SAFETY EVALUATION REPORT

DOCKET NO. 72-1032

HOLTEC INTERNATIONAL

HI-STORM FLOOD/WIND SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1032

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1 SUMMARY

By letter dated October 13, 2009, as supplemented December 18, 2009, April 13, June 4, and August 20, and October 14, 2010, Holtec International, Inc. (Holtec, the applicant) submitted an application to the U. S. Nuclear Regulatory Commission (NRC) to approve the HI-STORM Flood/Wind (FW) system Certificate of Compliance (CoC) - No. 1032 under 10 CFR 72 Subpart K, General License for Storage of Spent Fuel at Power Reactor Sites. The HI-STORM FW system consists of the following major components.

- HI-STORM FW Overpack
- Pressurized Water Reactor (PWR) Multi-Purpose Canister (MPC)- MPC-37
- Boiling Water Reactor (BWR) Multi-Purpose Canister (MPC)- MPC-89
- HI-TRAC Variable Weight (VW) Transfer Cask

This Safety Evaluation Report (SER) documents the review and evaluation of the proposed amendment. The SER uses the same Section-level format provided in NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” with some differences implemented for clarity and consistency.

The staff's assessment is based on whether Holtec meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection.

1.1 General Description

The objective of the review of the general description of the HI-STORM FW system is to ensure that Holtec has provided a description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.2 HI-STORM FW System General Description and Operational Features

HI-STORM (acronym for Holtec International Storage Module) FW system is a spent nuclear fuel storage system designed to be in full compliance with the requirements of 10 CFR Part 72. The model designation "FW" denotes this as a system which has been specifically engineered to withstand sustained Flood and Wind.

The HI-STORM FW system consists of a sealed metallic MPC contained within an overpack constructed from a combination of steel and concrete. The design features of the HI-STORM FW components are intended to simplify and reduce the on-site spent nuclear fuel (SNF) loading and handling work effort, to minimize the burden of in-use monitoring, to provide utmost radiation protection to the plant personnel, and to minimize the site boundary dose.

The HI-STORM FW system can store either PWR or BWR fuel assemblies, in the MPC-37 or MPC-89, respectively. The MPC is identified by the maximum number of fuel assemblies it can contain in the fuel basket. The MPC external diameters are identical to allow the use of a single overpack design, however the height of the MPC, as well as the overpack and transfer cask, are variable based on the SNF to be loaded.

The HI-STORM FW overpack is equipped with thru-wall penetrations at the bottom of the overpack and in its lid to permit natural circulation of air to cool the MPC and the contained SNF. The HI-STORM FW system is autonomous in-as-much as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any other facility, structures, or components at the site. The surveillance and maintenance required by staff is minimized by the HI-STORM FW system since it is completely passive and is composed of proven materials. The HI-STORM FW system can be used either singly or as an array at an Independent Spent Fuel Storage Installation (ISFSI). The site for an ISFSI can be located either at a nuclear reactor facility or an away-from a- reactor location.

The design features of the HI-STORM FW components are intended to simplify and reduce the on-site SNF loading and handling work effort, to minimize the burden of in-use monitoring, to provide utmost radiation protection to the plant personnel, and to minimize the site boundary dose. The HI-STORM FW overpack is stored at the ISFSI pad in a vertical orientation, which helps minimize the size of the ISFSI and leads to an effective natural convection cooling flow around the exterior and also in the interior of the MPC. The HI-STORM FW overpack features an inlet and outlet duct configuration engineered to mitigate the sensitivity of wind direction on the thermal performance of the system. More specifically, the HI-STORM FW overpack features a radially symmetric outlet vent (located in its lid) and inlet ducts arranged at 45-degree intervals in the circumferential direction to approximate an axisymmetric opening configuration, to the extent possible. A number of design measures are taken in the HI-STORM FW system to limit the fuel cladding temperature rise under a most adverse flood event (i.e., one that is just high enough to block the inlet duct):

a) The overpack's inlet duct is narrow and configured to block radiation efficiently even if the radiation emanating from the MPC is level (coplanar) with the duct penetration, therefore the MPC is not raised on a pedestal, which will allow floodwater to immediately come in contact with the bottom of the MPC and assist the ventilation air flow in cooling the MPC.

b) The overpack's inlet duct is tall, and the MPC stands on the overpack's baseplate, which is welded to the overpack's inner and outer shells. Thus, if the flood water rises high enough to block air flow through the inlet ducts, substantial surface area of the lower region of the MPC will be submerged in the water. Although heat transfer from the exterior of the MPC through air circulation is limited in such a scenario, the reduction is offset by convective cooling through the floodwater itself.

c) The MPCs are equipped with internal thermosiphon capability, which brings the heat emitted by the fuel back to the bottom region of the MPC as the circulating helium flows along the downcomer space around the fuel basket. This thermosiphon action places the heated helium in close thermal communication with the floodwater, further enhancing convective cooling via the floodwater.

The MPC-37 and 89 are similar as those used in the Holtec HI-STORM 100 Cask system and approved under CoC No. 1014. The HI-TRAC VW transfer cask is required for shielding and protection of the SNF during loading and movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. The MPC is placed inside the HI-TRAC VW transfer cask and moved into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC VW/MPC assembly is designed to prevent (contaminated) pool water from entering the narrow annular space between the HI-TRAC VW and the MPC while the assembly is submerged. The HI-TRAC VW transfer cask also allows dry loading (or unloading) of SNF into the MPC in a hot cell.

The HI-STORM FW system is designed to be operationally compatible with nearly all onsite ancillary components currently in use at sites deploying the HI-STORM 100 Cask systems. In particular, design criteria for optional ancillaries, such as a forced helium dehydration system (FHDS), as described in Appendices 2.B of the HI-STORM 100 SAR are compatible with the HI-STORM FW system.

2 PRINCIPAL DESIGN CRITERIAL EVALUATION

The objective of evaluating the principal design criteria related to the structures, systems, and components (SSCs) important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72.

2.1 Structures, Systems and Components Important to Safety

HI-STORM FW system SSCs important to safety are identified in Chapter 2 of the SAR. The safety classifications are based on the guidance in U.S. Nuclear Regulatory Commission, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” NUREG/CR-6407, INEL-95/0551, February 1996.

2.2 Design Basis for Structures, Systems and Components Important to Safety

The HI-STORM FW system design criteria summary includes the allowed range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.2.1 Spent Fuel Specifications

The HI-STORM FW Cask system is designed to store up to either 37 PWR fuel assemblies and up to 89 BWR fuel assemblies. Detailed specifications for the approved fuel assemblies are given in SAR Section 2.1. These include the maximum enrichment, maximum decay heat, maximum average burnup, minimum cooling time, maximum initial uranium mass, and detailed physical fuel assembly parameters. The limiting fuel specifications are based on the fuel parameters considered in the structural, thermal, shielding, criticality and confinement analyses.

2.2.2 External Conditions

SAR Section 2.2 identifies the bounding site environmental conditions and natural phenomena for which the HI-STORM FW system is analyzed.

2.3 Design Criteria for Safety Protection Systems

The principal design criteria for the HI-STORM FW system are identified in SAR Chapter 2.

2.3.1 Decommissioning

The decommissioning considerations of the HI-STORM FW system are provided in SAR Section 2.4.

2.4 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM FW system application, the staff finds the following:

F2.1 The staff concludes that the principal design criteria for the HI-STORM FW system are acceptable with regard to demonstrating compliance with the regulatory requirements of 10 CFR Part 72. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. More detailed evaluations of design criteria and assessments of compliance with those criteria are presented in SER Sections 3 through 9, and 11.

3 STRUCTURAL EVALUATION

The objectives of this review were to assess the safety analysis of the structural design features, the structural design criteria, and the structural analysis methodology used to evaluate the expected structural performance capabilities under normal operations, off-normal operations, accident conditions and natural phenomena events for those structures, systems and components important to safety included in this application.

The review was conducted against the appropriate regulations as described in 10 CFR 72.124 (a), 72.234 (a) and (b), 72.236 (b), (c), (d), (l), (g) and (h).

3.1 Structural Design of the HI-STORM FW System

The HI-STORM FW system is made up of three major components that are used in the dry spent fuel storage system and require structural evaluation: the MPC, the transfer cask HI-TRAC VW and the dry storage overpack/cask (HI-STORM FW).

3.1.1 Structural Design Features

3.1.1.1 Multi-Purpose Dry Storage Canisters

The HI-STORM FW system utilizes two MPCs as confinement vessels: the MPC-37 for PWR fuel and the MPC-89 for BWR fuel. There are no gasketed ports or openings in the MPC. The MPCs do not rely on any mechanical sealing arrangement except welding. The confinement boundary contains no valves or other pressure relief devices. The closure system for the MPCs consists of two components: the MPC lid and the closure ring. The MPC lid can be either a single thick circular plate continuously welded to the MPC shell along its circumference or two dual lids welded around their common periphery.

Both MPCs have a diameter of 75.5". Both consist of a cylindrical steel shell with a Metamic HT basket. While the diameter is constant the length of the basket can change slightly depending on the length of the fuel to be stored. The maximum length of fuel to be stored is 199.2 in for PWR fuel and 181.5 for BWR fuel (taking into account a damaged fuel canister).

3.1.1.2 HI-TRAC VW Transfer Cask

The transfer cask provides a missile and radiation barrier during transport of the MPC from the fuel pool to the HI-STORM FW overpack. The HI-TRAC VW is principally made of carbon steel and lead. The cask consists of two major parts: (a) a multi-shell cylindrical cask body, and (b) a quick connect/disconnect bottom lid. The cylindrical cask body is made of three concentric shells joined to a solid annular forging (top flange) and a solid annular plate (bottom flange) by circumferentially continuous welds. The innermost and the middle shell are fixed in place by longitudinal connector ribs that serve as radial connectors between the two shells. The radial connectors provide a continuous path for radial heat transfer and render the dual shell configuration into a stiff beam under flexural loadings. The space between these two shells is filled with lead, which provides the bulk of the transfer cask's gamma radiation shielding capability and accounts for a major portion of its weight. Between the middle shell and the outermost shell is the weldment that is referred to as the "water jacket." The water jacket is filled with water which provides most of the neutron shielding capability to the cask. The water jacket is outfitted with pressure relief devices to prevent over-pressurization in the case of an abnormal event that causes the water mass inside of it to boil.

3.1.1.3 HI-STORM FW Storage Overpack

The HI-STORM FW storage overpack is a steel cylindrical structure consisting of inner and outer low carbon steel shells, a lid, and a baseplate. Between the two shells is a thick cylinder of unreinforced (plain) concrete. Plain concrete is also installed in the lid to minimize skyshine. The testing and placement guidelines for the concrete are incorporated by reference from Appendix 1.D of the HI-STORM 100 FSAR. The storage overpack serves as a missile and radiation barrier, provides flow paths for natural convection, provides kinematic stability to the system, and acts as a shock absorber for the MPC in the event of a postulated tipover accident. The storage overpack is not a pressure vessel since it contains cooling vents.

3.1.2 Structural Design Criteria

The design loadings for the HI-STORM FW under Normal, Off-Normal, Accident, and Short-Term Operations are listed in section 2.2 of the SAR.

For the MPC, the principal design criterions are provided from its design as a Class 1 pressure vessel in accordance with Section III, Subsection NB of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The basket is designed according to displacement criteria in lieu of stress criteria.

For the HI-TRAC VW, the structural steel components are designed to meet the stress limits of Section III, Subsection NF, Class 3 of the ASME Code for normal and off-normal storage conditions. The threaded anchor locations for lifting and handling of the transfer cask are designed in accordance with the requirements of NUREG-0612 and Regulatory Guide 3.61 for interfacing lift points.

The structural steel weldment of the HI-STORM FW overpack is designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3 for normal and off-normal loading conditions and Regulatory Guide 3.61 for handling conditions. The concrete allowable compression/bearing resistance is defined and quantified in American Concrete Institute -318-05.

3.2 Weights and Centers of Gravity

Weights and centers of gravity for the HI-STORM FW system are discussed in SAR section 3.2.

3.3 Structural Analysis

3.3.1 Normal Conditions

The HI-STORM FW system is designed to withstand normal conditions of storage, which include dead weight, handling (lifting of loaded MPC, lifting and handling of HI-TRAC VW with loaded MPC, lifting and transfer to ISFSI of overpack with loaded MPC), pressure, temperatures, and snow and ice.

3.3.2 Off-Normal Conditions

The HI-STORM FW system is designed to withstand off-normal conditions, which include pressure, environmental temperatures, transient event temperatures, leakage of seal, and partial blockage of air inlets.

3.3.3 Accident-Level Events and Conditions

3.3.3.1 Cask drop and Tipover

A cask drop during handling is a non-credible event per the SAR. The lifting and handling equipment is required to have a built-in redundancy against uncontrolled lowering of the load.

LS-DYNA finite element analyses were performed for the postulated non-mechanistic tipover accident of a loaded HI-STORM FW storage cask. The analyses are based on a bounding ISFSI pad design and bounding cask dimensions to maximize the potential damage. Results of the LS-DYNA analyses demonstrated that after the tipover event, the cask closure lid remained attached to the overpack body and the overpack did not suffer a significant loss of shielding; the MPC remained in the HI-STORM FW overpack and the latter did not suffer ovalization affecting the removal of the MPC; the MPC confinement boundary was not breached and the fuel basket panels in the active fuel region did not experience any permanent deformation to change the spacing between stored fuel assemblies.

3.3.3.2 Explosive Overpressure

The explosive overpressure effect is bounded by the sliding or tipover effects for the loaded overpack and to the accident condition design external pressure for the MPC, which is 55 psig.

3.3.3.3 Fire

The HI-STORM FW Fire analysis considers a 50-gallon fuel tank fire surrounding the overpack, or in a separate case surrounding the HI-TRAC VW with an MPC. It is shown in Chapter 4 of the SAR that the fire accident has a small effect on the MPC temperatures because of the short duration of the fire accidents and the large thermal inertia of the storage overpack. Therefore, a structural evaluation of the MPC under the postulated fire event on the overpack is not required.

3.3.3.4 Flood

The postulated flood event has two potential structural consequences: stability of the HI-STORM FW system due to flood water velocity, and structural effects of hydrostatic pressure and water velocity induced lateral pressure. The hydrostatic pressure effect is bounded by the external pressure condition. The maximum acceptable water velocity for a moving flood water scenario is computed as 30.8 ft/sec, for the controlling event of sliding.

3.3.3.5 Tornado Winds

Appendix 3.A of the SAR addresses the HI-STORM FW response due to combined tornado winds and large missile impact. The case of large missile impact plus the instantaneous pressure drop due to the tornado passing the cask is also considered. Appendix 3.B of the SAR considers the cases of small and intermediate missiles. A calculation has been performed to assess sliding of the HI-STORM FW and the HI-TRAC VW after an impact from the design basis large tornado missile and tornado wind in Holtec Report HI-2094392; Appendices A and C. Subsection 3.4.4.1.3 of the HI-STORM FW SAR include a description of the sliding calculation. The analyses show that the HI-STORM FW can withstand the effects of tornado winds and missile strikes.

3.3.3.6 Earthquake

Section 3.4 of CoC Appendix B for HI-STORM FW requires a dynamic analysis to establish the system's kinematic stability on the storage pad if a static equilibrium-based inequality is not satisfied. However, the basis for this approach assumes a rigid storage pad, through which the accelerations of the cask center of gravity roughly equal the ground response spectra accelerations. The staff finds that the equilibrium-based inequality approach will be valid for rigid storage pads only.

For all non-rigid storage pads, or for earthquakes stronger than that defined by the inequalities in Subsection 2.2.3(g) of the FSAR in the case of rigid pads, it is necessary to perform a dynamic analysis. The staff does not find the use of NUREG/CR-6865 for determining cask rotation and sliding acceptable. The stability analysis should be a non-linear dynamic analysis for the cask subjected to 3-dimensional seismic accelerations. It should consider pad flexibility as appropriate, and it should follow the guidance provided in the latest revision of NUREG-0800. The forces on the cask used to determine cask stability shall be obtained from accelerations at the cask center of gravity.

3.3.4 Evaluation Findings

Based on the information provided in the SAR, by reference, and by supporting documentation, the staff concludes that the HI-STORM FW system meets the acceptance criteria specified in 10 CFR 72. The SAR adequately describes all structures, systems, and components (SSC) that are important to safety, providing drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

F3.1 The applicant has met the requirements of 10 CFR 72.24, "Contents of Application: Technical Information," with regard to information pertinent to structural evaluation.

F3.2 The applicant has met the requirements of 10 CFR 72.26, "Contents of Application," and 10 CFR 72.44(c), "License Conditions," with regard to Technical Specifications (TS) pertaining to the structures of the proposed cask system.

F3.3 The applicant has met the requirements of 10 CFR 72.122(b) and (c) and 10 CFR 72.24(c)(3). The structures, systems, and components important to safety are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask for various design loads are determined by analysis. Total stresses for the combined loads of normal, off normal, accident, and natural phenomena events are acceptable and are found to be within limits of applicable codes, standards, and specifications.

F3.4 The applicant has met the requirements of 10 CFR 72.124(a), “Criteria for Nuclear Criticality Safety”, and 10 CFR 72.236 (b), “Specific requirements for spent fuel storage cask approval.” The structural design and fabrication includes structural margins of safety for those SSC important to nuclear criticality safety. The applicant has demonstrated adequate structural safety for the handling, packaging, transfer, and storage under normal, off-normal, and accident conditions.

F3.5 The applicant has met the requirements of 10 CFR 72.236(l), "Specific Requirements for Spent Fuel Storage Cask Approval." The design analysis and submitted bases for evaluation acceptably demonstrate that the cask and other systems important to safety will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.6 The applicant has met the requirements of 10 CFR 72.120, "General Considerations," and 10 CFR 72.122, "Overall Requirements," with regard to inclusion of the following provisions in the structural design:
- design, fabrication, erection, and testing to acceptable quality standards
- adequate structural protection against environmental conditions and natural phenomena, fires, and explosions
- appropriate inspection, maintenance, and testing
- adequate accessibility in emergencies
- a confinement barrier that acceptably protects the spent fuel cladding during storage
- structures that are compatible with appropriate monitoring systems
- structural designs that are compatible with ready retrievability of spent fuel

F3.7 The applicant has met the specific requirements of 10 CFR 72.236(e)(f) (g)(h)(i)(j)(k) and (m), as they apply to the structural design for spent fuel storage cask approval. The cask system structural design acceptably provides for the following required provisions:
- redundant sealing of confinement systems
- adequate heat removal without active cooling systems
- storage of the spent fuel for a minimum of 20 years
- compatibility with wet or dry spent fuel loading and unloading facilities
- acceptable ease of decontamination
- inspections for defects that might reduce confinement effectiveness
- conspicuous and durable marking
- compatibility with removal of the stored fuel from the site, transportation, and ultimate disposition by the U.S. Department of Energy

4 HI- STORM FW SYSTEM THERMAL EVALUATION

The thermal evaluation ensures that the cask components and fuel material temperatures of the HI-STORM FW system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. These objectives include confirmation that the fuel cladding temperature will be maintained below specified limits throughout the storage period to protect the cladding against degradation that could lead to gross ruptures. This portion of the review also confirms that the cask thermal design has been evaluated using acceptable analytical techniques and/or testing methods. The review was conducted against the appropriate regulations as described in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified, as required by 10 CFR 72.236(a), so that the design basis and the design criteria that must be provided for the structures, systems, and components important to safety can be assessed under the requirements of 10 CFR 72.236(b). This application was also reviewed to evaluate that the HI-STORM FW design fulfills the acceptance criteria listed in Sections 2, 4 and 11 of NUREG-1536 as well as associated Interim Staff Guidance (ISG) documents.

4.1 Spent Fuel Cladding

The applicant adopted certain guidelines of NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” and ISG-11, Revision 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” to demonstrate the safe storage of the material content described in Chapter 2 of the SAR and in the CoC for those aspects relevant to the HI-STORM FW system design. The applicant demonstrates the HI-STORM FW system complies with the following requirements:

1. The fuel cladding temperature must meet the temperature limit appropriate to its burnup level and condition of storage or handling set forth in Table 4.3.1 of the SAR.

2. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions set forth in Table 2.2.1 of the SAR.

3. The temperatures of the cask materials shall remain below their allowable limits set forth in Table 2.2.3 of the SAR, under all scenarios.

4.2 Thermal Properties of Materials

Material property tables for the HI-STORM FW components are included in SAR Section 4.2. Materials present in the MPCs include Alloy X (defined in the SAR), Metamic-HT, aluminum, and helium. Materials present in the HI-STORM FW overpack include carbon steel and concrete. Materials present in the HI-TRAC VW transfer cask include carbon steel, lead, and demineralized water. Thermal properties provided in the SAR include thermal conductivity, density, heat capacity, gas viscosity, and emissivity. The temperature range for the material properties covers the range of temperatures encountered during the thermal analysis with some exceptions that were justified by the applicant. The staff finds the material properties used by the applicant in the thermal analyses of HI-STORM FW acceptable.

4.3 Specifications of Components

HI-STORM FW system materials and components designated as "Important to Safety" (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) are summarized in Table 2.2.3 of the SAR. For evaluation of HI-STORM FW thermal performance, material temperature limits for long term normal, short-term operations, and off-normal and accident conditions are provided in Table 4.3.1 of the SAR. Fuel cladding temperature limits included in Table 4.3.1 of the SAR are consistent with ISG-11. These limits are applicable to all fuel types, burnup levels, and cladding materials approved by the NRC for power generation.

4.4 HI-STORM FW System

4.4.1 Thermal Design Criteria

The thermal design and operation of the MPC in the HI-STORM FW system meet the intent of the review guidance contained in ISG-11, Revision 3. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

1. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

2. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burnup fuel (HBF) and 570ºC (1058ºF) for moderate burnup fuel.

3. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570ºC (1058ºF).

4. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65ºC (117ºF) and the number of excursions to less than 10.

The thru-thickness temperature limits for the plain concrete in the overpack for long term and short term temperatures are set identical to those in the HI-STORM 100 FSAR (see HI-STORM FW SAR Table 2.2.3). The allowable temperatures for the structural steel components are based on the maximum temperature for which material properties and allowable stresses are provided in Section II of the ASME Code. The specific allowable temperatures for the structural steel components of the overpack are provided in SAR Table 2.2.3. The overpack is designed for extreme cold conditions, as discussed in SAR Subsection 2.2.2. The brittle fracture assessment of structural steel materials used in the storage cask is considered in SAR Section 3.1. The overpack is designed to dissipate the maximum allowable heat load (shown in Tables 1.2.3 and 1.2.4 of the SAR) from the MPC.

The allowable temperatures for the HI-TRAC VW transfer cask structural steel components are based on the maximum temperature for material properties and allowable stress values provided in Section II of the ASME Code. The allowable temperatures for the structural steel and shielding components of the HI-TRAC VW are provided in Table 2.2.3 of the SAR. The HI-TRAC VW is designed for off-normal environmental cold conditions, as discussed in Subsection 2.2.2 of the SAR. The HI-TRAC VW is designed and evaluated for the maximum heat load analyzed for storage operations. The maximum allowable temperature of water in the HI-TRAC jacket is a function of the internal pressure. To preclude over- pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is restricted to be less than the saturation temperature at the shell design pressure.

4.4.2 Thermal Design Features

To ensure the permissible Peak Centerline Temperature limits are not exceeded, SAR Subsection 1.2 specifies the maximum allowable decay heat per assembly for each MPC model in the three-region configuration. The following tables summarize the heat load data for MPC-37 and MPC-89.

MPC-37 Heat Load Data			
Number of Regions: 3			
Number of storage cells: 37			
Maximum Heat Load: 47.05 kW			
Region Number	Decay Heat Limit per Cell, kW	Number of Storage Cells per Region	Decay Heat Limit per Region, kW
1	1.13	9	10.17
2	1.78	12	21.36
3	0.97	16	15.52

MPC-89 Heat Load Data			
Number of Regions: 3			
Number of storage cells: 89			
Maximum Heat Load: 46.36 kW			
Region Number	Decay Heat Limit per Cell, kW	Number of Storage Cells per Region	Decay Heat Limit per Region, kW
1	0.44	9	3.96
2	0.62	40	24.80
3	0.44	40	17.60

The staff finds the description, design criteria, and design features of the HI-STORM FW design acceptable.

4.5 HI-STORM FW System Thermal Model

The applicant used Fluent program to evaluate the thermal performance of the HI-STORM FW spent fuel storage system. Fluent is a finite volume computational fluid dynamics (CFD) program with capabilities to predict fluid flow and heat transfer phenomena in two and three dimensions. The thermal analysis model developed by the applicant has the following key attributes:

1. The Metamic-HT fuel basket is modeled in the same manner as the model described in the HI-STAR 180 FSAR. Fuel storage spaces are modeled as porous media.

2. In the case of a BWR CSF, the fuel bundle and the small surrounding spaces inside the fuel “channel” are replaced by an equivalent porous media having the flow impedance properties computed using a 3-Dimensional (3-D) CFD model. The space between the BWR fuel channel and the storage cell is represented as an open flow annulus. The fuel channel is also explicitly modeled. The channeled space within is also referred to as the “rodded region” that is modeled as a porous medium. The fuel assembly is assumed to be positioned coaxially with respect to its storage cell. In the case of the PWR CSF, the porous medium extends to the entire cross-section of the storage cell. The CFD model for both the BWR and PWR flow resistance calculations is prepared for the design basis fuel defined in Table 2.1.4 of the SAR in comprehensive detail, which includes grid straps, BWR water rods and PWR guide and instrument tubes (assumed to be plugged for conservatism).

3. Every MPC fuel storage cell is assumed to be occupied by design basis PWR or BWR fuel assemblies specified in Chapter 2 (SAR Table 2.1.4). The in-plane effective thermal conductivity of the design basis fuel assembly is obtained using ANSYS finite element models of an array of fuel rods enclosed by a square box. Radiation heat transfer from solid surfaces (cladding and box walls) is enabled in these models. Using these models the effective conduction-radiation conductivities are obtained and reported in SAR Table 4.4.1. For heat transfer in the axial direction an area weighted mean of cladding and helium conductivities are computed. Axial conduction heat transfer in the fuel pellets and radiation heat dissipation in the axial direction are conservatively ignored. Thus, the thermal conductivity of the rodded region, like the porous media simulation for helium flow, is represented by a 3-D continuum having effective planar and axial conductivities.

4. The internals of the MPC, including the basket cross-section, aluminum shims, bottom flow holes, top plenum, and circumferentially irregular downcomer formed by the annulus gap in the aluminum shims are modeled explicitly. For simplicity, the flow holes are modeled as rectangular openings with an understated flow area.

5. The inlet and outlet vents in the HI-STORM FW overpack are modeled explicitly to incorporate any effects of non-axisymmetry of inlet air passages on the system’s thermal performance.

6. The air flow in the HI-STORM FW/MPC annulus is simulated by the k-ω turbulence model with the transitional option enabled.

7. A limited number of fuel assemblies (up to 12 in MPC-37 and up to 16 in MPC-89) classified as damaged fuel are permitted to be stored in the MPC inside Damaged Fuel Containers (DFCs). A DFC can be stored in the outer peripheral locations of both MPC-37 and MPC-89 as shown in SAR Figures 2.1.1 and 2.1.2, respectively. DFC emplaced fuel assemblies have a higher resistance to helium flow because of the debris screens. However, DFC fuel storage does not affect temperature of hot fuel stored in the core of the basket because DFC storage is limited by TS for placement in the peripheral storage locations away from hot fuel. For this reason the thermal modeling of the fuel basket under the assumption of all storage spaces populated with intact fuel is justified.

8. As shown in HI-STORM FW drawings in SAR Section 1.5, the HI-STORM FW overpack is equipped with a heat shield to protect the inner shell and concrete from radiation heating by the emplaced MPC. The heat shield, inner and outer shells and concrete are explicitly modeled.

The staff finds the description HI-STORM FW thermal models acceptable.

4.6 Thermal Evaluation for Normal Conditions of Storage

The applicant used the 3-D model described in the previous section to determine temperature distributions under long-term normal storage conditions for both MPC-89 and MPC-37. SAR Tables 4.4.2, 4.4.3 and 4.4.5 provide key thermal and pressure results. From the presented results it can be concluded that that the temperature field in the HI-STORM FW system with a loaded MPC containing heat emitting SNF complies with all regulatory temperature limits (SAR Table 2.2.3). The thermal environment in the HI-STORM FW system is in compliance with SAR Chapter 2 Design Criteria. Per SAR Chapter 3, all HI-STORM FW storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode under a minimum temperature condition of -40°F.

The storage scenarios described above assumed the HI-STORM FW storage system is located at sea level. However, if an ISFSI is located at an elevation greater than sea level, the effect of altitude on the peak cladding temperature shall be quantified as part of the 10 CFR 72.212 evaluations for the site using the site ambient conditions.

SAR Table 4.4.4 presents a summary of the MPC free volumes determined for the fixed height MPC-89 and lowerbound height MPC-37 fuel storage scenarios. The applicant calculated the MPC maximum gas pressure for a postulated release of fission product gases from fuel rods into this free space. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. Based on fission gases release fractions (NUREG 1536 criteria),fuel rods' net free volume and initial fill gas pressure, maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in SAR Table 4.4.5. The maximum computed gas pressures reported in SAR Table 4.4.5 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in SAR Table 2.2.1.

4.7 Thermal Evaluation for Short-Term Operations

Prior to placement in a HI-STORM FW overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, dried, backfilled with helium and transported to the HI-STORM FW module. If the fuel needs to be returned to the spent fuel pool, these steps must be performed in reverse. Finally, if required, transfer of a loaded MPC between HI-STORM FW overpacks or between a HI-STAR transport overpack and a HI-STORM FW storage overpack must be carried out in a safe manner. All of the above operations are short duration events that would likely occur no more than once or twice for an individual MPC.

4.7.1 Drying

4.7.1.1 Vacuum

The vacuum drying option is evaluated for the two limiting scenarios defined in SAR Section 4.5.2.2 to address Moderate Burnup Fuel (MBF) under design basis heat load and HBF under threshold heat load defined in SAR Table 4.5.1. The principle objective of the analysis is to ensure compliance with ISG-11 temperature limits. For this purpose the applicant developed 3-D FLUENT thermal models of the MPC-37 and MPC-89 canisters, as described in SAR Section 4.5.2.2 and bounding steady state temperatures computed. The results are tabulated in SAR Tables 4.5.6 and 4.5.7. The results show that the cladding temperatures comply with the ISG-11 limits for moderate and high burnup fuel in SAR Table 4.3.1 by adequate margins.

4.7.1.2 Forced Helium Dehydration

A FHDS provides concurrent fuel cooling during the moisture removal process through forced convective heat transfer. The attendant forced convection-aided heat transfer occurring during operation of the FHDS ensures that the fuel cladding temperature will remain below the applicable peak cladding temperature limit in SAR Table 2.2.3.

4.7.1.3 On-Site Transfer

The applicant evaluated an MPC-37 situated inside a HI-TRAC VW under the design heat load defined in SAR Section 1.2. The MPC-37 is evaluated because it yields the highest fuel and cask temperatures (per SAR Table 4.4.2). This scenario is analyzed using the same 3-D FLUENT model of the MPC-37 described in SAR Section 4.4 for normal storage inside the HI-TRAC VW transfer cask. The applicant's approach is to assume steady state maximum temperatures are reached. The results of thermal analyses tabulated in SAR Table 4.5.2 show that the cladding temperatures are below the ISG-11 temperature limits of High and Moderate Burnup Fuel (SAR Table 4.3.1). Under HI-TRAC VW operations, the bulk temperature of water in the water jacket surrounding the HI-TRAC VW body remains below the temperature limit specified in SAR Table 2.2.3.

4.8 Off-Normal and Accident Events

4.8.1 Off-Normal Events

The applicant considered three off-normal events: off-normal pressure, off-normal environmental temperature, and partial blockage of air inlets. The MPC off-normal pressures are reported in SAR Table 4.6.7. The result is below the off-normal design pressure (SAR Table 2.2.1). The off-normal temperature results are provided in SAR Table 4.6.1 and 4.6.7. The results are below the off-normal condition temperature and pressure limits (SAR Tables 2.2.3 and 2.2.1). The computed temperatures for the partial blockage of air inlets are reported in SAR Table 4.6.1 and the corresponding MPC internal pressure in SAR Table 4.6.7. The results are confirmed to be below the temperature and pressure limits (SAR Table 2.2.3 and 2.2.1) for off-normal conditions.

4.8.2 Accident Events

The applicant considered five accident events: fire, jacket water loss, extreme environmental temperatures, 100% blockage of air ducts, and burial under debris. Accident analyses results are provided in SAR Tables 4.6.2, 4.6.3, 4.6.4, 4.6.5, 4.6.6, and 4.6.7. All predicted maximum temperatures and pressures remain below the accident limits defined in SAR Table 2.2.1 (accident design pressure) and Table 2.2.3 (accident temperature limit).

The staff finds the description, assumptions, and analysis results of normal, off-normal and accident events acceptable.

4.9 Thermal tests

HI-STORM FW thermal tests are described in Chapter 9 of the SAR. Per Condition No. 8 of the CoC (if required), the user must perform an annular air flow thermal test to measure and confirm the mass flow rate predictions obtained by applying Chapter 4 thermal models. Also, as described in SAR Chapter 10, the first manufactured HI-STORM FW MPC will be thermally tested to confirm the thermal models described in Chapter 4.

4.10 Confirmatory Analysis

The staff reviewed the applicant's models and calculation options to determine the adequacy of the proposed HI-STORM FW thermal design. Additionally, the staff performed selected confirmatory analyses using the FLUENT finite volume CFD code, as an independent evaluation of the thermal analysis and modeling options presented in the applicant's SAR.

Specifically the staff developed a 3-D CFD model of HI-STORM FW loaded with MPC-37. Using the staff's confirmatory analysis, applicants predictions were confirmed because the staff's calculated results were very similar to the SAR results. Also, to confirm the applicant's approach to deal with the effect of surrounding casks (as described in the SAR), the staff developed a CFD model which includes a cask located in the center of an array in proximity to other casks. In order to simplify the calculations and to obtain bounding results, the staff assumed symmetry conditions on the lateral air flow and imposed pressure boundary conditions for the top of the array. The purpose of this analysis was to determine if the presence of other casks (as in the real case) would have any effect on the temperature of the inlet vents and also to determine if heat transfer by radiation from other casks would increase the predicted peak cladding temperature. The staff observed that the presence of surrounding casks had a minor effect on the temperature at the inlet vents and on the peak cladding temperature. Therefore, the staff find's the vendor's approach to include the effect of surrounding casks acceptable.

4.11 Evaluation Findings

F4.1 Chapter 2 of the SAR describes structures, systems, and components (SSCs) important to safety to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

F4.2 The HI-STORM FW storage system is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. Except during short-term operations, the cask is designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The spent fuel cladding is protected against degradation leading to gross ruptures under long-term storage by maintaining cladding temperatures below 752°F (400°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.4 The spent fuel cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining cladding temperatures below 1058°F (570°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.5 The staff finds that the thermal design of the HI-STORM FW storage system is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5 CONFINEMENT EVALUATION

5.1 Confinement Boundary

Section 7.1 of the application lists the confinement boundary of the HI-STORM FW MPC as the MPC shell, base plate, lid, vent and drain port covers, closure ring, and associated welds. These components are constructed from 304, 316, 304LN, or 316LN austenitic stainless steel. The lid may be constructed from both austenitic stainless steel and carbon steel. There is no safety concern regarding the use of this two-piece configuration, as the carbon steel portion of the two-piece lid is non-structural. The confinement boundary welds are specifically described in Table 7.1.1 of the application and categorized in accordance with the ASME Code Section III, Subsection NB, Article-3351. Attachment welds (and temporary welds to the confinement boundary) will be examined in accordance with ASME Code Section V with acceptance criteria per ASME Code Section III, Subsection NB. These welds shall be repaired in accordance with the requirements of the ASME Code Section III, Subsection NB and examined after repair in the same manner as the original weld.

5.2 Shell Seams and Shell-to-Base Plate Shop Welds

The TS require that the MPC shop welds (shell seams and shell-to-base plate shop welds) shall be helium leak tested and found “leak-tight” in accordance with the requirements of ANSI N14.5 as part of the initial acceptance criteria. This is to ensure that the body of the MPC is leak tight.

5.3 MPC Lid to Shell Weld

After loading of the spent fuel into the MPC, the MPC lid is moved on to the MPC while underwater. The MPC is then lifted from the spent fuel pool and the water level in the MPC is lowered slightly. An inert gas is used to flood the area between the MPC lid and the submerged fuel assemblies as stated in Table 3-2 of the TS (Appendix A). An automated welding device is then attached to the top of the MPC lid which performs the multi-pass lid to shell weld. As required by Section 3.5 of the TS (Appendix B), the level of combustible gas in the MPC is monitored during welding of the MPC. Monitoring is necessary to prevent ignition of hydrogen which is created by corrosion of components internal to the MPC.

The staff finds the use of inert gas and hydrogen monitoring acceptable during welding of the MPC lid in compliance with the recommended guidance in Section 8.4.1 NUREG-1536, Revision 1A, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility Draft General Review Considerations."

The MPC lid-to-shell weld is then examined by multi-layer liquid penetrant examinations (PT) at each 3/8-inch weld deposition, and a pressure test is performed. Both the PT examination of the MPC lid-to-shell weld and subsequent pressure test is specified in the Table 3-1 of the TS (Appendix B). The MPC lid-to shell weld is not helium leakage tested since the weld meets the guidance of NRC ISG-15 and criteria of ISG-18, therefore leakage from the MPC lid-to-shell weld is not considered credible.

5.4 Drying

According to Table 3-1 of the TS (Appendix A), the MPC shall be dried using a vacuum drying system (VDS) or a FHDS. The acceptance criterion when using a VDS is a MPC cavity pressure less than three torr for at least 30 minutes, while isolated from the vacuum pump. The acceptance criterion when using an FHDS is the gas temperature exiting the demoisturizer shall be less than 21°F for greater than 30 minutes or the gas dew point exiting the MPC shall be less than 22.9°F for greater than 30 minutes. These acceptance criteria (for VDS and FHDS) are necessary to ensure adequate moisture removal from the MPC cavities.

5.5 Backfilling

Table 3-2 of the TS (Appendix A) requires that the MPC shall be backfilled with 99.995% pure helium and specifies the minimum and maximum helium backfills required by potential heat loads. A minimum helium purity and bounding limits on the helium pressure are required to ensure the anticipated heat transfer characteristics within the cask and to maintain a chemically inert environment. The staff finds the bounding conditions and the purity of helium acceptable for the application.

5.6 Vent and Drain Cover Plates

After the welding of the MPC lid, drying and backfilling, the vent and drain port cover plates are welded in place, examined by the liquid penetrant method and helium leakage test of the vent and drain port cover plates is performed. These welds are tested to the leak tight criteria of 10^{-7} cm^3/s as specified in Table 3-1 of the TS (Appendix A).

5.7 MPC Closure Ring

After helium leak-testing of the vent and drain port cover plates, the closure ring is placed on the MPC and welded in place. The root and final welds are partial penetration welds, as specified in Table 7.1.1 of the application. The MPC closure ring welds will be examined using PT in accordance with ASME Code Section III, Subsection NB.

5.8 Evaluation Findings

Based on the above statements, the staff has the following evaluation findings with respect to the containment analysis:

F5.1 There is no safety concern regarding the use of this two-piece configuration, as the carbon steel portion of the two-piece lid is non-structural. The use of ASME Code Section III, Subsection NB, with the exceptions listed in Table 2.2.14 of the Safety Analysis Report and Table 3-1 of the Technical Specifications (Appendix B), is an acceptable code of construction for the welding of the confinement boundary.

F5.2 The use of helium leak testing in accordance with the requirements of ANSI N14.5 is consistent with the guidance found in Draft NUREG-1536 Rev. 1C, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility."

F5.3 Monitoring of combustible gas during welding of the MPC lid will prevent the ignition of hydrogen created by corrosion of components internal to the MPC.

F5.4 The acceptance criteria for cask drying as identified in the TS is similar to that approved by the staff for the HI-STORM 100 Dry Cask System, CoC No. 1014.

F5.5 A minimum helium purity and bounding limits on the helium pressure are required to ensure the anticipated heat transfer characteristics within the cask to maintain a chemically inert environment.

F5.6 The testing of the vent and drain port cover plates to the leak tight criteria of 10^{-7} cm^3/s as specified in Table 3-1 of the TS (Appendix A) will verify the adequacy of containment.

F5.7 The MPC Closure ring provides redundant confinement and meets the requirements of 10 CFR 72.236(e).

The staff concludes that the containment design features for the HI-STORM FW comply with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the containment design provides reasonable assurance that the HI-STORM FW will allow safe storage of spent fuel. These findings are reached based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

6 SHIELDING EVALUATION

The applicant proposed a new spent fuel storage overpack design, designated the HI-STORM FW. The HI-STORM FW system consists of a sealed metallic MPC contained within an overpack constructed from a combination of steel and concrete. The MPC's are designed to store intact PWR and BWR fuel assemblies. Also, the HI-STORM FW system is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Both damaged fuel assemblies and fuel debris are required to be loaded into DFCs. PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs), neutron source assemblies (NSAs) or similarly named devices. According to the applicant, these non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM FW system has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices.

6.1 Shielding Design Description

6.1.1 Design Criteria

The applicant used as a basis for protection against direct radiation the fact that the off-site dose for normal operating conditions to a real individual beyond the controlled area boundary is limited by 10 CFR 72.104 (a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations. Also, the applicant stated since these limits are dependent on plant operations as well as site specific conditions, the determination and comparison of ISFSI doses to this limit are necessarily site-specific. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10 CFR 72.212.

The overpack is designed to limit the calculated surface dose rates on the cask for all MPC designs as defined in Subsection 2.3.5 of the SAR. According to the applicant, the overpack is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10 CFR Part 20. The calculated overpack dose rates are determined in Section 5.1 of the SAR.

The staff reviewed the design criteria and found them acceptable. Each user will be required to protect personnel and minimize dose in accordance with ALARA principles and the regulations of 10 CFR Part 20.

6.1.2 Shielding Design Features

The applicant states that the shielding from the gamma radiation of the HI-STORM FW overpack during loading, unloading, and transfer operations is provided by the stainless steel structure and the basket of the MPC and the steel, lead, and water in the HI-TRAC transfer cask. For storage, the applicant states that the gamma shielding is provided by the MPC, and the steel and concrete (“Metcon” structure) of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water or Holtite of the HI-TRAC is provided transfer cask during loading, unloading, and transfer operations. The applicant performed the shielding analysis for the HI-STORM FW overpack containing an MPC-37 and loaded with intact design-basis fuel and determined dose rates for the positions shown in Figure 5.I.1 of the SAR.

The shielding analyses were performed with MCNP5 computer code. The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5 system. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are Westinghouse 17x17 and the General Electric 10x10, for PWR and BWR fuel types, respectively. The acceptable fuel characteristics, including the acceptable maximum burnup levels and minimum cooling times for storage of fuel in the HI-STORM FW MPCs are specify in Subsection 2.1 of the SAR. Required site specific shielding evaluations will verify whether those assemblies and assembly parameters are appropriate for the site-specific analyses.

The staff finds the shielding design features to be acceptable. Based on information provided by the applicant, the staff has reasonable assurance that the shielding design features of the HI-STORM FW system can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72.

6.2 Source Specification

The design-basis source specifications for bounding calculations are presented in Section 5.2 of the SAR. The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE5 system. Sample input files for SAS2H and ORIGEN-S were provided in Appendix 5.A of the SAR. The applicant states that the gamma source term was comprised of three distinct sources. The first was a gamma source term from the active fuel region due to decay of fission products. The second source term was from ^{60}Co activity of the stainless steel structural material in the fuel element above and below the active fuel region. The third source was from (n,γ) reactions described below. Table 5.2.1 of the SAR provided a description of the design basis fuel for the source term calculations. The determination of the design basis fuel assemblies is discussed in detail in Subsection 5.2.5 of the SAR. In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Table 5.2.1 resulted in conservative source term calculations. The staff's reviewed of the source term analyses and found the calculations to be acceptable.

6.2.1 Gamma Source

The gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuel at the burnups and cooling times used for normal and accident conditions were provided in Tables 5.2.2 through 5.2.5 of the SAR.

All photons with energies in the range of 0.45 to 3.0 MeV were included in the shielding calculations. According to the applicant, based on previous analyses that were performed for the HI-STORM 100 Dry Cask System, CoC No. 1014 and due to the magnitude of the gamma source at lower energies, photons with energies as low as 0.45 MeV must be included in the shielding analysis, but photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant. This is due to the fact that the source of gammas in this range is extremely low.

6.2.2 Neutron Source

The applicant states that the low initial fuel enrichments of 3.2 and 3.6 wt% were chosen for the BWR and PWR design basis fuel assemblies under normal conditions, respectively. For the accident conditions, a fuel enrichment of 4.8 wt% was chosen to accommodate the higher burnups of the selected source terms.

The neutron source calculated for the design basis fuel assemblies for the MPCs and the design basis fuel are listed in Tables 5.2.11 through 5.2.14 in neutrons/s for the selected burnup and cooling times used in the shielding evaluations for normal and accident conditions. The neutron spectrum was generated in ORIGEN-S.

The staff's reviewed of the source term analyses and found the calculations to be acceptable.

6.2.3 Non-Fuel Hardware

The applicant stated that BPRAs, TPDs, CRAs, and APSRs are permitted for storage in the HI-STORM FW system as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted as specified in Subsection 2.1 of the SAR. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis W 17x17 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.6 of the SAR in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time. The HI-STORM FW SAR presents dose rates for both BPRAs and TPDs. The results indicate that BPRAs are bounding, therefore all dose rates in this chapter will contain a BPRA in every PWR fuel location. Subsection 5.4.4 discusses the increase in the cask dose rates due to the insertion of BPRAs into fuel assemblies.

6.3 Shielding Model Specifications

The HI-STORM FW shielding and source configuration is described in Section 5.I.3 as well as Sections 5.3 and 5.4 of the SAR. The shielding analysis of the HI-STORM FW system was performed with MCNP5. A sample input file for MCNP was provided in Appendix 5.A of the SAR.

In subsection 5.1.1 of the SAR, the applicant stated that off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Subsection 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Subsection 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

6.3.1 Shielding and Source Configuration

The drawings that describe the HI-STORM FW system, including the HI-TRAC transfer cask were provided in Chapter 1 of the SAR. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. The inlet and outlet vents were modeled explicitly; therefore, streaming through these components was accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. A cross sectional view of the HI-TRAC VW with the MPC-37 and MPC-89, respectively, as it was modeled in MCNP were showed in Figures 5.3.3 and 5.3.4 of the SAR. Figure 5.3.5 of the SAR shows a cross sectional view of the HI-STORM FW overpack with the as-modeled thickness of the various materials.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed on Page 5-42 of the SAR.

In term of potential streaming path, the applicant included the inlet and outlets vents in the MCNP model of the HI-STORM overpack, accounting for their streaming effect. Also, the top lid was modeled with its reduced diameter, which accounts for higher localized dose rates on the top surface of the HI-STORM. The MCNP model of the HI-TRAC transfer cask accounts for the fins through the HI-TRAC water jacket, as discussed in Subsection 5.4.1 of the SAR, as well as the open annulus.

In term of fuel configuration, the applicant modeled the active fuel region as a homogenous zone. The end fittings and the plenum regions were also modeled as homogenous regions of steel. The masses of steel used in these regions were shown in Table 5.2.1 of the SAR. The axial description of the design basis fuel assemblies was provided in Table 5.3.1 of the SAR.

The staff reviewed the applicant's shielding models used in the analyses. The staff checked the code input in the calculation packages and confirmed that the proper material properties and boundary conditions were used. The engineering drawings were also consulted to verify that proper geometry dimensions were translated to the analysis model. The material properties presented in the SAR, were reviewed to verify that they were appropriately referenced and used.

The staff performed shielding and source term calculations using SCALE6 to compare photon and neutron sources.

The staff concludes that the design of the shielding system for the HI-STORM FW system is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the HI-STORM FW system will provide safe storage of spent fuel. This finding is based on a review that considered the specifications in the SAR Revision 1, the regulations, appropriate regulatory guides, staff confirmatory (including calculations and modeling) analysis, and accepted engineering practices. The staff reviewed the external radiation levels under normal conditions of transport and hypothetical accident conditions and found reasonable assurance that they satisfy 10 CFR 72.

6.3.2 Material Properties

The applicant used the composition and density specified in Table 5.3.2 of the SAR. The shielding calculations used a minimum allowable concrete density shown in Table 5.3.2 of the SAR, to determine dose rates; however, SAR Section 5.3.2 notes that the concrete density can be increased up to 200 lb/ft^3 at the request of the user to improve the shielding characteristics of the system and address potential ALARA considerations. The concrete density shown in Table 5.3.2

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. The applicant mentioned that the HI-TRAC transfer cask may be equipped with a water jacket to provide radial neutron shielding. Demineralized water (borated water) will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

6.3.3 Staff Evaluation

The staff evaluated the shielding models and found them to be acceptable. Based on the statements and calculations presented by the applicant, the staff finds the model is valid for the HI-STORM FW system.

6.4 Shielding Analyses

6.4.1 Computer program

The applicant used the MCNP-5 code for all of the shielding analyses. The energy distribution of the source term was used explicitly in the MCNP model. A different MCNP calculation was performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term was described in Table 2.1.5 and Figures 2.1.3 and 2.1.4 of the SAR. The PWR and BWR axial burnup distributions were obtained from different references. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

The applicant determined the neutron source strength in each of the 10 axial nodes listed in Table 2.1.5 by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.5 of the SAR for the PWR and BWR fuels resulted to be 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6%, and 76.8% increase in the neutron source strength in the peak nodes for the PWR and BWR fuel, respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies. MCNP was used to calculate doses at the various desired locations.

The design basis dose rates adjacent to the HI-STORM overpack during normal conditions for the MPC types in Table 1.0.1 were provided in Tables 5.1.6 and 5.1.7 of the SAR. Table 5.1.8 of the SAR provided the design basis dose rates at one meter from the overpack containing the MPC-37. A detailed discussion of the normal, off-normal, and accident condition dose rates was provided in Subsections 5.1.1 and 5.1.2 of the SAR.

6.4.2 Flux-to-Dose-Rate Conversion

MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. The response functions used in these calculations were listed in Table 5.4.1 of the SAR and were taken from ANSI/ANS 6.1.1, 1977.

6.4.3 Dose Rates

6.4.3.1 Normal Conditions

The potential off-normal conditions and their effect on the HI-STORM FW system were discussed in Chapter 12. According to the applicant, none of the off-normal conditions had any impact on the shielding analysis. Thus, off-normal and normal conditions are identical for the purpose of the shielding evaluation. The 10 CFR 72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

> 1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.
>
> 2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation. 10 CFR Part 20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public, and Chapter 11 specifically addresses these regulations. In accordance with ALARA practices, design objective dose rates are established for the HI-STORM FW system in Table 11.3.2.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM FW system in Table 11.3.2 of the SAR.

The staff reviewed the dose rates of normal condition found them to be acceptable.

6.4.3.2 Occupational Exposures

The applicant states that the HI-TRAC VW transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10 CFR Part 20, while also maintaining the maximum load on the plant's crane hook to below the rated capacity of the crane. The HI-TRAC VW calculated dose rates for a set of reference conditions are reported in Section 5.1 of the SAR.

These dose rates were used to perform a generic occupational exposure estimate for MPC loading, closure, and transfer operations. A postulated HI-TRAC VW accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in this Section.

The annular area between the MPC outer surface and the HI-TRAC VW inner surface can be isolated to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC VW surfaces expected to require decontamination are coated with a suitable coating. The maximum permissible surface contamination for the HI-TRAC VW is in accordance with plant-specific procedures and ALARA requirements.

6.4.3.3 Off-Site Dose Calculation

The off-site dose for normal operating conditions to a real individual beyond the controlled area boundary is limited by 10 CFR 72.104(a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations. According to the applicant, since these limits are dependent on plant operations as well as site specific conditions, the determination and comparison of ISFSI doses to this limit are necessarily site-specific. Dose rates for a single cask and a range of typical ISFSIs using the HI-STORM FW system are provided in Table 5.4.7 of the SAR. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10 CFR 72.212.

The overpack is designed to limit the calculated surface dose rates on the cask for all MPC designs as defined in Subsection 2.3.5 of the SAR. The overpack is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10 CFR Part 20. The calculated overpack dose rates are determined in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC transfer operations and a site boundary dose assessment for a typical ISFSI, as described in Chapter 11.

6.4.3.4 Accident Conditions

The applicant states that structural evaluations, presented in Chapter 3 of the SAR, shows that a freestanding HI-STORM FW storage overpack containing a loaded MPC remains standing during events that could potentially lead to a tip-over event. Thus, the tip-over accident is not considered as part of the shielding evaluation. Design basis accidents which may affect the HI-STORM FW overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary, adjacent, and one meter doses for the loaded HI-STORM FW overpack for accident conditions are equivalent to the normal condition doses, which meet the 10 CFR 72.106 radiation dose limits.

The design basis accidents analyzed in Chapter 11 of the SAR have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the MPC’s design features. Localized damage of the HI-TRAC outer shell is possible but, according to the applicant, localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

6.5 Staff Evaluation

The staff reviewed the dose calculations for normal operations and found them acceptable. Dose rates were calculated for the FW loaded with design-basis contents. The staff has reasonable assurance that compliance with 10 CFR Part 20 and 10 CFR 72.104(a) from direct radiation can be achieved by general licensees. The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and distances. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities, such as reactor operations. Each general licensee is responsible to verify compliance with 10 CFR 72.104(a) in accordance with 10 CFR 72.212. In addition, a general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B and will demonstrate compliance with dose limits to individual members of the public and workers (including for excavation activities), as required, by evaluation and measurements. The staff notes that the system contents result in relatively significant direct radiation dose rates, which is a concern primarily for operations involving the transfer cask (i.e., loading, unloading, and transport) for the FW system. Thus, each user may be required to take additional ALARA precautions to minimize doses to personnel and to make additional use of realistic fuel characteristics and distances to demonstrate compliance with public dose limits in 10 CFR Part 20 and 10 CFR Part 72.

The staff reviewed the accident evaluation and found it acceptable for the design changes requested in the application. The staff has reasonable assurance that the direct radiation from the FW satisfies 10 CFR 72.106(b) at or beyond a controlled boundary of 100 meters from the design-basis accidents.

6.6 Evaluation Findings

Based on the NRC staff's review of information provided for the HI-STORM FW application, the staff finds the following:

F6.1 The SAR sufficiently describes shielding design features and design criteria for the structures, systems, and components important to safety.

F6.2 Radiation shielding features of the HI-STORM FW are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F6.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104 and 72.106 are the responsibility of each general licensee. The HI-STORM FW shielding features are designed to satisfy these requirements.

F6.4 The staff finds the design addresses construction activities involving excavation (for ISFSI expansion) adjacent to the (operating) FW system sufficient to ensure that the shielding features will continue to be sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F6.5 The staff concludes that the design of the radiation protection system of the HI-STORM FW can be operated in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM FW system will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant’s analyses, the staff’s confirmatory analyses, and acceptable engineering practices.

7 CRITICALITY EVALUATION

7.1 Criticality Design Criteria and Features

The staff viewed information in Sections 1, 2 and 6 of the HI-STORM FW SAR and verified that the information is consistent as well as all descriptions, drawings, figures and tables are sufficiently detailed to support an in-depth staff evaluation.

Criticality is controlled within the HI-STORM FW system based on the fixed geometry of the basket structure, and neutron poison (Boron-10) present within the basket material (Metamic-HT). The HI-STORM FW system has two different basket designs. The MPC-37 is designed for PWR fuel and the MPC-89 for BWR fuel. For the MPC-37 for PWR fuel, the applicant also relies upon the soluble boron within the MPC water during loading and unloading operations for criticality control. The staff confirmed that the applicant has appropriate TS limits and surveillance requirements for soluble boron concentrations within the PWR MPC during loading and unloading operations. Administrative control on the soluble boron concentration during loading and unloading of the MPC-37 consists of frequent and independent measurements and is described in the HI-STORM FW SAR Operating Procedures, Section 9. An accidental loss of soluble boron is therefore not credible and hence not considered. The staff finds this acceptable.

7.2 Fuel Specification

The staff examined the specifications for the types of SNF that will be stored in the HI-STORM FW system. The applicant proposes to store both PWR and BWR type light water reactor (LWR) fuels. Tables 2.1.2 and 2.1.3 of the HI-STORM FW SAR list the fuel specifications for each type of PWR and BWR fuel, respectively.

The applicant proposes 16 different PWR fuel types and 18 different BWR fuel types. Although this fuel may be vendor and utility specific fuel, it is defined using generic specifications. These include:

- Lattice
- Number of fuel rods
- Minimum Fuel Clad OD
- Maximum Fuel Clad ID
- Maximum Fuel Pellet Diameter
- Maximum Fuel Rod Pitch
- Maximum Active Fuel Length
- Number of Guide and/or Instrument Tubes (PWR)
- Maximum Guide and/or Instrument Tube Thickness (PWR)
- Number of Water Rods (BWR)
- Minimum Water Rod Thickness (BWR)
- Maximum Channel Thickness (BWR)

In addition to the above fuel specifications, there are foot notes to help further define each fuel type.

The applicant states that the above characteristics were found to be most reactive based on the assumption that all of the fuel assemblies to be stored in the HI-STORM FW are undermoderated. The applicant provided some calculations (Table 6.2.3 of the HI-STORM FW SAR) demonstrating that all of the fuel assemblies are undermoderated. Previous experience using analyses performed in support of the HI-STORM 100 Dry Cask system, CoC No. 1014 also demonstrate that the fuel specifications are limiting. In addition, the applicant performed further analysis with the PWR 17X17B and BWR 10x10A fuel assemblies showing that the above listed specifications to give the most reactive conditions for these assemblies. The staff finds this acceptable.

The applicant is using maximum planar averaged enrichments to specify BWR fuel rather than maximum enrichment. The applicant references the HI-STORM 100 Dry Cask system application and in this application they have shown that this is conservative for several 8x8 and 9x9 lattices within the geometry of the MPC-68. For the HI-STORM FW application, the applicant re-performed these calculations in the MPC-89 and presented the results in Table 6.2.4 of the HI-STORM FW SAR. The applicant also provided additional calculations using the higher enrichments (4.8% planar average) for a 10x10 fuel lattice for four different and bounding variable enrichment patterns. The results are shown in Table 6.2.2 of the HI-STORM FW SAR and show that assuming planar average enrichment is conservative or statistically equivalent to that of actual enrichment. The applicant did this for the 10x10A and although this assembly is not the bounding with respect to reactivity; the staff does not expect the conclusions to be significantly different with other assembly types. Any differences should be small and bounded by other conservative assumptions within the applicant's analysis. The staff finds the use of planar averaged enrichment acceptable for the HI-STORM FW.

The applicant uses a value of 10.686 g/cm^3 for UO_2 density. This is 97.5% of the theoretical density. This is a conservative value, since it corresponds to a very high pellet density of 99% or more of the theoretical density. The staff finds this acceptable.

For all PWR fuel types the applicant allows annular pellets in the top and bottom 12" of the active fuel length. The applicant performed calculations for several assembly classes with 12" of annular pellets with several different annulus sizes in the top and bottom of the active fuel length. The results of the calculations are shown in Table 6.2.6 of the HI-STORM FW SAR and show that there is no statistically significant effect on reactivity. The staff finds that assuming that the pellets are all solid is an acceptable assumption.

The applicant states for BWR assemblies that allow part-length rods (PLRs), they performed calculations with the PLRs absent and found that it was more reactive when they removed the PLRs all together, except for the 10x10G. For this configuration, removing the PLRs was not conservative and therefore all rods are assumed full length. The applicant also determined that there was not an intermediate length of the PLRs for the 10x10G that was more reactive than full-length. The analyses for the PLRs is discussed in Section 6.2.1 of the HI-STORM FW SAR and applies to the following fuel designs: 9x9A, 10x10A, 10x10B, 10x10F, and 10x10G. The staff finds it acceptable that there is no minimum PLR requirement for the above mentioned classes.

Some BWR assemblies do not have water rods, but have other geometries such as a "water cross" that performs the same moderating function for the assembly. In these cases the applicant models the feature as it is known and the specification of "minimum water rod thickness" refers to the thickness of the material used for the feature. This is applicable for assemblies 8x8F, 9x9G, 10x10C and 10x10G. The 9x9B and the 10x10B allow a minimum water rod thickness of zero. Section 6.2.1 of the HI-STORM FW SAR states that they were analyzed with the zircaloy tubes replaced by water and therefore the bounding water rod thickness is zero. The staff finds this acceptable.

The locations of the guide tubes are water rod locations are selected as the usual locations for the respective fuel assemblies and are shown in Appendix 6.B, Section 6.B.4 of the HI-STORM FW SAR. Several variations of known configurations have been analyzed. The staff finds that the locations of these assembly features are appropriate.

For all BWR assemblies that are to be stored in the HI-STORM FW system, the applicant allows that the water rods may also be sealed at both ends and contain zirconium based cladding material in lieu of water. The applicant states that they perform their criticality calculations with flooded water rods and that this is more conservative than sealed water rods because there is reduced moderator in the sealed rod configurations. This is discussed in Section 6.4.6 of the HI-STORM FW SAR. The staff finds this acceptable.

7.2.1 Non-fuel Hardware

The applicant allows for non-fuel hardware to be stored in the PWR MPC (MPC-37). This includes: BPRAs, TPDs, Control Rod Assemblies, Control Element Assemblies, Axial Power Shaping Rods, Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, vibration suppressor inserts, Instrument Tube Tie Rods (ITTRs), and components of these devices such as individual rods.

The applicant's k-eff calculations assume that the guide tubes are flooded with borated water. Although the hardware displaces moderator and provides some absorption, it also displaces soluble boron and therefore its effect on reactivity cannot be inferred. The applicant performed some additional calculations where the guide tubes were voided in the MPC-37 to show the effect that the presence of hardware has on reactivity. Table 6.4.5 of the HI-STORM FW SAR shows that voided guide tubes produce lower reactivity for a range of moderator densities for the 17x17B and that for the limiting assembly (15x15F) in the presence of damaged fuel (and higher boron concentrations) the voided guide tube configuration is less reactive. Table 6.4.10 of the HI-STORM FW SAR gives the results of all of the MPC-37 assemblies and shows that voided guide tubes produced lower reactivity than that of the flooded guide tubes for all but two assembly types. The assembly types for which there is an increase are the 14x14C and the 15x15I. The staff does not find that the increase is substantial (0.0005 in k-eff) and is statistically equivalent to that of the calculation with the flooded guide tubes. In addition these assembly types are not limiting assemblies and the applicant has other conservative assumptions that would negate any non-conservative uncertainty that this modeling assumption would have on reactivity. Therefore the staff finds the inclusion of hardware is acceptable.

The applicant allows for storage of start-up neutron sources. They state that a neutron source will not increase reactivity with a subcritical system. None of the neutron sources contain fissile material. Some of the neutron sources contain beryllium, which is a moderating material, however since a neutron source is only allowed in a single assembly within an MPC, its presence would only minimally impact the criticality of the system. The staff therefore finds the inclusion of neutron sources acceptable given this justification and other conservative assumptions used in the HI-STORM FW criticality analyses.

7.2.2 Fuel Condition

The applicant allows for the storage of damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris must be stored within a DFC. The applicant allows up to 12 DFCs to be stored within the MPC-37 in peripheral locations as denoted in Figure 2.1.1 of the HI-STORM FW SAR, and 16 DFCs to be stored in the MPC-89 in peripheral locations as denoted in Figure 2.1.2 of the HI-STORM FW SAR. The definition of damaged fuel is located in the Glossary of the HI-STORM FW SAR.

The damaged fuel is modeled as a bare array. It includes no cladding or structural material of any kind. The applicant performed various calculations to ensure that the model is conservative and maximizes reactivity. This includes varying the pitch and number of rods in an assembly. For configurations with soluble boron the applicant performed some calculations to show that replacing the cladding and structural material with borated water is still conservative.

The damaged fuel evaluations have an active fuel length of 150 inches. The staff finds this length bounds the actual length of all of the BWR and PWR fuel assemblies that are to be stored in the HI-STORM FW with the exception of the 17x17D and 17x17E PWR assembly classes. These assemblies have an active fuel length ≤ 170 inches. The applicant justifies the use of 150 inches for these assemblies by stating that the 15x15F assembly class bounds the reactivity of both the 17x17D and 17x17E. In addition sensitivity studies were performed on the 17x17B (Table 6.2.1 of the HI-STORM FW SAR) that show that changing the active fuel length only provides a small change in reactivity. The staff finds the applicant's justification acceptable and that any non-conservative uncertainty due to this assumption would be negated by other conservative analytical assumptions.

Section 6.4.4.1 of the HI-STORM FW SAR states that additional evaluations for damaged fuel assemblies with respect to missing rods, rod diameter variations, consolidated fuel assemblies with cladded rods, and enrichment variations were performed for the HI-STORM 100 Dry Cask system to further show that the above approach using arrays of bare fuel rods are bounding. The staff viewed the applicable sections of the HI-STORM 100 Dry Cask system FSAR and finds that this information further demonstrates that the damaged fuel configuration is bounding. The staff notes that the HI-STORM 100 evaluations may not directly be applicable in that the basket geometry and allowed contents are slightly different. However, the staff finds that any uncertainties associated with these differences are acceptable given the applicant's many conservative analytical assumptions.

Results of the damaged fuel evaluations are located in Table 6.1.4 and 6.1.5 of the HI-STORM FW SAR. The applicant performed 5 evaluations, each with a different assembly class representing the intact fuel assemblies. These assemblies were found to bound a group of assemblies allowed for storage in the HI-STORM FW such that all assemblies have been accounted for in the damaged fuel evaluations. Some assemblies have reduced enrichment requirements in the presence of damaged fuel (e.g. 10x10G) and the limiting assembly in the presence of damaged fuel was different than the limiting assemblies calculated in Tables 6.1.1 and 6.1.2 of the HI-STORM FW SAR. The PWR evaluations were performed using the minimum soluble boron concentrations necessary when DFCs are present. The staff finds that these calculations adequately represent all of the assemblies that are to be stored in the HI-STORM FW. The results of these calculations show that damaged fuel is acceptable for storage in the HI-STORM FW in the manor specified in the application.

7.3 Model Specification

The staff verified that the applicant included manufacturing tolerances important to criticality safety in the drawings in Chapter 1 of the SAR. The staff verified that the applicant considered the manufacturing tolerances of the basket when constructing their criticality model. This is discussed in Section 6.3.1 of the HI-STORM FW SAR. The applicant uses the minimum possible basket wall width. The applicant presents the results of an analysis using 10x10A fuel for the MPC-89 and 17x17B fuel for the MPC-37 in Table 6.3.2 of the HI-STORM FW SAR to demonstrate that the tolerances chosen are more conservative than the nominal dimensions. The staff finds the results of the analysis acceptable and that it would be representative of all assemblies that are to be stored in the HI-STORM FW.

The basket is manufactured from individual panels which are expected to be in direct contact with each other. However since there is a possibility that there could be small gaps between the panels, the applicant analyzed this effect. The applicant performed an analysis showing that the difference is very small. The staff finds that assuming no gaps in the panels is acceptable.

The applicant analyzed temperature variations using CASMO-4. The results of the applicant's analysis show that a minimum water temperature (hence maximum water density) is the most bonding. The results of the applicant's calculations are presented in Table 6.3.1 of the HI-STORM FW SAR. The staff finds this acceptable.

7.3.1 Configuration

The applicant provided a discussion of the potential changes to the basket due to accident conditions. The accident conditions mainly result in deflections of the basket walls. The staff finds that the applicant has addressed these changes and that the model that they use to evaluate the criticality safety of the package is represented during normal, off-normal and accident conditions.

The applicant provided tabular information in Tables 6.3.3 and 6.3.7 of the HI-STORM FW SAR showing the dimensions of the package used in the criticality analyses. The staff verified that these are consistent with the drawings and that they are conservative with respect to the tolerances and accident conditions discussed.

The applicant models each fuel type according to the specifications in Tables 2.1.2 and 2.1.3 of the HI-STORM FW SAR. These specifications are consistent with the fuel specifications required by the Technical Specifications. The applicant neglects all minor structural material and replaces it with borated or unborated water, as applicable. The staff finds that neglecting the structural material is acceptable as it provides modeling simplicity and its inclusion would only have negligible impact on criticality.

The applicant performed calculations addressing the positioning of the fuel assemblies within the basket cells. They performed a sensitivity study using the representative assemblies for both the MPC-37 and the MPC-89 where they move all of the assemblies toward the center of the basket, then toward the periphery. The results of the applicant's calculations are documented in Table 6.3.5 of the HI-STORM FW SAR. These calculations show that when the assemblies are moved toward the center of the basket that the reactivity increases, likewise when they are moved toward the periphery, reactivity decreases. Therefore the applicant states that they perform all of their criticality calculations with the assemblies moved toward the center of the basket. The staff finds this assumption is conservative and acceptable and is applicable for all assemblies that are to be stored in the HI-STORM FW.

The applicant provides information addressing configurations in which the cask is filled with partial density water or is partially filled with water. The applicant evaluates partial flooding for both the vertical and horizontal positions for both MPC designs. The applicant found that the fully flooded condition gives the highest reactivity. The results of the applicant's calculations are shown in Table 6.4.2 of the HI-STORM FW SAR. The staff finds that the fully flooded model is acceptable and applicable to all of the assembly configurations that are to be stored in the HI-STORM FW.

The applicant provides information addressing possible preferential flooding conditions. Section 6.4.2.4 of the HI-STORM FW SAR states that there is a possibility for the DFCs to remain partly filled with water while the remainder of the MPC is dry. The applicant performed an evaluation with just the DFCs flooded to determine the reactivity effect of this condition. The applicant found that the most reactive condition is when all assemblies within the MPC are fully flooded. The results of the applicant's calculations are shown in Table 6.4.4 of the HI-STORM FW SAR and therefore all calculations performed for damaged fuel evaluations are with a fully flooded

MPC. The staff finds that the fully flooded model is acceptable for damaged fuel configurations and applicable to all of the assembly configurations that are to be stored in the HI-STORM FW.

The applicant provides information addressing the differences in external moderator density. With a fully flooded cask, the applicant varied the external moderator for representative fuel designs, 10x10A for the MPC-89 and 17x17B for the MPC-37. The results of the applicant's calculations show that the maximum k-eff is independent from the external water density. Therefore all of the calculations are performed with full density water outside of the cask. The results of the applicant's calculations are shown in Table 6.4.1 of the HI-STORM FW SAR. These calculations show that the difference in k-eff is within the uncertainty of the calculation. The staff finds the use of full density external moderator acceptable.

The applicant considered flooding in the fuel rod pellet-to-clad gap regions with unborated water. The applicant found that it is conservative to assume that the pellet-to-clad gap regions are flooded and therefore for all cases, the pellet-to-clad gap regions are assumed flooded with unborated water. Table 6.2.3 of the HI-STORM FW SAR shows that for all assemblies to be stored in the HI-STORM FW that it is conservative to flood the pellet-to-clad gap. This is further shown in Table 6.4.3 of the HI-STORM FW SAR. The staff finds this assumption acceptable.

7.3.2 Material Properties

The staff verified that compositions and densities are provided for all materials used in the computational model. This information is provided in Table 6.3.4 of the HI-STORM FW SAR. The applicant did not provide the source of the material compositions listed in Table 6.3.4 of the HI-STORM FW SAR however the staff verified that the specifications for stainless steel, lead, zirconium, aluminum and concrete are standard compositions. Although there could be minor impacts on criticality based on differences in composition for these materials (such as absorption of neutrons in Fe), the staff does not find that that these variations would be enough to have a safety significant impact on the calculated reactivity and any uncertainties would be negated by the applicant's many conservative analytical assumptions.

Section 1.2.1.4.2 of the HI-STORM FW SAR states that the water jacket used to provide radial neutron shielding material in the transfer cask (HI-TRAC VW) may be fortified with ethylene glycol. This was not directly addressed in the criticality analyses however the applicant states that it is far from the fuel material and separated by absorbing and reflecting materials and that it would have no noticeable effect on criticality. The staff agrees with the applicant's assessment and finds the addition of ethylene glycol to have no negative effect on the results of the criticality analyses.

The applicant assumes 90% of the B-10 content for the Metamic-HT fixed neutron absorber. The staff verified that the applicant has appropriate acceptance testing and assessment of the material to ensure its continued efficacy. This information is located in Section 10.1.6.3 of the HI-STORM FW SAR. The staff finds this assumption acceptable.

The applicant performed an analysis that assumed that the absorber material received a constant neutron source equivalent to the design basis fuel as determined in Section 5.2 of the HI-STORM FW SAR and found there was no significant degradation of the B-10 for 60 years. The staff finds that this meets the requirements of 10 CFR 72.124(b) and finds this acceptable.

7.4 Criticality Analysis

7.4.1 Computer Programs

The applicant performs the criticality evaluations using the MCNP5 three-dimensional Monte Carlo code and continuous energy cross sections. The applicant lists the specific cross sections used in Table 6.A.4 of the HI-STORM FW SAR. These are mostly derived from ENDF/B-V and ENDF/B-VI data, with the exception of tin, where ENDF data is not available and so ENDL data is used. The MCNP5 code and the cross sections used by the applicant are widely used in these types of applications and the staff finds it is appropriate for this application.

7.4.2 Multiplication Factor

The staff examined the results of the k-eff calculations for the storage cask. These are in Tables 6.1.1 through 6.1.5 of the HI-STORM FW SAR. Tables 6.1.1 and 6.1.2 of the HI-STORM FW SAR show maximum k-eff values using the transfer cask configuration, Table 6.1.3 in the HI-STORM FW SAR has representative k-eff values for the storage cask configuration, and Tables 6.1.4 and 6.1.5 in the HI-STORM FW SAR show the maximum k-eff values when considering the maximum number of DFCs allowed. These values represent the highest k-eff that might occur during normal, off-normal and accident conditions. They are evaluated with the worst combination of manufacturing tolerances and include the calculation bias, uncertainties and calculation statistics.

The limiting criticality configurations are for the transfer cask (HI-TRAC VW) configuration since it is flooded it has the highest reactivity conditions. The limiting configuration for the MPC-37 (PWR fuel) is the 15x15F fuel with 5.0% enrichment and a minimum soluble boron concentration of 2000 ppm. The maximum k-eff value is 0.9455. For the MPC-89 (BWR fuel), the limiting configuration is the 10x10G fuel with 4.6% maximum planar average enrichment and the maximum k-eff value is 0.9466.

The applicant calculated “representative values” of k-eff for the storage cask (overpack) calculations and presented the results in Table 6.1.3 of the HI-STORM FW SAR. The staff notes that these results are only for the 17x17B assembly class within the MPC-37 and the 10x10A assembly class within the MPC-89. Although no effort has been made to find the most reactive assembly class for this configuration, the results show a large margin to criticality and the staff believes that any other assembly class would also show a large margin to criticality based on the similar calculated k-eff values for other assembly classes in Tables 6.1.1, 6.1.2, 6.1.4 and 6.1.5 of the HI-STORM FW SAR. Therefore the staff finds these results acceptable and provide adequate assurance that the HI-STORM FW storage cask is subcritical.

The staff verified that the applicant provided representative input files. The staff also verified that the information regarding the model configuration, material properties and cross sections is properly represented in the input files. The staff reviewed the key input data for the criticality calculations specified in the input files and finds them acceptable. The staff viewed the output files provided and determined that they have proper convergence and that the calculated k-eff values from the output files agree with those reported in the text.

All of the calculated k-eff values meet the sub-criticality criterion of k-eff<0.95 and therefore the staff finds them acceptable.

7.4.3 Independent Staff Calculations

The staff performed independent calculations to verify the k-eff of the HI-STORM FW. The staff constructed its model using design information found in the SAR. The staff used the KENO6 code with the 238-group cross section library derived from ENDF-VI data.

The staff performed calculations of the MPC-37 and the MPC-89 using the limiting fuel as determined in Table 6.1.1 and 6.1.2 of the HI-STORM FW SAR. For the MPC-37, the fuel is the 15x15F with 5.0% enrichment and 2000ppm boron. For the MPC-89 the fuel is the 10x10G with 4.6% average planar enrichment. The staff used the fuel dimensions as specified in Tables 2.1.2 and 2.1.3 of the HI-STORM FW SAR.

For the 10x10G fuel model, the staff makes the following assumptions:

- The staff preserves the geometry of the rods and water cross, however for simplicity the staff makes the conservative assumption that the water cross material is replaced by water
- The staff assumes all rods are full length, this assumption was found to be conservative by the applicant

For the 15x15F the staff uses the instrument/guide tube configuration from Appendix 6.B.4 of the HI-STORM FW SAR.

For simplicity the staff assumes that the bundles are centered within the basket cell for both the MPC-37 and MPC-89 models.

The staff assumes that all assemblies are intact. This assumption is based on the applicant's calculations that show that the intact fuel configurations bound that of the DFC configurations.

The staff's model was based on the HI-TRAC VW transfer cask since this cask would see flooding and provides a more reactive condition.

The staff's model assumes full flooding of the inside of the fuel assemblies, and basket, including pure water inside the gap between the fuel rod and the cladding. The staff's model also assumes full density water external moderator.

The staff used several simplifying assumptions similar to that of the applicant. The staff assumed no structural materials within the basket besides the fuel tubes. The staff assumed that there was no assembly hardware and modeled only the active fuel length of 150 inches as axially centered within the basket.

The results of the staff's calculations show that the k-eff of the MPC-89 with 10x10G fuel is 0.9466 and for the MPC-37 with 15x15F fuel is 0.9319. Both the MPC-37 and MPC-89 configurations are less than 0.95 and similar to that of the applicant. The staff finds that this helps to demonstrate that the system is subcritical and that the features important to criticality are sufficiently described and that the applicant has addressed the most reactive conditions and that the reported k-eff is conservative and that the applicant has appropriately modeled the cask geometry and materials.

7.4.4 Benchmark Comparisons

The staff reviewed the HI-STORM FW SAR to ensure that the computer codes used to evaluate the k-eff of the package are benchmarked against critical experiments and that an appropriate bias was calculated. The applicant stated that all benchmark calculations were performed using the same computer code, computer system and cross section data as was used to evaluate the k-eff of the cask system. The applicant only provided benchmark results for the MCNP5 code and did not provide benchmark results the CASMO-4 code. Since CASMO-4 was only used to determine reactivity effects from changes in temperature, the staff finds this acceptable.

The applicant discusses three significant parameters that affect criticality. These are (1) enrichment, (2) B-10 loading of the absorber plates, and (3) lattice spacing or water gap thickness. The applicant provides a discussion of each of these effects and provides critical experiments used to determine the bias for these effects.

7.4.4.1 Experiments and Applicability

The applicant provided a list of experiments used to determine the bias. The staff reviewed the experiments listed in Table 6.A.1 of the HI-STORM FW SAR to determine their applicability to the HI-STORM FW and its proposed contents.

These calculations bound the range of important design variables used in the cask designs. The staff ensured that the following design parameters for the HI-STORM FW are within the benchmark experiments cited by the applicant:

- Enrichment
- Type of fissile material
- B-10 loading

7.4.4.2 Bias Determination

The applicant determined that there were no trends in the k-eff results for the following effects: (1) enrichment, (2) B-10 loading, (3) reflector material and spacings, and (4) fuel pellet diameter and pitch. Since there are no trends in the data for any of these individual effects, the applicant chooses to represent all of the data by the “energy of the average lethargy causing fission” or EALF and performed linear regression of all of the data and determined a total calculational bias. This approach has been reviewed and approved previously by the staff and found acceptable for the HI-STORM 100 and the HI-STAR 100. The applicant shows a bias of 0.0030 and states that this was added to all of the calculated k-eff values in Chapter 6 of the HI-STORM FW SAR along with 2 standard deviations of the calculated k-eff and the standard error of the bias of 0.0008. The staff finds this acceptable.

7.5 Burnup Credit

The applicant does not take credit for burnup. All calculations are performed using fresh fuel assumptions. The staff finds this conservative and acceptable.

7.6 Evaluation Findings

Based on the above statements, the staff has the following evaluation findings with respect to the criticality analysis:

F7.1 Structures, systems, and components important to criticality safety are described in sufficient detail in the HI-STORM FW SAR to enable an evaluation of their effectiveness.

F7.2 The cask and its spent fuel transfer systems are designed to be subcritical under all credible conditions.

F7.3 The criticality design is based on favorable geometry, and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the term requested in the CoC application and there is no credible way for the fixed neutron poisons to significantly degrade during the requested term in the CoC application; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10 CFR 72.124(b).

F7.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for the term requested in the CoC application.

The staff concludes that the criticality design features for the HI-STORM FW are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the HI-STORM FW will allow safe storage of spent fuel. These findings are reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

8 MATERIALS EVALUATION

The HI-STORM FW incorporates one major material change that differentiates it from all previous Holtec storage canister designs. That change is the incorporation of their proprietary aluminum-based material, called Metamic HT, for the fuel basket structure.

The remaining materials used in the fabrication of the “FW” storage system have been used in a number of previously reviewed storage system designs. Since there are few changes to the balance of the materials used in the FW system, detailed evaluation is primarily provided only for changes from previous designs.

8.1 Metamic HT Spent Fuel Basket

Metamic HT is a Holtec proprietary aluminum-based material intended for dual purpose use in the Holtec HI-STORM FW spent fuel basket. Metamic HT is designed to be both a neutron poison for criticality control and also a load-bearing structural material. Previously, no material was used for such a dual purpose role in any storage canister fuel basket.

Staff review has found that the composition and properties of Metamic HT are unique. It is a powder metallurgy material composed of aluminum combined with aluminum oxide and boron carbide (Holtec uses the terminology “metal matrix composite” to generically describe Metamic HT). The aluminum oxide is a finely dispersed second-phase which provides enhanced room temperature, elevated temperature, and creep strength. The boron carbide is the neutron poison used for criticality control. Since Metamic HT evolved from a previously reviewed, non-structural neutron poison (“classical” Metamic), with closely similar neutronic properties, the neutronic properties of the new Metamic HT were already well characterized.

Discussion and evaluation of Metamic HT in this SER principally involves its structural characteristics. The neutronic properties of Metamic HT are essentially identical to previously reviewed classical Metamic. Therefore, the neutronic properties are not discussed in this SE.

Since Metamic HT is a new structural material, a comprehensive test program was necessary to fully determine its physical properties and characteristics. The testing program emulated that typically employed to qualify an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code structural material. Holtec provided extensive data from their testing program for staff review. The data and analyses were documented in the Holtec comprehensive proprietary report; “Metamic HT Qualification Sourcebook”. Since this document and data are proprietary, the staff summarizes the testing program and findings in lieu of excerpting detailed technical discussion from the applicants’ documents.

The testing program, employing a variety of standard American Society for Testing Materials (ASTM) test methods, including: yield strength, tensile strength, elongation, reduction in area, Young’s Modulus, Charpy impact strength, thermal conductivity, coefficient of thermal expansion, and emissivity. Additionally, isotropy was evaluated, thermal aging effects were examined, welding procedures developed, weld properties determined, irradiation effects evaluated, thermally induced microstructural alteration assessed, and accelerated creep testing was performed. Neutron attenuation was verified (as for previously accepted “classic” Metamic) and corrosion testing was conducted in a simulated borated pool water environment.

For each material property and evaluation, up to 30 samples were tested. Samples were taken from multiple lots of material to determine lot to lot variability, which was found to be minimal.

Overall, the staff found that the applicants test program was comprehensive in scope and supported the wide variety of property data needs for characterizing Metamic HT.

8.1.1 Mechanical Properties

Using the guidance of ASME Section II, Appendix 5, Holtec determined mechanical properties (as listed previously) at room temperature and also at several temperatures ranging from minus 40 degrees C to 500 degrees C. The test data was analyzed using statistical methods and minimum, average, and mean values of the various properties were determined. Additionally, a design value for the various properties was established which is called the Minimum Guaranteed Value (MGV). The MGV is a numerical value (for any given property) which was chosen to be below the lowest measured value from the test data. The MGV is then demonstrated (guaranteed) to be exceeded for every manufactured lot of Metamic HT through lot testing. Any lot which fails to meet one (or more) MGV values is discarded. Lot testing sample size is controlled by a statistical sampling approach. If any lot of material fails to meet any one of the various MGV’s, then an enhanced sampling plan is invoked for subsequent production lots.

The staff found the mechanical properties of Metamic HT to generally equal or exceed those of conventional high-strength aluminum alloys, especially with respect to high-temperature performance. The lab data demonstrated that the short and long-term high-temperature strength was significantly superior to any conventional aluminum alloy. These superior high temperature properties extended well beyond the normal ASME Code temperature limit of about 400°F for aluminum alloys. Metamic HT was shown to have very good long-term properties at 400°C, and acceptable properties at the design accident condition of 500°C. The staff found this performance to be unprecedented compared to any other aluminum alloy.

Since the strength of aluminum alloys is adversely affected by long times at elevated temperatures, the creep performance of Metamic HT was studied. As detailed later in this SER, the creep properties of Metamic HT were demonstrated to be significantly superior to any other aluminum alloy.

A common property of high strength aluminum alloys is reduced ductility. Metamic HT also displays this attribute. The design consequence of using a low ductility material requires the overall stresses to remain well below yield and local stresses and strains to remain near the elastic limit. The suitability of the material must therefore be considered in the context of the structural analysis and not based solely on the materials ductility. This limitation is detailed in the structural analysis section of this SER.

The staff finds that the mechanical property testing program adequately characterized the performance of Metamic HT for its intended use as a structural material.

8.1.2 Low Temperature Effects

Since Metamic HT is an aluminum-base material, it is expected that it would not be susceptible to ductile-brittle transformation or brittle fracture issues at low temperatures. To verify this assumption, coupons were tested at minus 40°C. The samples showed no degradation in tensile, elongation, or impact properties, compared to room temperature properties. Thus, it was demonstrated that no low temperature ductility issues affected Metamic HT.

The staff finds that Metamic HT is suitable for the lowest design temperatures of minus 20 and minus 40°F, as specified by 10 CFR Part 72.

8.1.3 Thermal Aging Effects on Mechanical Properties

All metals undergo changes in their mechanical properties when exposed to elevated temperatures. Aluminum-based materials typically exhibit a decline in properties at temperatures above about 200°F. These property changes are generally reversible after short duration exposure. Long duration, elevated temperature exposure usually results in permanent decreases in mechanical properties such as yield and tensile strength. Since the storage canister is designed for a 40-year license period, the long-term elevated temperature performance of Metamic HT is of primary interest. This is especially so since the fuel basket is designed to initially operate at up to 400°C, well above the typical ASME Code limit of around 400°F for aluminum alloys.

To determine the long-term, elevated temperature performance of Metamic HT, the applicant exposed more than 30 samples to temperatures up to 500°C in order to determine mechanical properties and also to “age” the material in an accelerated fashion. The accelerated aging technique is intended to duplicate the metallurgical and physical property changes that would occur in the material under design conditions, but on a faster time scale. This accelerated testing is done by exposing the material to a higher temperature (for a shorter time period) and using a mathematical model to equate this accelerated “aging” process to a lower temperature, longer duration exposure as would occur in normal service. After the thermal aging, the samples were tested and compared to as-produced (un-aged) material samples to determine if any permanent changes had occurred to the material properties.

Holtec asserted that since Metamic HT is not a heat-treatable material (very unlike typical high strength aluminum alloys), it should not exhibit any aging effects. The Holtec test data supported this contention. Aged samples exhibited some small changes in properties when compared to un-aged (room temperature) samples. The NRC staff judged the changes to be minor. Tensile and yield strength values dropped about 2 to 4 % whereas elongation slightly increased and reduction in area slightly decreased. Charpy impact strength was virtually unchanged. This is a unique response, as all other high-strength aluminum-based Code materials exhibit an aging effect, often severe. The fact that the aging temperature used for the Metamic HT samples was roughly 350°F higher than the temperature limit for other aluminum alloys makes the Metamic HT performance more notable.

The staff finds that the applicants' data supports the contention that thermal aging is not a significant factor at the design temperature. Thus, no long term, thermally induced, degradation would be expected in service.

8.1.4 Thermal Aging Plus Irradiation Effects on Mechanical Properties

Samples of Metamic HT were exposed to elevated temperatures plus radiation fluence levels beyond those to be found in service. The test results have not revealed any unexpected or significant adverse synergistic effects. Some increase in tensile and yield strength are noted along with a corresponding slight reduction in ductility and toughness. This result is consistent with the performance of other metals in similar service.

The staff finds the results of the thermal plus irradiation exposure show some small effect which is consistent with the experience for other metals. Thus no unanticipated response to service conditions (heat plus radiation) would result. The staff further notes that the fluence levels for a storage cask are a number of orders of magnitude lower than the accumulated dose in reactor operation. This provides an additional margin of safety against adverse effects (such as brittle fracture) in Metamic HT components in service.

8.1.5 Thermal Aging Effects On Microstructure.

Microstructural alteration is an important indicator of material property changes. To support the mechanical property data provided, the applicant microscopically examined as-extruded material and compared its microstructure to thermally aged samples. The thermally aged samples showed little or no microstructural alteration when compared to the as-extruded (e.g., new material) samples. This demonstrates a remarkable thermal stability which is not observed in other aluminum-based materials, especially at the maximum design operating temperature of 400°C.

The staff examined the photomicrographs of the as-extruded (new) and thermally aged materials and found no significant changes to the microstructure. Based upon this examination, the staff finds that Metamic HT is highly stable at the design temperature. This provides reasonable assurance that microstructural changes under service conditions will not occur, and thus the material properties of Metamic HT will remain unaffected during service.

8.1.6 Anisotropy

A significant issue with many high performance aluminum materials is anisotropy of mechanical properties. Potential anisotropy in Metamic HT was explored as part of the materials characterization project. A series of samples, including as-manufactured, artificially aged, and aged and irradiated, were tested in accordance with the compressive test technique of ASTM E9-89. Test results showed no significant variation in properties with respect to extrusion orientation. The applicant concluded that anisotropy in Metamic HT is negligible.

On review of the provided data, the staff finds that anisotropy in Metamic HT is small and may be neglected for the intended application.

8.1.7 Weld Properties

High performance aluminum alloys are generally unweldable. This is due to the fact that the heat of welding adversely alters the microstructure. The strengthening microstructure is obtained by special heat-treatments, which high strength aluminum alloys must undergo to attain their enhanced strength. Consequently, high strength aluminum alloy structures (such as aircraft) are always riveted, bolted, or bonded instead of welded.

Metamic HT utilizes a strengthening mechanism ("oxide dispersion strengthening") which is different from that employed in most aluminum-base materials. It does not depend on a heat treatment to achieve its strength. However, the Metamic HT strengthening mechanism is adversely affected by weld-zone, temperature-induced, microstructural alteration. Consequently, any Metamic HT welds are expected to have less than the original base material strength.

The applicant undertook a weld development program to optimize the strength of Metamic HT welds. They were able to achieve weld strength of approximately 60% of the original base material strength. This weld performance obviously affects weld design and restricts weld utilization. To compensate for this loss of strength, the Metamic HT basket welds are restricted to areas where stresses are low.

Welds with strengths which are lower than the base material strength would not be permitted in ASME Code-based design. The Metamic HT basket is not designed in compliance with the ASME Code. To compensate for the adverse weld properties, a strain-controlled design method was employed for the basket design. The adoption of a strain-controlled design and the placement of the welds in a low stress (hence low strain) location somewhat compensates for reduced weld strength. The structural evaluation of the basket welds are detailed in the structural section of this SER.

The staff finds that although Metamic HT is deficient with respect to normally accepted (e.g., ASME Code) weld strength requirements, the strain-limited design approach and the placement of the welds in a low stress (and strain) region of the basket assures adequate performance in service.

8.1.8 Creep Properties

Since the fuel basket operates above the ASME Code temperature limit of about 400 degrees F for aluminum-base Code materials, creep strength/performance was identified early in the

Metamic HT characterization program as a key issue. Consequently, a creep test program was established to determine the creep strength of Metamic HT at the design operating temperature.

Seven long-term creep samples were tested. Samples were tested at 300, 350, and 400°C. Stress levels used in the tests ranged from 200 to 1000 psi. The test conditions were designed to provide data which can be extrapolated to a creep life at design conditions. Although employing a limited number of creep test samples, the applicant attempted to bound design conditions sufficiently to ensure a large margin of safety.

The creep samples accumulated varying test times, between about 17,500 hours and 20,000 hours at the various temperature and load conditions. When the testing was discontinued, no specimens had failed. Cumulative creep strains were reported to vary between about 0.07 percent and 0.24 %, depending upon test conditions (time, stress, and temperature).

The staff observes that the creep testing was extremely severe compared to design conditions. Normally, a wide difference (such as the applicant employed) between the accelerated test conditions and the design conditions is avoided. This is due to metallurgical considerations. Typical creep tests are generally conducted at a slightly elevated temperature but at the same stress as the maximum design stress. The reason is to avoid metallurgical phenomena which could distort the result and lead to a significantly understated (but conservative) creep life prediction. While this would be safely conservative, it could be unnecessarily so by leading to much shortened creep life predictions.

The applicant used the accumulated creep data in conjunction with an established creep strain equation to produce a cumulative creep strain versus operating time relation. The applicant then compared the creep equation predictions with the data from the creep test samples. In every case, it was shown that the creep equation over predicted the actual measured creep strain by a comfortable margin. The applicant asserts that this demonstrates their equation is conservative due to over predicting the cumulative creep strain.

In conjunction with the creep equation, a limiting creep strain of 0.4% was adopted by the applicant as the maximum allowable creep strain in service. This limit is based on a foreign construction Code creep strain limit for aluminum components. Employing the applicants creep strain equation along with the 0.4% allowable creep strain limit yields a service life well beyond the normal license period.

Although this method is logical, the limiting creep strain is not adequately supported by the available data, because none of the creep test specimens were tested to failure. This means the creep strain at failure is undetermined. Thus, any creep strain limit based on failure strain is not supported. The applicant has demonstrated what the creep strain rate is for the several specimens. But absent failure data, there is no way to predict either creep strain at failure or maximum creep life, meaning time to failure. Adoption of the 0.4 percent creep strain limit may be conservative, based upon general creep failure strain knowledge, but it remains speculative absent failure strain data.

Another issue with the creep testing program involves the choice of test parameters (temperatures and stresses). It is recognized that a limitation of accelerated creep testing is the susceptibility to overstate the creep strain at failure, at design conditions. Higher temperature, or, especially, higher stress, accelerated creep testing is susceptible to yielding creep failure strains which are slightly larger than those which would be achieved under service conditions. The reasons are related to the activation of additional creep mechanisms due to the necessarily more severe conditions employed during testing. In the applicants' case, the test stresses were significantly higher than what is found in service for the fuel basket. Thus, the staff must view the achieved creep strains as possibly overstated. However, the test outcome, when the time (not strain) element is considered, is highly conservative with respect to predicting the useful creep life under design conditions. Thus, although the staff finds the measured creep strains as possibly overstated, the creep life (time) performance of the material when in operation is conservatively assured.

With respect to predicted creep life, as measured in hours instead of accumulated creep strain, the applicant applied a mathematical formula called the Larson-Miller (L-M) equation. The L-M equation is frequently employed to relate the time at accelerated test conditions to time at operating conditions. Again, as for the creep strain equation, the L-M equation cannot predict total creep life unless the creep data includes samples tested to failure. However, given that the test samples never failed, it is assured that a predicted service life based upon the test sample times will be conservative.

In a previous license application, the applicant provided an L-M calculation, based on the least conservative data, which showed that a 40-year continuous operating life was achievable. The staff, through an independent L-M calculation, verified that a 40-year operating life is amply supported by the applicants creep test data.

Despite some shortcomings of the test program, the staff finds that the creep testing has bounded the design conditions by a wide margin. The severity of the creep tests will tend to understate the predicted service life (as measured in hours), which is conservative at a predicted continuous service life of greater than 40 years. With respect to creep strain, the staff finds that due to the very elevated test conditions (stress), and the lack of failure strain data, the validity of the proposed 0.4% creep strain limit is unknown. However, the shortcoming of strain results and predictions becomes immaterial when the service life, expressed in hours, is employed instead.

In conclusion, the staff finds that the creep tests have conclusively demonstrated the ability of Metamic HT to adequately perform continuously at design conditions for at least 40 years. An added level of conservatism exists when it is recognized that no storage canister will ever operate continuously at maximum design temperature due to the continual decline in decay heat and consequent significant reduction in operating temperature.

8.1.9 Corrosion Resistance

Metamic HT was tested for compatibility with borated water, as would be typical for cask loading and unloading conditions. Aluminum alloys are very slightly corroded by borated water and Metamic HT performed similarly to other aluminum-based materials in immersion tests. The applicant concluded that Metamic HT has more than adequate corrosion resistance for the intended service.

The staff finds that Metamic HT is not susceptible to significant chemical or galvanic reactions and will perform adequately in accordance with 10 CFR 72.120(d).

8.1.10 Conclusions–Metamic HT

The applicant has provided data to quantify the performance and characteristics of Metamic HT in a manner somewhat emulating that of an ASTM/ASME material, but with several deviations from normal ASME Code practice (the staff uses the ASME Code as a benchmark).

One deviation is the establishment of the MGV for various properties. The MGV values adopted by the applicant differ from the "stress allowable" used by the ASME Code. The Code "stress allowable" is directly used for stress-based design calculations. The applicants MGV data is primarily used as the Metamic HT material production QA/QC acceptance criteria. MGV properties exceed the property values assumed in the design basis. Consequently, a direct comparison of "stress allowable" and MGV's is not valid.

A second major deviation in Metamic's performance from normal ASME Code requirements is the reduced weld strength. The ASME Code does not permit welds of a lower strength than the base material. The applicant recognized this characteristic and specified weld use only in low stress areas. However, since the basket is a strain-based design, the "stress allowable" method becomes immaterial and the adequacy of the welds must be evaluated by means of the structural strain analysis. As determined from the structural analysis (and discussed in the structural analysis section of this SER), the welds were found to provide an adequate margin of safety for their specific application.

The third deviation from ASME Code practice involves the creep test data. The ASME Code practice is to creep test materials to failure. The Metamic HT creep testing program never took a sample to failure. Thus, the data is incomplete in the sense that failure creep strain and ultimate time to failure is not determined. However, the applicant has adequately demonstrated that the Metamic HT will perform without failure for a license period of 40 years. No safety significant issue results from the "incomplete" creep test data.

Despite the above-discussed deviations from normal creep testing practice, the staff finds the applicants program of testing to be adequate to characterize the properties and performance of Metamic HT. The staff further finds that Metamic HT has adequate creep properties for the intended application.

The staff finds that the applicant has met the requirements of 10 CFR 72.124. Materials used for criticality control are adequately designed and specified to perform their intended function.

8.2 Other Materials of Construction

The balance of the HI-STORM FW system is fabricated from materials which have all been previously evaluated by the NRC staff for their suitability. The bill of materials in SAR drawings and chapter 2 provide details of each material type and specification. Since all the materials have been previously reviewed and employed for 10 CFR Part 71 packages, only brief synopses of materials related findings are summarized below.

8.2.1 Confinement Boundary

The fuel canister confinement is fabricated from one of several ASME grades of austenitic stainless steel, referred to by the applicant as "Alloy X". Alloy X assumes the least favorable property characteristics from among the several materials grades specified. These properties are use for all design calculations. The purpose is to allow for free interchange of the several grades of stainless steel. This provides the applicant with procurement flexibility while complying with all required design properties. This method of allowing for material substitution has been previously reviewed by the staff and found to be acceptable.

The use of austenitic stainless steel also means that the fuel canister is immune to brittle fracture issues.

8.2.2 Gamma and Neutron Shield

The radiation shield (concrete overpack) is composed primarily of un-reinforced concrete with a carbon steel liner plate on the inside. Since the overpack has no structural role, the lack of reinforcing steel is not a detriment. The lack of reinforcing steel is a deliberate exclusion in order to avoid the possibility of interior voids in the concrete which would degrade the shielding performance. This type of construction has been approved by the staff previously and found to be entirely satisfactory in service at numerous installations. For the design service conditions, there are no conditions which will result in a degradation of the materials performance for the duration of the license period (up to 40-years). Experience has shown that the materials should easily achieve 40 years of service with no loss of performance or adverse degradation.

8.2.3 Welding

All weld filler materials utilized in the welding of the confinement boundary comply with the provisions of the appropriate ASME Code Subsection.

All “non-Code welds” (e.g. welds not important to safety) will be made using weld procedures that meet ASME Section IX, AWS D1.1, D.1.2 or equivalent.

All non-destructive examinations will comply with Section V of the ASME Code, with acceptance criteria as specified by the code of construction for the specific component.

8.2.4 Chemical, Galvanic, or Other Reactions

Non-fuel hardware is included as allowable contents in the HI-STORM FW system. All non-fuel hardware materials have been previously evaluated and found to be acceptable for dry cask storage, including the wet loading operations. The materials are all either corrosion resistant or inert in the fuel pool environment. No adverse chemical or galvanic reactions involving the cask materials or contents would occur during wet loading or in dry storage.

The staff finds that the applicant has met the requirements of 10 CFR 72.236(h). The HI-STORM FW storage cask employs materials that are compatible with wet and dry SNF loading and unloading operations and facilities. These materials will not degrade over time or react with one another during any conditions of storage.

8.2.5 Conclusion Other Materials of Construction

Since the materials for the balance of the storage canister have all been previously reviewed for acceptability, and since the conditions of use are unchanged, the staff finds that the materials are acceptable for their specified uses.

8.3 Evaluation Findings

F8.1 SAR Chapter 8 adequately describes the materials used for SSCs important to safety and the suitability of those materials for their intended functions is sufficient detail to evaluate their effectiveness.

F8.2 The applicant has met the requirements of 10 CFR 72.122(a). The material properties of SSCs important to safety conform to quality standards commensurate with their safety function.

F8.3 The applicant has met the requirements of 10 CFR 72.122(h)(1) and 72.236(h). The design of the DSS and the selection of materials adequately protects the SNF cladding against degradation that might otherwise lead to damaged fuel.

F8.4 The applicant has met the requirements of 10 CFR 72.236(h) and 72.236(m). The material properties of SSCs important to safety will be maintained during normal, off-normal, and accident conditions of operation so the SNF can be readily retrieved without posing operational safety problems.

F8.5 The applicant has met the requirements of 10 CFR 72.236(g). The material properties of SSCs important to safety will be maintained during all conditions of operation so the SNF can be safely stored for the minimum required years and maintenance can be conducted as required.

F8.6 The staff concludes the material properties of the structures, systems, and components of the HI-STORM FW storage cask is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the material properties provides reasonable assurance the cask will allow safe storage of SNF for a licensed life of at least 40 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

9 OPERATING PROCEDURES EVALUATION

The operating procedures review ensures that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for identified key operations.

9.1 Areas of Review

The HI-STORM FW SAR was reviewed and the following operations were acceptably addressed:

Loading Operations

Fuel Specifications
Damaged Fuel
Subcriticality Features
ALARA
Offsite Releases
Draining and Drying
Filling and Pressurization

Welding and Sealing
Administrative Programs

Cask Handling and Storage Operations

Cask Unloading

Damaged Fuel
Cooling, Venting, and Reflooding
Fuel Crud
ALARA
Offsite Release

9.2 Staff Evaluation

The staff concludes that the generic procedures and guidance for operation of the HI-STORM FW system are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the system will enable safe storage of spend fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and acceptable practices.

9.3 Evaluation Findings

F9.1 The HI-STORM FW System can be wet loaded and unloaded. General procedure descriptions for these operations are summarized in Chapter 9 of the SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F9.2 The bolted closure plate and welded MPC of the cask allow retrieval of the spent fuel for further processing or disposal as required.

F9.3 The general operating procedures are designed to prevent contamination of the MPC and facilitate decontamination of the overpack. Routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F9.4 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading and unloading will be governed by the 10 CFR Part 50 license conditions.

F9.5 The general operating procedures described in the SAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F9.6 Section 10 of the SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F9.7 The staff concludes that the generic procedures and guidance for the operation of the HI-STORM FW System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

10 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The specific discipline acceptance tests and maintenance programs are evaluated in sections 3 through 9 and 11 of this SER. The results of the evaluation are captured in their applicable sections.

11 RADIATION PROTECTION EVALUATION

The objective of the review of this section is to ensure that the capability of the radiation protection design features, design criteria, and operating procedures, as appropriate, of the HI-STORM FW system, can meet regulatory dose requirements for the proposed contents. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d).

The proposed HI-STORM FW system was reviewed to determine if the radiation protection design features, as described in the SAR, are acceptable to the staff. The staff's conclusions, summarized below, are based upon the information provided in the application.

The applicant has established that the HI-STORM FW system has been designed in accordance with 10 CFR 72 and maintains radiation exposures ALARA consistent with 10 CFR Part 20 and the guidance provided in Regulatory Guides 8.8 and 8.10. According to the applicant, licensees using the HI-STORM FW system will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10 CFR Part 20 are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM FW system, and be familiarized with the expected dose rates around the MPC, HI-STORM overpack and HI-TRAC VW during all phases of loading, storage, and unloading operations. Dose rate limits at the HI-TRAC VW and HI-STORM overpack surfaces are provided in Chapter 13 of the SAR. These rates were calculated to ensure that the HI-STORM FW system is operated within design basis conditions and those ALARA goals will be met. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10 CFR Part 20 standards for radiation protection are met in accordance with the site's written commitments.

11.1 Radiation Protection Design Criteria and Design Features

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. As required by 10 CFR Part 20 and 10 CFR 72.212, each general licensee is responsible for demonstrating site-specific compliance with these requirements. In addition, proposed TS 5.3 establishes direct radiation dose rate limits and other radiation protection criteria for the cask system. These criteria are based on bounding dose rate values, which are used to determine occupational and off-site exposures and other design-specific factors important in the radiation protection system. The radiation protection design features are described in Chapter 11 of the SAR.

The staff reviewed the design criteria and found them acceptable. Section 11 of this SER discusses staff's review of the capability of the shielding and confinement features during off-normal and accident conditions, as appropriate.

11.2 Radiation Protection Features in the System Design

Each site licensee will apply its additional site-specific ALARA objectives, policies, procedures, and practices for members of the public and personnel. The design of the HI-STORM FW system components has been principally focused on maximizing ALARA during the short-term operations as well as during long-term storage. Some of the key design features engineered in the system components to minimize occupational dose and site boundary dose are summarized in Table 11.2.1 of the SAR. The design measures listed in Table 11.2.1 have been incorporated in the HI-STORM FW system to effectively reduce dose in fuel storage applications.

The staff considered ALARA assessment for the HI-STORM FW and found it acceptable for the described dose rates. Operational ALARA objectives, policies, procedures, and practices are the responsibility of the site licensee, as required by 10 CFR Part 20 and 10 CFR 72.104(b). The staff noted that the allowable contents result in relatively significant direct radiation dose rates. For the HI STORM FW system, high dose rates are of particular concern for operations involving the transfer cask (i.e., loading, unloading, and transport) and not the storage overpack, which has low dose rates. Therefore, each user may be required to take additional ALARA precautions during these operations to minimize doses to personnel and to make additional use of realistic fuel characteristics and distances to demonstrate compliance with public dose limits in 10 CFR Part 20 and 10 CFR Part 72.

11.3 Estimated On-Site Cumulative Dose Assessment

The staff reviewed the overall occupational dose estimates and found them acceptable. The occupational dose exposure estimates provide reasonable assurance that occupational limits in 10 CFR Part 20, Subpart C can be achieved. The staff expects actual operating times and personnel exposure rates will vary for each system, depending on site-specific operating conditions, including detailed procedures and special measures taken to maintain exposures ALARA. The collective exposures will be distributed among multiple personnel responsible for various tasks. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20, Subpart B. In addition, each licensee will demonstrate compliance with occupational dose limits in 10 CFR Part 20, Subpart C and other site-specific 10 CFR Part 50 license requirements with evaluations and measurements.

The staff noted that the unique design of the FW introduces possible streaming paths in addition to the inlet and outlet vents. The applicant evaluated these two features, including performing dose rate calculations. The applicant determined that these features are not significant streaming paths and do not pose an additional radiological concern. Although the HI-STORM FW system requires only minimal maintenance during storage, maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Since most of the maintenance is expected to occur outside the actual cask array, a dose rate of 10 mr/hr is estimated.

11.4 Estimated Controlled Area Boundary Dose Assessment

The applicant estimated offsite direct radiation dose rates at the site boundary for a FW overpack. Based on Table 11.4.1 in the SAR, the analyses indicated that a single FW overpack (assuming design-basis fuel and full occupancy) can meet the annual dose limit of 25 mrem at 100 meters.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each general licensee. The general licensee using the HI-STORM FW system must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance with 10 CFR 72.104(a). The actual doses to an individual beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, distances, and use of engineered features. In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each general licensee. The NRC may inspect the site-specific use of the HI-STORM FW system for compliance with radiological requirements.

The general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B and will demonstrate compliance with dose limits to individual members of the public, as required in 10 CFR Part 20, Subpart D, by evaluations and measurements.

Based on its shielding analyses, the applicant developed criteria for the radiation protection program in its proposed TS 3.2 for the FW contents. The criteria include the requirements for the cask user to (1) establish cask specific surface dose rate limits based on its 10 CFR 72.212 analyses; (2) assure maximum surface dose rates are below values based on the bounding shielding calculations for the top of the overpack; (3) measure dose rates at specific locations on the cask; and (4) implement specific corrective actions if measured dose rates during operations exceed the limits. The dose rate limits and measurements for the FW were proposed based upon considerations of what is necessary to ensure shielding effectiveness and with regard to importance to occupational and public dose.

Staff reviewed the proposed limits and measurements with consideration of these parameters and finds that they are acceptable.

11.5 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM FW system application, the staff finds the following:

F11.1 The SAR sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F11.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F11.3 The FW is designed to provide redundant sealing of the confinement system.

F11.4 The FW is designed to facilitate decontamination to the extent practicable.

F11.5 The SAR adequately evaluates the FW and its systems important to safety to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F11.6 The SAR sufficiently describes the means for controlling and limiting occupational exposures for the proposed contents within the dose and ALARA requirements of 10 CFR Part 20.

F11.7 Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The FW is designed to assist in meeting these requirements.

F11.8 The staff concludes that the design of the radiation protection system of the FW is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the FW will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering

12 ACCIDENT ANALYSIS EVALUATION

The specific discipline accident analyses are evaluated in sections 3 through 9 and 11 of this SER. The results of the evaluation are captured in their applicable sections.

13 TECHNICAL SPECIFICATION S AND OPERATING CONTROLS AND LIMITS EVALUATION

13.1 Objective

The technical specifications and operating controls and limits review ensures that the operating controls and limits of the TS, including their bases and justification, meet the requirements of 10 CFR Part 72. The evaluation is based on information provided by the applicant in the HI-STORM FW SAR Chapter 13 as well as accepted practices and any commitments discussed in other chapters of the SAR or other subsequent correspondence.

13.2 Evaluation Findings

F.13.1 The staff concludes that the conditions for use of the HI-STORM FW system identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

14 QUALITY ASSURANCE EVALUATION

The purpose of this review and evaluation is to determine whether Holtec has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G. Holtec's QA program has been previously evaluated in the review of the HI-STORM 100 Cask system, CoC No. 1014 application and subsequent amendments.

14.1 Areas Reviewed

QA Organization
QA Program
Design Control
Procurement Document Control
Instructions, Procedures, and Drawings
Document Control
Control of Purchased Material, Equipment, and Services
Identification and Control of Materials, Parts, and Components
Control of Special Processes
Licensee Inspection
Test Control
Control of Measuring and Test Equipment
Handling, Storage, and Shipping Control
Inspection, Test, and Operating Status
Nonconforming Materials, Parts, or Components
Corrective Action
QA Records
Audits

NUREG-1536 provides the criteria for evaluating the above 18 areas. In a number of cases, the description of, or actions to be taken by, personnel involved in quality activities were incorporated by reference to the applicable sections of the Holtec's Quality Assurance Manual (HQAM). It was therefore necessary to review such referenced sections in the HQAM to determine whether the QA program, as submitted, met the requirements of 10 CFR Part 72, Subpart G. While this evaluation determined that the QA program is acceptable, proper implementation of the QA program will be assessed during future NRC inspections.

14.2 Evaluation Findings

F14.1 The QA program describes the requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G, and 10 CFR Part 21, Reporting of Defects and Noncompliance.

F14.2 The structure of the organization and assignment of responsibility for each activity ensure that designated parties will perform the work to achieve and maintain specified quality requirements.

F14.3 Conformance to established requirements will be verified by qualified personnel and groups not directly responsible for the activity being performed. These personnel and groups report through a management hierarchy which grants the necessary authority and organizational freedom and provides sufficient independence from economic and scheduling influences.

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