9 CONFINEMENT EVALUATION

9.1 Conduct of Review

The staff reviewed the confinement evaluation presented in the SAR (Private Fuel Storage Limited Liability Company, 2000). The staff reviewed Sections 3.3.2.1, 3.4, 4.2.1.2.2, 4.2.1.5.5, and 8.2.7 of the SAR. The staff also reviewed the applicable sections of the proposed Technical Specifications.

The Facility will use the HI-STORM 100 Cask System, which has been approved by NRC for use under the general license provisions of 10 CFR Part 72. The design specifications for the confinement function of the HI-STORM 100 cask are addressed in Chapter 7 of the HI-STORM 100 Cask System FSAR (Holtec International, 2000).

Based on the statements in the HI-STORM 100 FSAR, the applicant conducted an analysis of a hypothetical radiological release. The confinement evaluation submitted by the applicant relies on the analyses performed by Holtec International to demonstrate compliance with 10 CFR Part 72, and includes a discussion of radiological release calculations and an evaluation of stored material degradation. The applicant provided no information in the PFS Facility SAR on chemical composition and mechanical properties of materials for construction of critical cask components; this information is provided in the referenced HI-STORM 100 Cask System FSAR.

This review was conducted in accordance with the guidance presented in Chapter 9 of NUREG–1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities.”

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 information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR 72.24(l) requires that the licensee provide a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents, including the means to maintain radioactive materials in effluents as low as is reasonably achievable.

- 10 CFR72.44(c) requires that the licensee define Technical Specifications for design features that would have a significant effect on safety if altered or modified.

- 10 CFR 72.122(a) requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.

- 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 limit to ALARA levels the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions.

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

The HI-STORM 100 Cask System is designed for long-term confinement and dry storage of PWR and BWR spent nuclear fuel. The design of the HI-STORM 100 Cask System is discussed in detail in Section 4.2.1.4 of the SAR. The major components of the HI-STORM 100 Cask System that are classified as important to safety include the sealed MPC and the storage cask. The MPC is designed to maintain a confinement barrier under all normal, off-normal, and accident conditions.

The confinement boundary for the Holtec HI-STORM 100 includes the MPC shell, the bottom baseplate, the MPC lid (including the vent and drain port cover plates), the MPC closure ring, and the associated welds. The welds forming the confinement boundary are described in detail in Section 7.1.3 of the HI-STORM 100 FSAR. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME code, Section III, Subsection NB, to the maximum extent practicable. The MPC lid and MPC closure ring seal welds are designed to maintain confinement under normal and design basis accident conditions.

In Section 6.5 of the SAR, the applicant states that no releases of any type of radioactive material will occur during normal operations. This statement is in agreement with the statements in Section 7.2.3 of the HI-STORM 100 FSAR. Thus, leakage of the MPCs under normal conditions was not considered.

In Section 8.2.7 of the SAR, leakage from the HI-STORM cask under hypothetical accident conditions was evaluated. Following the methodology in the HI-STORM 100 FSAR, and in accordance with Interim Staff Guidance Number 5, the applicant calculated the dose to an individual continuously present for 30 days at the location nearest to the Canister Transfer Building on the owner controlled area boundary. This hypothetical, worst-case calculation yielded a total effective dose equivalent of 2.68 mrem from a single leaking MPC. The accident dose rates (dose due to direct and scattered radiation and to a hypothetical release) for the HI-STORM 100 Cask System do not exceed limits specified in 10 CFR 72.106(b).

While a hypothetical accident condition leakage calculation was performed for the HI-STORM 100 cask, the applicant expects that there will be no release of radioactive materials in effluents during normal and all credible accident conditions. This is supported by the applicant's analyses which demonstrate that the MPC would maintain its confinement integrity under the design basis normal, off-normal, and accident conditions (including earthquake, tornado, flood, explosions, fire, and cask tipover). Based on the results of the applicant's analyses, the staff agrees that the MPC confinement integrity would be maintained under the design basis normal, off-normal, and accident conditions. The staff further investigated the acceptability of the applicant's conclusion that the HI-STORM 100 cask would not leak under normal, off-normal, and accident conditions by performing a risk assessment of the MPC confinement system welds. The probabilities of a leak through the various welds in the MPC are summarized in Table 9-1.

Table 9-1: Number of Through-Wall Flaws per Weld for each weld used to construct and seal the HI-STORM MPC

Weld	Probability of a Through-Wall Flaw
MPC Lid to Shell Weld	5.8×10^{-9} (P1)
Closure Ring to Shell	3.6×10^{-4} (P2)
Closure Ring to Lid Weld	2.8×10^{-4} (P3)
Vent and Drain Cover Plate Weld	1.2×10^{-6} (P4)
Circumferential and Axial Seam Weld	7.1×10^{-6} (P5)
Shell to Baseplate Weld	2.6×10^{-6} (P6)

Applying the probabilities outlined in Table 9-1, the probability of forming a leak in the cask lid seal welds is:

$$(P1 \times P2) + (P1 \times P3) + [(P4 \times P2) + (P4 \times P3)] \times 2 =$$

$$[P1 + (2 \times P4)] \times (P2 + P3) = 1.5 \times 10^{-9}$$

The total probability of a leak in the MPC is calculated as the probability from the shell to base plate weld (P6), plus the probability of the seam weld (P5), plus the probability of the cask lid welds. The total probability is:

$$P6 + P6 + [(P1 + P4) \times (P2 + P3)] = 9.7 \times 10^{-6}$$

The probabilities above do not credit helium leak testing performed by the MPC manufacturer. Consequently, the probabilities are expected to be lower.

Using the assumption that the cask leaks at a flow rate and with an effluent source term analyzed in the HI-STORM 100 FSAR (maximum leak rate permitted by the HI-STORM 100 Technical Specifications and validated not to be exceeded by the Technical Specification leak tests), the radiological consequence at 100 meters from the cask, assuming plateout and settling of radionuclides, is 8.08×10^{-5} mrem/year. The risk from one cask is the product of the probability of a leak times the consequences (mrem/year). The risk from one HI-STORM leaking cask is 8×10^{-10} mrem/year.

Assuming a population of 4000 casks, Table 9-2 identifies the probability of one or more (k) casks leaking and the associated risk, in mrem/year.

Table 9-2: Risk at PFS from Leaking Cask(s) at 100 meters

Number of Casks (k) Leaking From a Population of 4000 Casks	Probability of k Casks Leaking at the Same Time	Risk from k Casks Leaking (mrem/year)
1	4×10^{-2}	3×10^{-6}
2	8×10^{-4}	1×10^{-7}
3	1×10^{-5}	2×10^{-9}
4	9×10^{-8}	3×10^{-11}
5	7×10^{-10}	3×10^{-13}

The staff's analysis shows that more than 3 out of 4000 casks leaking at PFS is not a credible event (i.e., probability less than 1×10^{-6}). At 100 meters from the cask, the radiological consequence of radioactive effluents leaking from three casks is negligible (i.e., 2×10^{-4} mrem/year, which is orders of magnitude below normal background radiation).

The staff, therefore, concludes that:

- The risk to the public from radioactive effluent released at the Facility with 4000 casks is 3×10^{-6} mrem/year (calculated by summing the risks in the third column of Table 9-2).

- Should three in 4000 casks leak, the consequence at 100 meters is insignificant (2×10^{-4} mrem/year).

- Reasonable assurance exists that the risk from radioactive effluents released to the general public from storing 4000 HI-STORM casks at the Facility is insignificant.

- Stainless steel welded casks (with redundant welds in the lid enclosure of the cask) manufactured and inspected according to ASME code, as approved by the staff, are not expected to release radioactive effluents.

In addition to the investigation described above, the staff reviewed the applicable chapters of the SAR, and found that the conclusions that were made by the applicant were in agreement with the Holtec HI-STORM 100 FSAR, and are acceptable. The staff also reviewed the site Technical Specifications proposed by the applicant, and found those portions related to the confinement integrity of the HI-STORM cask to be acceptable.

9.1.2 Confinement Monitoring

The staff's review of this section focused on two areas. These areas included the continuous monitoring of closure seal effectiveness and the measure of radionuclides released to the environment during normal and accident conditions. These areas are discussed in Sections

4.3.1, 4.3.5, 4.3.8.4, and 4.3.11 of the SAR. The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR 72.24(l) requires that the licensee provide a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents, including the means to maintain radioactive materials in effluents as low as is reasonably achievable.

- 10 CFR 72.44(c) requires that the licensee define Technical Specifications for surveillance requirements to ensure that the conditions for safe storage be 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.

- 10 CFR 72.126(c)(1) requires that means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions be provided for storage and handling systems.

- 10 CFR 72.128(a)(3) requires that spent fuel storage systems be designed with a capability to test and monitor components important to safety.

The final seal welds and leak testing of the MPC will be performed at the originating nuclear power plant. Welder qualifications, welding procedures, nondestructive examination, and leak testing of the welds will be in accordance with NRC-accepted industry standards (i.e., the ASME Boiler and Pressure Vessel Code and ANSI-N14.5).

The MPCs are loaded into the shipping casks at the originating nuclear power plant and are not opened at the Facility. Each MPC will arrive on the PFS Facility site in a shipping cask, and is then transferred to the HI-STORM 100 storage cask in the canister transfer building, using the HI-TRAC transfer cask.

Based on the staff's assessment of welded cask enclosures, as stated in NUREG–1536, "Standard Review Plan for Dry Cask Storage Systems," Chapter 7, Section V.2, the MPC, which is the confinement system for the HI-STORM 100 Cask System, provides reasonable assurance that no effluents will be released and, therefore, requires no monitoring of the MPC for leakage. The seal weld will be inspected and tested in accordance with the requirements in Section 8.1.5 of the HI-STORM 100 FSAR. These requirements were reviewed during the certification of the HI-STORM 100 storage cask, and were found to be acceptable by the staff.

The staff finds the applicant's proposal to provide no monitoring of the confinement barrier for the HI-STORM 100 casks acceptable, because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures.

9.1.3 Protection of Stored Materials from Degradation

Review of this section of the SAR was performed to establish that the fuel cladding would not experience significant degradation during the licensed storage period of 20 years. 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 spent fuel 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 the loading of the MPC, the main lid is welded and a helium leak test is performed on the seal weld. The MPC cavity is then vacuum dried and filled with helium fill gas. The vent and drain ports are then welded into place and a helium leak test is conducted on the vent and drain port covers. These steps are described in detail in the HI-STORM 100 FSAR. The helium back-fill procedure ensures that the presence of oxidizing gasses in the MPC cavity will be minimized.

The thermal analysis of the HI-STORM 100 cask indicates that the fuel cladding temperature will not exceed the limits established to prevent fuel clad degradation during storage.

The staff verified that the applicant's SAR was consistent with the information provided in the HI-STORM 100 FSAR. The staff reviewed the proposed Technical Specifications and found the portions related to the protection of stored materials from degradation in the HI-STORM cask to be acceptable.

9.2 Evaluation Findings

Evaluation of confinement of spent nuclear fuel stored at the Facility assumed that only the HI-Storm 100 cask will be used. Based upon the staff's review of the applicant's submittal and the applicable Technical Specifications the staff has made the following findings.

- The radionuclide confinement analysis for the Holtec HI-STORM 100 cask and the PFS site has met the requirements of 10 CFR 72.24(l) by providing a description of how radioactive materials in gaseous and liquid effluents will be controlled such that they are ALARA. 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 MPC lid is welded and tested in accordance with ASME code and is not expected to leak under normal, off-normal, and accident conditions, the staff finds that the requirements of 10 CFR 72.122(h)(3), 10 CFR 72.126(d),10 CFR 72.128(a)(3), and 10 CFR 72.122(a) have been met.

- The staff concludes that the HI-STORM 100 cask, which contains the MPC which has been welded and tested in accordance with ASME code, is not expected to leak and therefore does not require confinement monitoring. Based on this finding, the requirements of 10 CFR72.44(c), 10 CFR 72.122(h)(4), 10 CFR 72.126(c)(1), and 10 CFR 72.128(a)(3) are met.

- The staff concludes that the proposed Technical Specifications are sufficient to protect the stored materials from degradation in accordance with 10 CFR 72.24(g). The staff also finds that the proposed methods to protect the stored materials from degradation are acceptable to protect the spent fuel cladding from gross ruptures in accordance with 10 CFR 72.122(h)(1).

9.3 References

American Society of Mechanical Engineers. ASME Boiler and Pressure Vessel Code, Section III. NY: American Society of Mechanical Engineers. 1998.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System). Volumes I and II*. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2000

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