

**HOLTEC INTERNATIONAL**  
**HI-STORM 100 CASK SYSTEM**  
**SAFETY EVALUATION REPORT**

ML003711779

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## **INTRODUCTION**

By letter dated October 26, 1995, as supplemented, Holtec International (Holtec) submitted an application to the Nuclear Regulatory Commission, in accordance with 10 CFR Part 72 for the review and approval of the HI-STORM 100 Cask System. In support of this application, Holtec submitted a Topical Safety Analysis Report (SAR) for the HI-STORM 100 Cask System<sup>1</sup>. February 4, 2000, Holtec submitted Revision 10 to the SAR.

This Safety Evaluation Report (SER) documents the review and evaluation of Revision 10 to the SAR. The SAR follows the format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.<sup>2</sup> This SER uses essentially the same Section-level format, with some differences implemented for clarity and consistency.

The review of the SAR addresses the handling and dry storage of spent fuel in a single dry storage cask design, the HI-STORM 100 Cask System. The cask would be used at an Independent Spent Fuel Storage Installation (ISFSI) that would be licensed under 10 CFR Part 72<sup>3</sup> at a reactor site operating with a 10 CFR Part 50 license.

The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. Decommissioning, to the extent that it is treated in the SAR, presumes that, as a bounding case, the HI-STORM 100 Cask System is unloaded and subsequently decontaminated before disposition or disposal.

### **References**

1. Topical Safety Analysis Report for the HI-STORM 100 Cask System, Rev. 10.
2. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."
3. U.S. Code of Federal Regulations. "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

## 1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the HI-STORM 100 Cask System is to ensure that Holtec has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

### 1.1 General Description and Operational Features

The HI-STORM 100 Cask System is a dry cask storage system for spent light water reactor fuel. The system comprises three discrete components: the multi-purpose canister (MPC), the HI-TRAC transfer cask, and the HI-STORM 100 storage overpack.

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. It is made entirely of stainless steel except for the neutron absorbers and aluminum heat conduction elements. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. The honeycombed basket, which is equipped with Boral neutron absorbers, provides criticality control.

There are three types of MPCs: the MPC-24, which holds up to 24 pressurized water reactor (PWR) fuel assemblies; the MPC-68, which holds up to 68 boiling water reactor (BWR) fuel assemblies; and the MPC-68F, which holds up to 68 BWR fuel assemblies, some of which are in the form of fuel debris. These MPCs are the same canisters that have been evaluated and approved for storage and transport in the HI-STAR 100 Cask System (Docket Numbers 72-1008 and 71-9261). All three MPCs have an outer diameter of 68-3/8 inches and an overall length of 190-1/2 inches. Figure 1-1 gives a cross-sectional view of an MPC.

The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the spent fuel pool to the storage overpack. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a water jacket attached to the exterior. Two types of HI-TRAC transfer casks are available: the 125-ton HI-TRAC and the 100-ton HI-TRAC. The weight designation is the maximum weight of a loaded transfer cask during any loading, unloading, or transfer operation. Both transfer cask types have identical cavity diameters (68-3/4 inches). The 100-ton HI-TRAC has an outer diameter of 91-1/4 inches and an overall height of 196-1/4 inches (with pool lid). The 125-ton HI-TRAC, which has thicker lead and water shielding, has an outer diameter of 94-5/8 inches and overall height of 201-1/2 inches (with the pool lid). Figure 1-2 gives a cross-sectional view of the 125-ton HI-TRAC transfer cask with the transfer lid.

The HI-STORM 100 overpack provides shielding and structural protection of the MPC during storage. The overpack is a heavy-walled, steel and concrete, cylindrical vessel. Its side wall consists of 26-3/4 inches of plain concrete that is enclosed between two carbon steel shells. The inner shell is 1-1/4 inches thick and the outer shell is 3/4 inches thick. The overpack has four air inlets at the bottom and four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has channels attached to its interior surface. The channels guide the MPC during insertion and removal. The channels also

provide a flexible medium to absorb impact loads, while still allowing cooling air to circulate through the overpack. The overpack cavity has a height of 191-1/2 inches and, after allowance for the attached channels, an inner diameter of 69-1/2 inches to accommodate the MPC. The overall external dimensions of the overpack are 132-1/2 inches in diameter and 239-1/2 inches in height. The overpack weighs approximately 135 tons without the MPC and a maximum of 180 tons with the MPC. Figure 1-3 gives a cross-sectional view of the HI-STORM 100 overpack.

The basic sequence of operations for the HI-STORM 100 Cask System is as follows: (1) the transfer cask, with the MPC inside, is lowered into the spent fuel pool and the MPC is loaded; (2) the transfer cask and MPC are removed from the spent fuel pool and the MPC is drained, dried, backfilled, and leak tested; (3) the transfer cask is placed on top of the overpack and the MPC is lowered into the overpack; and (4) the overpack, with the MPC inside, is moved to the storage pad. A loaded HI-TRAC transfer cask can be handled vertically or horizontally; a loaded HI-STORM 100 overpack can only be moved vertically.

MPC transfer between the transfer cask and overpack can be performed inside or outside a 10 CFR Part 50 controlled structure (e.g., a reactor building). Section 2.3.3 of the SAR provides detailed design criteria for the auxiliary equipment and structures that would be used to perform an MPC transfer outside of a 10 CFR Part 50 controlled structure. The purpose of these design criteria is to prevent, during such transfers, the cask system from experiencing loads more severe than those considered in the structural and thermal analyses. The cask transfer facility requirements are also specified in Appendix B to the Certificate of Compliance.

## **1.2 Drawings**

Section 1.5 of the SAR contains the non-proprietary drawings for the HI-STORM 100 Cask System, including drawings of the structures, systems, and components important to safety. The drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the entire system. Specific structures, systems, and components are evaluated in Sections 3 through 14 of this SER.

## **1.3 Cask Contents**

The approved contents for the MPC-24 are intact PWR fuel assemblies with maximum initial enrichments of 4.0 to 4.6 weight percent <sup>235</sup>U. The approved contents for MPC-68 are intact or damaged BWR fuel assemblies with maximum planar average initial enrichments of 2.7 to 4.2 weight percent <sup>235</sup>U. The approved contents for the MPC-68F are intact or damaged BWR fuel assemblies and BWR fuel debris with a maximum planar average initial enrichment of 2.7 weight percent <sup>235</sup>U. The BWR fuel may be shipped with or without the fuel channels. Damaged fuel assemblies and fuel debris must be placed in damaged fuel containers (DFCs).

The specific enrichments and physical, thermal, and radiological characteristics of the approved contents are given in Section 2.1 of the SAR. The SAR (Table 1.0.1) also provides definitions for intact fuel assemblies, damaged fuel assemblies, and fuel debris.

As previously stated, the MPC-24, MPC-68, and MPC-68F are the same canisters that are approved for use in the HI-STAR 100. Likewise, their respective contents are the same

contents that have been evaluated and approved for storage and transport in the HI-STAR 100 Cask System.

#### **1.4 Qualification of the Applicant**

Holtec International provides the design, analysis, licensing support, and quality assurance for the HI-STORM 100 Cask System. Fabrication of the system is done by one or more qualified fabricators under Holtec's quality assurance program. Section 1.3 of the SAR adequately details Holtec's technical qualifications and previous experience in the area of spent fuel storage design and licensing.

#### **1.5 Quality Assurance**

The quality assurance program, and implementing procedures, are evaluated in Section 13 of this SER.

#### **1.6 Evaluation Findings**

- F1.1** A general description and discussion of the HI-STORM 100 Cask System are presented in Section 1 of the SAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.
- F1.2** Drawings for structures, systems, and components important to safety are presented in Section 1.5 of the SAR. Specific structures, systems, and components are evaluated in Sections 3 through 14 of this SER.
- F1.3** Specifications for the spent fuel to be stored in the dry cask storage system are provided in Section 1.2.3 of the SAR. Detailed specifications are presented in Section 2.1 of the SAR and Appendix B to the Certificate of Compliance.
- F1.4** The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.3 of the SAR and are acceptable to the NRC staff.
- F1.5** The quality assurance program and implementing procedures are described in Section 13 of the SAR and are evaluated in Section 13 of the SER.
- F1.6** The HI-STORM 100 Cask System was not reviewed in this SER for use as a transportation cask.
- F1.7** The staff concludes that the information presented in this section of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is based on a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry cask storage practices detailed in NUREG-1536.

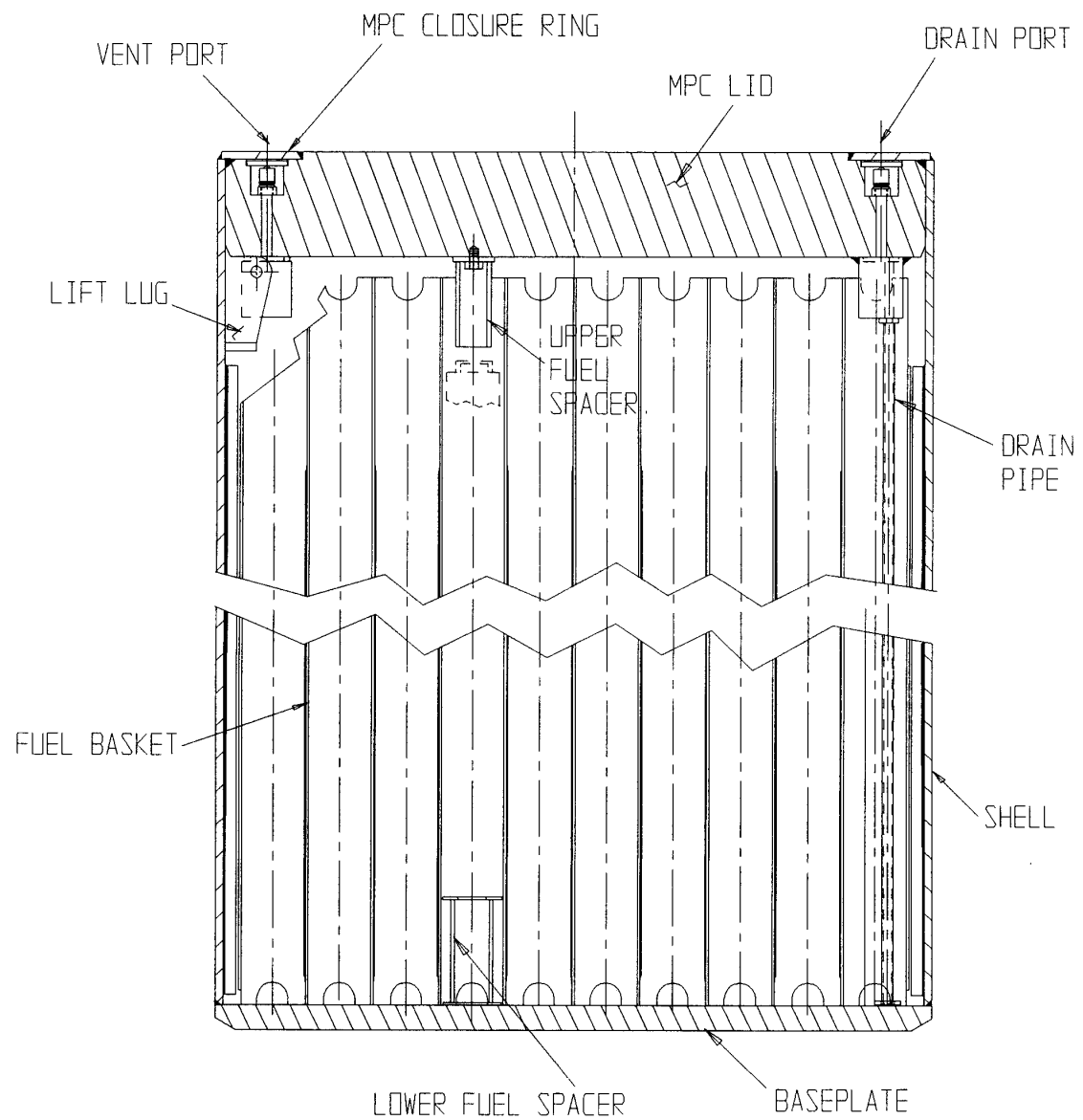


Figure 1-1 Cross- Sectional View of an MPC

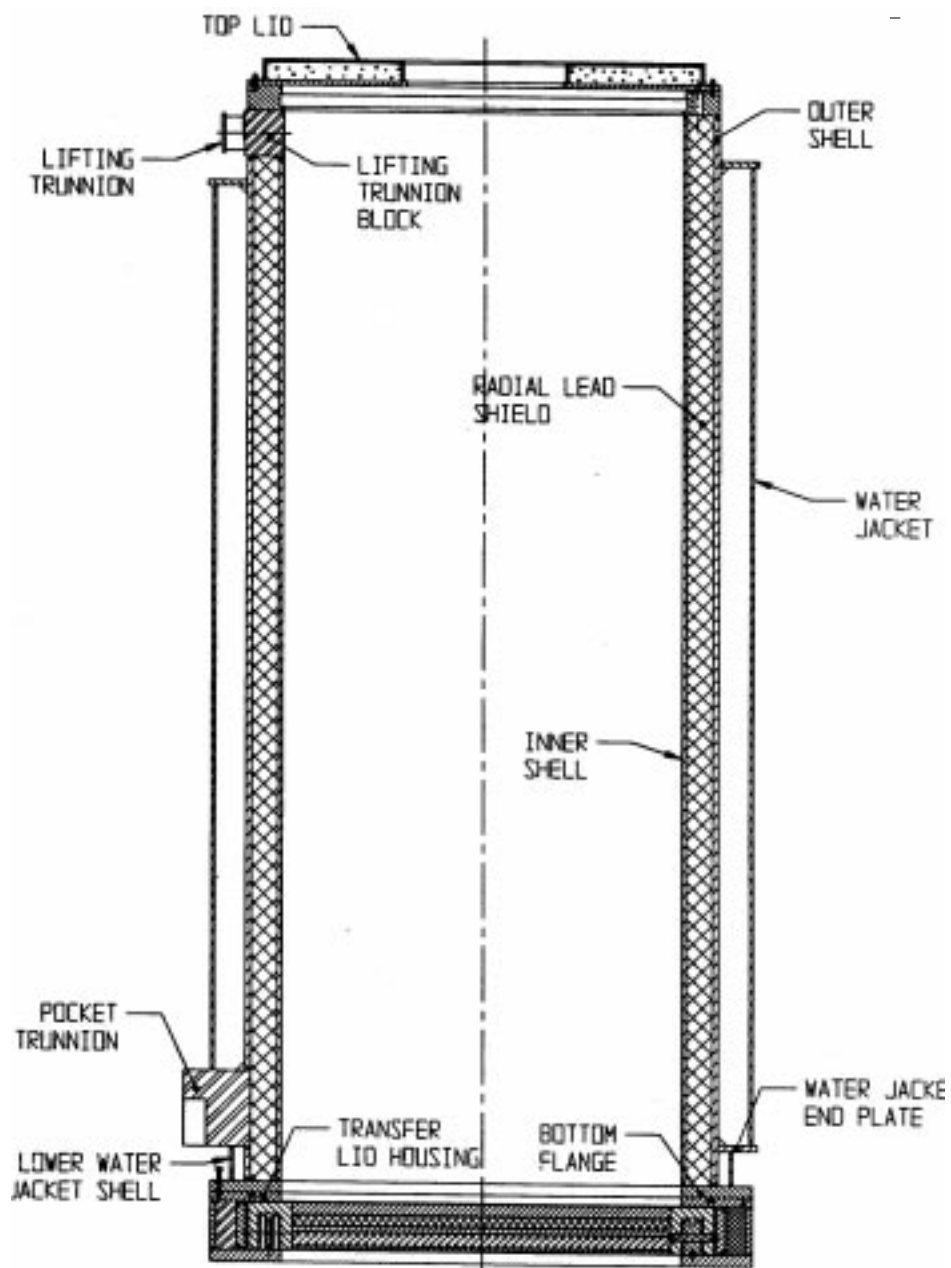


Figure 1-2 Cross-Sectional View of HI-TRAC Transfer Cask



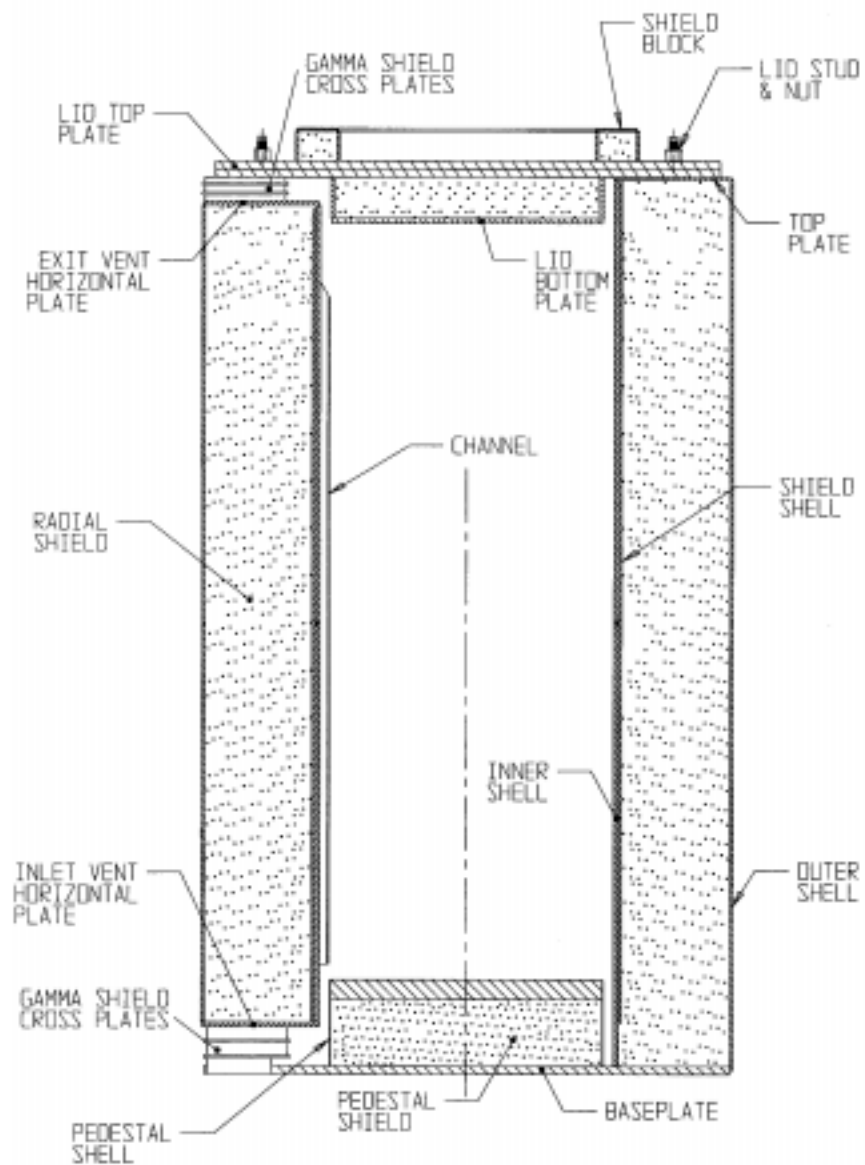


Figure 1-3 Cross-Sectional View of the HI-STORM 100 Storage Overpack

## **2.0 PRINCIPAL DESIGN CRITERIA**

The objective of evaluating the principal design criteria related to the structures, systems, and components 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**

The structures, systems, and components important to safety are summarized in Table 2.2.6 of the SAR. In this table, each component is assigned a safety classification. The safety classifications are based on the guidance in NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

Table 2.2.6 also identifies the function and governing code of the components. The governing code for the structural design of the MPC, the transfer cask, and the metal components in the overpack is the ASME Boiler and Pressure Vessel Code (ASME Code). The governing code for the concrete in the overpack is American Concrete Institute (ACI) 349. Exceptions to these Codes are delineated in the SAR.

### **2.2 Design Bases for Structures, Systems and Components Important to Safety**

The HI-STORM 100 Cask 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 100 Cask System is designed to store up to 24 PWR fuel assemblies and up to 68 BWR fuel assemblies. Detailed specifications for the approved fuel assemblies are given in Section 2.1 of the SAR. 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**

Section 2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the HI-STORM 100 Cask System is analyzed. These are evaluated in Sections 3 through 14 of this SER.

### **2.3 Design Criteria for Safety Protection Systems**

The principal design criteria for the MPC, the HI-STORM overpack, and the HI-TRAC transfer cask are summarized in SAR Tables 2.0.1, 2.0.2, and 2.0.3, respectively. The codes and standards of the design and construction of the system are specified in Section 2.2 of the SAR.

### **2.3.1 General**

Section 2 of the SAR states that the design life of the HI-STORM 100 Cask System is 40 years. The adequacy of the MPC, transfer cask, and overpack for this design life is discussed in Sections 3.4.11 and 3.4.12 of the SAR. The system is approved for a 20-year storage period.

### **2.3.2 Structural**

The structural analysis is presented in Section 3 of the SAR. The HI-STORM 100 Cask System components are designed to protect the cask contents from significant structural degradation, preserve retrievability, provide adequate shielding, and maintain subcriticality and confinement under the design basis normal, off-normal, and accident loads. The design basis normal, off-normal, and accident conditions are defined in Section 2.2 of the SAR. The load combinations for which the MPC, transfer cask, and overpack are designed are defined in Section 2.2.7 of the SAR.

### **2.3.3 Thermal**

The thermal analysis is presented in Section 4 of the SAR. The HI-STORM 100 Cask System is designed to passively reject decay heat. Heat removal, by conduction, radiation, and natural convection, is independent of intervening actions under normal, off-normal, and accident conditions. The thermal design criteria include maintaining fuel cladding integrity and ensuring that temperatures of materials and components important to safety are within the design limits.

### **2.3.4 Shielding/Confinement/Radiation Protection**

The shielding and confinement analyses and the radiation protection capabilities of the HI-STORM 100 Cask System are presented in Sections 5, 7, and 10 of the SAR. Confinement is provided by the MPC, which has a welded closure. The MPC's confinement function is verified through hydrostatic testing, helium leak testing and weld examinations. Radiation exposure is minimized by the neutron and gamma shields and by operational procedures.

### **2.3.5 Criticality**

The criticality analysis is presented in Section 6 of the SAR. The design criterion for criticality safety is that the effective neutron multiplication factor, including statistical biases and uncertainties, does not exceed 0.95 under normal, off-normal and accident conditions. The design features relied upon to prevent criticality are the fuel basket geometry and permanent neutron-absorbing Boral plates. The continued efficacy of the Boral over a 20-year storage period is assured by the design of the system. Depletion of the  $^{10}\text{B}$  in the Boral is negligible because the neutron flux in the MPC over the storage period is low.

### **2.3.6 Operating Procedures**

Generic operating procedures are described in Section 8 of the SAR. This section outlines the loading, unloading, and recovery operations and provides the basis and general guidance for more detailed, site-specific procedures.

### **2.3.7 Acceptance Tests and Maintenance**

The acceptance test and maintenance program are presented in Section 9 of the SAR, including the commitments, industry standard, and regulatory requirements used to establish the acceptance, maintenance, and periodic surveillance tests.

### **2.3.8 Decommissioning**

Decommissioning considerations for the HI-STORM 100 Cask System are presented in Section 2.4 of the SAR and evaluated in Section 14 of this SER.

## **2.4 Evaluation Findings**

- F2.1** The staff concludes that the principal design criteria for the HI-STORM 100 Cask 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 Sections 3 through 14 of this SER.

## **3.0 STRUCTURAL EVALUATION**

This section evaluates the structural designs of the HI-STORM 100 Cask System. Structural design features and design criteria are reviewed, and analyses related to structural performance under normal, off-normal, accident, and natural phenomena events are evaluated.

### **3.1 Structural Design Features and Design Criteria**

#### **3.1.1 Structural Design Features**

The HI-STORM 100 Cask System comprises of the HI-STORM 100 overpack, the HI-TRAC transfer cask, and the MPCs. A description of these components is given in Section 1.1 of this SER.

The HI-STORM 100 overpack includes both concrete and structural steel components. The concrete is enclosed in inner and outer carbon steel shells that are connected to each other by four radial plates, and top and bottom plates. The main structural function is provided by the carbon steel components. The structural steel components are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF for Class 3 plate and shell components. The concrete, which is not reinforced, functions mainly as shielding. The concrete is made of Type II Portland Cement and has a compressive strength of 4000 psi and a minimum density of 146 lbs/cubic ft. For lifting the overpack, four threaded anchor blocks are welded to the radial plates which are, in turn, welded to the inner and outer steel shells and the baseplate. Lifting lugs are then inserted into the blocks. The overpack may also be lifted from the bottom using specially designed lifting transport devices.

The HI-TRAC transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. The main structural function is provided by the carbon steel. The transfer cask is not a pressure vessel. However, the structural steel components of HI-TRAC are designed for the stress limits of the ASME Code, Section III, Subsection NF, Class 3. The lifting trunions on the transfer cask are designed to meet the design safety factor requirements for a single-failure proof lifting equipment.

As previously stated, the MPCs and their respective contents are the same canisters and content that have been evaluated and approved for storage and transportation in the HI-STAR 100 Cask System. The structural analyses (storage and transportation) of the MPCs and contents in the HI-STAR 100 Cask System bound the structural analysis of the MPCs and contents in the HI-STORM 100 Cask System. This is due to the lower design g-load for the HI-STORM 100 Cask System (i.e., 45g for HI-STORM versus 60g for HI-STAR, as discussed below).

#### **3.1.2 Structural Design Criteria**

The design criteria for the HI-STORM 100 Cask System are summarized in Tables 2.0.1, 2.0.2 and 2.0.3 of the SAR. Other industry standards and regulatory guidance documents that apply

for the purposes of determining loads and load combinations are: ANSI 57.9, ANSI A58.1, NUREG-0800, and Regulatory Guides 1.76, 1.60, and 1.61.

For the overpack, the principal structural design criteria for normal, off-normal, accident, and natural phenomena events are based on ASME Code, Section III and ACI 318 and 349. The structural steel components are designed to Subsection NF of the ASME Code. The threaded holes in the overpack steel blocks that are used for critical lifts are designed to the criteria of ANSI N14.6. While ASME Code, Section III, was intended for the design and fabrication of reactor vessels, the NRC accepts its use, to the extent practical, for spent fuel storage systems. Table 2.2.15 of the SAR lists the exceptions to the ASME Code for the overpack. These exceptions to the Code design and fabrication criteria were reviewed and found acceptable.

For the transfer cask, the principal structural design criteria for normal and accident events are based on ASME Code, Section III, Subsection NF, Class 3. Only Service Levels A and D requirements for the normal and accident conditions are applicable to the transfer cask. The transfer cask lifting trunions are designed in accordance with NUREG-0612 and ANSI 14.6.

The individual loads and loading combinations for which the overpack and transfer cask are evaluated are listed in Tables 2.2.13 and 2.2.14, respectively, of the SAR. The structural evaluation of the overpack and the transfer cask under these loads are discussed in Sections 3.4 and 3.5, respectively, of this SER.

The allowable stresses for the HI-TRAC transfer cask are listed in Table 2.2.12 of the SAR. The allowable stresses are based on the appropriate ASME subsections and service levels. Table 3.3.2 of the SAR contains the numerical values of the allowable stress/stress intensities for the steel structural material, SA-516 Grade 70, according to temperatures. The staff concludes that these values meet the ASME allowable stresses, based on the appropriate ASME subsection and service levels.

### **3.2 Weights and Center of Gravity**

Section 3.2 of the SAR presents the weights for the HI-STORM 100 Cask System. This section lists the weights of the individual components as well as the total system weights. Section 3.2 also presents the locations of the centers of gravity (CGs). Because the non-axisymmetric effects of the contents are neglected, all CGs are located on the cask centerline.

### **3.3 Structural Materials**

The primary structural material in the overpack and transfer cask is carbon steel. In the MPC, the primary structural material can be any one of the stainless steel alloys specified in Appendix 1.A of the SAR. Section 3.3 of the SAR provides the mechanical properties used in the structural design and evaluation of the HI-STORM 100 Cask System. The properties include yield stress, ultimate stress, modules of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. Values are presented for a range of temperatures which envelopes the maximum and minimum temperatures under all service conditions.

For all structural materials, the stress limits have been defined at or below the maximum temperature allowed by the ASME Code, Section II, Part D. The information provided on structural materials is consistent with the application of the accepted design Code, ASME Section III, selected for the overpack. The staff concluded that the material properties used are appropriate for all load conditions. The staff verified that the SAR clearly references acceptable sources, primarily the ASME Code, for all material properties.

The staff concluded material properties and characteristics needed to satisfy safety requirements will be maintained over the approval period. The life cycle may include conditions experienced during loading, transport, emplacement, storage, transfer, retrieval, and decommissioning. Service conditions include normal, off-normal operations, accidents, and natural phenomena. The staff concluded the materials of construction used for the HI-STORM 100 Cask System are compatible with the environment during all design configurations.

### **3.3.1 Concrete**

Table 3.3.5 provides the mechanical properties of the plain concrete, although it is not considered a structural component of the overpack. Allowable bearing stress for plain concrete for normal loading conditions is in conformance with ACI 318. The requirements of ACI 349, which provide for the concrete mix quality, have been included in the design. The use of plain concrete, while not included in NUREG-1536 and an exception to ISFSI design practice, has been found to be acceptable due to the special design features of the overpack.

### **3.3.2 Brittle Fracture of Materials**

Paragraph 3.1.2.3 of the SAR discusses test requirements of ferritic steels to prevent brittle fracture at low temperatures. The lowest service temperature (LST) or minimum ambient temperature for handling the loaded overpack is restricted to 0°F. ANSI N14.6 requires a nil ductility transition temperature (NDT) of 40°F lower than the LST. Two parts of the overpack are involved with the lifting via overhead crane or other lifting device. They are the anchor block fabricated from SA 350 LF3 and the radial plate fabricated from SA 516 Grade 70. Table 3.1.18 of the SAR states that the radial plate is exempt from testing because the thickness is 0.75 inches. However, the anchor block is greater than 0.75 inches thick and is thus subject to testing according to paragraph NF-2331 and Section IIA of the ASME Code. The acceptance criteria in the SAR Table 3.1.18 are given as Table NF-2311(a)-2 of the ASME B&PV Code. This table states that for steels with less than 55 ksi yield strength (the SA 350 LF3 yield strength is 37.5 ksi) a minimum Charpy V-notch absorbed energy of 25 ft-lb is required. Table 5 of the Section IIA of the Code gives the standard impact test temperature of the LF3 grade as -150°F. The staff has reviewed the Holtec SAR and concurs that the ASME B&PV Code acceptance criteria are adequate.

Components of the HI-TRAC transfer cask, such as lifting trunnion, lifting trunnion block, and pocket trunnion have thickness greater than 2 inches. The HI-TRAC lifting trunnion is fabricated from SB-637 steel, a high strength nickel alloy material. This material has a high resistance to fracture at low temperature. All other structural components in HI-TRAC are made of SA516 Grade 70 steel which possesses sufficient fracture toughness to preclude brittle fracture.

### **3.3.3 Material Compatibility**

Discussion of potential chemical and galvanic reactions is given in SAR Section 3.4.1. The MPC is constructed primarily of stainless steel and has previously been evaluated and accepted for use in the HI-STAR 100 Cask System. The overpack and transfer cask each combine low alloy and nickel alloy steels, carbon steels, neutron and gamma shielding materials, and bolting materials. All of these materials have a long history of non-galvanic behavior within close proximity of each other. Also, paint used on the transfer cask (Carboline 890, Thermaline 450) do not chemically react with borated water. The staff agrees with the applicant's conclusion that the HI-STORM 100 Cask System is constructed of materials which will not produce a significant chemical or galvanic reaction and the attendant corrosion or combustible gas generation.

### **3.3.4 Welds**

The MPC welds were evaluated and found to be acceptable in the HI-STAR 100 Cask System storage and transportation reviews. The staff's conclusions in those reviews are applicable to the use of the MPCs in the HI-STORM 100 Cask System.

The requirements and structural analyses for the overpack welds are included in the SAR. This includes commitments on welder qualification, weld quality and welding procedures, weld inspection by description and by citation of applicable codes and standards, and the exceptions to the applicable code (ASME Code, Section III, Subsection NF). The overpack weld designs, commitments to weld material and fabrication, commitments to weld inspection and test, and structural analyses of weld joints in the SAR were reviewed and compared with the requirements of the codes as accepted for use with the overpack. The designs and other commitments are considered acceptable. The welds are considered to provide acceptable margins of safety for meeting the important-to-safety functional requirements.

### **3.3.5 Bolting Materials**

Material properties of the bolting materials used in the overpack and transfer cask lifting trunnions are given in Table 3.3.4 of the SAR. These were found to be acceptable.

### **3.3.6 Lead**

Lead is not considered to be a structural material in the HI-TRAC transfer cask. Its loading capacity is neglected in the structural analysis, except in the analysis of a tornado missile strike where it acts as a missile barrier. Mechanical properties of lead are provided in Table 3.3.5 of the SAR.

## **3.4 Structural Analysis of HI-STORM 100 Overpack**

### **3.4.1 Normal and Off-Normal Conditions**

Structural analysis in the SAR for the overpack steel components and welds complies with the requirements of ASME Code, Section 3, NF for Class 3 plate and shell supports. Equipment



and system components used for lifting were analyzed in accordance with ANSI N14.6. Use of these codes for the overpack structural analysis is considered to be acceptable and to comply with 10 CFR 72.24(d). The concrete structural analysis, where appropriate, complies with the requirements of ACI 318-95 for plain structural concrete. Use of plain concrete is an exception to ISFSI design practice that all concrete be reinforced. Its use was requested by Holtec and accepted by the staff on the basis of special features of its use in the overpack.

The SAR overpack analyses for normal conditions considered deadweight (including fully loaded MPC, HI-TRAC, and both, where appropriate), operating temperature, and handling loads associated with loaded overpack transfer on site, and with loading and unloading the MPC.

During normal operations and conditions, the lowest safety factors for the overpack structure occur when the overpack is lifted with a full MPC. Both top and bottom lifting were analyzed (computations at SAR Appendix 3.D, summarization at SAR Section 3.4.3.5). All factors of safety are shown to exceed 1.0. The staff determined that the analytical approach and computations are acceptable; that the sections, components, and welds for which factors of safety were determined include those that should have the lowest factors of safety; and that the resulting factors of safety are acceptable.

The following additional normal condition handling situations were analyzed in the SAR: static load of HI-TRAC with maximum weight MPC atop the overpack (SAR Section 3.4.4.3.2.1); and lift of the overpack lid (computations at SAR Appendix 3.AC, summarization at SAR Section 3.4.3.7). These analyses and their results have been evaluated by the staff and have been determined to be acceptable.

The off-normal case considered for structural analysis was off-normal temperature. Acceptability of the description and analysis of the potential consequences of the condition is addressed in the evaluation at SER Section 11. The potential for loading the overpack structure due to temperature differences between the MPC and the overpack was determined not to be feasible (SAR Section 3.4.4.2 and Appendices 3.U, 3.W, and 3.AF) for normal and off-normal (and accident level) temperatures. The analyses of temperatures are at SAR Section 4 and are evaluated at SER Section 4. The computations and their results were acceptable.

The potential for off-normal handling loads was addressed and determined not to be significant. The principal concern would be impairment of ready retrieval of stored materials due to a jam of the MPC during insertion into the overpack. A jam could be due to misalignment or insufficient clearance between the MPC and the overpack interior. Misalignment is precluded by the close tolerances between HI-TRAC and the MPC, with HI-TRAC serving as an alignment guide during MPC insertion into the overpack.

The potential for jam due to insufficient clearance between the MPC exterior and overpack interior is avoided by the 3/4 inch smaller inside diameter of HI-TRAC relative to the minimum inside clearance within the overpack. This difference is considered to exceed potential stackups of manufacturing tolerances under the quality control commitments in the SAR. If a jam due to stackup of manufacturing tolerances were to occur, the problem should occur upon insertion of the MPC into HI-TRAC (prior to fuel loading). The HI-TRAC-MPC radial clearances during

handling operations are less than the radial clearances with the MPC and overpack. Further, a jam due to temperature differences upon loading (e.g., a hot MPC into an overpack at the minimum temperature at which loading operations are permitted (0°F)) should be self-clearing, when the overpack is warmed by the MPC and the MPC shell is cooled by the ambient air flow.

The staff reviewed analyses presented in the SAR and concluded that the overpack design and planned operational use allow acceptable structural margins of safety during normal operations and expected operational occurrences during the life of the ISFSI, in compliance with 10 CFR 72.24(d)(1).

### **3.4.2 Accident Conditions**

The SAR includes analysis of the overpack design for adequacy to prevent or mitigate the consequences of accidents, including natural and man made phenomena and events, in compliance with 10 CFR 72.24(d)(2), 10 CFR 72.122(b) and (c). Acceptability of the accident descriptions and analysis of potential consequences is addressed in the evaluation of the accident analysis (SER Section 11).

The SAR includes the following load combinations for accident conditions for the HI-STORM 100 overpack; (1) deadweight, operating temperature, and handling accident, (2) deadweight, operating temperature, and accident external pressure, (bounds the external pressure due to explosion), (3) deadweight, operating temperature, plus the following (as separate cases)-- earthquake, tornado wind, tornado missile, and flood.

Additional potential accident level conditions and events that do not have significant structural margin of safety implications are included in the accident analysis (SAR Section 11) and are, therefore, not addressed in this structural evaluation. Structural concerns for loss of steel ductility at extremely low environmental temperatures are avoided by the operating limit of 0°F for transfer operations.

The SAR accident condition structural analyses for the overpack steel components and welds comply with the requirements of ASME Code, Section 3, NF for Class 3 plate and shell supports. Margins of safety were determined for Service Level D. Further analysis was performed to determine that any residual deformation would be acceptable. Concern with permanent deformation allowed by Service Level D stress limits is primarily related to the potential for significant impairment of ready retrievability of the stored materials.

The SAR does not include structural analyses for the overpack for the following accident level events and conditions (described and discussed in the indicated SAR sections): HI-TRAC transfer cask handling accident (Section 11.2.1), partial blockage of MPC basket vent holes (Section 11.2.5), 100% fuel rod rupture (Section 11.2.9), confinement boundary leakage (Section 11.2.10), lightning (Section 11.2.12), 100% blockage of air inlets (Section 11.2.13), burial under debris (Section 11.2.14), and extreme environmental temperature (Section 11.2.15).

The first five of these accidents (11.2.1, 11.2.5, 11.2.9, 11.2.10, and 11.2.12) have no structural consequences. The structural consequences for the overpack for 100% blockage of air vents or

burial under debris are within the envelopes of other analyzed accident level conditions or events and are, therefore, not separately analyzed. Structural consequences of the extreme environmental temperature are not considered to be significant.

Staff review of rationale for omitting separate structural analysis for the above accidents for the overpack has determined that such omissions are acceptable. The accidents should have no significant structural effects. Thermal analyses show that the temperatures of the structural materials should remain within allowables. The accidents having thermal consequences should result in less differential thermal expansion than computed for the analyzed (SAR Appendix 3.AF) situation of insertion of a hot MPC into a cold overpack.

The staff evaluation of the acceptability of the overpack design for each of the accident level conditions subject to structural analysis is given below.

### Seismic Event

The seismic event is analyzed in SAR Section 3.4.7.1. The HI-STORM 100 Cask System is subjected to the design-basis seismic event consisting of three orthogonal statistically independent acceleration time-histories. The HI-STORM system is conservatively considered as a rigid body subject to a set of horizontal and vertical inertia forces to determine the design-basis earthquake that will not cause cask tipover. The analysis assumed the vertical seismic load is only a fraction of the horizontal seismic load and it is always acting in the upward direction. Based on this conservative static tipover analysis, the applicant determines the permissible seismic input to ensure against incipient tipping. These seismic input are then used to assess the sliding of the HI-STORM system. The dynamic analysis indicates that the HI-STORM system could slide an average of 0.12 inches with the coefficient of friction value equal to 0.25. The static analysis indicates that the HI-STORM system would not slide with a static coefficient of friction value of 0.53. Because the coefficient of dynamic friction is approximately 25% smaller than the coefficient of static friction<sup>1</sup>, the staff concludes that use of 0.25 as the coefficient of friction in the dynamic analysis is acceptable. Based on the analyses performed, the acceptable design-basis earthquake input on the top surface of the ISFSI pad is listed below:

#### Design-Basis Earthquake Input on the Top Surface of an ISFSI Pad

Horizontal g-level in Each of Two Orthogonal Directions	Horizontal g-level Vector Sum	Corresponding Vertical g-Level (upward)
0.32g	0.445g	$0.5 \times 0.32g = 0.16g$
0.3g	0.424g	$0.667 \times 0.3g = 0.2g$
0.28g	0.397g	$0.89 \times 0.28g = 0.25g$

The staff concludes that the seismic analysis is adequate and acceptable. The design-basis earthquake input may be used for determining the acceptability of a site.

## **Tornado Wind and Missile Loading**

Tornado and tornado missile effects on the system safety are discussed in SAR Section 3.4.8 and are the subjects of analysis in SAR Appendices 3.C and 3.G. The analyses are based on the worst tornado and tornado missile parameters used for the design and analysis of U.S. nuclear facilities subject to NRC approval. The analyses address the following: (1) the potential for overpack and minimum weight MPC overturning due to tornado winds and due to simultaneous combination of tornado winds (or pressure drop) and worst case missile strike; (2) the potential for MPC penetration due to the most effective missile strikes (two different nominal missiles at the side, top, and vents); and (3) overpack permanent deformation as might risk significant impairment of ready retrievability of stored materials due to missile strike.

Staff evaluation has determined that the SAR analytical approaches, assumptions, and results are acceptable. The overturning analyses indicate that the worst case loadings result in overpack rotation that is only about 1/10th that needed to reach the point of incipient overturning. The analysis of missile penetration indicates that the worst case may result in penetration of the overpack outer steel shell but not penetration of the concrete shield and would not cause stresses in the inner shell to exceed the elastic limit. There would be no risk of penetration of the confinement barrier of the MPC. There would be no transient or permanent loading of the MPC due to overpack penetration or deformation. The analysis of the potential for permanent deformation of the overpack such as may impair ready retrievability determined that there would be no permanent deformation of the overpack inner shell and, therefore, no impairment of retrievability.

The staff concludes that the tornado and tornado missile analyses are adequate and acceptable. The phenomena analyzed are considered to envelop the corresponding phenomena at all points on U.S. territory.

## **Flood Loading**

The SAR structural analysis of the overpack for flood is in Section 3.4.6. The analysis determines factors of safety for sliding and overturning due to a uniform flood current of 15 ft/sec applied to a fully submerged overpack with the minimum weight MPC. The computed factors of safety exceed the required minimum factor of safety of 1.1 for sliding and overturning given in ANSI ANS 57.9-84, Section 6.17.4.1, for other accident level phenomena.

The staff determined that the analytical approaches, computations, results, and acceptance criteria are acceptable. The staff concludes that the flood analysis is adequate.

## **Tipover**

The overpack will not tipover as a result of a credible natural phenomenon, including tornado wind and a tornado-generated missile, a seismic event, or a flood. However, to demonstrate the defense-in-depth features of the design, a non-mechanistic tipover scenario per NUREG-1536 was analyzed.

The tipover is described in SAR Section 3.4.10. The SAR structural analysis of a tipover of an overpack with MPC is in Appendix 3.B. This analysis is to determine if the overpack structure would deform (dynamically or permanently) to an extent that would impose additional loads on the MPC or interfere with possible MPC removal. Appendix 3.A provides the analysis of the maximum decelerations that may be experienced by the overpack and MPC within the overpack as a result of tipover. Appendix 3.K provides a structural analysis of the integrity of the overpack lid during tipover. Appendix 3.L provides structural analysis of the studs securing the overpack lid to the overpack during tipover. The analyses of the lid structural integrity and securing studs are discussed in SAR Section 3.4.4.3.2.2. Summary factors of safety are shown in SAR Table 3.4.5.

The structural analyses of tipover in the SAR conclude that the overpack would maintain safety, that the factors of safety of the most critically loaded elements would be above 1.0, and that any deformation would not impose loads on the MPC or significantly impair ready retrievability of the stored materials. The computed decrease in diameter of 0.11 inches is compared (in SAR Appendix 3.B) to a stated radial clearance of 0.1875 inches (Note: The clearance between the maximum MPC diameter and the channels/shims attached to the inner shell of the overpack is 13/16 inches, providing an average radial clearance of about 0.4 inches).

The staff review determined that the analytical approaches, computations, results, and acceptance criteria are acceptable. The assumptions relating to the receiving surface are acceptable. The pad used is the reference pad used in the HI-STAR 100 Cask System SAR. This pad is identical to that used for the Lawrence Livermore National Laboratory "Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads and Application to Generic ISFSI Storage Cask for Tipover and Side Drop" (UCRL-ID-126295). The factors of safety determined are considered to include the lowest factors of safety associated with tipover of the overpack.

The SAR tipover analyses for the overpack do not include the following: possibly denting the overpack outer shell and disruption of the adjacent concrete shielding along the line of impact, affect of the reaction causing MPC deceleration on the supporting channel(s) at the overpack interior, or affect of MPC translation toward the top of the overpack following impact (due to centrifugal acceleration). Staff consideration of these is summarized below.

Local denting of the overpack outer shell and possible fracturing of the adjacent concrete shield material ( between the inner and outer overpack shells) would not affect safety or retrievability and, if significant, would be visually evident.

Failure of a channel supporting the MPC at time of tipover impact would not affect retrievability or significantly affect sidewall clearances of the MPC within the overpack (a consideration for cooling). Any failure is expected to be by bending of the web and buckling of the flanges possibly accompanied by some local weld failures. The extent of deformation would be limited due to the progressive assumption of load by adjacent channels.

The MPC may translate into contact with the overpack lid upon ground impact following tipover. When vertical, there is only about 1 inch clearance between the top of the MPC and the overpack lid. This does not permit development of significant relative velocity between the MPC

and the overpack in the tipover event for the maximum centrifugal acceleration (when the overpack is essentially horizontal and the MPC is only restrained longitudinally by steel on stainless steel friction). Due to the finite distance of MPC travel necessary for it to impact the overpack lid, any such impact would not be concurrent with maximum deceleration due to ground impact. Therefore, the tensile load to the lid restraining studs would not occur at the time of maximum shear in those studs.

As a result of the above considerations, the staff concludes that the scope of the SAR tipover analysis is acceptable.

### **Accidental Drop**

The SAR structural analysis of accidental drops of an overpack with a full MPC is in Appendix 3.M. This analysis determines factors of safety for the overpack structural components in the load path associated with an 11 inch vertical drop onto the reference pad. Appendix 3.A provides the determination of the maximum height (11 inches) that the overpack and MPC within the overpack may be dropped with overpack longitudinal axis vertical without imposing more than 45g deceleration on the MPC. This drop height is used as a limiting condition of use for the height that the overpack with MPC may be above a receiving surface. Summary minimum factors of safety are shown in SAR Tables 3.4.5 and 3.4.9.

The analyses of the overpack structural elements in the SAR determine that the factors of safety of the most critically loaded elements would be above 1.0, and that any deformation would not impose loads on the MPC or impair ready retrievability of the stored materials.

The staff review determined that the analytical approaches, computations, results, and acceptance criteria are acceptable. The assumptions relating to the receiving surface (used for both drop and tipover) are acceptable. The pad is the reference pad used for the HI-STAR 100 Cask System SAR. This pad is identical to that described in UCRL-ID-126295. The factors of safety determined are considered to include the lowest factors of safety associated with tipover of the overpack.

The SAR drop analysis does not include examination of a corner drop or drop with the overpack longitudinal axis horizontal. The analyses also do not include the stresses in all of the welds or all of the component members of the overpack body or lid weldments.

A corner drop with the center of gravity over the point of impact is considered to be most likely to cause local permanent distortion of the overpack. A drop from the maximum design drop height of 11 inches would, however, result in greater penetration of the receiving surface, reducing the maximum decelerations experienced by the MPC. The effects of subsequent overturning would be within the effects determined for the non-mechanistic tipover event. Any simultaneous deformation of the overpack would further reduce the deceleration. Any significant permanent deformation of the overpack at the point of impact would be readily observable following the event. The restraint of 11 inches vertical height for overpack handling should preclude a situation in which the full overpack were raised sufficiently to permit a corner drop with c.g. vertically over a tangent to the base plate edge.

Effects of a drop of 11 inches with the overpack longitudinal axis horizontal would be enveloped by the effects of the non-mechanistic tipover (in which the c.g. drops about 6.5 feet in elevation). As previously stated, the overpack cannot be handled in a horizontal orientation.

The overpack body and lid include welds and component pieces which are not in the primary load carrying path. This was determined by analysis or inspection for the different members. As the overpack structures are integral weldments (body and lid), these would typically experience some stresses associated with elastic response of the main elements and, thereby, may carry part of the load. This could possibly reduce stresses in the main members. These secondary members are identified as "non-structural" (SAR Bill of Materials BM-1575).

Failure of the "non-structural" members or of their welds would not affect safety of the system. It is assumed that they make no contribution to system integrity in the calculations of stresses of the main load carrying members. The staff concurs in the approach used and in the design of the respective welds of these secondary elements, as prequalified welds under the American Welding Society, AWS D1.1, "Structural Welding Code-Steel." The welds associated with the load carrying path are required to comply with the design, fabrication, and inspection criteria of ASME Code, Section 3, NF, for Class 3 plate and shell supports.

As a result of the above considerations, the staff concludes that the scope of the SAR accidental drop analysis is acceptable.

## **Fire**

The SAR structural analysis of the fire accident (described in SAR Section 11.2.4.2.1) is in Section 3.4.4.2.2. The analysis determines the maximum temperatures that may occur in the overpack structural components, compares these with allowables for accident conditions, and describes mitigating features of the design. All materials are determined to have maximum temperatures below allowables, with the exception that concrete temperature within 1 inch (of a total thickness of 26-3/4 inches) of the outer shell may exceed the temperature limit for accidents.

Staff review has determined that the scope, approach, analysis, and results of the structural analysis for the fire accident are acceptable and comply with 10 CFR 72.122(c). The staff concludes that the fire accident structural analysis is acceptable.

## **Explosion**

The SAR analysis of overpack safety under explosion is in Section 3.4.7.2. The analysis establishes limits for maximum overpressure and dynamic pressure resulting from explosions. The analysis determines steady-state and pulse dynamic pressure limits for incipient overturning and determines the potential for overpack distortion that might impair retrievability of the stored material. The limits determined in the SAR are a steady state pressure differential across the overpack diameter of 5 psi, or a rectangular pressure pulse of 10 psi acting for one second. Structural analysis of stresses associated with the pressure limits show that the stresses would be trivial and that there would be no distortion affecting retrievability of the stored materials.

The staff review has determined that the pressure limit for the overpack of 5 psi from explosions is acceptable. If a credible explosion that would apply more than 5 psi to the outer surface of the overpack is identified for a site, the site must address this issue in its 10 CFR 72.212 evaluation.

### **3.4.3 Supplemental Data**

Additional codes and standards referenced in the HI-STORM 100 Cask System design and fabrication are listed in SAR Section 3.6.1. Three computer codes have been utilized to perform structural and mechanical analysis for the overpack. These codes are ANSYS, DYNA3D, and WORKING MODEL. The ANSYS and DYNA3D Codes are public domain codes with a well-established history of usage in the nuclear industry. WORKING MODEL V3.0/V4.0 is a computer aided engineering tool and is commercially available. The staff found these codes acceptable during its review of the HI-STAR 100 Cask System application.

## **3.5 Structural Analysis of the HI-TRAC Transfer Cask**

### **3.5.1 Lifting Devices**

The HI-TRAC transfer cask trunnions are part of a non-redundant lifting system. Based on the requirements of Section 5.1.6(3) of NUREG-0612, the lifting trunnions are able to support a load of 10 times the actual lifted load without exceeding the material ultimate strength. The lifted load also includes a dynamic load factor of 1.15. Stresses in the trunnions are compared to the allowable strength per NUREG-0612, and stresses in trunnion block are compared with the appropriate allowable strengths from Subsection NF of the ASME Code.

Structural evaluation of the lifting trunnions for the 125-ton HI-TRAC transfer cask and the 100-ton HI-TRAC transfer cask are analyzed in Appendix 3.E and 3. AE of the SAR. Analytical results are presented in Sections 3.4.3.1, 3.4.3.2, 3.4.3.3, and 3.4.3.4 of the SAR. During lifting of the transfer cask from the spent fuel pool, the transfer cask pool lid supports the weight of a loaded MPC plus water. The transfer cask pool lid and transfer lid are analyzed in Appendices 3.AB, 3.AC, and 3.AD of the SAR. During a lifting operation, the transfer cask body supports the weight of a loaded MPC, the transfer lid or pool lid, and the water in the transfer cask and MPC. In either case, the load is transferred to the bottom flange of the transfer cask through the bolts. Appendix 3.AE of the SAR provides the evaluation of the bottom flange of the transfer cask and has demonstrated that the required limits on stress are maintained. Analytical results are presented in Sections 3.4.3.8, 3.4.3.9, and 3.4.3.10 of the SAR. The primary stresses in the top trunnions are less than the ultimate strength of the trunnion material. Local bearing stress, thread shear stress, and weld stress in the top trunnion support block do not exceed the allowable limits. Stresses in the lids and bolts are also shown to be below yield under three times the applied lifted load.

### **3.5.2 Stress Calculations**

The purpose of the stress analyses is to provide assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive materials, unacceptable radiation levels, or impairment of ready retrievability of fuel from the MPC and the MPC from the HI-TRAC transfer



cask. MPC stress analyses performed by the applicant focus on the basket and canister stress distributions due to the design basis load of 45 g, imposed by a horizontal drop of the transfer cask. The HI-TRAC transfer casks and the HI-STAR 100 overpack inner diameters are identical. The analysis of the MPC in the HI-STAR 100 overpack under 60 g for side impact (Docket 72-1008) bounds the analysis of the MPC in the HI-TRAC transfer cask under 45 g.

### **3.5.3 MPC Stress Calculations**

The MPCs used in the HI-STORM 100 Cask System are the same MPC designs evaluated and approved for use in the HI-STAR 100 Cask System. The applicable MPC analyses provided in the HI-STAR 100 Cask System SAR are referenced in the HI-STORM 100 Cask System SAR.

### **3.5.4 Differential Thermal Expansion**

Tables 4.4.15 and 4.5.4 of the SAR provide the temperatures used for the differential thermal expansion analyses of the MPC in the HI-TRAC transfer cask. Evaluation results demonstrated that a physical interference between discrete components of the HI-STORM 100 Cask System will not develop due to differential thermal expansion during any operating condition.

### **3.5.5 Transfer Cask Stress Calculations**

The function of the transfer cask is to provide radiation and structural protection of the MPC during loading, unloading, and transfer operations. In the event of a postulated accident, the transfer cask must not suffer permanent deformation to the extent that the ready retrievability or confinement integrity of the MPC is compromised.

All structural analyses were performed for the 125-ton HI-TRAC transfer cask which bounds the 100-ton HI-TRAC transfer cask. The analyses presented in the SAR demonstrate the ability of components of the transfer cask to perform their structural functions during the transfer mode. Load combinations are given in Table 3.1.5. Appendices 3.AA and 3.AI of the SAR provide the rotation trunnions weld analyses for the 125-ton and 100-ton HI-TRAC transfer casks. Analytical results show that the stresses are within the allowable strengths of ASME Section III, Subsection NF.

Appendix 3.AD of the SAR provides the 125-ton HI-TRAC transfer lid stress analysis when the lid is subject to the acceleration loads of a horizontal drop. It is shown in Appendix 3.AD that the cask body, under a deceleration of 45 g, will not separate from the transfer lid. The stresses meet level D stress limits for NF Class 3 components.

Appendix 3.AG of the SAR provides the stress analysis and evaluations of the connecting welds of the water jacket and the transfer cask. Analytical results show that the stresses are within the allowable limits with adequate safety factor.

Appendix 3.AH of the SAR evaluates the potential for top lid separation under a deceleration of 45 g during a horizontal drop event. It is concluded that the 12 lid bolts, together with the short tongue and groove connection, provide acceptable protection against top lid separation.

### **3.5.6 Tornado Wind and Missile Impact**

The HI-TRAC transfer cask is assessed for the intermediate and large missile penetration. The penetration potential of an 8-inch missile strike on the transfer cask is evaluated in Appendix 3.H of the SAR. Two locations are evaluated: the lead backed outer shell of the transfer cask and the flat transfer lid consisting of multiple steel plates with a layer of lead backing. In each case, it is shown that there is no penetration that would lead to a radiological release.

The applicant evaluated the effect of a large tornado missile strike on the side of a loaded transfer cask. The applicant used the finite element code DYNA3D. The finite element model includes the loaded MPC, the transfer cask inner and outer shells, the water jacket, the lead shielding, and the transfer cask lid. The large tornado missile was simulated by an impact force-time pulse applied on an area representing the frontal area of an 1800-kg vehicle. The results are presented in Appendix 3.AN of the SAR. The results demonstrate that the retrievability of the MPC in the wake of a large tornado missile strike is not adversely affected since the inner shell of the transfer cask does not experience any plastic deformation. The maximum primary stress intensity is below the ASME Code Level D allowable limit for NF, Class 3 structures.

### **3.5.7 Drop Events**

The transfer cask may be transported horizontally with the transfer lid in place. The applicant performed analyses to demonstrate that under a postulated carry height of 42 inches above the ground the design basis load of 45 g is not exceeded. The analyses were performed using two different models. A simplified model of the drop event was performed using a two-dimensional computer model that assumed that the transfer cask and MPC acted as a single rigid body. A second model of the drop event used the DYNA3D finite element computer code. For this analysis, the transfer cask and MPC were modeled as individual bodies. The results are presented in Appendix 3.AN and show the maximum interface force at the top lid and MPC interface. These results are used as input to the analysis in Appendix 3.AH which demonstrates that the top lid of the HI-TRAC transfer casks will contain the MPC during and after a handling accident.

## **3.6 Evaluation Findings**

Based on the information provided in SAR, by reference, the supporting documentation, and the staff's own confirmatory analyses, the staff concludes that the HI-STORM 100 Cask System meets the acceptance criteria specified in NUREG-1536.

- F3.1** The SAR adequately describes all structures, systems, and components that are important to safety, providing drawings and text in sufficient detail to allow evaluation of their structural effectiveness.
- F3.2** The applicant has met the requirements of 10 CFR 72.24, "Contents of Application: Technical Information," with regard to information pertinent to structural evaluation.

- F3.3** The applicant has met the requirements of 10 CFR 72.26, "Contents of Application," and 10 CFR 72.44(c), "License Conditions." All the supporting calculations which document the structural integrity of the MPC, HI-STORM 100 overpack, and the HI-TRAC transfer cask are adequate.
- F3.4** 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 load are determined by analysis. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and found to be within the limits of applicable codes, standards, and specifications.
- F3.5** 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 of the dry cask storage system includes structural margins of safety for those structures, systems, and components 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.
- F3.6** The applicant has met the requirements of 10 CFR 72.236(l), "Specific Requirements for Spent Fuel Storage Cask Approval." The design analyses 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.
- F3.7** 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:
- F3.7.1** design, fabrication, erection, and testing to acceptable quality standards;
  - F3.7.2** adequate structural protection against environmental conditions and natural phenomena, fires, and explosions;
  - F3.7.3** appropriate inspection, maintenance, and testing;
  - F3.7.4** adequate accessibility in emergencies;
  - F3.7.5** a confinement barrier that acceptably protects the spent fuel cladding during storage;
  - F3.7.6** structures that are compatible with appropriate monitoring systems; and
  - F3.7.7** structures that are compatible with ready retrievability of spent fuel.

**F3.8** 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:

**F3.8.1** redundant sealing of confinement systems;

**F3.8.2** adequate heat removal without active cooling systems;

**F3.8.3** storage of the spent fuel for a minimum of 20 years;

**F3.8.4** compatibility with wet or dry spent fuel loading and unloading facilities;

**F3.8.5** acceptable ease of decontamination;

**F3.8.6** inspections for defects that might reduce confinement effectiveness;

**F3.8.7** conspicuous and durable marking; and

**F3.8.8** compatibility with removal of the stored fuel from the site, transportation, and ultimate disposition by the U.S. Department of Energy.

### **3.9 References**

1. Beer and Johnston, "Vector Mechanics for Engineers: Static and Dynamic," McGraw-Hill Book Co., 1962.

## 4.0 THERMAL EVALUATION

The thermal review ensures that the cask component and fuel material temperatures of the HI-STORM 100 Cask 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 analytic techniques and/or testing methods.

### 4.1 Spent Fuel Cladding

The thermal design criteria for preventing fuel cladding degradation are presented in Section 4.3 of the SAR. The applicant used the method developed by the Commercial Spent Fuel Management (CSFM) at Pacific Northwest National Laboratory<sup>1</sup> to establish long-term dry storage temperatures limits for zircaloy clad spent nuclear fuel. The applicant assumed bounding values of zircaloy fuel rod cladding oxide thickness that correspond to a maximum allowable HI-STORM PWR and BWR fuel burnup of 45,000 MWD/MTU.

The applicant calculated the peak rod internal gas temperature based on its analysis of the MPC internal temperature distribution. The analyses assumed bounding fuel pin pressures at end-of-life for the peak power rod at in-core operating condition, based on PNL-6189 report (CSFM), dated May 1987. Since publication of that report, new analytic techniques and data have shown that the peak fuel pin pressure is significantly lower than the reactor operating pressure (for example, the peak pressure for the peak power rod would be about 1600 psia for PWRs and about 900 psia for BWRs; however, the applicant used 2416 psia for the PWR fuel rod pressure and 1094 psia for the BWR fuel rod pressure). Use of the PNL-6189 operating pressures, as implemented by Holtec, alleviates the need for calculating the fission product gases released by the fuel pellets during plant operation and significantly bounds the calculated internal fuel rod pressure at dry storage conditions. In addition, the applicant's analyses assumed bounding design basis decay heat generated by the fuel. The rod internal gas temperature for the peak rod was calculated by averaging the pellet outer edge temperature and rod gas plenum outer edge temperature axial distribution. The active fuel region temperatures were averaged over 24 axial segments for the pellet stack region and two axial segments for the plenum region. The trapezoidal rule was used to calculate an average temperature for each axial segment. The bulk average rod gas temperature was then calculated by the applicant over the total plenum and gap volumes. The peak fuel rod internal pressure recommended in the CSFM method was then adjusted, using average rod gas temperature and the ideal gas law to determine the rod internal gas pressure. The rod internal gas pressure was then used to calculate the cladding hoop stress in the CSFM method. The staff reviewed the Holtec calculations of the peak rod pressures and found them acceptable.

The applicant identified the bounding PWR and BWR fuel design that provided the highest calculated zircaloy cladding hoop stress. This was determined by evaluating the design characteristics of each specific fuel design that impact the CSFM method results. The long-term

zircaloy cladding dry storage temperature limits, calculated by the applicant and confirmed by the staff, are presented in Table 4-6 below.

The applicant evaluated the long-term dry storage temperature limit for stainless steel clad spent nuclear fuel based on in-reactor irradiation and wet pool and dry storage experience coupled with an assessment of failure mechanisms for irradiated stainless steel cladding, as presented in an Electric Power Research Institute (EPRI) report<sup>2</sup>. This report concludes that long-term stainless steel clad spent fuel temperatures of 806°F (430°C) in dry storage will not result in cladding failure for storage times of up to 50 years. The applicant also noted that the stainless steel clad spent nuclear fuel has a longer cooling time and lower decay heat than the design basis zircaloy clad spent fuel. Thus, based on its lower decay heat and a higher long-term dry storage cladding temperature limit, the applicant concluded that the zircaloy clad fuel temperature limits, listed on Table 4-6, are bounding for fuels with stainless steel clad. The staff finds the applicant's evaluation acceptable.

The applicant selected a short-term fuel cladding temperature limit of 1058°F (570°C) for all zircaloy and stainless steel clad spent nuclear fuel. This limit is consistent with the criteria listed in NUREG-1536. Test data for a time period of 740 to 1,000 hours cited by the applicant also corroborates this value of short-term cladding temperature limit.

## **4.2 Cask System Thermal Design**

The cask system thermal design for the HI-STORM 100 overpack containing a loaded MPC is presented in Sections 1.2, 2.1.6, and 4 of the SAR.

### **4.2.1 Design Criteria**

The applicant addressed the HI-STORM 100 Cask System design criteria developed to meet 10 CFR Part 72 requirements for 20 years of storage of spent nuclear fuel. These design criteria encompass normal, off-normal, and postulated accident conditions.

The thermal design criteria for the HI-STORM 100 overpack with the loaded MPC are given in Section 2.2.1.5 of the SAR. Table 4-1 lists the design temperature limits for the concrete and steel components of the HI-STORM spent fuel storage system.

**Table 4-1**  
**HI-STORM 100 Component Normal, Off-normal, and Accident Temperature Limits**

<b>HI-STORM 100 Component</b>	<b>Normal Condition Design Temperature Limit, °F</b>	<b>Off-Normal and Accident Condition Temperature Limit, °F</b>
Overpack Outer Steel Shell	350	600
Overpack Concrete	200	350
Overpack Inner Steel Shell	350	400
Overpack Lid Top Steel Plate	350	550
Remainder of Overpack Steel Structure	350	400

#### **4.2.2 Design Features**

The thermal design features of the HI-STORM 100 overpack with the loaded MPC consist of:

- (a) internal labyrinthine air flow passage with four air inlets and four air outlets;
2. carbon steel inner and outer cylindrical shells;
3. carbon steel baseplate;
4. carbon steel top plate;
5. concrete encased within steel cylindrical shells; and
6. helium backfill gas in the basket.

The helium backfill gas used in the MPC provides superior heat conduction from the fuel to the basket wall, when compared to other inert gases, as well as an inert atmosphere. The effectiveness of the helium gas was demonstrated on full scale casks at the Idaho National Engineering and Environmental Laboratory. The internal air passage of the overpack (formed by the MPC outer surface and the baseplate and inner cylindrical shell) provides the primary means of MPC decay heat removal by natural convection cooling. The cooling is passive in that it uses differential air density buoyancy to drive the air flow past the MPC outer wall. The four air inlets and four air outlets provide a redundant and geometrically separate means of supplying air and discharging air from the internal passage. Along with natural convection, radiation and conduction heat transfer occurs from the MPC outer surface across the air passage to the inner liner of the HI-STORM 100 overpack.

The concrete overpack is encased in carbon steel cylindrical shells. The heat that is not removed by the air passage is primarily conducted through the steel cylindrical shells. The concrete mass also represents a large thermal inertia (heat capacity) that, for some accident scenarios such as blocked air inlets and fire, introduces a significant time delay before material temperature limits are reached in the HI-STORM 100 Cask System.

The staff verified that all methods of heat transfer internal and external to the HI-STORM 100 Cask System are passive. Sections 1.5 and 4 of the SAR provide information relative to materials of construction, general arrangement, dimensions of principal structures, and description of all structures, systems, and components important to safety, in sufficient detail to support a finding that the design will satisfy the design bases with an adequate margin, as required by 10 CFR 72.24(c)(3).

### 4.3 Thermal Load Specification/Ambient Temperature

The thermal load specifications for an overpack loaded with the MPC are given in Sections 2.2 and 4.4 of the SAR. Table 4-6 lists the maximum allowable decay heat load that can be stored in the MPC-24 and in the MPC-68 as a function of time following removal of the fuel assemblies from the reactor core (e.g., fuel decay time). These limits on decay heat loads are based on the calculated maximum cladding temperature limits for normal conditions. Solar thermal loads, as listed in 10 CFR 71.71, were also incorporated into the analysis, as appropriate. The thermal loads apply to normal, off-normal, and accident conditions except for the fire accident. During a postulated fire accident, the thermal loads on the overpack include heat generated from the enveloping fire that is added to the MPC decay heat.

The ambient temperatures assumed as design bounding values for the thermal evaluation of the HI-STORM 100 system are listed in Table 4-2. The 80 °F normal annual average temperature assumed in the evaluation exceeds the highest annual average temperature recorded in the continental USA. The staff validated that number through data collected by NOAA and found it acceptable.

**Table 4-2**  
**HI-STORM 100 System Design Ambient Temperatures**

Condition	Temperature (°F)
Normal Annual Average	80
Normal Soil Annual Average	77
Off-Normal Maximum 3-Day Average	100
Off-Normal Minimum 3-Day Average	-40
Accident Maximum 3-Day Average	125

The staff has reviewed and confirmed the design basis decay heat load for the specific fuel designs. The staff has also verified that the bounding decay heats have been properly calculated.



## 4.4 Model Specification

The thermal model specification is presented in Section 4.4.1 of the SAR and discussed below.

### 4.4.1 Configuration

The thermal-fluid dynamic analysis was performed using the FLUENT<sup>3</sup> computer code that models the HI-STORM 100 system in three-dimensional space. The model includes the MPC volume, the HI-STORM steel encased concrete overpack, and a surrounding cylindrical tank region. The overall model of the HI-STORM 100 system consists of 3,933 asymmetric elements. The MPC is modeled as solid with 1,188 axisymmetric elements and temperature dependent thermal conductivity. The internal air passages, air inlets, and air outlets were simulated by hydraulically equivalent resistance porous media. The external surface of the overpack is enclosed in a cylindrical tank region which models the effect of adjacent casks on an ISFSI pad. The model configuration includes a conduction heat transfer path from the overpack through the ISFSI concrete pad and to the soil below.

The equivalent thermal conductivity of regions within the MPC was calculated using the ANSYS computer code. ANSYS was also used to model the thermal transient response of the HI-STORM 100 overpack to the postulated fire accident. In addition, the ANSYS fuel assembly effective thermal conductivity analysis was used to benchmark the FLUENT computer code analysis of fuel assembly effective thermal conductivity.

The staff reviewed the analytic assumptions used by the applicant in modeling the components of the MPC, the HI-STORM overpack, and the HI-TRAC transfer cask. The assumptions maximized the resistance for heat transfer through the MPC, overpack, and transfer cask. The applicant used bounding assumptions in its analyses. Some of the more significant conservatisms assumed in this analysis included: (1) neglect of convective cooling of the helium gas inside the MPC, thereby maximizing the calculated peak clad temperature; (2) for the maximum average ambient temperature, the applicant assumed 125 °F ambient temperature with maximum solar heat input (insolation) for a period of three days<sup>\*</sup>; (3) maximizing the thermal resistance between two materials (e.g., imposing a 2 mil air gap between Boral and its casing, using the properties of air rather than helium where only microscopic gaps would exist in the MPC, imposing a uniform gap of air between the lead and steel in the HI-TRAC transfer cask); and (4) assuming all of the fuel assemblies are at the maximum design thermal limit.

Other conservatisms are addressed in the SAR, and were reviewed by the staff and found acceptable.

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\* Note: It takes three days for the HI-STORM System to reach steady-state thermal conditions. This bounding assumption exceeds the highest short-term NOAA recorded continental USA temperature of 124 °F. The NOAA data indicates that the temperature variation for that hot day was 40 °F. In other words, the high ambient temperature for the day was 124 °F and the low was 84 °F. Assuming a 125 °F ambient condition for three days with a high solar energy input (sun never sets assumption) clearly bounds the consequences of the postulated event.

#### **4.4.2 Material Properties**

The material properties used in the thermal analysis are presented in Section 4.2 of the SAR. This section identifies the temperature dependent thermal conductivity, emissivity, density, heat capacity, and viscosity (for gases) for all the materials used in the HI-STORM 100 system. The three material properties used in the overpack are air (present within the internal air passage), carbon steel, and concrete. The MPC is composed of helium, stainless steel, aluminum alloy 1100, zircaloy, uranium dioxide, and Boral. When a range of possible numerical values for these material properties was available from suitable references, the applicant selected the value(s) that resulted in the most bounding thermal calculation results.

#### **4.4.3 Boundary Conditions**

The boundary conditions in the thermal analysis are specified in Section 4.4 of the SAR. The off-normal and accident high ambient temperature cases include a solar insolation boundary condition on the top and side surfaces of the overpack.

The cylindrical tank region around the overpack in the FLUENT model configuration provides an external boundary condition for the overpack that simulates the presence of adjacent casks on an ISFSI pad. This region models the hydraulic resistance from nearby casks which effects flow to the air inlets. In addition, this tank region reflects back all heat which is radiated from the outside surface of the overpack, thereby, simulating the effect of adjacent casks radiating heat back to an interior cask in an ISFSI array. This boundary condition conservatively precludes any heat loss from the overpack surface by radiation heat transfer.

The soil below the overpack is assumed to be at a constant temperature commensurate with the high numerical value presented in Section 4.3 of this SER. This high constant soil temperature results in a bounding low heat loss from the HI-STORM 100 system to the earth below it.

The applicant uses bounding thermal boundary conditions for the postulated fire accident analysis in accordance with NUREG-1536. The fire is postulated to have a duration of 3.6 minutes based on 50 gallons of diesel fuel, the maximum allowed by Appendix B of the Certificate of Compliance at an ISFSI site. The applicant assumed bounding surface convection heat transfer coefficients, fire dimensions, flame temperature, surface emissivity, and overpack ventilation passage air temperature both during and after the fire. The staff concludes that the total energy available from burning 50 gallons of combustible fuel would have insignificant impact on the HI-STORM overpack and HI-TRAC transfer cask, given the large heat capacity of those designs.

### **4.5 Thermal Analysis**

#### **4.5.1 Computer Programs**

The FLUENT and ANSYS<sup>4</sup> computer programs are used in thermal analyses of the HI-STORM 100 system. FLUENT is a finite volume computational fluid dynamics computer code which is capable of both steady state and transient analyses. The applicant has previously used FLUENT

in the HI-STAR 100 Cask System application that has been reviewed and accepted by the staff. ANSYS is a three-dimensional finite element heat transfer and stress computer code which is also capable of both steady state and transient analyses. ANSYS is cited in NUREG-1536 as an acceptable computer code for thermal evaluation of dry spent fuel storage cask systems.

#### **4.5.2 Temperature Calculations**

The results of temperature calculations for normal, off-normal, and accident conditions are presented in Sections 4.4, 11.1, and 11.2 of the SAR for both the PWR MPC-24 and the BWR MPC-68. The normal and off-normal temperature calculations were performed at the two different assumed ambient temperatures which were discussed in Section 4.3 of this SER. The accident temperature calculations were performed for an extreme ambient temperature, as discussed in Section 4.3 of this SER, and for a hypothetical maximum fire enveloping a loaded HI-STORM overpack. All cases assumed the maximum design basket-specific decay heat load. Key calculated or assumed HI-STORM 100 system component temperatures under normal, off-normal, and accident conditions for both MPC designs are in Table 4-3.

All the calculated component material temperatures for normal, off-normal, and accident conditions remain below their respective material temperature limits with the exception of the outer 1-inch layer of the concrete overpack which exceeds the concrete short term temperature limit for the fire accident scenario. This transient thermal response of a small fraction of the concrete is allowable for a fire condition as discussed in NUREG-1536. It should be noted that the concrete in the HI-STORM overpack is not a structural component. Therefore, exceeding the temperature limit does not present a safety issue. The staff reviewed the applicant's bounding analyses and found the consequences superficial (see Figure 11.2.2 in the SAR) and acceptable.

The applicant calculated the effect of the numerical value of zircaloy fuel cladding surface thermal emissivity on calculated maximum cladding temperature. The applicant assumed a value of 0.8 for this emissivity, which is based on several references cited in the SAR. To assess the impact of a bounding low emissivity of 0.4, the applicant calculated the maximum cladding temperature with this lower value. The resulting peak cladding temperature increased by approximately 5°C. This small change in calculated maximum cladding temperature indicates that the selected zircaloy emissivity is adequate for this analysis. The staff notes that the emissivity of stainless steel fuel cladding may be lower than that of zircaloy because of its higher corrosion resistance. However, any increase in calculated maximum cladding temperature for stainless steel clad spent fuel due to lower emissivity would be small in comparison to the decrease in cladding temperature due to the fact that the design basis stainless steel clad fuel has a lower burnup, longer cooling time, and therefore lower decay heat than the design basis zircaloy clad fuel.

The applicant performed an analysis of the off-normal condition of partial blockage of air inlets and an accident analysis assuming full blockage of all air inlets. Both cases were analyzed with an assumed 80°F ambient air temperature and the same models and methodology that were used for the normal condition thermal analyses. For the bounding off-normal, partial air inlet blockage scenario of three of the four air inlets completely blocked, the calculated maximum component temperatures are presented in Table 4-4. The applicant also analyzed the thermal response of a complete blockage of all air inlets to identify the time when a component material

temperature limit is exceeded. Results for this accident up to the time when the concrete short-term limit is reached (i.e., about 33 hours) are also presented in Table 4-4.

**Table 4-3**  
**Calculated Maximum HI-STORM 100 System Component Temperatures**

<b>Component</b>	<b>Normal MPC-24 °F</b>	<b>Normal MPC-68 °F</b>	<b>Extreme (125°F) Ambient °F</b>	<b>Maximum Fire °F</b>
Fuel Cladding	692	742	774 (PWR) 790 (BWR)	730 (PWR) 746 (BWR)
MPC Basket	657	722	770	690 (PWR) 726 (BWR)
Basket Periphery	417	366	NR	NR
MPC Outer Shell	295	301	352	NR
Overpack Inner Shell	166	171	217	300
Average Concrete	149	151	NR	184
Overpack Outer Shell	131	131	176	570
Overpack Bottom Plate (Max.)	183	183	NR	NR
Overpack Lid Top Plate	157	159	NR	NR
Air Inlet	80	80	125	300*
Air Outlet	179	185	231	300*

\* Analytical assumption; NR=Not reported or not required for the evaluation of these conditions.

**Table 4-4**  
**Calculated Maximum HI-STORM 100 Component Temperatures**  
**for Air Inlet Blockage Accidents**

<b>Component</b>	<b>Partial Blockage - 3 Inlet Ducts Blocked (°F)</b>	<b>Complete Blockage of all Air Inlets, at 33 Hours (°F)</b>
Fuel Cladding	778	846
Overpack Inner Shell (Maximum Concrete)	232	348
Overpack Outer Shell	149	145

#### **4.5.3 Pressure Analysis**

The applicant presented HI-STORM 100 system MPC calculated pressures for normal, off-normal, and accident conditions in Section 4.4.4 of the SAR. The maximum internal pressure was calculated using the free volume of the MPC, ideal gas law, and accounted for the backfill helium gas along with a fraction of the stored fuel helium fill gas and fission product gas. The normal, off-normal, and accident conditions were differentiated by the assumption of the fraction of stored spent fuel which contributed fill gas and fission gas to the MPC. These fractions were 1%, 10%, and 100%, respectively for the normal, off-normal, and accident cases, which are in agreement with NUREG-1536. In each case, 100% of the fuel rod fill gas and 30% of all fission product gases were assumed to be released to the MPC interior volume. The resulting MPC-24 and MPC-68 pressures are summarized in Table 4-5.

**Table 4-5**  
**Calculated Maximum MPC Pressures for Normal, Off-Normal, and Accident Conditions**

<b>Condition</b>	<b>MPC-24 Pressure (psig)</b>	<b>MPC-68 Pressure (psig)</b>
Normal (1% fuel failure)	59.3	57.6
Off-Normal (10% fuel failure)	62.8	60.3
Accident (100% fuel failure)	97.6	87.4

The calculated maximum pressure for both MPC designs and all conditions remains below its appropriate design pressure.

#### 4.5.4 Confirmatory Analysis

The staff's review of the HI-STORM application encompassed the inputs, assumptions, methodology, and results of the applicant's temperature and pressure analyses which were submitted in support of the SAR, including the MPC, the transfer cask, and the overpack. All the assumptions were found to be in compliance with NUREG-1536, Section 4.V.5.(c). Input parameters are consistent with design values for the MPC, the HI-STORM overpack, and the HI-TRAC transfer cask. The staff finds that the applicant selected suitably bounding and appropriate boundary conditions for normal, off-normal, and accident conditions. In addition, the staff reviewed the results of a validation of the computer code and analytic method used by Holtec in the HI-STORM analyses; this validation compared the code results with test data performed by DOE and the Energy Power Research Institute (EPRI) on a full scale spent fuel cask instrumented and tested at the Idaho National Engineering and Environmental Laboratory. The results of Holtec's analytic method showed good agreement with the DOE/EPRI test data. Based on the staff's review, these validation results, and the FLUENT code's recognized value as an analytic tool in conducting thermal analyses, the staff finds that the applicant's analytic methods for calculating the thermal responses of the MPC, overpack, and transfer cask are acceptable. In addition, although the HI-STAR analyses and staff evaluation hereof are not relied upon in the thermal evaluation in this SER, the staff notes that previous staff evaluation of the applicant's HI-STAR 100 SAR's FLUENT computer code results, using the ANSYS finite element computer code, confirmed the temperature calculation results shown by Holtec's analysis, thus confirming Holtec's ability to utilize the FLUENT code correctly.

The staff also reviewed the form loss and friction loss coefficients used by the applicant to simulate the hydraulic characteristics of the internal air passage. The applicant's form loss coefficients were found to be suitably bounding and applicable to the specific geometry of the HI-STORM 100 air passages.

The staff evaluated and accepted the applicant's selected heat transfer coefficients. The temperature and pressure results were found to be correctly calculated using the identified inputs, assumptions, and methodology.

The staff evaluated the applicant's peak fuel rod internal gas average temperature calculation, used to determine the long-term dry storage temperature limits for zircaloy clad fuel rods. To calculate the maximum fuel rod temperature limit for long-term storage, the applicant volume-averaged the temperature of the gases within the gap and plenum of the limiting fuel rod assuming bounding fuel pin pressures, as identified in the PNL-6189 CSFM report. Using the derived pressure, a corresponding cladding stress was calculated and a fuel age dependent temperature limit was identified. The CSFM method has been used and accepted by the staff in previous ISFSI license applications. The staff performed confirmatory calculations for the dry storage temperature limits. Table 4-6 lists the permissible Fuel Temperature and Allowable Heat Loads for the MPC-24 and MPC-68.

**Table 4-6**  
**Maximum Allowable MPC Decay Heat Limits and Heat Load As**  
**a Function of Fuel Decay Time**

<b>Fuel Decay Time (years)</b>	<b>PWR MPC-24 Fuel Temperature Limit (°C)</b>	<b>PWR Maximum MPC-24 Allowable Decay Heat Load (kW)</b>	<b>BWR MPC-68 Fuel Temperature Limit (°C)</b>	<b>BWR Maximum MPC-68 Allowable Decay Heat Load (kW)</b>
5	366.6 (692 °F)	20.88	394.4 (742 °F)	21.52
6	358.6 (677 °F)	20.17	379.2 (714 °F)	20.31
7	335.6 (636 °F)	18.18	354.8 (671 °F)	18.41
10	330.2 (626 °F)	17.72	348.8 (660 °F)	17.95
15	323.8 (615 °F)	17.17	342.1 (648 °F)	17.45

The staff concludes that the MPC decay heat limits in Table 4-6 assure that all material temperature limits are not exceeded and no gross ruptures would occur in a dry helium storage environment for the license period of 20 years.

#### **4.6 HI-TRAC Thermal Review**

The HI-TRAC transfer cask is a short-term container used to load and unload the HI-STORM concrete storage overpack. The HI-TRAC transfer cask is used for various plant operations, such as, normal onsite transport of spent nuclear fuel, MPC cavity vacuum drying, post-loading wet transfer operations, and MPC cooldown and reflood required for unloading spent nuclear fuel. Holtec designed the HI-TRAC transfer cask to ensure that fuel integrity is maintained through adequate rejection of decay heat from the spent nuclear fuel. Heat generated from the MPC outer surface is transmitted across an air gap to an inner shell steel liner, through a lead-to-steel air gap, through a lead shield, through an outer shell steel liner, through a water jacket, through the enclosure shell of the water jacket, and to the atmosphere. Where uncertainties exist, bounding assumptions are made. For example, a maximum gap distance between the MPC and the HI-TRAC inner shell is assumed for degrading the heat transfer characteristics of the design. Thermal expansion that could minimize the gap is not credited. The water jacket, used for neutron shielding, surrounds the cylindrical steel wall. The water jacket is composed of carbon steel channels with welded connecting enclosure plates. An artificial gap was assumed between the steel inner shell and the lead shielding material to maximize the thermal resistance of the HI-TRAC transfer cask.

In the vertical position, the bottom face of the HI-TRAC transfer cask is in contact with the supporting surface. Heat transfer from the bottom face is not credited. The remaining outer surfaces are insulated using 10 CFR Part 71 insulation criteria, averaged on a 24-hour basis. The staff reviewed the assumptions used by the applicant in modeling the HI-TRAC transfer cask and found them acceptable.

The thermal characteristic of the HI-TRAC transfer cask is documented in Section 4 of the SAR. The use of the FLUENT computer code to evaluate the temperature distributions for onsite transport conditions is acceptable. Use of the FLUENT code on spent fuel cask designs was validated with comparison to data from a full scale storage cask loaded with 24 canisters of consolidated PWR spent fuel assemblies. The thermal heat generated by the spent fuel was 23 kW. The tests were performed by the Pacific Northwest National Laboratory and the Idaho National Engineering and Environmental Laboratory. The results from the FLUENT thermal code showed good agreement with the data.

In Appendix B of the Certificate of Compliance, the minimum ambient temperature for onsite transport operations is limited to 0°F. HI-TRAC was analyzed under 0°F conditions. Further, procedures were developed that ensure safe operations (e.g., requiring the addition of antifreeze).

#### **4.6.1 Loading of the MPC with Spent Nuclear Fuel**

HI-TRAC transfers the MPC to the spent fuel pool for loading. Once loaded, the HI-TRAC system removes the MPC from the pool for vacuum drying and filling with helium fill gas. From the time the HI-TRAC is removed from the spent fuel pool, plant procedures require that the vacuum drying operation be initiated prior to the water temperature in the MPC reaching saturation. An adiabatic temperature rise calculation is performed to determine the maximum time limit before the water in the MPC reaches saturation temperature during wet transfer operation. The maximum allowable time for wet transfer is a function of initial temperature of the water inside the MPC. Table 4-7 lists the allowable time durations for wet transfer operations under design load conditions.



**Table 4-7**  
**HI-TRAC Allowable Time Duration for Wet Transfer Operations**  
**Under Design Load Conditions**

<b>Initial Water Temperature (°F) for Wet Transfer of SNF</b>	<b>Maximum Allowable Time Duration for Wet Transfer Operation (hr)</b>
115	35.2
120	33.4
125	31.5
130	29.7
135	27.9
140	26.1
145	24.3
150	22.5

In the event that the maximum allowable time identified above is found to be insufficient to complete all wet transfer operations, forced water circulation will be initiated and maintained to remove the decay heat from the MPC cavity.

The staff reviewed the applicant's analytic assumptions and results for loading of the MPC and finds the wet transfer evaluations acceptable.

#### **4.6.2 Vacuum Drying Operation**

Long term storage of spent nuclear fuel is done with the MPC filled with inert helium gas. HI-TRAC removes the MPC from the refueling pool for decontamination and vacuum drying. The vacuum drying of the MPC is performed with the annular gap between the MPC and the HI-TRAC filled with water. The water in the gap between the MPC and HI-TRAC will maintain the MPC shell temperature around the saturation temperature of the water in the annular gap. Using the FLUENT code, a thermal analysis of the MPC during vacuum drying was performed to assess the peak clad temperature at design basis heat loads. Table 4-8 lists the results of the thermal analysis under vacuum conditions.

**Table 4-8**  
**HI-TRAC Thermal Analysis Under Vacuum Conditions**

<b>Component Under Vacuum Conditions</b>	<b>MPC-24 (°F)</b>	<b>MPC-68 (°F)</b>
Flue Clad	827	822
MPC Basket	759	786
MPC Basket Periphery	442	315
MPC Outer Shell Surface	232	232

The calculated temperatures are below the maximum short-term limits. The staff reviewed the methods and assumptions used by the applicant in support of its vacuum drying procedures and finds these results acceptable.

#### **4.6.3 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation**

Holtec evaluated the consequences of cask cooldown and reflood procedures to support fuel unloading from a dry condition. The procedures for cask cooldown and reflooding the MPC were developed to ensure that uncontrolled thermal stressing and failure in structural members would not occur and that injection of water would not result in significant steam formation that leads to significant over-pressurization of the confinement boundary. This is accomplished through gradually cooling of the helium by a forced flow helium circulation system (e.g., Cool-Down System). The Cool-Down System uses an external water chiller as the heat sink. Once the Cool-Down System cools the MPC internals to less than 200 °F, water can be injected into the MPC without concern of significant boiling and excessive thermal stress.

The Technical Specifications for cask cooldown prevent filling the MPC with water if the helium temperature exceeds 200 °F. In the event that the Cool-Down System fails to reduce the helium temperature to below 200 °F, LCO 3.1.3 was established to ensure that the MPC and the overpack remain in a safe condition. As the operators attempt to restore the gas temperature to within the 200 °F temperature limit, the operators must also ensure proper cooling of the MPC. Should the overpack be placed in a relatively open area, such as an unobstructed refueling floor, no additional actions are necessary since adequate cooling is maintained by ambient conditions. However, if the overpack is located in a structure such as a decontamination pit or fuel vault, additional actions may be necessary depending on the heat load of the stored fuel. Acceptable actions include, removal of the overpack from the pit or vault and placing it in an open area, such as a refueling floor with a reasonable amount of clearance around the cask and not near a significant source of heat, or by supplying nominally 1000 SCFM of ambient (or cooler) air to the space inside the vault at the bottom of the overpack. These measures ensure that the fuel cladding remains below the short term temperature limit. The staff reviewed the applicant's analytic methods and assumptions used in the unloading operations and finds them acceptable.

#### 4.6.4 Maximum Temperatures Under Onsite Transport Conditions

Holtec analyzed the maximum temperatures under onsite transport conditions for the HI-TRAC design. A bounding steady-state analysis of the HI-TRAC transfer cask was performed using design-basis insolation levels. Table 4-9 summarizes the calculated maximum temperatures for the HI-TRAC transfer cask and MPC.

**Table 4-9**  
**Calculated Maximum Temperatures for the HI-TRAC Transfer Cask and MPC**

Component	Temperature (°F)
Fuel Clad	902
MPC Basket	884
Basket Periphery	527
MPC Outer Shell Surface	459
HI-TRAC Inner Surface	323
Water Jacket Inner Surface	315
Enclosure Shell Outer Surface	223
Water Jacket Bulk Water	269
Axial Neutron Shield	175

The staff reviewed the applicant's analytic methods and assumptions used in support of onsite transport and finds the results of these analyses acceptable.

#### 4.6.5 Maximum Internal Pressure

Following fuel loading and vacuum drying, but prior to installing the MPC closure ring, the MPC is initially filled with helium. During handling in the HI-TRAC transfer cask, the gas temperature within the MPC rises to its maximum operating temperature. The gas pressure inside the MPC will increase accordingly. The maximum MPC internal pressure was calculated for normal onsite transport conditions, as well as off-normal conditions that assume 1% and 10% failed fuel rods (in accordance with NUREG-1536). The calculated peak pressures are listed in Table 4-10. All pressures were within the design limit. The staff reviewed the applicant's analytic methods and assumptions used to evaluate the internal pressure of the MPC and finds the results acceptable.

**Table 4-10**  
**HI-TRAC Pressure Calculations and Associated Design Pressures**

Condition	Calculated Pressure (psig)	Design Pressure (psia)
MPC-24: Initial Backfill (at 70°F) Normal Condition With 1% Rods Ruptured With 10% Rods Ruptured	28.3 66.6 67.0 70.0	100
MPC-68: Initial Backfill (at 70 °F) Normal Condition With 1% Rods Ruptured With 10% Rods Ruptured	28.5 67.0 67.3 70.8	100

#### 4.7 Evaluation Findings

10 CFR Part 72 requires an analysis and evaluation of the dry cask storage system thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years. This section reviewed the thermal design and performance of the long-term storage overpack (HI-STORM 100) and the associated spent fuel transfer cask (HI-TRAC) used to load and unload the dry cask storage system and for various plant operations, such as onsite transport of SNF, including loading and unloading operations of SNF in the MPC. The staff concludes that the HI-STORM overpack and HI-TRAC transfer cask designs fulfill the following acceptance criteria:

1. Fuel cladding temperature at the beginning of the dry cask storage is below the anticipated damage-threshold temperatures for normal conditions.
2. Fuel cladding temperatures (zircaloy) are maintained below 570 °C (1058 °F) for short-term accident conditions, short-term off-normal conditions, and fuel transfer operations (e.g., vacuum drying of the cask or dry transfer).
3. The maximum internal pressure of the cask remains within the design pressures for normal, off-normal, and accident conditions assuming rupture of 1%, 10%, and 100% of the fuel rods, respectively. Assumptions for pressure calculations include release of 100% of the fill gas and 30% of the significant radioactive gases in the fuel rods.
4. Cask and fuel materials are maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions in order to enable components to perform their intended safety functions.
5. For each fuel type proposed for storage, the dry cask storage system provides reasonable assurance that the degradation will not lead to gross ruptures or the fuel must be otherwise

confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage.

6. Fuel cladding degradation resulting from creep cavitation are limited to 15 percent of the original cross-sectional area during dry storage.
7. The cask system is passively cooled.
8. The thermal performance of the cask is within the allowable design criteria specified in Section 2 (e.g., materials, decay heat specifications) and Section 3 (e.g., thermal stress analysis) of the SAR for normal, off-normal, and accident conditions.

The following summarizes the staff's finding on the HI-STORM 100 Cask System:

- F4.1** Structures, systems, and components important to safety are described in sufficient detail in Sections 1.2 and 2.3 of the SAR to enable an evaluation of their thermal effectiveness. Structures, systems, and components important to safety remain within their operating temperature ranges.
- F4.2** The HI-STORM 100 overpack with the loaded MPC-24 or MPC-68 is designed with a heat-removal capability that is verifiable and reliable, consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.
- F4.3** The staff finds, in accordance with 10 CFR 72.122(h), that the spent fuel cladding is protected against degradation leading to gross ruptures by maintaining the cladding temperature for zircaloy and stainless steel clad below the temperature limits listed in Table 4-6 in a helium gas environment. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.
- F4.4** The staff concludes that the thermal design in the SAR 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 HI-STORM 100 overpack with the loaded MPC-24 or MPC-68 cask will allow safe handling and storage of spent fuel for a certified life of 20 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.

#### **4.7 References**

1. Pacific Northwest Laboratory, "Recommended Temperature Limits For Dry Storage Of Spent Light Water Reactor Zircaloy-Clad Fuel Rods In Inert Gas," PNL-6189, May 1987.
2. Electric Power Research Institute, "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," EPRI TR-106440, April 1996.
3. FLUENT, Inc., "FLUENT Computational Fluid Dynamics Software."
4. Swanson Analysis Systems, Inc., "ANSYS Finite Element Modeling Package," 1993.

## **5.0 SHIELDING EVALUATION**

This section evaluates the capability of the HI-STORM 100 Cask System shielding features to provide adequate protection against direct radiation from its contents. The regulatory requirements for providing adequate radiation protection to 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). Because 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits is evaluated in Section 10 (Radiation Protection) of the SER.

The shielding review focuses on the calculation of the dose rates from both direct gamma and neutron radiation at locations near the overpack and transfer cask, and at assumed distances away from the overpack. Section 10 of the SAR presents estimated occupational exposures and off-site dose rates that are based on the dose rates calculated in Section 5 of the SAR.

### **5.1 Shielding Design Features and Design Criteria**

#### **5.1.1 Shielding Design Features**

The HI-STORM 100 Cask System is designed to provide shielding of gamma and neutron radiation during loading, storage, and unloading operations. The concrete overpack provides shielding during storage of the MPC. The 100-ton and 125-ton HI-TRAC transfer casks consist of a lead and steel overpack surrounded by a water jacket and provide shielding during handling and transfer operations of the MPC. The 100-ton transfer cask contains less shielding material than the 125-ton HI-TRAC to meet crane capacity and/or fuel floor loading limitations at potential general licensee reactor sites. The generic HI-TRAC design also consists of two bottom lid designs to be compatible with loading and unloading operations. A low weight bottom pool lid is used primarily for HI-TRAC operations within the spent fuel pool. The pool lid is replaced by a heavier-shielded transfer lid for transfer operations of the MPC to and from the concrete overpack.

The MPC shielding design consists of a 0.5-inch thick steel canister with a 2.5-inch thick steel baseplate and a 9.5-inch thick steel lid for the MPC-24 or a 10-inch lid for the MPC-68. The overpack radial shield design consists of 26.75 inches of concrete encased in inner and outer steel shells with a total thickness of 2.75 inches. The top of the overpack is shielded by 10.5 inches of concrete encased in top and bottom steel lid plates with a total thickness of 5.25 inches. The bottom of the overpack is shielded by the pedestal shield and baseplate with a total of 7 inches of steel and 17.0 inches of concrete. The overpack also has internal labyrinthine air passages with steel cross plates that mitigate direct streaming of radiation through the air inlets and outlets.

The 100-ton HI-TRAC radial shield design consist of 1.75 inches of steel, 2.875 inches of lead, and a 5-inch thick water jacket. The 125-ton HI-TRAC radial shield design consists of 1.75 inches of steel, 4.5 inches of lead, and a 5.36-inch thick water jacket. The 100-ton top lid consists of a 1-inch thick steel plate with a 27-inch diameter opening for MPC access. The 125-

ton top lid consists of 3.25 inches of Holtite-A (similar to NS-4-FR) encased in steel plates with a total thickness of 1.5 inches. The 100-ton HI-TRAC pool lid consist of 2.5 inches of steel and 1.5 inches of lead. The 125-ton HI-TRAC pool lid consists of 3 inches of steel and 2.5 inches of lead. The 100-ton HI-TRAC transfer lid consist of 2.75 inches of steel and 1.5 inches of lead, and the 125-ton HI-TRAC transfer lid consists of 3.5 inches of steel, 2.5 inches of Holtite-A, and 2.0 inches of lead.

The staff evaluated the HI-STORM 100 Cask System shielding design features of the overpack and HI-TRAC designs and found them acceptable. The SAR analysis provides reasonable assurance that the shielding design meets the regulatory requirements of 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

### **5.1.2 Shielding Design Criteria**

Sections 2.0.2, 2.0.3, and 5.1 specify design criteria for the surface dose rates on the concrete overpack and the HI-TRAC designs. The criteria for the overpack is a surface dose rate on the side to be less than 40 mrem/hr, the surface dose rate on the top to be less than 10 mrem/hr, and the air inlet and outlet dose rates to be less than 60 mrem/hr. Specific dose rate criteria for the HI-TRACs are not given. However, proposed Technical Specifications in Appendix 12A establish surface dose limits on the concrete overpack, 100-ton HI-TRAC, and 125-ton HI-TRAC that are based on calculated dose rate values used to determine occupational and off-site exposures. The overall radiological protection design criteria for the HI-STORM 100 system are the regulatory dose requirements in 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), and maintaining occupational exposures ALARA.

The staff evaluated the HI-STORM 100 system shielding design criteria and found them acceptable. The SAR analysis provides reasonable assurance that the shielding design criteria meet the regulatory requirements of 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Surface dose rate limits for the overpack and HI-TRAC casks are incorporated into LCOs 3.2.1 and 3.2.3 of the Technical Specifications. The overall radiological protection design features and design criteria for the HI-STORM 100 system are evaluated in Section 10 of the SER.

## **5.2 Source Specification**

The source specification is presented in Section 5.2 of the SAR. All gamma and neutron source term calculations were performed with the SAS2H and ORIGEN-S modules of SCALE 4.3, using the 44-group cross-section library<sup>1</sup>. Four design basis fuel types are identified for zircaloy and stainless steel clad fuel in both the MPC-24 and MPC-68 configurations. In addition, design basis fuel types are specified for damaged BWR and MOX BWR fuel. The design basis fuel types are determined by calculating source terms for the fuel types presented in Tables 2.1.1 and 2.1.2, and specifying the fuels with the highest source strengths as the design-basis fuel types. Subsequent source terms are calculated for each fuel type with design basis burnup and cooling times that bound the burnup and cooling time parameters requested for storage in the overpack.

The Babcock & Wilcox 15x15 assembly is identified as the design basis fuel for zircaloy-clad PWR fuel (MPC-24). The design basis burnup and cooling time for the overpack shielding analysis is 45,000 MWD/MTU for 5 years. The design basis burnup and cooling times for the HI-TRAC transfer cask are 35,000 MWD/MTU for 5 years and 45,000 MWD/MTU for 9 years. The initial uranium loading is approximately 10 kgU higher than the maximum PWR uranium loading specified for the package contents.

The Westinghouse 15x15 stainless-steel-clad fuel assembly is identified as the design-basis fuel for the stainless-steel-clad PWR fuel (MPC-24). The design-basis burnup and cooling time is 40,000 MWD/MTU for 8 years.

The General Electric 7x7 assembly is identified as the design-basis fuel for the zircaloy-clad BWR fuel (MPC-68). The design-basis burnup and cooling time used in the overpack shielding analysis is 45,000 MWD/MTU for 5 years. The design-basis burnup and cooling times for the HI-TRAC shielding analysis are 30,000 MWD/MTU for 5 years and 45,000 MWD/MTU for 12 years. The fuel assembly is also shown to have bounding source characteristics for the zircaloy-clad 6x6 MOX fuel, the zircaloy-clad damaged BWR fuel, and the zircaloy-clad BWR fuel debris (MPC-68F) for both normal and accident conditions. The design-basis burnup and cooling time for these BWR fuels are 30,000 MWD/MTU for 18 years.

The Allis Chalmers 10x10 assembly is identified as the design-basis fuel for the stainless-steel-clad BWR fuel (MPC-68). The design-basis burnup and cooling time is 22,500 MWD/MTU for 10 years.

The staff evaluated the basis and methodology for determining design-basis fuel types and found it acceptable. Each design-basis fuel type has the highest uranium loading among other fuels in its category and has burnup and cooling times that bound the burnup and cooling time parameters requested for storage in the HI-STORM 100 system.

### **5.2.1 Gamma Source**

Gamma source terms are calculated for each design-basis fuel and are listed in Tables 5.2.5 through 5.2.9 and Table 5.2.22. The SAR presents a dose rate analysis to evaluate the effect of each gamma energy group on calculated surface dose rates, including overpack inlet and outlet vent dose rates. The analysis demonstrates that fuel gammas with energies from 0.45 MeV to 3.0 MeV comprise more than 99% of the external gamma dose. Therefore, only this gamma energy range is applied in the gamma shielding evaluation.

The gamma source term includes Cobalt-60 gamma radiation from activated assembly hardware that is listed in Tables 5.2.12 and 5.2.13. The activated hardware source terms are calculated assuming hardware masses listed in Table 5.2.1 and reactor flux scaling factors listed in Table 5.2.10. All hardware material is assumed to be Inconel or steel with a cobalt-59 impurity of 1000 parts-per-million (ppm). The SAR states information gathered from utilities and fuel vendors indicates that industry cobalt reduction programs have resulted in cobalt impurities in Inconel and steel to be less than 500 ppm since the late 1980s. A reference document cited in the SAR indicates fuel assemblies manufactured in the 1970s have cobalt impurities in Inconel ranging from 2000 to 2200 ppm. The SAR states the analysis assuming 1000 ppm is



bounding because discharged fuel with design-basis cooling times (5 years) will have impurities significantly lower than 1000 ppm. Discharged fuel with potentially higher cobalt impurities will have corresponding longer cooling times and calculated dose rates will, therefore, be bounded by the dose rates assuming 1000 ppm and design-basis cooling times. In addition, the SAR states recent fuel assemblies incorporate zircaloy grid spacers rather than Inconel grid spacers, which reduces total cobalt content of the hardware. Appendix D of Section 5 compares the dose rates around the overpack and HI-TRAC of design-basis fuel with an Inconel cobalt impurity of 1000 ppm and design-basis fuel with an Inconel cobalt impurity of 4700 ppm and an additional 4 years of cooling.

### **5.2.2 Neutron Source**

Neutron source terms calculated for each design-basis fuel are listed in Tables 5.2.16 through 5.2.20 and Table 5.2.23. The neutron source terms are calculated using the initial enrichments listed in Table 5.2.24 of the SAR for each burnup range examined in the SAR analysis. These initial enrichment values are lower than the average initial enrichment values for each corresponding burnup range specified in EIA Service Report SR/CNEAF/96-01<sup>2</sup>. The SAR recognizes that a small number of fuel assemblies may exist with enrichments below the values assumed in the source term analysis and states such assemblies will have an insignificant effect on the calculated dose rates. The SAR also states each general licensee should consider fuel enrichments when performing site-specific dose analyses. Analyses in Section 5.4 of the SAR indicates that the neutron component accounts for about five percent of the total off-site dose from the overpack.

### **5.2.3 Confirmatory Analysis**

The staff reviewed the analyses for gamma and neutron source terms and finds it acceptable. The staff performed confirmatory analysis for selected fuel types and burnup conditions. The staff used SAS2H and ORIGEN-S of SCALE 4.4 using the 44-group cross section library and the DOE Characterization Data Base<sup>3,4</sup>. The staff examined the proposed contents listed in Tables 2.1.3 and 2.1.4 of the SAR, and has reasonable assurance that the design-basis gamma and neutron source terms for the MPC-24 and MPC-68 configurations are acceptable for the shielding analysis.

The staff finds the flux scaling factors to be bounding or appropriate for determination of hardware source terms. The staff also finds the cobalt impurity assumption of 1000 ppm for Inconel to be acceptable for the HI-STORM 100 shielding analysis. Some reference documents indicate Inconel impurity values in older fuel may be higher than 1000 ppm. However, the staff has reasonable assurance that the gamma source terms used in the shielding analysis are bounding or appropriate for the HI-STORM 100 based on historical statements and dose rate comparisons presented in the SAR. The following factors are considered in this determination: (1) Cobalt-60 has a 5.27 year half-life and high energy gamma fission products significantly decay within the same time frame, (2) analyses in Appendix D of Section 5 of the SAR indicate fuel manufactured before the mid 1980's will have decayed significantly longer than design-basis cooling times and the corresponding dose rates will be comparable, (3) the design-basis burnup and initial uranium loading used in the SAR shielding analysis are significantly higher than the maximum burnup and uranium loading allowed in the Certificate of Compliance, (4)

maximum exterior dose rates are limited in the Certificate of Compliance, and (5) each general licensee will perform site-specific dose evaluations to demonstrate compliance with 10 CFR Part 72 radiological requirements, operate the HI-STORM 100 under a 10 CFR Part 20 radiological program, and monitor dose rates during ISFSI operations.

The staff also accepts that the lower-than-average initial enrichment values used to determine neutron source terms are adequate for each burnup level examined in the HI-STORM 100 shielding analysis. The values bound a significant portion of discharged fuel and the design-basis source terms are calculated for burnups significantly higher than those allowed in the Certificate of Compliance. In addition to general licensee considerations discussed in the above paragraph, the total source term and exterior dose rates are adequately controlled by limits in the Certificate of Compliance such as maximum burnup, minimum cooling time, maximum initial uranium loading, and maximum dose rates. Therefore, license conditions for minimum enrichment are not required for the HI-STORM 100 system. Each general licensee should consider minimum fuel enrichments when performing 10 CFR 72.212 radiological evaluations.

## **5.3 Shielding Model Specifications**

### **5.3.1 Model Specification**

The HI-STORM 100 system shielding and source configuration is presented in Section 5.3.1 of the SAR. The shielding analysis of the HI-STORM 100 was performed with MCNP-4A, a three-dimensional Monte Carlo transport code<sup>5</sup>. The shielding model for normal and accident conditions consist of a three dimensional representation of the overpack and HI-TRAC, including all shielding materials, the spent nuclear fuel source, gamma shield cross plates, and internal air passage streaming pathways based on the Design Drawings in Section 1.5 of the SAR. Radial and axial views of the overpack and HI-TRAC shielding models are presented in Figures 5.3.2 through 5.3.17 of the SAR.

As discussed in Section 5.4.3 of the SAR, the off-site dose models employ a two-stage MCNP modeling technique that consists of first generating an overpack surface source file and then using it for subsequent off-site dose calculations. The off-site dose models include a single cask modeled with an infinite earth slab surrounded by dry air at a density equivalent to the density of air at 20°C. The model includes a 700 meter high air volume and 500 meters to 1050 meters of horizontal air volume surrounding the cask. The air is extended at least 50 meters beyond calculated dose points in order to account for backscattering. Dose points are modeled at various distances. The off-site dose models also include a model with two infinite rows of casks in order to simulate the side dose contribution from the second row of casks in an array, which are partially shielded by the front row of casks. The dose contributions from both models are used to determine the total dose for various array configurations.

#### **5.3.1.1 Source Configuration**

The shielding source is divided into five axial regions: lower-end piece, fuel, gas plenum springs, gas plenum spacer, and upper-end piece. Axial views of the relative position of the source term within the shielding configurations are depicted in Figures 5.3.8 and 5.3.9 for the

MPC-24 and MPC-68, respectively. The fuel region is modeled as a homogenous zone, and the end fittings and plenum regions are modeled as homogenous regions of steel. The SAR demonstrated that homogenization of the fuel assembly versus explicit modeling does not affect the accuracy of the shielding results. The axial distribution of the gamma source is assumed to follow the relative burnup profiles listed in SAR Table 2.1.11 for PWR and BWR fuel assemblies. The axial distribution of the neutron source was assumed to follow the relative burnup profiles raised to the power of 4.2. This adjustment was performed to account for the non-linear buildup of neutron source terms (primarily Cm-244) as a function of burnup. The total integrated neutron source increased by 15.6% and 36.9% above the neutron source values calculated for the average burnup of the PWR and BWR fuel, respectively. The source MPC region is also modeled with a number of simplifications and bounding assumptions that effectively reduce the amount of actual shielding and result in higher calculated dose rates.

### **5.3.1.2 Streaming Paths and Regional Densities**

The shielding models include steaming paths through the inlet and outlet vents of the overpack and the lifting trunnions, pocket trunnions, and top opening in the HI-TRAC transfer cask. As discussed in Section 5.3.1.2 of the SAR, the cask design eliminates other potential streaming paths. For example, the MPC lid is designed with a block of steel directly beneath the MPC vent port to prevent streaming.

The composition and densities of the materials used in the shielding analysis are presented in SAR Tables 5.3.2 and 5.3.3. The homogenized fuel assembly region accounts for the uranium dioxide, zircaloy, and stainless steel present in each assembly. Mixed oxide fuel specifically accounts for the presence of plutonium isotopes mixed with uranium isotopes in the fuel pellets. The Boral plates are explicitly modeled with a lower  $^{10}\text{B}$  mass fraction than specified by the SAR bill of materials. Standard density and composition concrete and carbon steel are used in the overpack model. Sections 4.4 and 4.5 of the SAR demonstrate that all materials used in the HI-STORM 100 remain below the design temperatures specified in SAR Table 2.2.3 during normal conditions. Therefore, the shielding analysis does not address changes in material density or composition from temperature variations. The bounding accident condition for shielding assumes complete loss of the HI-TRAC water jacket. Therefore, the water in the water jacket is replaced with a void for the accident shielding analyses.

The staff evaluated the SAR shielding model and found it acceptable. The composition and density of materials in the shielding analysis is appropriate and/or bounding. The model dimensions and material specifications provide reasonable assurance that the HI-STORM 100 was adequately modeled in the shielding analysis. The material integrity of the shielding materials is evaluated in Section 3 of the SER.

## **5.4 Shielding Analyses**

### **5.4.1 Shielding Analyses**

The shielding analyses is performed with MCNP-4A and presented in Section 5.4 of the SAR. The individual cross-section libraries used for each nuclide are based on ENDF/B-V

cross-section data. The SAR provides references for MCNP photon and neutron benchmarking problems against experimental data. Appendix 5C of the SAR contains a sample MCNP input file used in the shielding analysis. The SAR uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose-rate conversion factors to calculate dose rates in the shielding analysis.

#### **5.4.1.1 Normal Conditions**

The SAR presents calculations for normal condition design-basis dose rates for the MPC-24 and MPC-68 at the overpack locations shown in Figures 5.1.1. Calculated dose rates for design-basis fuel loaded into the overpack are listed in SAR Tables 5.1.2 through 5.1.6. A summary of calculated dose rates for the overpack is listed in SER Table 5.4-1. SAR Table 5.4.8 presents the calculated PWR and BWR stainless steel-clad dose rates on the side of the overpack.

The calculated total dose rates for the air inlets, air outlets, and side of the overpack are dominated by the gamma dose component and the top dose rate is dominated by the neutron dose component. This is expected because the concrete neutron shielding is 10.5 inches thick on the top and 26.75 inches thick on the side. The calculated dose rates are below the dose rate design criteria specified in the SAR.

The SAR also presents calculations for normal condition dose rates surrounding the MPC-24 100-ton and 125-ton HI-TRAC locations shown in Figures 5.1.2 and 5.1.4 on the surface and at one meter. As shown in SAR Tables 5.1.7, 5.4.9 and 5.4.10, the maximum dose rates calculated for the side, top, and bottom of the MPC-24, 100-ton HI-TRAC configuration are approximately 875 mrem/hr, 355 mrem/hr, and 3640 mrem/hr, respectively. For the MPC-68 100-ton configuration, the maximum dose rates are 710 mrem/hr, 220 mrem/hr, and 4100 mrem/hr, respectively. The 125-ton HI-TRAC has significantly more shielding on the side and bottom of the cask and dose rates are bounded by 100-ton HI-TRAC dose rate values.

Section 5.4.1 of the SAR indicates the peak dose rate at the pocket trunnion is approximately 1500 mrem/hr. The SAR indicates that exposure from the localized peak will be mitigated by the actual locations of personnel and the placement of temporary shielding during operations described in Section 8 of the SAR. SAR Tables 5.4.2 and 5.4.3 present 100-ton HI-TRAC dose rates for the fully flooded MPC for both a full and empty water jacket to demonstrate the reduction in dose rates during various operational phases described in Section 8 of the SAR. Because HI-TRAC contributes significantly to occupational exposure, the SAR presents dose profiles for the top, bottom, and sides of the 100-ton HI-TRAC at various distances. The dose profiles are shown in Figures 5.1.5 through 5.1.11. These profiles indicate that dose rates significantly decrease from peak locations to the edges of the top, bottom, and sides of HI-TRAC. The dose rates also significantly decrease as physical distance is increased from the top, bottom, and side of HI-TRAC. For example, Figure 5.1.6 demonstrates the dose rate at the bottom edge of HI-TRAC, where occupational personnel may be located, is more than a factor 20 lower than the peak dose rate value of 3.5 rem/hr at the center of the lid.

**Table 5.4-1**  
**SAR Maximum Calculated HI-STORM 100 Overpack Total Dose Rates (mrem/hour)**

Dose Rate Location*	Intact zircaloy Clad PWR	Intact zircaloy Clad BWR	Intact Stainless Steel Clad PWR	Intact Stainless Steel Clad BWR
Air Inlet, Contact	13.51	14.52	(BBZ)**	(BBZ)
Air Inlet, 1-Meter	8.21	8.66	(BBZ)	(BBZ)
Center Overpack Side, Contact	33.58	34.92	38.10	17.90
Center Overpack Side, 1-Meter	17.42	17.30	19.43	9.15
Air Outlet, Contact	8.59	7.23	(BBZ)	(BBZ)
Air Outlet, 1-Meter	5.03	4.58	(BBZ)	(BBZ)
Top Center, Contact	4.91	4.31	(BBZ)	(BBZ)
Top Center, 1-Meter	1.70	1.35	(BBZ)	(BBZ)

\* Dose rates are calculated over average surface areas for each location

\*\* (BBZ) = dose rate bounded by the intact zircaloy clad fuel category calculated dose rate

#### **5.4.1.2 Accident Conditions**

The SAR does not identify an accident that significantly degrades the shielding capability of the concrete overpack. The design-basis shielding degradation for HI-TRAC is assumed to be complete loss of the water jacket. SAR Table 5.1.10 shows that the maximum dose rate at one meter from the side of HI-TRAC is approximately 1090 mrem/hr. The one meter dose rate is extrapolated to 100 meters and is calculated to be less than 1 mrem/hr.

#### **5.4.1.3 Occupational Exposures**

The SAR analysis uses the MPC-24 configuration with the design-basis zircaloy clad fuels for the overpack, 100-ton HI-TRAC, and 125-ton HI-TRAC to estimate occupational exposures during HI-STORM 100 system operations. Section 10 of the SAR presents the estimated occupational exposures that are based on dose rate calculations in Section 5 of the SAR.

#### **5.4.1.4 Off-site Dose Calculations**

Sections 5.1 and 5.4.3 of the SAR estimate the offsite dose rates from various cask arrays. SAR Tables 5.1.9 and 5.4.7 and SAR Figure 5.1.3 present the total calculated offsite annual dose rate for these arrays at distances of 100 to 400 meters based on an 8,760 hour (i.e. 100% occupancy) exposure time. The SAR calculations show an offsite annual dose of 25 mrem, the

10 CFR 72.104(a) limit, at about 180 meters for one cask and at about 310 meters for a 2x5 cask array with the bounding intact zircaloy clad PWR source term. SAR Table 10.4.2 show the normal condition dose rate at 100 meters from the 100-ton HI-TRAC is approximately 0.3 mrem/hr. The generic off-site calculations demonstrate that the HI-STORM 100 system is capable of meeting the offsite dose criteria for postulated cask arrays of one or more casks at distances greater than 100 meters.

#### **5.4.2 Confirmatory Calculations**

The staff performed a confirmatory analysis of selected overpack dose rates with MCNP-4B2<sup>6</sup> and manual calculation methods. The staff based its confirmatory evaluation of the overpack on the design features and model specifications discussed above in Sections 5.1, 5.2, and 5.3, coupled with an analysis of the applicant's SAR calculations. The staff's calculated dose rates were in close agreement with the SAR values. The staff found that the calculated surface dose rates and estimated dose rates beyond the controlled area boundary are acceptable for the overpack. The staff determined through confirmatory calculations that off-site dose rates increase as the assumed air density decreases. Site specific dose analysis should therefore account for appropriate atmospheric conditions. The staff also performed independent calculations with MCNP-4B2 that statistically demonstrate the two-stage MCNP modeling technique is acceptable for determining off-site dose rates. In addition, confirmatory calculations of the design-basis zircaloy-clad PWR fuel were performed to demonstrate that an assumed minimum enrichment of 3.3 weight percent increases the off-site dose rate by less than 1% as compared to the SAR assumed minimum enrichment of 3.6 weight percent. The staff evaluation of the off-site dose estimations finds that the SAR has adequately demonstrated that the HI-STORM 100 system is designed to meet the offsite dose criteria of 10 CFR 72.104(a) for a minimum controlled area distance of 100 meters as stated in 10 CFR 72.106(b).

The staff performed a confirmatory analysis of selected dose rates around the 100-ton and 125-ton HI-TRACs with MCBEND-9D<sup>7</sup>. The confirmatory calculations were based on the design features and model specifications discussed above in Sections 5.1 and 5.3. The staff found that the calculated surface dose rates and one meter dose rates are acceptable. All dose rates remain below the applicable dose rate limits of the SAR. The fuel characteristics in SAR Tables 2.1.3 and 2.1.4 and the burnup, cooling time, and initial uranium loading parameters listed in the proposed Technical Specifications in Appendix 12A are included as conditions in the Certificate of Compliance. The calculated overpack and HI-TRAC dose rates, as discussed above, are also included as conditions in the Certificate of Compliance.

Section 10 of the SER evaluates the overall off-site dose rates from the HI-STORM 100 Cask System. 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 100 Cask 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 individuals beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configuration, topography, demographics, atmospheric conditions, and use of engineered features (e.g., berm). 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 72.104(a) is the responsibility of each applicant for a site license.

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 by 10 CFR Part 20, Subpart D, by evaluations and measurements.

A license condition has been included regarding engineered features used for radiological protection. The license condition states that engineering features (e.g., berms and shield walls) used to ensure compliance with 10 CFR 72.104(a) are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

## **5.5 Evaluation Findings**

- F5.1** The SAR sufficiently describes shielding design features and design criteria for the structures, systems, and components important to safety in sufficient detail to allow evaluation of their effectiveness.
- F5.2** Radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104 and 10 CFR 72.106.
- F5.3** Operational restrictions 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 HI-STORM 100 Cask System shielding features are designed to assist in meeting these requirements.
- F5.4** The staff concludes that the design of the shielding system for the HI-STORM 100 Cask 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 100 Cask System will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

## **5.6 References**

1. L.M. Petrie, et al. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-4, Revision 5, 1995.
2. Energy Information Administration, U.S. Department of Energy, "Spent Nuclear Fuel Discharges from U.S. Reactors 1994," February 1996.
3. Oak Ridge National Laboratory Radiation Shielding Information Center, "SCALE4.4; Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstation and Personal Computer," CCC-545, September 1998.
4. TRW Environmental Safety Systems, Inc., "DOE Characteristics Data Base, User Manual for the CDB\_R," November 16, 1992.

- 5.. J.F. Briesmeister, Ed. "MCNP - A General Monte Carlo N-Particle Transport Code Version 4A," Los Alamos National Laboratory, LA-12625-M (1993).
6. Oak Ridge National Laboratory Radiation Shielding Information Center, "MCNP4B2: Monte Carlo N-Particle Transport Code System," CCC-660, January 1998.
7. AEA Technology, "MCBEND - A Monte Carlo Program for General Radiation Transport Solutions, User Guide for Version 9," Issue 6, June 1997.



## 6.0 CRITICALITY EVALUATION

### 6.1 Review Objective

The staff reviewed the HI-STORM 100 Cask System criticality analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the HI-STORM 100 Cask System meets the following regulatory requirements: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g). The SAR was also reviewed to determine whether the cask system fulfills the following acceptance criteria listed in Section 6 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems:

5. The multiplication factor, including all biases and uncertainties at a 95 percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.
6. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.
7. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.
4. Criticality safety of the cask system should not rely on use of the following credits:
  - a. burnup of the fuel
  - b. fuel-related burnable neutron absorbers
  3. more than 75 percent for fixed neutron absorbers when subject to standard acceptance tests.

The staff's evaluation is summarized below.

### 6.2 Criticality Design Criteria and Features

The design criterion for criticality safety is that the effective neutron multiplication factor,  $k_{\text{eff}}$ , including statistical biases and uncertainties, shall not exceed 0.95 for all postulated arrangements of fuel within the cask system under normal, off-normal, and accident conditions.

The design features relied upon to prevent criticality are the fuel basket geometry and permanent neutron-absorbing Boral panels. In all three MPC designs, the fuel assemblies are placed in baskets with square fuel cells and Boral panels fixed to the fuel cell walls. The MPC-24 basket has a minimum flux trap size of 1.09 inches and the minimum  $^{10}\text{B}$  content of 0.0267 g/cm<sup>2</sup> in the Boral panels. The MPC-68 and MPC-68F baskets have a minimum pitch of

6.43 inches between the fuel cells. The minimum  $^{10}\text{B}$  content is  $0.0372 \text{ g/cm}^2$  in MPC-68 Boral panels and  $0.01 \text{ g/cm}^2$  in the MPC-68F Boral panels.

The staff reviewed Sections 1, 2, and 6 of the SAR and verified that the design criteria and features important to criticality safety are clearly identified and adequately described. The staff verified the consistency of the information between Sections 1, 2, and 6. The staff verified that the SAR contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

The staff also verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the HI-STORM 100 Cask System design meets the “double contingency” requirements of 10 CFR 72.124(a).

### **6.3 Fuel Specification**

The HI-STORM 100 Cask System is designed to store up to 24 PWR or 68 BWR spent fuel assemblies. The fuel assemblies must fit into one of 36 PWR or BWR fuel assembly classes defined by the applicant. The classes of fuel assemblies that are approved for storage in the HI-STORM 100 Cask System are listed in Tables 2.1-3 and 2.1-4 of the SAR. In those tables, the fuel specifications important to criticality safety are:

- maximum initial enrichment (PWR) or maximum planar average initial enrichment (BWR)
- number of fuel rods, including number of partial length rods (BWR)
- minimum clad outer diameter
- maximum clad inner diameter
- maximum pellet diameter
- fuel rod pitch
- number of guide tubes (PWR) or number of water rods (BWR)
- minimum guide tube thickness (PWR) or minimum water rod thickness (BWR)
- maximum channel thickness (BWR)

The parameters listed above represent the limiting or bounding parameters for the fuel assemblies. A fuel assembly having these actual specifications would be the most reactive or bounding assembly in that class. In some assembly classes, the bounding assembly corresponds to an actual assembly design. However, in most assembly classes, the bounding assembly is an artificial assembly having the bounding parameters from several different assemblies within a specified class and is, therefore, more reactive than any real assembly design in that class.

The applicant referenced the SAR for the HI-STAR 100 Cask System for determining the most reactive fuel assemblies since the two cask systems are essentially identical, except for the overpack materials, and the allowed fuel assemblies are identical. The staff previously reviewed the HI-STAR 100 Cask System analyses for determining the most reactive fuel assemblies within each class and found that they represent the limiting or bounding parameters for fuel assemblies in a cask.

In terms of criticality safety, the most important fuel specification is the fuel enrichment. The MPC-24 may contain 24 PWR fuel assemblies with maximum initial enrichments varying from 4.0 to 4.6 weight percent  $^{235}\text{U}$  depending on the fuel assembly class. The MPC-68 may contain 68 BWR fuel assemblies with maximum planar average initial enrichments varying from 2.7 to 4.2 weight percent  $^{235}\text{U}$  depending on the fuel assembly class. The MPC-68F may contain BWR fuel assemblies from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes with a maximum planar average initial enrichment of 2.7 weight percent  $^{235}\text{U}$ .

The maximum planar average initial enrichment is specified for the BWR fuel assemblies because these assemblies typically have fuel rods with varying enrichments. The maximum planar average enrichment is the simple average of the distributed fuel rod enrichments within a given axial plane of the assembly lattice. The applicant's criticality analyses for BWR fuel assumed that the entire fuel assembly was at the maximum planar average initial enrichment. The applicant presented the calculational results in Appendix 6B of the SAR which demonstrates that this is more conservative than explicitly modeling pin-by-pin enrichments.

Specifications on the condition of the fuel are also included in the SAR. The HI-STORM 100 Cask System is designed to accommodate intact fuel assemblies, damaged fuel assemblies, and fuel debris as defined in the SAR. The MPC-24 can only contain intact PWR fuel assemblies. The MPC-68 can only contain intact UO<sub>2</sub> BWR fuel assemblies from the approved classes; intact or damaged BWR MOX fuel assemblies from the 6x6 B class; or damaged BWR UO<sub>2</sub> fuel assemblies from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes. The MPC-68F may contain intact or damaged BWR fuel assemblies or fuel debris from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes. The damaged fuel and fuel debris must be placed in DFCs which are designed to confine gross fuel particulates to a known, subcritical geometry.

The staff reviewed the fuel specifications considered in the criticality analyses and verified that they are consistent with or bound the specifications given in Sections 1, 2, and 12 of the SAR. The staff agrees that fuel assembly parameters listed above represent the limiting or bounding parameters for fuel assemblies in a cask that does not rely on borated water for criticality control.

The staff reviewed the applicant's calculations that compared the use of planar-averaged enrichment versus explicit pin-by-pin enrichments in BWR fuel assemblies. Based on the results of these calculations and the information in Appendix 6.B of the SAR, the staff agrees that using the maximum planar average initial enrichment in the criticality analyses of BWR fuel assemblies is appropriate.

## **6.4 Model Specification**

### **6.4.1 Configuration**

The HI-STORM 100 Cask System consists of a transfer cask, a concrete storage overpack, and an MPC. The three MPC designs used with the HI-STORM 100 Cask System, as well as their respective contents, are identical to those previously reviewed by the staff for the HI-STAR 100 Cask System. The only difference between the HI-STAR 100 Cask System and the HI-STORM 100 Cask System are the reflector materials in the transfer cask and overpack. These

reflector materials do not significantly affect the reactivity of the system. The HI-STAR 100 overpack is constructed mainly of steel; whereas the HI-STORM 100 storage overpack is concrete and the HI-TRAC transfer cask is lead with a water jacket.

The applicant used three-dimensional calculational models in its criticality analyses. Sketches of the models are given in Section 6.3 of the SAR. The models are based on the engineering drawings in Section 1.5 of the SAR and consider the dimensional worst-case tolerance values. As previously stated, the design-basis off-normal and accident events do not affect the design of the cask from a criticality standpoint. Therefore, the calculational models for the normal, off-normal, and accident conditions are the same.

To determine the most reactive MPC basket dimension combinations, considering manufacturing tolerances, the applicant performed two-dimensional CASMO-3 and three-dimensional MCNP4a calculations. These calculations were taken from the HI-STAR 100 Cask System criticality safety analysis. Based on the results, the MPC-24 was modeled using the nominal fuel cell pitch (10.777 inches), the minimum box inner dimension (8.81 inches), the nominal box wall thickness (5/16 inch), and the minimum flux trap size (1.09 inches). The MPC-68 (includes MPC 68F) was modeled using the minimum fuel cell pitch (6.43 inches), the minimum box inner dimension (5.993 inches), and the nominal box wall thickness (1/4 inch).

The calculational models also conservatively assumed the following:

- fresh fuel isotopics (i.e., no burnup credit)
- 75 percent credit for the  $^{10}\text{B}$  loading in the Boral panels for the MPC- 24 and MPC-68 and only 67 percent credit for the MPC-68F
- the Boral panels are only as long as the fuel assembly active length which is 150 inches maximum instead of the actual panel length of 156 inches
- the Boral panels located on the periphery of the MPC-24 are only 5 inches wide, although the engineering drawings specify 12 of the peripheral panels to be 6.25 inches and all other panels to be 7.5 inches wide
- flooding of the fuel rod gap regions with pure water whenever the cask contains water
- the maximum planar average enrichment of the fuel assemblies in the MPC-68F is 3.0 weight percent  $^{235}\text{U}$  even though the maximum permitted is 2.7 weight percent  $^{235}\text{U}$

The fuel assemblies were modeled explicitly. For BWR fuel assemblies, the water channels were appropriately included in the model. The models for damaged fuel assemblies and fuel debris considered lost or missing fuel rods, collapsed fuel assemblies, and powdered fuel.

The applicant considered various levels of external (interspersed) and internal moderation to determine the most reactive moderating conditions (optimum moderation). For all MPC designs, the applicant determined that optimum internal moderation occurs when flooded with 100 percent density unborated water. The applicant also determined that the reactivity of a fully flooded single package is insensitive to the degree of interspersed moderation.

Normally, preferential or uneven flooding within the MPC is not a concern because the MPC baskets are designed such that the volume inside and outside the fuel cells will flood and drain at the same rate. For damaged fuel in DFCs, however, uneven draining may be possible

because the DFC drainage holes are covered with 250 mesh debris screens. The water surface tension in a 250 mesh screen may be capable of supporting water. Thus, the DFCs may hold water or may not drain at the same rate as the rest of the MPC cavity. The applicant did not consider a case in which the DFCs retained water while the rest of the MPC cavity was drained. However, the staff performed an independent analysis considering this scenario for the HI-STAR 100 cask which uses the same MPCs and DFCs. In this analysis, the staff assumed that the entire internal volumes of the DFCs were filled with water while the rest of the MPC cavity was dry. This analysis resulted in a maximum  $k_{eff}$  of approximately 0.9 for the most reactive damage fuel assembly class (the 6x6 C). In comparison, the applicant's analysis shows a  $k_{eff}$  of approximately 0.8 for the 6x6 C assembly when the entire MPC cavity is fully and evenly flooded. Although there is a significant increase in  $k_{eff}$ , it still remains well below 0.95. Thus, the staff concludes that even if preferential or uneven flooding is possible within the DFCs, it does not present a criticality concern for the fuel assemblies considered.

Based on the results of the applicant's evaluation and the staff's independent confirmatory calculations, the staff concludes that the most reactive moderating conditions have been considered.

The off-normal and accident condition events will not adversely affect the design features important to criticality safety. Therefore, in terms of reactivity and criticality control, the configuration of the cask after an off-normal or accident event will be identical to or bounded by the normal condition configuration.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in Sections 1 and 2, including engineering drawings.

The staff reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances were incorporated into the calculational models.

For its confirmatory analyses, the staff independently modeled the cask using the engineering drawings and bills of material presented in Section 1.5 of the SAR. The staff's fuel assembly models were based on the fuel assembly parameters given in Sections 2 and 6 of the SAR. The staff found its models of the cask and contents to be compatible with the applicant's.

#### **6.4.2 Material Properties**

The composition and densities of the materials considered in the calculational models are provided in Table 6.3.4 the SAR.

One of the most important materials in the HI-STORM 100 Cask System is the Boral neutron absorber in the MPC basket. In Section 1.2.1.3.1 of the SAR, the applicant provided a detailed description of the characteristics, historical applications, service experience, and manufacturing quality assurance of the Boral material. The minimum required  $^{10}\text{B}$  content is verified through the acceptance testing program described in Section 9.1.5.3. As previously stated, a maximum of only 75 percent credit is taken for the  $^{10}\text{B}$  content in the Boral panels.

The continued efficacy of the Boral, over a 20-year storage period, is assured by the design of the HI-STORM 100 Cask System. The applicant demonstrated that the neutron flux from the irradiated fuel results in a negligible depletion of the  $^{10}\text{B}$  content in the Boral. In addition, a structural analysis was performed which demonstrates that the Boral panel will remain in place during accident conditions.

The staff reviewed the composition and number densities presented in Table 6.3.4 of the SAR and found them to be acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Based on the information provided on the Boral material, the staff agrees that the continued efficacy of the Boral poison can be assured by the design of the HI-STORM 100 Cask System, and a surveillance or monitoring program is not necessary.

The staff reviewed the neutron absorber acceptance test described in Section 9.1.5.3 of the SAR. The staff's acceptance of the neutron absorber test described in this section is based, in part, on the fact that the criticality analyses assumed only 75 percent of the minimum required  $^{10}\text{B}$  content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

## **6.5 Criticality Analysis**

### **6.5.1 Computer Programs**

The applicant's principal criticality analysis code was MCNP4a, a three-dimensional, continuous-energy, Monte Carlo N-Particle code. The MCNP4a calculations used the continuous-energy cross-section data distributed with the code. This cross-section data is based on ENDF/B-V cross-section library. The applicant also performed independent verification of its MCNP4a calculations using the KENO-Va code in the SCALE 4.3 system. The KENO-Va calculations used the 238-group cross-section library.

CASMO-3, a two-dimensional transport theory code, was used to assess the incremental reactivity effects of manufacturing tolerances. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances. Based on the results of the CASMO-3 calculations, the worst-case combination of manufacturing tolerances was determined and incorporated into the three-dimensional MCNP4a and KENO-Va models.

The staff performed confirmatory analyses with the CSAS/KENO.Va modules of SCALE developed at Oak Ridge National Laboratory. The code is a standard in the industry for performing criticality analyses.

The staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel system.

### 6.5.2 Multiplication Factor

Results of the applicant's criticality analyses show that the  $k_{\text{eff}}$  in the HI-STORM 100 Cask System will remain below 0.95 for all fuel loadings. The results of the applicant's MCNP4a criticality calculations for the bounding assemblies are given in Tables 6.1.1, 6.1.2, and 6.1.3 of the SAR. The maximum  $k_{\text{eff}}$  calculated for each MPC design are summarized in the table below. These results have been adjusted to include all biases and uncertainties at a 95 percent confidence level.

Most Reactive Fuel Assembly Class*	Maximum $k_{\text{eff}}$ HI-STAR	HI-TRAC $k_{\text{eff}}$	MPC Design
15x15 F	0.9478	0.9468	MPC-24
10x10 A	0.9457	0.9448	MPC-68
6x6 C	0.8021	0.8024	MPC-68F

\*The most reactive fuel assembly class was determined from the HI-STAR 100 Cask System criticality analysis.

The staff reviewed the applicant's calculated  $k_{\text{eff}}$  values and agrees that they have been appropriately adjusted to include all biases and uncertainties at a 95 percent confidence level or better.

The staff performed independent criticality calculations for the HI-TRAC transfer cask. The results of the staff's confirmatory calculations were in close agreement with the applicant's results for the corresponding fuel assembly class.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the HI-STORM 100 Cask System will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

### 6.5.3 Benchmark Comparisons

The applicant performed benchmark calculations on selected critical experiments, chosen, as much as possible, to bound the range of variables in the HI-STORM 100 Cask System design. The three most important parameters are the fuel enrichment, the  $^{10}\text{B}$  loading of the neutron absorbers, and the fuel cell spacing (MPC-68) or flux trap size (MPC-24). Parameters such as reflector material and spacing, fuel pellet diameter and fuel rod pitch, and MOX fuel, have a smaller effect but were also considered in selecting the critical experiments.

Results of the benchmark calculations show that there are no trends in the bias. The benchmark analysis yielded the following calculational biases:  $0.0021 \pm 0.0006$  for MCNP4a, and  $0.0036 \pm 0.0009$  for KENO-Va. These biases were determined by truncating to 1.000 any

calculated  $k_{\text{eff}}$  that exceed unity. The uncertainty associated with each bias has been multiplied by the one-sided K-factor for 95 percent probability at the 95 percent confidence level ( $\sim 2.05$  for the number of cases analyzed).

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations.

The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining the calculational bias acceptable and conservative. The staff also verified that only biases which increase  $k_{\text{eff}}$  have been applied.

## **6.6 Supplemental Information**

The spent fuel assembly classes that can be loaded into the HI-STORM 100 Cask System without compromising criticality safety requirements are listed in Section 2.0 of Appendix B to the Certificate of Compliance. All supportive information has been provided in the SAR, primarily in Sections 1, 2, and 6.

## **6.7 Evaluation Findings**

Based on the information provided in the SAR and the staff's own confirmatory analyses, the staff concludes that the HI-STORM 100 Cask System meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

- F6.1** Structures, systems, and components important to criticality safety are described in sufficient detail in Sections 1, 2, and 6 of the SAR to enable an evaluation of their effectiveness.
- F6.2** The HI-STORM 100 Cask System is designed to be subcritical under all credible conditions.
- F6.3** 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 20-year storage period. In addition, there is no credible way to lose the fixed neutron poisons; therefore, there is no need to provide any further means to verifying their continued efficacy as required by 10 CFR 72.124(b).
- F6.4** The analysis and evaluation of the criticality design and performance have demonstrated that the cask will provide for the safe storage of spent fuel for 20 years with an adequate margin of safety.
- F6.5** The staff concludes that the criticality design features for the HI-STORM 100 Cask System 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 100 Cask System will allow safe storage of



spent fuel. This finding considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

## 7.0 CONFINEMENT EVALUATION

The confinement features and capabilities review ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

### 7.1 Confinement Design Characteristics

The applicant has clearly identified the confinement boundary. The confinement boundary includes the MPC shell, the bottom baseplate, the MPC lid (including the vent and drain port cover plates), the MPC closure ring, and the associated welds. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code, Section III, Subsection NB to the maximum extent practicable. Exceptions to the ASME Code are listed in the SAR Table 2.2.15. The MPC lid (with the vent and drain port cover plates welded to the lid) and closure ring are welded to the upper part of the MPC shell at the loading site. This provides redundant sealing of the confinement boundary. The welds forming the confinement boundary are described in detail in Subsection 7.1.3 of the SAR. The redundant closures of the MPC satisfy the requirements of 10 CFR 72.236(e) for redundant sealing of confinement systems.

The applicant provided procedures for drying and evacuating the cask interior during loading operations. The staff reviewed these procedures and finds that this design, if fabricated properly according to the SAR, will maintain the confinement boundary. Maintaining the stable pressure of 3 torr for greater than 30 minutes with the vacuum pump disconnected, assures that an acceptably low quantity of water remains in the MPC.

The applicant's testing is performed to a leak rate of  $5 \times 10^{-6}$  cm<sup>3</sup>/sec helium and confirms that the amount of helium lost from the MPC over the approved period due to the hypothetical accident conditions leakage rate is limited to less than 2.5% of the backfilled amount. This ensures that an adequate amount of helium remains in the MPC to maintain an inert atmosphere and to support the heat transfer over the lifetime of the cask.

For normal storage conditions, the MPC uses multiple confinement barriers provided by the fuel cladding and the MPC enclosure vessel to assure that there is no release of radioactive material to the environment. The MPC is backfilled with an inert gas (helium) to protect against cladding degradation. Section 3 of the SER shows that all confinement boundary components are maintained within their code-allowable stress limits during normal storage conditions. Section 4 of this SER shows that the peak confinement boundary component temperatures and pressures are within the design-basis limits for normal conditions of storage. Weld examinations, including multiple surface and volumetric examinations, hydrostatic testing, and leakage rate testing on the MPC lid weld; multiple surface examinations; and leakage rate testing of the vent and drain port cover plate welds assure the integrity of the MPC closure. Holtec described the MPC inspection and test acceptance criteria in SAR Table 9.1.1. MPC closure weld examination and acceptance criteria are included in Section 3.1.1 of the Technical Specifications and Section 3.3 of Appendix B to the Certificate of Compliance. The all-welded construction of the MPC with

redundant closure provided by the fully welded MPC closure ring, and extensive inspection and testing, ensures that no release of radioactive material for normal storage and transfer conditions will occur.

## **7.2 Confinement Monitoring Capability**

For cask systems using canisters with seal weld closures, continuous monitoring of the weld closures is not necessary because there is no known plausible, long-term degradation mechanism which would cause the seal welds to fail. Continuous monitoring of the cask, including periodic surveillance, inspection, and survey requirements, as well as preexisting radiological and environmental monitoring programs of licensees throughout the lifetime use of canisters with seal weld closures, are such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions.

## **7.3 Nuclides with Potential for Release**

The quantity of radioactive nuclides postulated to be released to the environment and the applicable bounding calculational method have been assessed as discussed in NUREG/CR-6487, "Containment Analysis of Type B Packages Used to Ship Various Contents" and ANSI N14.5-1997, "Leakage Tests on Packages for Shipment." The applicant used isotopes that contributed greater than or equal to 0.1% of the total activity inventory based on bounding shielding fuel source terms. This activity cutoff point is acceptable because the analysis captures over 97% of the dose at the site boundary. The release fractions used for the calculations are consistent with NUREG/CR-6487. Design -basis leakage rate of  $5 \times 10^{-6}$  cm<sup>3</sup>/sec is specified in SAR Table 7.1.1. This leak rate was derived from a detailed analysis that implements 10 CFR Part 71, Appendix A criteria. For the confinement analysis, the leak rate was conservatively increased to  $7.5 \times 10^{-6}$  cm<sup>3</sup>/sec (e.g., 50% increase) for calculating an equivalent break flow diameter for assessing offsite consequences for normal, off-normal and hypothetical accident conditions. The staff finds this acceptable.

## **7.4 Confinement Analysis**

Since the confinement boundary is welded and the temperature and pressure of the MPC are within the design- basis limits, no discernable leakage is credible. However, to demonstrate that the HI-STORM 100 Cask System meets the requirements of 10 CFR 72.104(a), the applicant performed detailed analyses with the following assumptions.

**Table 7.1**  
**Analytic Assumptions for Calculating Offsite Radiological Consequences**

	Normal Operating Conditions	Off-Normal Operating Conditions	Accident Conditions
% Failed Fuel	1%	10%	100%
Breathing Rate	$3.3 \times 10^{-4} \text{ m}^3/\text{sec}$	$3.3 \times 10^{-4} \text{ m}^3/\text{sec}$	$3.3 \times 10^{-4} \text{ m}^3/\text{sec}$
$\chi/Q$	Reduction factor 50	Reduction factor 50	No Reduction Factor
Wind Speed	1 m/sec	1 m/sec	1 m/sec
Dispersion Factor	F-Stability Diffusion	F-Stability Diffusion	F-Stability Diffusion

The staff's confirmatory audit evaluation applied the same postulated break flow diameter ( $4.96 \times 10^{-4} \text{ cm}$ ) as the applicant with the following variation on the  $\chi/Q$  value. For normal and off-normal conditions, the staff applied Stability Curve D<sup>1</sup> with a wind speed of 5 m/sec. In addition, it was assumed that the wind direction was limited to 1/3 the circumferential direction. Good agreement was obtained between the staff's and Holtec's calculations and confirmed compliance with the requirements of 10 CFR 72.104(a). For postulated accident conditions, a good agreement was also obtained between the staff's and Holtec's calculations (44.1 mrem and 44.0 mrem, respectively) and confirmed compliance with the requirements of 10 CFR 72.106(b).

## 7.5 Supportive Information

Supportive information or documentation includes justification of assumptions and analytical procedures, computer spreadsheets, drawings of the MPC confinement boundary and applicable pages from referenced documents.

## 7.6 Evaluation Findings

- F7.1** Section 7 of the SAR describes confinement structures, systems, and components important to safety in sufficient detail to permit evaluation of their effectiveness.
- F7.2** The design of the HI-STORM 100 Cask System adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of this SER discusses the staff's relevant temperature considerations.
- F7.3** The design of the HI-STORM 100 Cask System provides redundant sealing of the confinement system closure joints using dual welds on the MPC lid and the MPC closure ring.

- F7.4** The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. No instrumentation is required to remain operational under accident conditions. Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.
- F7.5** The quantity of radioactive nuclides postulated to be released to the environment has been assessed as discussed above. In Section 10 of the SER, the dose from these releases is added to the direct dose to show that the HI-STORM 100 Cask System satisfies the regulatory requirements of 10 CFR 72.104(a) (e.g., during normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area does not exceed 25 mrem to the thyroid and 25 mrem to any other critical organ) and 10 CFR 72.106(b) (e.g., any individual located on or beyond the nearest boundary of the controlled area will not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent will not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem).
- F7.6** The confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.
- F7.7** The staff concludes that the design of the confinement system of the HI-STORM 100 Cask System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the HI-STORM 100 Cask System will allow safe storage of spent fuel. This finding 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.

## **7.7 References**

1. U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1989.

## **8.0 OPERATING PROCEDURES**

The review of the operating procedures is to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for key operations.

### **8.1 Cask Loading**

Detailed loading procedures must be developed by each user.

The loading procedures described in the SAR include appropriate key prerequisite, preparation, and receipt inspection provisions to be accomplished before loading. These include verification of lift yoke load test certifications, visual inspection of key components, and reference to the site's heavy load handling procedures. The procedures also verify that tests, inspections and verifications, and cleaning procedures required in preparation for loading are specified. The procedure descriptions include actions necessary to ensure that appropriate levels of fluid (e.g., demineralized water in the annulus region between the MPC and the HI-TRAC transfer cask and the spent fuel pool water within the MPC) are properly controlled. The procedures describe the activities sequentially in the anticipated order of performance.

#### **8.1.1 Fuel Specifications**

The procedures described in the SAR provide for fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading have been pre-selected for loading into the MPC. After pre-selected fuel assemblies have been loaded in the MPC, a confirmatory post-loading visual verification is performed. Exact fuel specifications for fuel that is permitted to be loaded into an MPC is specifically designated in Section 2.0 of Appendix B to the Certificate of Compliance. Detailed site-specific procedures are necessary to ensure all fuel loaded in the cask meets the fuel specifications as delineated in the certificate. These procedures are subject to evaluation on a site-specific basis through the inspection process rather than during licensing review.

#### **8.1.2 ALARA**

The loading procedures incorporate general as low as reasonably achievable (ALARA) principles and practices. ALARA practices include periodic monitoring of dose rates, the use of annulus sealing equipment to reduce occupational exposure and limit MPC contamination, the use of temporary shielding, and the use of special tools to reduce occupational exposure. ALARA principles include warnings and notes that precede steps and identify potential radiological hazards. The procedures incorporate LCOs 3.2.1, 3.2.2 and 3.2.3 of the Technical Specifications, which specify limits for surface dose rates and radionuclide contamination. Each cask user will need to develop detailed loading procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

### 8.1.3 Draining and Drying

Section 8 of the SAR contain the operating procedures for use in draining and drying the MPC. These procedures clearly describe the process for removing water vapor and oxidizing material to an acceptable level.

After refilling the MPC with demineralized or spent fuel pool water, helium or nitrogen at a pressure less than 20 psig is used to drain as much water as practicable from the cask. The volume of water removed is recorded for later use. The vacuum drying system (VDS) is used with a stepped approach to reduce the potential for blockage of the evacuation system as a result of icing during evacuation. After the VDS reduces the MPC pressure to less than 3 torr, the pump is isolated, and a 30-minute holding period begins. In accordance with the Technical Specifications, the MPC remains at  $\leq 3$  torr for  $\geq 30$  minutes. This dryness verification test is included as LCO 3.1.1 in the Technical Specifications.

The 3 torr value described above is consistent with methodology in NUREG-1536, which references PNL-6365<sup>1</sup>. Moisture removal is inherent in the vacuum drying process, and levels at or below those evaluated in PNL-6365 are expected if the vacuum drying is performed as described in the SAR. This will serve to reduce the amount of oxidants to below the levels where significant cladding degradation is expected.

The MPC is backfilled with helium on top of the spent fuel pool water for applicable leak testing and then filled with water for the hydrostatic test. After the hydrostatic test, nitrogen or helium is used to force the spent fuel pool water from the cask, effectively removing contaminants. The cask is then re-evacuated using the VDS, tested to ensure ability to maintain a vacuum as described above, and backfilled with helium before final closure. A suitable inert cover gas ( $\geq 99.995\%$  pure helium) is specified to minimize this source of contaminants in accordance with the recommendations of PNL-6365. The operating procedures provide for repetition of the evacuation and repressurization cycles using the VDS if the vacuum is lost during the vacuum drying process, as could occur during loading.

### 8.1.4 Welding and Sealing

Section 8 of the SAR describes the use of the Automated Welding System Robot for the MPC lid closure weld. Remote welding helps ensure the dose to welders will be ALARA. Prior to welding, approximately 120 gallons of water are removed from the MPC to keep moisture away from the weld region. In addition, a vacuum pump is connected to keep moist air from condensing on the MPC lid weld area. Section 8 also describes the nondestructive examination (NDE) to be done on closure welds, including visual examination (VT), root and final pass dye penetrant examination (PT), ultrasonic examination (UT), leak testing, and hydrostatic tests. All NDE will be performed in accordance with the applicable sections of the ASME Code, Sections III and V, with the NDE requirements, applicable code, and acceptance criteria described in detail in SAR Table 9.1.3. Multi-layer PT examination may be used in lieu of UT on the MPC lid weld. Leak testing will be performed with a mass spectrometer leak detector (MSLD) in accordance with the manufacturer's instructions and ANSI N14.5<sup>2</sup>. The allowable leakage is  $5.0 \times 10^{-6}$  atm cm<sup>3</sup>/s helium, in accordance with Section 9 of the SAR and the Technical Specifications. The SAR also includes acceptable provisions for correction of weld defects

(repair in accordance with the site's approved weld repair procedure, additional NDE, and any additional drying and purging that may be necessary. Similar welding procedures are described for the MPC closure ring and vent and drain port cover plates. The staff recognizes that the MPC lid-to-shell weld may be performed manually. The staff concludes these procedures provide for acceptable welding and NDE of the closure welds.

## **8.2 Cask Handling and Storage Operations**

All accident events applicable to the transfer of the cask to the storage location are bounded by the design events described in Sections 2 and 11 of the SAR. All conditions for lifting and handling methods are bounded by the evaluations in Sections 3 and 4 of the SAR. Section 5.0 of the Technical Specifications require that a Cask Transport Evaluation Program be established, implemented, and maintained. The program provides a means for evaluating various on-site transport configurations and route conditions to ensure that the design basis drop limits are met.

Inspection, surveillance, and maintenance requirements that are applicable during ISFSI storage are discussed in Section 8.2 of the SAR. These surveillance requirements include a requirement to periodically verify the performance of the passive heat removal system. This requirement is also documented as LCO 3.1.2 of the Technical Specification. Maintenance requirements are discussed in Section 9 of the SAR. The staff determined that these discussions were acceptable.

Occupational and public exposure estimates are evaluated in Section 10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

## **8.3 Cask Unloading**

Detailed unloading procedures must be developed by each user.

The unloading procedures describe the general actions necessary to prepare the MPC for unloading in a reactor spent fuel pool, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC transfer cask and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations.

### **8.3.1 Cooling Venting & Reflooding**

The MPC is cooled using a closed-loop heat exchanger to reduce the MPC internal temperature to allow water flooding. The cool-down process utilizes helium to gradually reduce the cladding temperature to less than 200° F such that the MPC may be flooded with water without thermally shocking the fuel assemblies or over-pressurizing the MPC from the formation of steam.



Procedures for obtaining a gas sample are included to provide for assessment of the condition of the fuel assembly cladding. This allows for detection of potentially damaged or oxidized fuel. The procedures include ALARA caution steps to prevent the possible spread of contamination.

### **8.3.2 ALARA**

The unloading procedures incorporate general ALARA principles and practices. ALARA practices include periodic monitoring of dose rates, the use of annulus sealing equipment to reduce occupational exposure and limit MPC contamination, gas sampling of the MPC volume to identify potential clad damage, the use of temporary shielding, and the use of special tools to reduce occupational exposure. ALARA principles include warnings and notes that precede steps and identify potential radiological hazards. Each cask user will need to develop detailed unloading procedures that incorporate the ALARA objectives of their site-specific radiation protection program.

### **8.3.3 Fuel Crud**

The ALARA practices and procedures provide for the mitigation of the possibility of dispersal of fuel crud particulate material. However, experience with wet unloading of BWR fuel after transportation has involved handling significant amounts of crud. This fine crud includes  $^{60}\text{Co}$  and  $^{55}\text{Fe}$  and will remain suspended in water or air for extended periods. The dry cask reflood process during unloading of BWR fuel has the potential to disperse crud into the fuel transfer pool and the pool area atmosphere thereby creating airborne exposure and personnel contamination hazards. Therefore, detailed procedures incorporating provisions to mitigate the possibility of fuel crud particulate dispersal must be developed by each cask user.

## **8.4 Evaluation Findings**

- F8.1** The HI-STORM 100 Cask System can be wet loaded and unloaded. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.
- F8.2** The bolted closure plate and welded MPC of the cask allow retrieval of the spent fuel for further processing or disposal as required.
- F8.3** 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.
- F8.4** 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.
- F8.5** 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.

- F8.6** 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.
- F8.7** The staff concludes that the generic procedures and guidance for the operation of the HI-STORM 100 Cask 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.

## **8.5 References**

1. R.W., Knoll, et al., Pacific Northwest Laboratory, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, DE88 003983, November 1987.
2. American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," ANSI N14.5-1987, January 1987.

## **9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM**

The objective of the review of the acceptance tests and maintenance program is to ensure that Holtec's SAR includes the appropriate acceptance tests and maintenance programs for the HI-STORM 100 Cask System.

### **9.1 Acceptance Tests**

The acceptance tests inspections to be performed on the HI-STORM 100 Cask System are discussed in detail in Section 9.1 of the SAR. The testing and inspection acceptance criteria for the MPC, the HI-STORM 100 overpack, and the HI-TRAC transfer cask are listed in SAR Tables 9.1.1, 9.1.2, and 9.1.3, respectively. These inspections and tests are intended to demonstrate that the HI-STORM 100 Cask System has been fabricated, assembled, and examined in accordance with the design criteria given in Section 2 of the SAR.

The following is a summary of the HI-STORM 100 Cask System acceptance tests.

#### **9.1.1 Visual and Nondestructive Examination Inspections**

Nondestructive examination (NDE) commitments for the HI-STORM 100 Cask System are given in Table 9.1.3 of the SAR. This table specifies the NDE location, examination type, applicable Code, and acceptance criteria.

The MPC confinement boundary is fabricated and inspected in accordance with ASME Code, Section III, Subsection NB, with certain exceptions listed in the SAR. The MPC lid-to-shell and closure welds are not full penetration welds. The vent and drain port welds and closure ring welds will be examined by the PT method. ASME code exceptions are contained in Section 3.3 of Appendix B to the Certificate of Compliance. The lid and closure rings are welded independently to provide a redundant seal. Also, the MPC lid to the MPC closure ring weld is examined by the PT method to ensure acceptable weld integrity.

The NDE of weldments is well-characterized on drawings, using standard NDE symbols and/or notations in accordance with AWS 2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination." Fabrication inspections include VT, PT, magnetic particle, UT, and radiographic examinations (RT), as applicable.

Structural and confinement boundary weld examinations and acceptance criteria meet the applicable requirements of ASME Code, Section III. Volumetric examination of the MPC lid closure weld will be done via UT\* with acceptance criteria per ASME Section III, 1995 Edition, with 1997 Addenda, Paragraph NB-5332. Additional PT, leak testing, and VTs are also performed on the MPC lid to shell closure weld to ensure structural integrity and ability to perform its confinement function. The staff finds that this combination of examinations meets

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\* Multi-layer PT examination is permitted in lieu of UT on the MPC lid weld.

the staff's position that the closure weld for the outer cover plate may be inspected using either volumetric or multiple-pass dye penetrant techniques subject to the following conditions:

- 1) PT may only be used in lieu of volumetric examination only on austenitic stainless steels. PT should be done in accordance with ASME Code, Section V, Article 6, "Liquid Penetrant Examination."
- 2) If PT alone is used, at a minimum, it must include the root and final layers and sufficient intermediate layers to detect critical flaws.
- 3) For multiple layer PT, the maximum undetectable flaw size is less than the critical flaw size. The critical flaw size shall be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. Flaws in austenitic stainless steel are not expected to exceed the thickness of one weld bead.
- 4) The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code, Section III, NB-5350 for PT and NB-5332 for UT.
- 5) If PT alone is used, a design stress-reduction factor of 0.8 or less must be applied to the weld design.
- 6) The results of the PT examination, including all relevant indications, shall be made a permanent part of the licensee's records by video, photographic, or other means providing a retrievable record of weld integrity. Video or photographic records should be taken during the final interpretation period described in ASME Code, Section V, Article 6, T-676.

Welds which do not meet the acceptance criteria will be repaired and reexamined in accordance with the original examination method and associated acceptance criteria. The applicant committed to ASME Code sections for material procurement, design, fabrication, and inspection for the MPC confinement boundary, fuel baskets, fuel basket supports, damaged fuel container, overpack helium retention boundary, overpack components, and trunnions in Table 2.2.7 of the SAR. The staff concludes the applicant's choice of ASME Code Section III, Subsections NB, NF, and NG as described in Table 2.2.7 is appropriate.

### **9.1.2 Structural and Pressure Tests**

Structural and pressure tests are subdivided into four areas: (1) the transfer cask lifting trunnions, (2) hydrostatic testing of the transfer cask water jacket and the MPC confinement boundary, (3) materials testing, and (4) leakage testing. All testing must be performed in accordance with written and approved procedures. The test results must be documented and the documentation will become part of the final quality documentation package.

### **9.1.2.1 Lifting Trunnions**

To ensure that the lifting trunnions do not have any hidden material flaws and that the trunnions are properly installed, the trunnions are tested at 300% of the maximum design lifting load (i.e., 3x250,000 lbs = 750,000 lbs for the 125-ton HI-TRAC and 3x200,000 lbs = 600,000 lbs for the 100-ton HI-TRAC) in accordance with ANSI N14.6. Any evidence of deformation, distortion or cracking of the trunnion or the adjacent HI-TRAC transfer cask areas will require replacement of the trunnion and/or repair of the transfer cask. Following any repairs and/or replacements, the load testing will be repeated and components reexamined.

### **9.1.2.2 Hydrostatic Testing**

#### **a. HI-TRAC Transfer Cask Water Jacket**

The water jackets on the 125-ton HI-TRAC and the 100-ton HI-TRAC will be hydrostatically tested to 75 psig (+3, -0) and 71 psig (+3, -0), respectively. The test will be performed after the water jacket has been welded together. After completion of the hydrostatic test, the water jacket exterior will be visually examined for cracks or deformation. Also, accessible welds will be liquid penetrant or magnetic particle tested.

#### **b. MPC Confinement Boundary**

Hydrostatic testing of the MPC confinement boundary will be performed in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000, when field welding of the MPC lid-to-shell is completed. The hydrostatic pressure for the test is 125 psig, +5,-0 psig, which is 125% of the design pressure of 100 psig. Following completion of the 10-minute hold period at the hydrostatic test pressure, and while maintaining a minimum test pressure of 125 psig, the surface of the MPC lid-to-shell weld will be visually examined for leakage and then reexamined by PT. If a leak is discovered, the test pressure will be reduced, the MPC cavity water level lowered, the MPC cavity vented, and the weld will be examined to determine the cause of the leakage and/or cracking. Repairs to the lid-to-shell weld will be performed in accordance with an approved written procedure prepared in accordance with the ASME Section III, Subsection NB, NB-4450.

### **9.1.2.3 Materials Testing**

The majority of materials in the transfer cask and a portion of the materials in the overpack are ferritic steels. Ferritic steels used in the transfer cask and overpack are tested to assure that these materials are not subjected to brittle fracture failures.

The transfer cask and overpack materials are Charpy V-notch tested in accordance with the ASME Code, Section III, Subsection NF, Articles NF-2300 and NF-2430. Table 3.1.18 in Section 3.1 of the SAR provides the fracture toughness test criteria and temperature for the Charpy V-notch test.

The overpack concrete is mixed, poured, and tested as described in Appendix 1.D of the SAR. The testing shall verify that the composition, compressive strength, and density of the concrete meet design requirements. Concrete testing shall be performed for each lot of concrete and shall comply with ACI 349.

#### **9.1.2.4 Leakage Testing**

The MPC confinement boundary is leak tested with a helium mass spectrometer leak detector in accordance with ANSI N14.5. The MPC design helium leak rate is  $5 \times 10^{-6}$  atm-cm<sup>3</sup>/sec helium. The applicant committed to using an MSLD with a sensitivity of  $2.5 \times 10^{-6}$  atm-cm<sup>3</sup>/sec. The staff finds this practice acceptable. The staff concludes the leakage criteria meets or exceeds those specified in the principal design criteria in Section 2 of the SAR.

#### **9.1.3 Shielding Tests**

Fabrication and testing controls for each shielding material are described in Section 9.1.2.3 and 9.1.5.1 of the SAR. The concrete utilized in the construction of the HI-STROM overpack shall be mixed, poured, and tested in accordance with procedures in SAR Appendix 1.D and dimensions in the Design Drawings. Concrete testing shall be performed for each lot of concrete and comply with ACI 349 and test specimens shall be in accordance with ASTM C39. The installation of the lead shield in the HI-TRAC transfer cask designs shall be performed using written and qualified procedures in order to ensure voids are minimized. Each lot of lead shall be tested for chemical composition. The installation of the Holtite-A neutron shielding material will be performed according to written and qualified procedures. Each lot of the Holtite-A material will be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density meet the requirements in the Bill of Materials.

The effectiveness of the lead pours in the HI-TRAC transfer cask body, pool lid, and transfer lid doors shall be verified during fabrication by performing gamma scanning on all accessible surfaces of the cask in the lead pour region. Gamma scanning shall be performed in accordance with written and approved procedures. After first loading of each HI-TRAC transfer cask and HI-STORM 100 overpack, radiation measurements will be performed to verify shielding effectiveness and to verify compliance with dose limits in the Technical Specifications. Neutron and gamma effectiveness tests will be performed using approved written procedures and with calibrated neutron and gamma dose meters.

The staff reviewed the shielding fabrication testing and controls and effectiveness tests and found them acceptable. Each cask user will need to develop site-specific, detailed tests that incorporate the shielding effectiveness tests described in this section.

#### **9.1.4 Neutron Absorber Tests**

After manufacturing, a statistical sample of each lot of Boral is tested using wet chemistry and/or neutron attenuation techniques to verify the minimum <sup>10</sup>B content at the ends of the Boral panel. The minimum allowable <sup>10</sup>B content is 0.0267 g/cm<sup>2</sup> for the MPC-24 Boral panels, 0.0372 g/cm<sup>2</sup> for the MPC-68 Boral panels, and 0.01 g/cm<sup>2</sup> for the MPC-68F Boral panels. Any panel with a

$^{10}\text{B}$  loading less than the minimum allowed will be rejected. Tests will be performed using written and approved procedures. Results will be documented and become part of the HI-STORM 100 Cask System quality records documentation package.

The staff's acceptance of the neutron absorber test described above is based, in part, on the fact that the criticality analyses assumed only 75% of the minimum required  $^{10}\text{B}$  content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

Installation of the Boral panels into the fuel basket shall be performed in accordance with written and approved instructions. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a Boral panel in accordance with Holtec International Drawing Nos. 1395 and 1401.

### **9.1.5 Thermal Tests**

Following the loading and placement on the storage pad of the first HI-STORM 100 Cask System placed in service, the operability of its natural convective cooling will be verified by performing an air temperature rise test. This test is described in operating procedures, in Section 8.1 of the SAR. In addition, the overpack air inlet and outlet vents are periodically surveyed or, optionally, an overpack air temperature program is implemented to verify continued operability of the heat removal system.

### **9.1.6 Cask Identification**

The cask shall be marked with a model number, unique identification number, and empty weight. This information will appear on a data plate, which is detailed in drawings in Section 1.5 of the SAR. In addition, the exterior of shielding casks or other structures that may hold the confinement cask while it is in storage shall be marked. This marking provides a unique, permanent, and visible number to permit identification of the cask stored therein.

## **9.2 Maintenance Program**

### **9.2.1 Inspection**

The HI-STORM 100 Cask System maintenance program schedule is described in Table 9.2.1 of the SAR. The program includes a schedule for initial and periodic visual inspections, shielding effectiveness tests, and relief valve maintenance. The staff concludes these inspections are acceptable for initial and continued operation of the HI-STORM 100 Cask System. As discussed in Section 6 of the SER, the SAR does not include procedures for periodic testing of neutron poison (Boral) effectiveness. The material has a proven history of nuclear service in various reactors and in other transportation and storage casks; therefore, periodic testing is not necessary. The staff concludes that the acceptance inspections and tests are sufficient.

### **9.3 Evaluation Findings**

- F9.1.** Section 9.1 of the SAR describes the applicant's proposed program for preoperational testing and initial operations of the HI-STORM 100 Cask System. Section 9.2 discusses the proposed maintenance program.
- F9.2** Structures, systems, and components important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Tables 2.2.6 and 8.1.6 of the SAR identify the safety importance of structures, systems, and components. Tables 2.2.6 and 2.2.7 present the applicable standards for their design, fabrication, and testing.
- F9.3** The applicant/certificate holder/licensee will examine and/or test the HI-STORM 100 Cask System to ensure that it does not exhibit any defects that could significantly reduce its confinement and shielding effectiveness. Section 9.1 of the SAR describes this inspection and testing.
- F9.4** Holtec will mark the cask with a data plate indicating its model number, unique identification number, and empty weight. Drawing 1561, Sheet 5 of 5, in Section 1.5 of the SAR illustrates and describes this data plate.
- F9.5** The staff concludes that the acceptance tests and maintenance program for the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. 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.



## **10.0 RADIATION PROTECTION EVALUATION**

This section evaluates the capability of the radiation protection design features, design criteria, and the operating procedures of the HI-STORM 100 Cask System to meet regulatory dose requirements. 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).

Occupational exposures from the HI-STORM 100 Cask System are based on the direct radiation dose rates calculated in Section 5 of the SAR and the operating procedures discussed in Section 8 of the SAR. Doses to individuals beyond the controlled area boundary (members of the public) are determined from the direct radiation (including skyshine) dose rates calculated in Section 5 of the SAR and the dose rates from design-basis atmospheric releases calculated in Section 7 of the SAR.

### **10.1 Radiation Protection Design Criteria and Design Features**

#### **10.1.1 Design Criteria**

Section 10.1.2 of the SAR defines the radiological protection design criteria as the limits and requirements in 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106, and the guidance in Regulatory Guide 8.8<sup>1</sup>. This is consistent with NRC guidance. 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. The Technical Specifications also establish surface dose limits on the concrete overpack and the 100-ton and 125-ton HI-TRAC transfer casks that are based on calculated dose rate values used to determine occupational and off-site exposures. The Technical Specifications also establish exterior contamination limits for the overpack and accessible portions of the MPC.

#### **10.1.2 Design Features**

Section 10.1.2 of the SAR presents radiological protection design features which provide radiation protection to operational personnel and members of the public. These radiation protection design features include the following:

- the thick-walled concrete overpack that provides shielding of gamma and neutron radiation
- the thick MPC lid that provides occupational shielding during loading and unloading procedures
- the confinement system that consists of multiple welded barriers to prevent atmospheric release of radionuclides and is designed to maintain confinement of fuel during accident conditions

- the system designs optimize the placement of temporary shielding and maintain water in the MPC and transfer cask annulus space to reduce dose rates during closure operations
- the low-maintenance design that reduces occupational exposures during ISFSI operation
- the implementation of ALARA principles into the cask design and operating procedures that reduce occupational exposures
- the smooth surfaces that minimize decontamination time and the overpack annulus seal that prevents contamination of the MPC exterior

Section 10.1.2 of the SAR also discusses design features that address process instrumentation and controls, control of airborne contaminants, decontamination, radiation monitoring, and other ALARA considerations. In addition, Sections 10.1.3 and 10.1.4 of the SAR address operational considerations and describe auxiliary shielding devices to minimize occupational and public doses. Section 10.1.4 of the SAR specifies each shielding device as either mandatory or optional for both the 100-ton and 125-ton HI-TRAC. The annulus shield, vent duct shield inserts and the transfer step are auxiliary shielding devices required to maintain exposures ALARA for the HI-STORM 100 system using either the 100-ton or 125-ton HI-TRAC transfer casks. The temporary shield ring is also required when using the 100-ton HI-TRAC.

The NRC staff evaluated the radiation protection design features and design criteria for the HI-STORM 100 system and found them acceptable. The SAR analysis provides reasonable assurance that use of the HI-STORM 100 storage cask can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Sections 5, 7, and 8 of the SER discuss staff evaluations of the shielding features, confinement features, and operating procedures, respectively. Section 11 of the SER discusses staff evaluations of the capability of shielding and confinement features during off-normal and accident conditions.

## **10.2 ALARA**

Section 10.1 of the SAR presents evidence that the HI-STORM 100 radiation protection design features and design criteria address ALARA requirements, consistent with 10 CFR Part 20 and Regulatory Guides 8.8<sup>1</sup> and 8.10<sup>2</sup>. The SAR states each site licensee will apply its existing site-specific ALARA policies, procedures, and practices for cask operations to ensure that personnel exposure requirements in 10 CFR Part 20 are met. Because the 100-ton HI-TRAC transfer cask contains less shielding than the 125-ton HI-TRAC, occupational exposure rates may be significantly higher from the 100-ton HI-TRAC. As stated in Section 2.0.3 of the SAR, the ALARA considerations dictate the general licensee should use the 125-ton HI-TRAC provided the licensee is capable of utilizing it. However, sites may not be capable of utilizing the 125-ton HI-TRAC due to crane capacity limitations, floor loading considerations, or space envelope limitations in the fuel pool or air lock. As with other dose reduction-based plant modifications, individual users who cannot accommodate the 125-ton HI-TRAC due to plant design limitations must perform a cost-benefit analysis of the modifications which would be necessary to use the 125-ton HI-TRAC. The cost of the modifications would be weighed against

the value of the projected reduction in radiation exposure and a decision made based on each plant's particular ALARA implementation philosophy.

The NRC staff evaluated the ALARA assessment of the HI-STORM 100 using both the 100-ton and 125-ton HI-TRAC and found it acceptable. Section 8 of the SER discusses the staff evaluation of the operating procedures with respect to ALARA principles and practices. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. In addition, conditions in the Technical Specifications establish limits for the surface dose rates and surface contamination on the HI-STORM 100.

### **10.3 Occupational Exposures**

Section 8 of the SAR discusses general operating procedures that general licensees will use for fuel loading, cask operation, and fuel unloading. Section 10.3 of the SAR discusses the estimated number of personnel, the estimated dose rates, and the estimated time for each task. The estimated occupational doses are based on the direct radiation calculations in Section 5 of the SAR and the generic operating procedures in Section 8 of the SAR. The estimated doses received by personnel during loading, storage, and unloading are presented in Tables 10.3.1, 10.3.2, and 10.3.3. The dose estimates assume the use of the temporary shield ring. Approximate locations of personnel during loading and unloading are depicted in Figures 10.3.1a through 10.3.1e. The dose estimates indicate that the total occupational dose in loading a single cask with design-basis fuel (shielding) into the MPC-24 is approximately 1800 person-mrem using the 100-ton HI-TRAC and 600 person-mrem using the 125-ton HI-TRAC. The estimated occupational dose for unloading the cask is 950 person-mrem using the 100-ton HI-TRAC and 350 person-mrem using the 125-ton HI-TRAC. The estimated occupational exposure from transferring the MPC from a HI-STAR 100 transportation cask into the overpack is approximately 650 person-mrem using the 100-ton HI-TRAC and 325 person-mrem using the 125-ton HI-TRAC. The yearly estimated doses for cask surveillance and cask maintenance of the HI-STORM 100 are approximately 90 person-mrem and 300 person-mrem, respectively.

The NRC staff reviewed the overall occupational dose estimates and found them acceptable. The occupational exposure dose 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 cask depending on site-specific operating conditions, including detailed procedures and special measures taken to maintain exposures ALARA, and evolving experience with the HI-STORM 100 system. 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. Staff evaluation of the operating procedures is presented in Section 8 of the SER.

### **10.4 Public Exposures From Normal and Off-Normal Conditions**

Section 10.4.1 of the SAR summarizes the calculated dose rates to individuals beyond the controlled area (members of the public), as presented in Sections 5 and 7 of the SAR. The SAR evaluates and concludes that the confinement functions of the MPC are not affected by normal

and off-normal conditions. While no effluents are expected, Section 7.2 of the SAR presents dose rate calculations for normal and off-normal conditions assuming a continuous atmospheric release of radionuclides based on an MPC maximum leak rate. The SAR analyses indicates yearly off-site exposures are negligible for normal and off-normal conditions.

Section 5.4.3 of the SAR presents calculated direct radiation dose rates at distances beyond 100 meters from sample cask-array configurations loaded with design basis fuel. Figure 5.1.3 depicts estimated dose rate versus distance curves for various array configurations. Table 10.4.1 specifies distances at which the regulatory design limit of 25 mrem/yr can be achieved for various array configurations. Table 10.4.2 presents estimated dose rates at various distances from a 100-ton HI-TRAC transfer cask loaded with design basis fuel. As shown in Table 10.4.1, a single cask loaded with design-basis fuel is below regulatory limits at approximately 200 meters. A two-by-five cask array is below regulatory limits at approximately 350 meters.

The staff evaluated the public dose estimates from direct radiation and the assumed atmospheric release from normal and off-normal (anticipated occurrences) conditions and found them acceptable. A discussion of the staff's evaluation and confirmatory analysis of the shielding and confinement dose calculations are presented in Sections 5 and 7 of the SER. The staff concludes that the calculated dose rates from design-basis confinement releases are insignificant compared to the dose rates from direct radiation. Therefore, direct radiation (including skyshine) is the primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions.

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 100 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 individuals beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configuration, topography, demographics, atmospheric conditions, and use of engineered features (e.g., berm). 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 72.104(a) is the responsibility of each applicant for a site license.

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 by 10 CFR Part 20, Subpart D, by evaluations and measurements.

A license condition, in Appendix B of the Certificate of Compliance, has been included regarding engineered features used for radiological protection. The license condition states that engineering features (e.g., berms and shield walls) used to ensure compliance with 10 CFR 72.104(a) are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

## **10.5 Public Exposures From Design-Basis Accidents and Natural Phenomena Events**

### **10.5.1 Design-Basis Public Exposures**

Section 10.4.2 of the SAR summarizes the calculated dose rates (from Sections 5 and 7) for accident conditions and natural phenomena events to individuals beyond the controlled area (members of the public). The SAR evaluates and concludes that the confinement function of the MPC is not affected by design-basis accidents or natural phenomena events. Section 7.3 of the SAR presents dose rate calculations for an assumed 30-day atmospheric release of radionuclides based on 100% fuel failure after the design basis accident and the maximum MPC leak rate. The maximum estimated dose at 100 meters is approximately 45 mrem for the 30-day atmospheric release. Specific organ doses are also within the regulatory limits specified in 10 CFR 72.106.

The SAR analysis indicates the worst-case shielding consequences for the concrete overpack results in a negligible effect on dose at the controlled area boundary. The worst case shielding consequence for the HI-TRAC transfer casks is complete loss of the 100-ton HI-TRAC water jacket. The estimated dose rate at 100 meters is less than 1 mrem/hr and the calculated accident dose for an assumed 30-day accident is less than the 5 rem regulatory limit specified in 10 CFR 72.106(b). Section 11 of the SAR discusses recovery actions for each design-basis accident. Recovery actions include installation of temporary shielding, if necessary, to mitigate potential off-site exposures.

The staff evaluated the public dose estimates from direct radiation and atmospheric release from accident conditions and natural phenomena events and found them acceptable. A discussion of the staff's evaluation and confirmatory analysis of the shielding and confinement dose calculations are presented in Sections 5 and 7 of the SER, respectively. A discussion of the staff's evaluation of the accident conditions and recovery actions are presented in Section 11 of the SER. The staff has reasonable assurance that the combined effects of direct radiation and atmospheric releases from bounding design-basis accidents and natural phenomena will be below the regulatory limits for the whole body and specific organs in 10 CFR 72.106(b).

## **10.6 Evaluation Findings**

**F10.1** The SAR sufficiently describes radiation protection design bases and design criteria for the structures, systems, and components important to safety.

**F10.2** 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.

**F10.3** The HI-STORM 100 Cask System is designed to provide redundant sealing of confinement systems.

**F10.4** The HI-STORM 100 Cask System is designed to facilitate decontamination to the extent practicable.

- F10.5** The SAR adequately evaluates the HI-STORM 100 Cask System and its systems important to safety, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.
- F10.6** The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.
- F10.7** Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.06 are the responsibility of the site licensee. The HI-STORM 100 Cask System is designed to assist in meeting these requirements.
- F10.8** The staff concludes that the design of the radiation protection system for the HI-STORM 100 Cask System is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM 100 Cask System will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

## **10.7 References**

1. U.S. Nuclear Regulatory Commission, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.
2. U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, May 1997.

## **11.0 ACCIDENT ANALYSIS EVALUATION**

The purpose of the review of the accident analyses is to evaluate the applicant's identification and analysis of hazards, as well as the summary analysis of system responses to both off-normal and accident or design-basis events. This ensures that the applicant has conducted thorough accident analyses, as reflected by the following factors:

1. identified all credible accidents
2. provided complete information in the SAR
3. analyzed the safety performance of the cask system in each review area
4. fulfilled all applicable regulatory requirements

### **11.1 Off-Normal Events and Conditions**

Off-normal conditions are Design Event II as defined in ANSI/ANS 57.9-1984. These events can be expected to occur with moderate frequency or on the order of once per year. Table 2.2.13 of the SAR outlines the notation for design loadings for normal, off-normal, and accident conditions. Table 2.2.14 defines the applicable load cases and load combinations for each component. For off-normal events concerning the HI-STORM 100 overpack, the SAR considered deadweight, off-normal operating temperature, and handling loads. For the HI-TRAC transfer cask, off-normal operating temperatures and off-normal handling are also addressed. The NRC staff reviewed the analyses for these conditions and found them to be acceptable. There is no adverse impact on the cask integrity from any off-normal event.

#### **11.1.1 Off-normal Environmental Condition**

The applicant analyzed the impact of off-normal ambient temperatures of -40°F to 100°F on the calculated component temperatures using the same computer code and methodology as was used in the normal condition thermal analyses. The thermal results for this off-normal event show that all calculated material temperatures are less than their respective short term temperature limits.

Use of the HI-STORM 100 Cask System for storage is restricted to sites with a lower bound temperature of -40°F. For ambient temperatures between 0°F and 32°F, a 25% ethylene glycol solution is added to the demineralized water in the HI-TRAC water jacket to prevent freezing. Any handling of a loaded overpack or transfer cask, including all lifting operations, is restricted to a minimum of 0°F ambient temperature. This is to prevent the possibility of brittle fracture of the ferritic steel parts of the overpack and the HI-TRAC transfer cask. Staff evaluation of the applicant's analysis accepts the thermal results as bounding for this scenario.

#### **11.1.2 Partial Blockage of the Air Inlets**

The applicant analyzed the effect of partial blockage of the air inlets on calculated component temperatures using the same computer code and methodology as was used in the normal condition thermal analyses. The applicant considered two cases: two of the four air inlets completely blocked and three of the four air inlets completely blocked. The ambient temperature

for the off-normal condition was 80°F with full insolation. The thermal results for this off-normal event show that all calculated material temperatures are less than their respective short term temperature limits. Staff evaluation of the applicant's analysis accepts the thermal results as bounding for this scenario.

### **11.1.3 Off-Normal Handling of HI-TRAC**

During upending and or downending of the transfer cask, the total lift weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions supports approximately one-quarter of the total weight. If the lifting device goes slack, the total weight would be applied to the lower pocket trunnions only. Under this off-normal condition, the pocket trunnions would each be required to support one-half of the total weight of the transfer cask. The applicant evaluated this condition and demonstrated that the pocket trunnions have sufficient strength to support the increased load. All stresses in the pocket trunnions are within the allowable limits. This analysis is provided in Appendices 3.AA and 3.AI of the SAR.

### **11.1.4 Off-normal Pressures of MPC**

Since the MPC remains in tact during the off-normal handling events, the pressure within the MPC remains within the allowable limits.

### **11.1.5 Dose Limits for Off-Normal Events**

Section 11.1 of the SAR examines the dose consequences for the identified off-normal events. The SAR determined that the confinement function of the MPC is not affected by off-normal conditions. However, Section 7 of the SAR analyzes a continuous atmospheric release during normal conditions. The SAR shows that off-normal events do not result in any significant dose consequences and that direct radiation conditions are the same as normal conditions analyzed in Section 5 of the SAR. A summary of the estimated occupational and public doses from off-normal events is presented in Section 10 of the SAR.

The staff reviewed the off-normal event analyses with respect to 10 CFR 72.104(a) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual beyond the controlled area will not exceed the limits in 10 CFR 72.104(a) during off-normal conditions (anticipated occurrences). Sections 5, 7, and 10 of the SER further evaluate the radiological doses during off-normal events.

## **11.2 Accident-Level Events and Conditions**

Accident level events and conditions are Design Events III and IV as defined in ANSI/ANS 57.9-1984. They include natural phenomena and human-induced low-probability events. The applicant provided analyses to demonstrate design adequacy for the accident-level events discussed below. The staff concurs that all accident-level events and conditions have been identified and all potential safety consequences considered.



## **11.2.1 100% Blockage of Air Inlets**

### **11.2.1.1 Cause of 100% Blockage**

The likely cause of a full blockage of the air inlets is a cask burial associated with a seismic event or landslide. The event is analyzed as a bounding condition and is not considered credible.

### **11.2.1.2 Consequence of 100% Blockage**

The applicant analyzed the effect of 100% blockage of the air inlets on calculated component temperatures using the same computer code and methodology as was used in the normal condition thermal analyses. This analysis was used to demonstrate the time period at which any short-term material temperature limit is exceeded. The applicant's analysis indicated that the peak concrete temperature reaches its short-term limit of 350°F at approximately 33 hours after the 100% blockage of all air inlets. The MPC confinement boundary and fuel cladding temperatures remain below their short-term temperature limits at 72 hours. Based on this results, the Technical Specifications require a 24-hour periodic surveillance to verify that the overpack inlet and outlet ducts are free of blockage.

The applicant also calculated the peak MPC pressure during this event and illustrated that the resultant pressure is 120.4 psig. This falls below the design pressure of 125 psig.

The staff accepts the applicant's analysis of this accident and its thermal consequences.

## **11.2.2 HI-TRAC/MPC Horizontal Drop**

During the operation, the loaded HI-TRAC transfer cask can be moved to the ISFSI in vertical or horizontal position. The loaded HI-TRAC transfer cask is typically transported by a heavy-haul vehicle that cradles HI-TRAC horizontally or by a device with redundant drop protection that holds HI-TRAC vertically. A cask drop from the horizontal handling height limit is a credible accident, and a vertical drop of the loaded HI-TRAC transfer cask is not a credible accident. The handling accident analysis evaluates the effects of dropping the loaded HI-TRAC in the horizontal position. The analysis of the handling accident is provided in Section 3. The analysis shows that limited localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket may occur as a result of the handling accident. The HI-TRAC transfer cask top lid and transfer lid housing are demonstrated to remain attached by withstanding the maximum deceleration of 45 g. The transfer lid doors are also shown to remain closed during drop. Limiting the inertia loading to 60 g or less ensures the fuel cladding remains intact. The applicant has demonstrated that the 45 g limit for the HI-TRAC transfer cask is met and ensures that the fuel cladding remains intact. The structural evaluation of the MPC for 45 g is provided in Section 3.4 of the SAR. As discussed in this section, the MPC stresses resulting from the HI-TRAC side drop are all within allowable values. Analysis of the lead in the transfer cask is performed in Appendix 3.F of the SAR. The analysis shows that there is no appreciable change in the lead shielding.

The temperatures of the MPC in the HI-TRAC transfer cask due to loss of water in the water jacket are listed on Table 11-1.

**Table 11-1**  
**Maximum Temperatures Caused by Loss of Water From the HI-TRAC Water Jacket (°F)**

<b>Temperature Location</b>	<b>Normal</b>	<b>Calculated Without Water in the Water Jacket</b>	<b>Accident Condition Design Temperature Limits (°F)</b>
Fuel Cladding	902	914	1058 short-term
MPC Basket	884	896	950 short-term
MPC Basket Periphery	527	541	950 short-term
MPC Outer Shell Surface	459	476	775 short-term
HI-TRAC Inner Shell	323	345	600 short-term
HI-TRAC Water Jacket Inner Surface	315	329	350 long-term
HI-TRAC Enclosure Shell Outer Surface	223	221	350 long-term
Axial Neutron Shield	175	177	300 long-term

For this accident condition, the temperatures fall below the limits for accident conditions.

### **11.2.3 HI-STORM/MPC Overpack Vertical Drop**

#### **11.2.3.1 Cause of Vertical Drop**

Although a handling accident is unlikely, a drop event from the handling height is regarded as a required accident to be analyzed.

#### **11.2.3.2 Consequences of Vertical Drop**

The vertical drop accident, described in Section 3.4.10 of the SAR, is an event which assumes the longitudinal axis of the cask vertically impacts a reference pad from a height of 11 inches. The maximum acceleration of the loaded overpack due to the vertical drop was determined to be 44 g and is, therefore, bounded by the MPC design acceleration of 45 g. The staff evaluation of the analysis is given in Section 3 of this SER. The staff concluded that the drop analysis is

adequate for certification and licensing at those sites where it is shown that handling of the overpack will not be greater than 11 inches and that the receiving surface hardness does not exceed that analyzed in the SAR.

Under this postulated accident, all stresses remain within allowable values, thereby assuring that the confinement boundary remains in tact.

#### **11.2.4 HI-STORM 100 Storage Overpack Tipover**

##### **11.2.4.1 Cause of Tipover**

Although analyses have shown that the overpack will not tip over as a result of severe natural phenomena, such as earthquakes and tornadoes, a tipover analysis is required as a bounding design event to demonstrate the defense-in-depth of the design.

##### **11.2.4.2 Consequences of Tipover Accidents**

The tipover is described in Section 3.4.10 of the SAR. Analyses included a structural analysis of the tipover event, the determination of maximum accelerations that may be experienced, an analysis of the integrity of the overpack lid during the event, and an analysis of the studs securing the lid to the overpack. Staff review of the structural analyses are in Section 3 of this SER. The maximum acceleration of the MPC inside the overpack was shown to be 43.2 g as a result of the tipover event. This acceleration is bounded by the 45 g acceleration for which the MPC has been designed and analyzed. The structural analyses of tipover in the SAR concluded that the overpack would maintain safety, that the factors of safety of the most critically loaded elements would be above 1.0, and that deformations of the overpack would not impose loads on the MPC or impair retrievability following a tipover event. The staff review determined that the analytical approaches, computations, results, and acceptance criteria are acceptable.

#### **11.2.5 Burial Under Debris**

##### **11.2.5.1 Cause of Burial**

Natural phenomena that could lead to burial of the cask under man-made or earthen material.

##### **11.2.5.2 Consequences of Burial**

The applicant analyzed the effects of a postulated accident in which the cask is buried under debris which would act as an additional thermal resistance to heat removal from the cask surface as well as 100% blockage of all air inlets. This scenario satisfies the requirement of NUREG-1536 to perform an adiabatic heatup calculation. The thermal effect of debris was modeled as adiabatic insulation on the overpack along with complete air inlet blockage. The results of this analysis show that the short-term cladding temperature limit would not be reached until more than 100 hours. As in the case of the 100% air inlet blockage accident, the concrete short-term limit of 350°F would be expected to be reached at approximately 33 hours. This accident analysis demonstrates that the 24 hour surveillance interval for the cask air inlets is

acceptable for identifying and initiating corrective action in the case of a burial under debris accident so that no material short-term thermal limits are exceeded. The staff accepts the applicant's analysis of this accident and its thermal consequences.

## **11.2.6 Wind and Tornado Missiles**

### **11.2.6.1 Cause of Wind and Missiles**

The overpack and transfer cask will be placed on an unsheltered concrete pad at an ISFSI and will be subject to extreme weather conditions which could include extreme winds and missiles from a tornado.

### **11.2.6.2 Consequences of Wind and Tornado Missiles**

The applicant performed tornado and missile effects analyses which are discussed in Section 3.4.8 of the SAR. High-velocity wind from passing tornadoes will exert an external pressure load on the overpack and could also generate large missiles that have the potential for striking and damaging the overpack. Potential effects include cask tipover and penetration of the overpack.

Cask stability was evaluated for a design basis wind velocity of 360 mph and a pressure drop of 3 psi. The cask was shown to not tip over or slide as a result of the 360 mph wind.

Analyses showed that the loaded overpack may slide a short distance as a result of a missile impact. The applicant determined that the overpack will remain upright under simultaneous tornado wind and missile loadings and that the tornado missiles will not penetrate the overpack inner shell. No spalling of the concrete resulting from a missile impact is possible due to the design of the overpack. Staff review of tornado missile impacts is in Section 3 of the SER.

The analysis in Section 3.4 of the SAR show that tornado missiles do not penetrate the transfer cask. Similar to the overpack, the effect of a tornado missile impact on the transfer cask is limited to shielding damage.

## **11.2.7 Flood**

### **11.2.7.1 Causes of Floods**

A flood at an ISFSI may be caused by external events such as unusually high water from a river, dam break, seismic event, tsunami, and severe weather (e.g. hurricanes).

### **11.2.7.2 Consequences of Floods**

Flooding is one of the accidents that must be evaluated as required by 10 CFR 72.122(b)(2). The analysis of flood effects on the overpack are presented in SAR Section 3.4.6. The overpack was evaluated for a 125 ft static head of water and a flood water velocity of 15 ft/sec. The staff

review of flooding effects is discussed in Section 3 of this SER. The staff agrees that the overpack will neither slide nor overturn under the design.

The requirements of 10 CFR Part 72, Subpart E, with respect to flooding have been met.

### **11.2.8 Earthquake**

#### **11.2.8.1 Cause of Earthquake**

A postulated earthquake of the ISFSI site.

#### **11.2.8.2 Consequences of Earthquake**

The earthquake accident must be evaluated as required by 10 CFR 72.122(b)(2). The applicant performed a seismic analysis to determine the effects on the overpack of a design-basis earthquake which has ground accelerations as specified in the table in Section 3.4.7.1 of the SAR. The acceleration values were applied to the loaded overpack while in the storage mode to determine if the a tipover could occur. The seismic event is analyzed in SAR Section 3.4.7.1. The evaluation of this analysis is discussed in Section 3 of this SER. The staff concluded that the overpack would maintain stability under these applied loading conditions.

### **11.2.9 Confinement Boundary Leakage**

Resulting dosage to the general public due to confinement boundary leakage is analyzed in Section 7 of the SAR. The calculated maximum whole body dose is 44.1 mrem. The calculated dose to the thyroid is 4.1 mrem. These values are below the regulatory limit of 5 Rem.

### **11.2.10 Explosion**

#### **11.2.10.1 Cause of Explosion**

Explosions involving combustible materials shipped to reactor sites and on transportation links near nuclear power plant sites.

#### **11.2.10.2 Consequences of Explosions**

The analysis of explosion effects of the overpack are presented in SAR Section 3.4.7.2. The external pressure wave generated by a credible explosion is on the order of 5 psig. The structural evaluation is discussed in Section 3 of this SER. The staff agrees that the overpack will neither slide nor overturn as a result of the postulated explosion.

## **11.2.11 Lightning**

### **11.2.11.1 Cause of Lightning**

The overpack will be stored on an unsheltered ISFSI concrete pad, and there is the potential for lightning to strike the overpack.

### **11.2.11.2 Consequences of Lightning**

As discussed in Section 11.2.12 of the SAR, any lightning strike on the overpack will discharge through the steel shell of the overpack to the ground. Such an occurrence will have no adverse effect on the overpack.

## **11.2.12 Fire**

### **11.2.12.1 Causes of Fire**

A postulated rupture of the transporter vehicle fuel tank and subsequent ignition of the spilled fuel.

### **11.2.12.2 Consequences of Fire**

As described in SAR Section 11.2 an analysis was performed which assumed a bounding, hypothetical fire caused by 50 gallons of spilled diesel fuel which encompass the overpack and burn at a temperature of 1475°F for a period of 217 seconds. The staff review of this analysis is discussed in Section 4 of this SER. Suitably bounding assumptions were made regarding fire heat load, boundary conditions, and heat transfer coefficients. The ANSYS computer code was used to calculate the transient response of the overpack. The analysis showed that all component temperatures remained below their short-term thermal limits with the exception of the outer one-inch of the concrete which exceeds its short-term limit during the fire accident. NUREG-1536 accepts this consequence from a fire accident.

A fire analysis was also performed for the HI-TRAC transfer cask. Bounding parameters used in the analysis are listed in Table 11.2.3 of the SAR. The duration of the postulated fire was calculated as 4.8 minutes. This duration is limited by the 50-gallon diesel fuel container that the Certificate of Compliance limits to the site. The resulting MPC-24 and MPC-68 internal pressures rose to 70.3 and 70.7 psig, respectively. With the assumption of 100% rupture of the fuel, the MPC-24 and MPC-68 internal pressures were 114.9 and 106.3 psig. These pressures are below the design limit. The calculated peak clad, basket and MPC shell temperatures were 947°F, 572°F, and 504°F, respectively. Should the water jacket be postulated to fail and spill its content, then the temperatures would increase by less than 20°F. These conditions lie within the design temperatures limits. There are no structural consequences to the transfer cask as a result of the fire accident.

### **11.2.13 Extreme Environmental Conditions**

#### **11.2.13.1 Cause of Extreme Environmental Conditions**

Weather events which subject the overpack to a 125°F ambient temperature with full solar insolation and maximum heat load.

#### **11.2.13.2 Consequences of Extreme Environmental Conditions**

The applicant analyzed the impact of a postulated accident ambient temperature of 125°F, plus incident solar insolation, on the calculated component temperatures using the same computer code and methodology as was used in the normal condition thermal analyses. The thermal results for this accident show that all calculated material temperatures are less than their respective short term temperature limits. Staff evaluation of the applicant's analysis accepts the results as bounding for this scenario.

### **11.2.14 Dose Limits for Design-Basis Accidents and Natural Phenomena Events**

Section 11.2 of the SAR examines the dose consequences for the identified design-basis accidents and natural phenomena events. Section 11 of the SAR determined that the confinement function of the MPC is not affected during design-basis accident conditions. However, Section 7 of the SAR analyzes an assumed 30-day atmospheric release from a design-basis accident event and calculates a total dose of approximately 45 mrem. The SAR does not identify an accident that significantly degrades the shielding capability of the concrete overpack. The design-basis shielding degradation for HI-TRAC is assumed to be complete loss of the water jacket. Section 5.1.2 of the SAR shows that the maximum dose rate at one meter from the side of HI-TRAC is approximately 1090 mrem/hr. The one meter dose rate is extrapolated to 100 meters and is calculated to be approximately 0.8 mrem/hr. The SAR determined that the radiological exposure at 100 meters will not exceed the 10 CFR 72.106(b) whole body and organ-specific dose limits. The SAR also indicates the estimated occupational exposure for recovery of a damaged HI-TRAC is less than 2000 person-mrem and can meet 10 CFR Part 20 limits. A summary of the estimated doses for design-basis accidents is presented in Section 10 of the SAR.

The staff reviewed the design-basis accident analyses with respect to 10 CFR 72.106(b) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual beyond the controlled area boundary will not exceed the limits in 72.106(b) and occupational exposures from cask recovery will not exceed 10 CFR Part 20 requirements. Sections 5, 7, and 10 of the SER further evaluate the estimated radiological doses during accident conditions.

## **11.3 Criticality**

As discussed in SER Section 6, the applicant has shown, and the staff has verified, that the spent fuel remains subcritical ( $k_{\text{eff}} < 0.95$ ) under all credible normal, off-normal, and postulated accident events. The design-basis off-normal and accident events do not adversely affect the

design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the HI-STORM 100 Cask System design meets the “double contingency” requirements of 10 CFR 72.124(a).

#### **11.4 Post-Accident Recovery**

Section 11.2 of the SAR discusses corrective actions for each accident event. There are no credible, design-basis accident that would adversely affect the canister confinement boundary or significantly damage the cask system at a level that could result in undue risk to public health and safety.

The staff reviewed the design-basis accident analyses with respect to post-accident recovery and found them to be acceptable. The staff has reasonable assurance that the site licensee can recover the HI-STORM 100 from the analyzed design-basis accidents and that the generic corrective actions outlined in the SAR are appropriate to protect public health and safety.

#### **11.5 Evaluation Findings**

**F11.1** Structures, systems, and components of the HI-STORM 100 Cask System are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.

**F11.2** The spacing of casks, discussed in Section 1.4 of the SAR, ensures accessibility of the equipment and services required for emergency response.

**F11.3** The applicant has evaluated the HI-STORM 100 Cask System to demonstrate that it will reasonably maintain confinement of radioactive material under off-normal and credible accident conditions.

**F11.4** A design-basis accident or a natural phenomena event will not prevent the ready retrieval of spent fuel for further processing or disposal.

**F11.5** The spent fuel will be maintained in a subcritical condition under accident conditions.

**F11.6** Because instrumentation and control systems are not required, no instruments or control systems are required to remain operational under accident conditions.

**F11.7** The applicant has evaluated off-normal and design-basis accident conditions to demonstrate with reasonable assurance that the HI-STORM 100 Cask System radiation shielding and confinement features are sufficient to meet the requirements in 10 CFR 72.104(a) and 10 CFR 72.106(b).

**F11.8** Table 12.1 of the SER lists the Technical Specifications for the HI-STORM 100 Cask System. These Technical Specifications are further discussed in Section 12 of the SER and are appended to the Certificate of Compliance.



**F11.9** The staff concludes that the accident design criteria for the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

## **12.0 CONDITIONS FOR CASK USE —TECHNICAL SPECIFICATIONS**

The purpose of the review of the conditions for cask use is to determine whether the applicant has fully evaluated the Technical Specifications and to ensure that the SER incorporates any additional operating controls and limits that the staff deems necessary.

### **12.1 Conditions for Use**

The conditions for use of the HI-STORM 100 Cask System are fully defined in the Certificate of Compliance and the Technical Specifications and Approved Contents and Design Features specifications that are appended to it.

### **12.2 Technical Specifications**

Table 12-1 lists the Technical Specifications for the HI-STORM 100 Cask System. The staff has appended these Technical Specifications to the Certificate of Compliance for the HI-STORM 100 Cask System.

### **12.3 Evaluation Findings**

**F12.1** Table 12-1 of the SER lists the Technical Specifications for the HI-STORM 100 Cask System. These Technical Specifications are further discussed in Section 12 of the SAR and are part of the Certificate of Compliance.

**F.12.2** The staff concludes that the conditions for use of the HI-STORM 100 Cask System identify necessary Technical Specifications to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The Technical Specifications 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.

TABLE 12-1  
HI-STORM 100 CASK SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION
1.1	Definitions
1.2	Logical Connectors
1.3	Completion Times
1.4	Frequency
2.0	(intentionally left blank)
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY/SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SPENT FUEL STORAGE CASK (SFSC) Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2	SFSC Radiation Protection
3.2.1	TRANSFER CASK Average Surface Dose Rates
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	OVERPACK Average Surface Dose Rates
Table 3-1	MPC Model-Dependent Limits
4.0	(intentionally left blank)
5.0	ADMINISTRATIVE CONTROLS
5.1	Programs
5.1.1	Cask Transport Evaluation Program
5.1.2	Radioactive Effluent Control Program
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements

## **13.0 QUALITY ASSURANCE**

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.

### **13.1 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.

### **13.2 Evaluation Findings**

- F13.1** 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."
- F13.2** 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.
- F13.3** 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.

- F13.4** The QA program is well-documented and provides adequate control over activities affecting quality, as well as structures, systems, and components important to safety, consistent with their relative importance to safety (graded approach).
- F13.5** Holtec's QA program complies with the applicable NRC regulations and can be implemented for the design, fabrication, testing, modification, and use of the HI-STORM 100 Cask System.
- F13.6** This SAR can be referenced without further QA review in a license application to receive and store spent fuel under 10 CFR Part 72, provided that the applicant applies its NRC-approved QA program meeting the requirements of 10 CFR Part 50, Appendix B, to the design, construction, and use of structures, systems, and components of a spent fuel storage installation that are important to safety.

## **14.0 DECOMMISSIONING**

The purpose of the review of the conceptual decommissioning plan for the HI-STORM 100 Cask System is to ensure that it provides reasonable assurance that the owner of the cask can conduct decontamination and decommissioning in a manner that adequately protects the health and safety of the public. Nothing in this review considers, or involves the review of, or ultimate disposal of spent nuclear fuel.

### **14.1 Decommissioning Considerations**

The conceptual decommissioning plan for the HI-STORM 100 Cask System is provided in Section 2.4 of the SAR. While the applicant clearly anticipates that the HI-STORM 100 Cask System could be used as part of a final geologic disposal system, the ability to decommission the HI-STORM 100 Cask System is also considered. For example, Table 2.4.1 of the SAR provides the quantities of the major nuclides which the applicant has determined would exist after 40 years of irradiation of the MPC and concrete overpack. The material activation results presented in Table 2.4.1 confirm that total system activation is low. Therefore, the MPC and concrete overpack could be disposed in a near-surface facility as low specific activity material. The applicant states that both the overpack and the MPC can be decontaminated using existing mechanical or chemical methods.

### **14.2 Evaluation Findings**

- F14.1** Holtec's proposed cask design includes adequate provisions for decontamination and decommissioning. As discussed in Section 2.4 of the SAR, these provisions include facilitating decontamination of the HI-STORM 100 Cask System, if needed; storing the remaining components, if no waste facility is expected to be available; and disposing of any remaining low-level radioactive waste.
- F14.2** Section 2.4 of the SAR also presents information concerning the proposed practices and procedures for decontaminating the cask and disposing of residual radioactive materials after all spent fuel has been removed. This information provides reasonable assurance that the applicant will conduct decontamination and decommissioning in a manner that adequately protects public health and safety.
- F14.3** The staff concludes that the decommissioning considerations for the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72.

## **CONCLUSIONS**

The staff has reviewed Revision 10 to the Topical Safety Analysis Report for the HI-STORM 100 Cask System. Based on the statements and representations contained in the SAR, and the conditions given in the Certificate of Compliance, the staff concludes that the HI-STORM 100 Cask System meets the requirements of 10 CFR Part 72.

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