dose equivalent (TEDE) at the controlled-area boundary. A TEDE of less than 3.2 μSv/yr [0.32 mrem/yr] is estimated from all other nearby facilities and operations. Independent analyses by the staff showed the TRIGA fuel (108 fuel elements) to yield the highest dose for normal operating conditions. The staff analyses followed guidance in NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000), and used the RSAC–5 computer code (Wenzel, 1993), assuming that 1 percent of the SNF was available for release, the HEPA filtration system was operable, and release fractions matched those recommended in Interim Staff Guidance Number 5 for normal operations (U.S. Nuclear Regulatory Commission, 1999). The staff's independent analyses of this scenario yielded a maximum TEDE of 0.049 μSv/yr [0.0049 mrem/yr]. Both the SAR analyses and the staff's independent analyses result in TEDE values that do not exceed the 0.25-mSv/yr [25-mrem/yr] operational dose limit in 10 CFR §72.104(a).

Although no releases are anticipated from the confinement system, Section 8.1 of the SAR states that releases from the FPA during off-normal events are bounded by analyses of the hypothetical accident scenario described in Section 8.2.4.5 of the SAR that estimates a TEDE of 0.2 μSv/yr [0.02 mrem/yr] at the controlled-area boundary.

Independent analyses conducted by the staff showed the TRIGA fuel (108 fuel elements) to yield the highest dose for off-normal conditions. The staff analyses followed guidance in NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000) and used the RSAC–5 computer code (Wenzel, 1993), assuming that 10 percent of SNF was available for release, the HEPA filtration system was operable, and the release fractions matched those recommended in

Interim Staff Guidance Number 5 for off-normal events (U.S. Nuclear Regulatory Commission, 1999). The staff's independent analyses of this scenario yielded a maximum TEDE of 0.49 μSv/yr [0.049 mrem/yr]. The SAR analyses and the staff's independent analyses result in TEDE values that do not exceed the 0.25-mSv/yr [25-mrem/yr] operational dose limit in 10 CFR §72.104(a).

In Section 8.2.4.5 of the SAR, leakage from the confinement system during hypothetical accident conditions was evaluated. Following methodology in accordance with Interim Staff Guidance Number 5 (U.S. Nuclear Regulatory Commission, 1999) and ANSI (American National Standards Institute, 1998), the applicant calculated the dose to an individual continuously present at the controlled-area boundary for 30 days at the location nearest to the proposed ISF Facility. This hypothetical, worst-case calculation yielded a TEDE of 0.2 μSv [0.02 mrem] from a HEPA failure in the FPA for an accident involving Peach Bottom Core 2 fuel. Independent analyses conducted by the staff also showed the Peach Bottom Core 2 fuel to yield the highest dose for hypothetical accident conditions (a maximum of 10 Peach Bottom Core 2 fuel elements are contained in the transfer casks). The staff's analyses followed guidance in NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000) and used the RSAC–5 computer code (Wenzel, 1993), assuming 100 percent of the SNF was available for release during transfer operations, the HEPA filtration system was inoperable, and the release fractions matched those recommended in Interim Staff Guidance Number 5 for accidents (U.S. Nuclear Regulatory Commission, 1999). The staff's independent analyses of this more conservative scenario yielded a maximum TEDE of 70 μSv [7 mrem]. The SAR analyses and the staff's independent analyses result in TEDE values that do not exceed the 50-mSv [5-rem] accident dose limit in 10 CFR §72.106(b).

Although a hypothetical accident condition leakage calculation was performed for the confinement system, the applicant expects there will be no release of radioactive materials in effluents during normal and all credible accident conditions. This expectation is supported by the applicant's analyses that demonstrate the confinement system would maintain its confinement integrity for the design basis normal, off-normal, and accident conditions (including vehicular collision with transporter; transfer cask drop during hoisting operations; transfer cask tipover; cask trolley collision events; drop of spent fuel container during handling; drop of ISF basket during handling; canister trolley movement in raised position; ISF canister drop; transverse movement of the canister handling machine with an ISF canister partially inserted; adiabatic heatup; loss of shielding; building structural failure onto structures, systems, or components; fire and explosion; maximum hypothetical dose accident; loss of external power for an extended interval; earthquake; flood; extreme wind; lightning; accidents at nearby sites; basaltic lava flow; and aircraft impact). Based on the results of the applicant's analyses, the staff agrees that confinement integrity would be maintained during the design basis normal, off-normal, and accident conditions.

The staff, therefore, concludes with reasonable assurance the risk from radioactive effluents released to the general public from storing the specified spent nuclear fuel in up to 246 metal storage tubes at the ISF Facility is insignificant and meets the requirements of 10 CFR §72.106(b). The staff also concludes that the SNF canister (with double-seal welds) and storage tube (with double metallic seal rings), if manufactured and inspected according to the ASME International code, as approved by the staff, will not release radioactive effluents, and thereby meet the requirements of 10 CFR §72.122(b), §72.126(d), and §72.128(a)(3).

The staff reviewed the applicable chapters of the SAR and found those portions related to the confinement integrity of the confinement system to be acceptable, thereby meeting the requirements of 10 CFR §72.122(h), §72.126(d), and §72.128(a)(3).

9.1.2 Confinement Monitoring

The staff's review of this section focused on two areas: the continuous monitoring of closure seal effectiveness and the measurement of radionuclides released to the environment during normal and accident conditions. The staff reviewed Sections 4.2.3.2.2, "Description of the Storage Tube Assembly and Associated Interfacing Equipment;" 4.3.2.1.2, "Instrumentation and Control;" 5.1.1.2.8, "Perform ISF Canister Lid Closure Weld;" 5.1.1.2.9, "Canister Vacuum Dry, Inert, and Leak Check;" 5.1.1.3, "Canister Handling;" 7.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation;" and 7.6.1, "Effluent and Environmental Monitoring Program," of the SAR (Foster Wheeler Environmental Corporation, 2003). The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR §72.24(l)(1) requires a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences. The description must identify the design objectives and the means to be used for keeping levels of radioactive material in effluents released to the environment as low as is reasonably achievable and within the exposure limits stated in §72.104. The description must include an estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal ISFSI operations.

- 10 CFR §72.44(c)(3)(iv) requires confirmation that the limiting conditions required for safe storage are met.

- 10 CFR §72.122(h)(4) requires that storage confinement systems have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.

- 10 CFR §72.126(c)(1) requires as appropriate for the handling and storage system, that effluent systems be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions must be provided for these systems. A means of measuring the flow of the diluting medium, either air or water, must also be provided.

- 10 CFR §72.128(a)(1) requires that spent fuel storage systems be designed with a capability to test and monitor components important to safety.

Based on the staff's assessment of the SNF canister double-seal welds and storage tube double metallic seal o-rings, as stated in Chapter 7, Section V.2 of NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997), the confinement system provides reasonable assurance that no

effluents will be released. The sealed SNF canisters and storage tubes will be purged, vacuum dried, inerted with helium, inspected, and tested for leaks. The SNF canister double-seal welds and storage tube double metallic seal rings are designed to maintain confinement during normal and design basis accident conditions.

The staff finds the applicant's proposal to provide continuous air and radiation area monitoring in the storage area acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate cask design requirements, meeting the requirements of 10 CFR §72.122(h)(4) and §72.126(c)(1).

9.1.3 Protection of Stored Materials from Degradation

The application was reviewed to establish that the fuel will be confined such that degradation of the fuel during the anticipated 30-year storage period does not pose operational safety problems with respect to its removal from storage. The Peach Bottom Core 1 fuel elements are known to be damaged. To account for the damaged fuel elements, Foster Wheeler Environmental Corporation conservatively assumed that no credit would be taken for the existing cladding for any of the fuel types. The staff reviewed Sections 3.1.1, "Materials to be Stored;" 3.3.2.1, "Confinement Barriers and Systems;" 4.2.3.3.6, "Heat Transfer and Thermal Evaluation;" 4.7.3.2.12, "Vacuum Drying and Helium Fill System;" 6.3.2, "Equipment and System Description;" 6.4.4, "Characteristics, Concentrations, and Volumes of Solidified Wastes;" and 6.4.6, "Storage Facilities," of the SAR (Foster Wheeler Environmental Corporation, 2003). The information presented has been reviewed for conformance with the following regulatory requirements.

- 10 CFR §72.24(g) requires that the license application include an identification and justification for the selection of those subjects that will be probable license conditions and technical specifications.

- 10 CFR §72.122(h)(1) requires that the SNF cladding be protected during storage against degradation that leads to gross ruptures or be otherwise confined such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage.

Following loading of the SNF canister, the main lid is welded and inspected. The canister cavity is then vacuum dried twice and backfilled with helium. The lid closure weld, vent plug seal, and vent plug seal weld are also inspected and tested. The sealed SNF storage canister is placed and sealed within a storage tube. The storage tube is also vacuum dried twice and backfilled with helium, inspected, and tested for leaks. These steps are described in detail in the SAR. The helium backfill procedure ensures the presence of oxidizing gasses in the canister cavity will be minimized.

The staff reviewed the proposed technical specifications and finds acceptable the portions related to the confinement of stored materials, thereby meeting the requirements of 10 CFR §72.122(h)(1).

9.2 Evaluation Findings

Based on the staff's review of the applicant's submittal and the applicable technical specifications, the staff made the following findings:

- The radionuclide confinement analysis for the confinement system proposed for the ISF Facility meets the requirements of 10 CFR §72.24(i)(1) by providing a description of how radioactive materials in gaseous and liquid effluents will be controlled so they are as low as is reasonably achievable. The requirements of 10 CFR §72.44(c) have been met, based on the staff's review of the technical specifications that have been submitted by the applicant. Because the SNF canister lid is welded and tested in accordance with ASME International code and is not expected to leak during normal, off-normal, and accident conditions, the staff finds that the requirements of 10 CFR §72.122(h)(3), §72.126(d), §72.128(a)(3), and §72.122(a) have been met.

- The staff concludes that the confinement system, which has been welded and tested in accordance with the ASME International code, is not expected to leak and the proposal to monitor the storage area with continuous air and radiation area monitors is acceptable. Based on this finding, the requirements of 10 CFR §72.44(c), §72.122(h)(4), §72.126(c)(1), and §72.128(a)(3) are met.

- The staff concludes that the proposed technical specifications are sufficient to confine the stored materials in accordance with 10 CFR §72.24(g). The staff also finds that the proposed methods are sufficient to confine the stored materials such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage in accordance with 10 CFR §72.122(h)(1).

9.3 References

American National Standards Institute. *Airborne Release Fractions at Non-Reactor Nuclear Facilities*. ANSI/ANS–5.10–1998. New York, NY: American National Standards Institute. 1998.

ASME International. *ASME Boiler and Pressure Vessel Code, Section III*. New York City, NY: ASME International. 2001.

Foster Wheeler Environmental Corporation. *Idaho Spent Fuel Facility Safety Analysis Report*. ISF–FW–RPT–0033. Docket 72-25. Amendment 03. Morris Plains, NJ: Foster Wheeler Environmental Corporation. November 2003.

U.S. Nuclear Regulatory Commission. NUREG–1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities.* Washington, DC: U.S. Nuclear Regulatory Commission. 2000.

U.S. Nuclear Regulatory Commission. *Confinement Evaluation*. Interim Staff Guidance Document–5. Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission. June 1999.

U.S. Nuclear Regulatory Commission. NUREG–1536, *Standard Review Plan for Dry Storage Systems.* Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

Wenzel, D.R. *The Radiological Safety Analysis Computer Code (RSAC–5) User's Manual*. Idaho Falls, ID: Westinghouse Idaho Nuclear Company, Incorporated. 1993.