FINAL

SAFETY EVALUATION REPORT

DOCKET NO. 72-1014

HOLTEC INTERNATIONAL

HI-STORM 100 CASK SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1014

AMENDMENT NO. 5

TABLE OF CONTENTS

FINAL
SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HI-STORM 100 CASK SYSTEM
HOLTEC INTERNATIONAL
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 5

SUMMARY

By letter dated December 30, 2004, Holtec International (Holtec) submitted an application to the United States Nuclear Regulatory Commission (NRC) to amend Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Cask System (License Amendment Request 1014-3, Revision 0), in accordance with U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste and Reactor-Related Greater than Class C Waste," Title 10, Part 72 (10 CFR Part 72). This application was resubmitted February 22, 2005, to satisfy proprietary withholding requirements of 10 CFR 2.390, as the applicant had requested portions of the application be withheld from public disclosure.

The complexity of the material submitted, the uniqueness of the proposed underground system, and the knowledge of prior unresolved technical issues associated with the request for an increase in the licensed thermal capacity obligated the NRC staff, hereafter referred to as the staff, to perform a technical "acceptance review." As a result of issues identified during the staff's technical acceptance review Holtec requested that the staff suspend technical review in order to make improvements to the HI-STORM 100U Cask System Design. Holtec submitted revised License Amendment Request 1014-3 (LAR 1014-3) on May 16, 2005. By letter dated June 14, 2005, the staff informed Holtec that the revised LAR 1014-3 application contained sufficient information for the staff to begin a technical review.

The staff notified Holtec by letter dated November 18, 2006, that the review had been discontinued due to issues identified during the course of the technical review and the inability of the staff to reach any conclusions and findings based on the information provided by Holtec. Subsequently in a letter dated November 29, 2006, Holtec requested that the HI-STORM 100U design be withdrawn from consideration for approval. Holtec submitted LAR 1014-3, Revision 2, on December 22, 2006, removing all reference to the HI-STORM 100U design such that the staff could move forward with review of the request for an increase in the maximum thermal decay heat load and other changes to the CoC.

The application, as modified by the December 22, 2006, Revision 2, submittal and as supplemented by submittals dated March 20, 2007, March 30, 2007, May 4, 2007, May 22, 2007, June 15, 2007, July 17, 2007, and September 6, 2007, requested changes to the Certificate of Compliance (CoC), Technical Specifications (TS) and Final Safety Analysis Report (FSAR) to modify the HI-STORM 100 Cask System. The revised application proposed to increase the maximum licensed thermal capacity of the HI-STORM 100 Cask System to 34 kW (uniform loading) and 36.9 kW (regionalized loading). Holtec also removed reference to analyses that did not directly support the licensing basis. Specifically, the proposed changes include:

1. Deletion of the requirement to perform thermal validation tests on thermal systems.
2. An increase in the design basis maximum decay heat loads. Additionally a new decay heat regionalized scheme is introduced.
3. An increase in the maximum fuel assembly weight for BWR fuel in the Multi-Purpose Canister (MPC) -68 from 700 to 730 lb.
4. Changes to the assembly characteristics of Pressurized Water Reactor (PWR)16x16 fuel assemblies to be qualified for storage in the HI-STORM cask system which include an increase in maximum fuel assembly weight for PWR fuel for assemblies that do not require upper and lower fuel spacers, a change in the Fuel Rod Clad ID and a change in minimum Guide/Instrument Tube Thickness, and a change in minimum soluble boron concentration for Array Class 16x16A for all intact fuel assemblies.
5. A change in the fuel storage locations in the MPC-32 for fuel with Axial Power Shaping Rod Assemblies (APSRs) and in the fuel storage locations in the MPC-24, MPC-24E, and MPC-32 for fuel with Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), and Control Element Assemblies (CEAs);
6. Elimination of the restriction that fuel debris can only be loaded into the MPC-24EF, MPC-32F, MPC-68F, and MPC-68FF canisters.
7. Requirement that all MPC confinement boundary components and any MPC components exposed to spent fuel pool water or the ambient environment be made of stainless steel or, for MPC internals, neutron absorber or aluminum.
8. Addition of a threshold heat load below which operation of the Supplemental Cooling System (SCS) would not be required and modification of the design criteria to simplify the system.
9. Minor editorial changes to include clarification of the description of anchored casks, correction of typographical/editorial errors, clarification of the definitions of Loading Operations, Storage Operations, Transport Operations, Unloading Operations, Cask Loading Facility, and Transfer Cask in various locations throughout the CoC and FSAR.
10. Modification of the definition of non-fuel hardware to include the individual parts of items defined as non-fuel hardware.

This Safety Evaluation Report (SER) documents the review and evaluation of the proposed revised FSAR, supplemental materials, and proposed CoC changes, as revised December 22, 2006, and as supplemented by submittals dated March 20, 2007, March 30, 2007, and May 4, 2007. The proposed FSAR follows the format similar to that of the U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997 (NUREG-1536) with differences implemented for clarity and consistency.

The staff's evaluation 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. The staff's evaluation focused only on modifications requested in the amendment and did not reassess previously approved portions of the CoC, TS, and the FSAR or those areas of the FSAR modified by Holtec as allowed by 10 CFR 72.48.

Drawings for structures, systems, and components important to safety presented in Section 1.5 of the proposed FSAR were not reviewed for this amendment unless specifically referenced by other changes. Specific structures, systems, and components are evaluated in Sections 3 through 14 of this SER, as necessary.

Specifications for the spent fuel to be stored in the dry cask storage system are provided in Section 1.2.3 of the proposed FSAR. Detailed specifications for the spent fuel are presented in Section 2.1 of the proposed FSAR and Appendix B to the CoC.

The technical qualifications of the applicant to engage in the proposed activities were reviewed and approved previously for CoC 1014 and were not reviewed for this amendment.

The quality assurance (QA) program and implementing procedures are described in Chapter 13 of the proposed FSAR. Specific changes to the QA program and program description are evaluated in Section 13 of this SER.

1.0 GENERAL DESCRIPTION

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

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 MPC, the HI-TRAC transfer cask, and the HI-STORM 100 storage overpack.

1.1.1 Multi-Purpose Canister

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. The confinement boundary components are made entirely of stainless steel. All MPC components that may come into contact with spent fuel pool water or the ambient environment, with the exception of neutron absorber, aluminum seals on vent and drain port caps, and optional aluminum heat conduction elements in early-vintage MPCs, are constructed of stainless steel. 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 neutron absorbers, provides criticality control. There are eight approved MPC designs; MPC-24, MPC-24E, and MPC-24EF which can contain a maximum of 24 pressurized water reactor (PWR) fuel assemblies; the MPC-32 and MPC-32F which can contain a maximum of 32 PWR fuel assemblies; and the MPC-68, MPC-68F, and MPC-68FF which can contain a maximum of 68 boiling water reactor (BWR) fuel assemblies. Vibration suppressors are considered integral non-fuel hardware consisting of zircaloy or stainless steel tubes.

The MPC designs, called the MPC-24EF, MPC-32F, and MPC-68FF, contained features required to classify them as a secondary containment for permitting transportation of fuel debris per the requirements of 10 CFR Part 71. Changes to 10 CFR Part 71 eliminated the need for secondary containment of fuel debris, permitting the use of non-F type MPCs, i.e., MPC-24E, MPC-32, and MPC-68, for storage and transport of fuel debris. Any contents that used to require loading into an MPC-24EF, MPC-32F or MPC-68FF are now permissible for loading into an MPC-24E, MPC-32, or MPC-68 (including the MPC-68F). The NRC staff finds this change to be acceptable.

1.1.2 HI-TRAC Transfer Cask

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 HI-TRAC was previously reviewed and approved by the staff for the original application. No significant design changes were made to the HI-TRAC and as such the staff only reviewed the HI-TRAC with respect to whether it was affected by the modifications to the other components.

1.1.3 HI-STORM 100 Overpack

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. In addition to the HI-STORM 100 overpack, there are three additional variations including the HI-STORM 100S, HI-STORM 100A, and HI-STORM 100SA. The HI-STORM 100S is a shorter version of the HI-STORM 100. To accommodate the height change, the location of the air ducts and MPC pedestal height were modified previously. The HI-STORM 100A and 100SA are similar to the HI-STORM 100 and 100S overpacks except that they have a baseplate that is anchored to the concrete pad at the independent spent fuel storage installation (ISFSI). The HI-STORM 100A and 100SA overpacks may be used to store fuel in high seismic areas. The HI-STORM 100S, 100A, and 100SA overpacks were approved under Amendment 1 to CoC 1014.

1.1.4 Basic Operation

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, welded closed, inspected, and backfilled with an inert gas; (3) the transfer cask is placed on top of the overpack and the MPC is lowered into the overpack; and (4) if necessary 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, 100S, 100A, and 100SA, 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).

1.2 Drawings

Section 1.5 of the proposed FSAR contains the drawings for the HI-STORM 100 Cask System and includes 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, as necessary.

1.3 Cask Contents

The amendment requested several additions and changes to the contents that may be stored in the HI-STORM 100 Cask System. These include storage of:

1. fuel debris into any of the eight approved MPC designs except the MPC-24,
2. BWR fuel with maximum fuel assembly weight of up to 730 lb in the MPC-68,
3. 16x16 PWR assemblies with increased maximum fuel assembly weight of up to 1,720 lb for assemblies not requiring fuel spacer, revised Fuel Rod Clad ID minimum Guide/Instrument Tube Thickness, and minimum soluble boron concentration for Array Class 16x16A for all intact fuel assemblies,
4. APSRs in new locations in the MPC-32 and CRAs, RCCAs, and CEAs in new locations in the MPC-24, MPC-24E and MPC-32, and
5. SNF with up to 34 kW (uniform loading) and 36.9 kW (regionalized loading) respectively.

1.4 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F1.1 A general description and discussion of the design changes to the HI-STORM 100 Cask System are presented in Chapter 1 of the proposed FSAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff concludes that the information presented in this Chapter of the proposed FSAR 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.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

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

Structures, systems, and components important to safety are annotated in Table 2.2.6 of the proposed FSAR. In this table, each component is assigned a safety classification. The safety classifications are based on the guidance in U.S. Nuclear Regulatory Commission, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," NUREG/CR-6407, INEL-95/0551, February 1996.

Table 2.2.6 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 American Society of Mechanical Engineers (ASME) Code. The governing code for the concrete in the overpack is American Concrete Institute (ACI) 349. Alternatives to these Codes are delineated in the proposed FSAR as provided in Table 2.2.15 and Appendix 1.D.

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 either 24 or 32 PWR fuel assemblies and up to 68 BWR fuel assemblies. Detailed specifications for the approved fuel assemblies, as modified by this amendment, are given in Section 2.1 of the proposed FSAR. 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 proposed FSAR identifies the bounding site environmental conditions and natural phenomena for which the HI-STORM 100 Cask System is analyzed. Changes to Section 2.2 were made for consistency with those changes described in greater detail elsewhere in the proposed FSAR.

2.3 Design Criteria for Safety Protection Systems

The principal design criteria for the MPC and the HI-STORM 100 overpack designs and the Transfer cask, are summarized in proposed FSAR Tables 2.0.1, 2.0.2, and 2.0.3. This amendment requested changes to Tables 2.0.1, 2.0.2, and 2.0.3 to be consistent with those changes described in greater detail elsewhere in the proposed FSAR. The codes and standards of the design and construction of the system and changes to the design criteria are specified in Section 2.2 of the proposed FSAR. The Cask Transfer Facility (CTF) that is not under the requirements of 10 CFR Part 50 and is to be designed, developed and operated by the cask system user at the site location of their choice, depending upon site-specific needs and capabilities, is described as one of three types. A stand-alone, above ground facility, an underground facility, combined with a mobile lifting device, or an underground facility, combined with a cask transporter/crawler. The confinement barrier and systems of the storage system shall not be compromised by the equipment used in the transfer operations that are identified as ancillary equipment that includes the CTF. In meeting the general specifications for the CTF as identified in Section 2.3.3 of the proposed FSAR, the cask system user will verify that use of one of the underground CTF options will not change the potential environmental and loading conditions to create unanalyzed conditions on the cask system during the transfer operations.

2.3.1 General

Chapter 2 of the proposed FSAR was modified, per this amendment, to include changes associated with the MPC and overpacks. The major changes include: (1) an increase in the maximum fuel assembly weight for BWR fuel in the MPC-68 from 700 to 730 lb. (2) changes to the assembly characteristics of 16x16 PWR assemblies approved for storage, including a maximum fuel assembly weight of up to 1,720 lb for assemblies not requiring fuel spacers, (3) elimination of the restriction that fuel debris can only be loaded into the MPC-24EF, MPC-32F, MPC-68F, and MPC-68FF canisters, (4) an increase in the design basis maximum decay heat loads and a new decay heat regionalized scheme is introduced, and (5) a change in the permissible storage locations in the MPC-32 for fuel with Axial Power Shaping Rod assemblies (APSRs) and a change in the permissible storage locations in the MPC-24, MPC-24E and MPC-32 for fuel with Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), and Control Element Assemblies (CEAs).

2.3.2 Structural

The structural analysis is presented in Chapter 3 of the proposed FSAR. 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 proposed FSAR. Changes made to the structural design criteria under this amendment are described in Section 3 of this SER.

2.3.3 Thermal

The thermal analysis is presented in Chapter 4 of the proposed FSAR. The HI-STORM 100 Cask System is designed to passively reject decay heat when on the ISFSI pad. Heat removal, by conduction, radiation, and natural convection, is independent of intervening actions under normal, off-normal, and accident conditions for storage of spent nuclear fuel in the HI-STORM 100 Cask System. 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 Chapters 5, 7, and 10 of the proposed FSAR.

Confinement is provided by the MPC, which has a welded closure. The MPC's confinement function is verified through pressure testing, helium leakage testing, and weld examinations. Radiation exposure is mitigated by the neutron and gamma shields and by operational procedures.

2.3.5 Criticality

The criticality analysis is presented in Chapter 6 of the proposed FSAR. 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 materials. The continued efficacy of the neutron-absorbing materials over a 20-year storage period is assured by the design of the system. Depletion of the 10B in the neutron-absorbing materials 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 Chapter 8 of the proposed FSAR. 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 Chapter 9 of the proposed FSAR, 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 proposed FSAR. The decommissioning features of the HI-STORM 100 Cask System did not change with this amendment and were not reevaluated by the staff.

2.4 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

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

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

The review was conducted against the appropriate regulations as described in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified as required by 10 CFR 72.236(a) so that the design basis and the design criteria that must be provided for the structures, system and components important to safety can be assessed under the requirements of 10 CFR 72.236(c) - (l). This amendment was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfills the acceptance criteria listed in Section 3 of NUREG-1536.

The amendment requests that have a direct bearing on the structural aspects of the spent fuel cask storage system include the small increase in the weight of fuel assemblies (700 lb to 730 lb for BWR assembles and 1680 lb to 1720 lb for PWR assemblies not requiring fuel spacers), and increased fuel assembly lengths. This amendment introduces new terminology within the existing criteria for safety protection systems as discussed in Section 2.3 of the proposed FSAR. This terminology is used in the description of the important-to-safety (ITS) equipment that may be used as ancillary or support equipment for ISFSI implementation related to the handling and movement of the MPC, the transfer cask and the storage casks on-site that are outside the 10 CFR Part 50 structures. For the HI-STORM 100 Cask System, such equipment/structures are encompassed by and known as the Cask Transfer Facility (CTF). This amendment, in referring to the existing design criteria for such ancillary equipment/structures, identifies the specific types of installations that could be used to execute cask handling and MPC transfers.

This section presents the results of the structural design review of the amendment request for the HI-STORM 100 Cask System.

3.1 Structural Design of the Additional Components for the HI-STORM 100 Cask System

The HI-STORM 100 Cask System is made up of three major components that are used in the dry spent fuel storage system: the MPC, the HI-TRAC transfer cask, and the HI-STORM dry storage overpack/cask. This structural portion of the SER addresses the MPC and the overpack designs that were impacted by this amendment.

3.1.1 Structural Design Features

No structural physical design feature was proposed for modification as a result of the amendment.

3.1.2 Structural Design Criteria

No changes were proposed in the structural design criteria under this amendment however as a result of increased temperatures resulting from this amendment the previously identified off-normal and accident condition temperature limits have increased. Stress limits as a function of material temperature therefore were modified and used to adjust previously calculated results. None of the resulting changes were significant enough to require modification of the physical dimensions of the components.

3.2 Weights and Center of Gravity

The small increases in fuel assembly weights (700 lb to 730 lb and 1680 lb to 1720 lb were considered in the revisions made in the proposed amendment by linear scaling of the results from the previous calculations. The scaling factors were 1.043 for the BWR fuel and 1.024 for the PWR fuel where fuel spacers are not necessary. This is an acceptable methodology for the elastic loading conditions and response.

3.3 Structural Analysis of HI-STORM Cask System

There were no proposed changes in the structural analysis methods or the conditions to be considered. The actual temperatures to be considered were higher as a result of the higher heat load of the spent fuel and this was reflected in the results. The terminology for the partial blocked vents under the off-normal condition was changed from two of four vents to state that the blockage considered will be 50%. This is acceptable and can correctly define the intent of the loading condition. As noted in Section 3.2 above, the analysis results also reflected the consideration for the small increases in the weight of the spent fuel assemblies.

3.4 Special Topics

3.4.1 Lifting Devices

No changes were proposed in this amendment.

3.4.2 Differential Thermal Expansion

The resulting increased thermal expansion from the increased temperatures associated with this amendment can be accommodated by the current design based on the analyses presented in Section 4.4.6 and Table 4.4.10 of the proposed FSAR.

3.5 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F3.1 The proposed FSAR revision adequately describes all changes to structures, systems, and components that are important to safety and provides drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

F3.2 Any modifications to SSCs as a result of the amendment important to safety are described for the HI-STORM 100 Cask System in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident and natural phenomena events.

F3.3 The HI-STORM 100 Cask System is designed to allow handling and retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel.

F3.4 The HI-STORM 100 Cask System is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.5 The cask and its systems important to safety are evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.6 The staff concludes that the structural design 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 structural evaluation provides reasonable assurance that the HI-STORM 100 Cask System will enable safe storage of spent nuclear fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable industry codes and standards, accepted practices and confirmatory analysis.

4.0 THERMAL EVALUATION

The thermal review ensures that the cask components 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 analytical techniques and/or testing methods.

The review was conducted against the appropriate regulations as described in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified, as required by 10 CFR 72.236(a), so that the design basis and the design criteria that must be provided for the structures, systems, and components important to safety can be assessed under the requirements of 10 CFR 72.236(c) - (l). This application was also reviewed to determine whether the HI-STORM 100 Cask System fulfills the acceptance criteria listed in Sections 4 and 11 of NUREG-1536 as well as associated Interim Staff Guidance (ISG) documents.

4.1 Description of Proposed Changes

The most significant changes proposed by the applicant in the FSAR that effect the thermal performance of the HI-STORM 100 Cask System are listed as follows:

1. Permissible MPC heat load is increased to 34 kW (uniform loading) and by 28% from 28.74 to 36.9 kW (regionalized loading).

2. Generalized regionalized storage approach is added to permit a continuum of fuel storage configurations over a range X, where X is the ratio of inner region to outer region fuel storage cells heat load limits. The permitted fuel loading regions for each MPC type are defined in Appendix B (Approved Contents) of the CoC.

3. MPC operating pressure is raised to 7 atmospheres (absolute).

4. A new three-dimensional (3-D) thermal model is added to evaluate fuel integrity under normal (long-term storage) conditions.

5. A new Section 4.6 is added to group all thermal analyses in support of off-normal and accident events evaluated in Chapter 11 of the proposed FSAR.

4.2 Spent Fuel Cladding

The applicant adopted certain guidelines of NUREG-1536 and ISG-11, Rev. 3, to demonstrate the safe storage of the material content described in Chapter 2 of the proposed FSAR and in the CoC of the HI-STORM 100 Cask System. The applicant has designed the HI-STORM 100 Cask System to comply with all the following eight criteria:

1. The fuel cladding temperature for long-term storage shall be limited to 752°F (400°C).

2. The fuel cladding temperature for short-term operations shall be limited to 752°F (400°C) for high burnup fuel (HBF) and 1058°F (570°C) for moderate burnup fuel (MBF).

3. The fuel cladding temperature should be maintained below 1058°F (570°C) for accident and off-normal event conditions.

4. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions.

5. The cask materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions.

6. For fuel assemblies proposed for storage, the cask system should ensure a very low probability of cladding breach during long-term storage.

7. The HI-STORM 100 Cask System should be passively cooled.

8. The thermal performance of the cask shall be in compliance with the design criteria specified in proposed FSAR Chapters 1 and 2 for normal, off-normal, and accident conditions.

The staff reviewed the criteria and found it acceptable.

4.3 Thermal Properties of Materials

Material property tables for the HI-STORM 100 Cask System components are included in the proposed FSAR. The temperature range for the material properties covers the range of temperatures encountered during the thermal analysis with some exceptions that were justified by the applicant. The staff reviewed the thermal properties of used materials and found them acceptable.

4.4 Specifications for Components

For evaluation of HI-STORM 100 Cask System thermal performance, material temperature limits for long-term normal, short-term operations, and off-normal and accident conditions are provided in Table 4.3.1 of the proposed FSAR. Fuel cladding temperature limits included in Table 4.3.1 of the proposed FSAR are adopted from ISG-11, Rev. 3. These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.

The proposed FSAR stated that the Pacific Northwest National Laboratory (PNNL) has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperature for MBF delineated in PNNL White Paper “Estimated Maximum Cladding Stresses for Bounding PWR Fuel Rods During Short Term Operations for Dry Cask Storage,” Lanning and Beyer, January 2004. PNNL’s study concluded that hydride reorientation is not credible during short-term operations involving low to MBF (up to 45 GWD/MTU). Accordingly, a higher temperature limit is applied to MBF, as specified in Table 4.3.1 of the proposed FSAR.

The staff reviewed the component specifications and found them acceptable.

4.5 General Description of the HI-STORM 100 Cask System

HI-STORM 100 Cask System consists of a sealed MPC located inside a vertically-oriented, ventilated storage overpack. Air inlet and outlet ducts that allow for air cooling of the stored MPC are located at the bottom and top of the overpack. The Spent Nuclear Fuel (SNF) assemblies reside inside the MPC, which is sealed with a welded lid to form the confinement boundary. The MPC contains a stainless-steel honeycomb fuel basket structure with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. Each fuel basket panel, with the exception of exterior panels on the MPC-68 and MPC-32, is equipped with a thermal neutron absorber panel sandwiched between an Alloy X steel sheathing plate and the fuel basket panel, along the entire length of the active fuel region. The MPC is backfilled with helium up to the design-basis initial fill level specified in the proposed FSAR. This provides a stable, inert environment for long-term storage of the SNF. Heat is rejected from the SNF in the HI-STORM 100 Cask System to the environment by passive heat transport mechanisms only. Within the MPC the pressurized helium environment sustains a closed loop thermosiphon action, removing SNF heat by an upward flow of helium through the storage cells. On the outside of the MPC, a ducted overpack construction with a vertical annulus facilitates an upward flow of air by buoyancy forces. The annulus ventilation flow cools the hot MPC surfaces and transports heat to the outside environment.

The applicant evaluated the HI-STORM Cask System (i.e., HI-STORM overpack, HI-TRAC transfer cask, and MPC) under normal storage (HI-STORM overpack), during off-normal and accident events, and during short term operations in a HI-TRAC.

4.6 Thermal Model

The applicant developed 3-D FLUENT thermal models of the MPC and overpack cask to evaluate fuel temperatures under normal (long-term storage), off-normal and accident conditions. The applicant used a two-dimensional (2-D) thermal model of the HI-TRAC transfer cask to perform the evaluation under short-term operations, off-normal, and hypothetical accidents. The staff has previously reviewed the HI-TRAC 2-D thermal model for Amendment 1 to the HI-STORM 100 Cask System CoC effective July 15, 2002.

The fuel basket composite cell walls (made up of Alloy X panels, neutron absorber panels and Alloy X sheathing) are represented by a homogeneous panel with equivalent orthotropic (thru-thickness and parallel plates direction) thermal conductivities. The applicant stated that it is impractical to model every fuel rod in every stored fuel assembly explicitly. Therefore, the cross section bounded by the inside of the storage cell (inside of the fuel channel in the case of BWR MPCs), which surrounds the assemblage of fuel rods and the interstitial helium gas (also called the “rodded region”), is replaced with an “equivalent” square homogeneous section characterized by an effective thermal conductivity. Since the effective thermal conductivity of the rodded region includes radiation heat transfer, the effective thermal conductivities will be a strong function of temperature because radiation heat transfer (a major component of the heat transport between the fuel rods and the surrounding basket cell metal) rises as the fourth power of absolute temperature. Therefore, in effect, the effective conductivity of the equivalent square section (depending on the coincident temperature) will be different throughout the basket. For thermal-hydraulic simulations, each fuel assembly in its storage cell is represented by an equivalent porous medium. For BWR fuel, the presence of the fuel channel divides the storage cell space into two distinct axial flow regions, namely, the in-channel (rodded) region and the square prismatic annulus region.

4.6.1 3-D Thermal Model (Storage Overpack and MPC)

The 3-D model implemented by the applicant to analyze the HI-STORM 100 overpack configuration for storage during normal, off-normal, and accident conditions, has the following characteristics:

a. The composite walls in the fuel basket consisting of the Alloy X structural panels, the aluminum-based neutron absorber, and the Alloy X sheathing, are represented by an orthotropic homogeneous panel of equivalent thermal conductivity in the three principal directions.

b. In the case of a BWR commercial spent fuel (CSF) loaded in the MPC-68, the fuel bundle and the small surrounding spaces inside the fuel “channel” are replaced by an equivalent porous media having the flow impedance properties computed using a 3-D FLUENT Computational Fluid Dynamics (CFD) model documented in Ref. [4.4.2] of the proposed FSAR. The space between the BWR fuel channel and the storage cell is represented as an open flow annulus. The fuel channel is also explicitly modeled. The porous medium within the channel space is also referred to as the “rodded region.” The fuel assembly is assumed to be positioned coaxially with respect to its storage cell.

   In the case of the PWR CSF loaded in an MPC-24 or MPC-32, the porous medium extends to the entire cross-section of the storage cell. As described in Ref. [4.4.2] of the proposed FSAR, the CFD model for both the BWR and PWR case is prepared for the design basis fuel in comprehensive detail, which includes grid straps, BWR water rods, and PWR guide and instrument tubes.

c. Every MPC fuel storage cell is assumed to be occupied by design basis (fuel assembly designs which are found to govern the thermal qualification criteria) PWR or BWR fuel assemblies specified in Chapter 2 (Table 2.1.5) of the proposed FSAR.

d. The internals of the MPC (MPC-24, MPC-32, or MPC-68), including the basket cross section, bottom mouse holes, top plenum, and circumferentially irregular downcomer are modeled explicitly. For modeling simplicity, the mouse holes are represented as rectangular openings.

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

f. The air flow in the HI-STORM/MPC annulus is simulated by a k-omega turbulence model with the transitional option enabled.

4.6.2 Fuel Assembly Flow Resistance Model

The applicant evaluated the HI-STORM 100 overpack for storage of bounding PWR (W-17x17) and BWR (GE-10x10) fuel assemblies. During fuel storage helium enters the MPC fuel cells from the bottom plenum and flows upwards through the open spaces in the fuel storage cells and exits in the top plenum. Helium flow in the fuel storage cells and MPC spaces is characterized as laminar flow because of low velocities.

To characterize the flow resistance of fuel assemblies inside square envelopes (fuel channel for BWR fuel or fuel storage cell for PWR fuel), the applicant developed 3-D models of Westinghouse 17x17 and General Electric 10x10 fuel assemblies using the FLUENT CFD program.

Using the 3-D fuel assembly models the applicant computed flow solutions by imposing pressure boundary conditions at the two extremities of the fuel storage cell at reference conditions (7 atmosphere absolute pressure and a temperature of 450°F). The applicant post-processed the results of the 3-D flow solutions to obtain equivalent porous media flow resistances as described below.

The applicant used two approaches to obtain the fuel assembly flow resistances: The first approach is the pressure drop method. This method is suitable when a zone is characterized by irregular geometries and the objective is to obtain a lumped resistance to duplicate the pressure drop. The second method is the shear stress method, which is suitable for flow zones characterized by regular geometries. The applicant adopted the pressure drop method for the inactive regions which are located below and above the active fuel region. For the active region, the applicant applied the shear stress approach because this region is characterized by an ordered array of entities (rods and grids). This method uses area-averaged wall shear stresses post-processed from the active region of the fuel assembly. Using hydraulic flow principles, the applicant mapped the wall shear stresses to flow resistance coefficient.

The staff reviewed the applicant’s fuel assembly flow resistance modeling approach and found it acceptable because the approach was generally equivalent to the approach used by the staff when performing confirmatory analysis. The applicant’s calculated flow resistance values were in general equivalent to the measured data obtained by the Office of Regulatory Research for a prototypic BWR fuel assembly. Since the FLUENT calculated values of a prototypic BWR fuel assembly were in general equivalent to the measured data, the staff used this comparison as a FLUENT validation to calculate flow resistance parameters for other fuel assembly types.

4.7 Thermal Evaluation for Normal Conditions of Storage

The applicant used the 3-D thermal model of HI-STORM 100 Cask System to determine temperature distributions under long-term normal storage conditions for an array of cases covering PWR and BWR fuel storage in uniform and regionalized loading configurations as described in the proposed FSAR. Under regionalized loading, the applicant analyzed an array of runs covering a range of regionalized storage configurations specified in Chapter 2 of the proposed FSAR (X=0.5 to X=3, where the parameter X, as defined in the FSAR, is the ratio of inner region to outer region fuel storage cells heat load limits). Based on this array of runs the applicant determined the fuel storage condition corresponding to X = 0.5 to be limiting for both PWR and BWR MPCs. Accordingly HI-STORM MPC and overpack temperatures are reported for this storage condition in the proposed FSAR.

The staff inspected the temperature field obtained from the applicant's thermal analysis of HI-STORM 100 Cask System and made the following observations:

1. The fuel cladding temperatures are below the regulatory limit under all storage scenarios (uniform and regionalized) in all MPC types.

2. The maximum temperature of the basket structural materials are within their design limits.

3. The maximum temperature of the neutron absorbers are below their design limits.

4. The maximum temperatures of the MPC pressure boundary materials are below their design limits

5. The maximum temperatures of concrete is within the guidance of the governing ACI Code.

The applicant's analyses for low service temperature (-40°F) of the HI-STORM is provided in Chapter 3 of the proposed FSAR. The applicant stated that all HI-STORM storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode at this minimum temperature condition.

The reduced ambient pressure at site elevations above the sea level will act to reduce the ventilation air mass flow, resulting in a net elevation of the peak cladding temperature. The applicant evaluated the effect of elevation on peak fuel cladding temperatures by performing calculations for a HI-STORM 100 Cask System situated at an elevation of 1500 feet. The peak cladding temperatures are calculated for a bounding configuration (non-uniform storage at X = 0.5). The results show that for the conditions evaluated in the proposed FSAR, the peak cladding temperature (including the effects of site elevation) is below the acceptable temperature limit of 752°F (400°C) for the MPC operating conditions of 7 atm (absolute) and a total decay heat of 36.9 kW. In light of the above evaluation, it is not necessary to place any ISFSI elevation constraints for HI-STORM deployment at elevations up to 1500 feet. If, however, an ISFSI is sited at an elevation greater than 1500 feet, the proposed FSAR states that the effect of altitude on the cladding temperature should be quantified as part of the 10 CFR 72.212 evaluation for the site using the site ambient conditions.

The MPC is initially filled with dry helium after fuel loading and drying prior to installing the MPC closure ring. During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined using the ideal gas law. The maximum computed gas pressures reported in the proposed FSAR are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in Chapter 2 of the proposed FSAR.

After reviewing the applicant's analyses and calculated results, the staff finds the applicant's thermal evaluation for normal conditions of storage acceptable because the maximum calculated temperatures remain below the allowable limit specified in ISG-11, Rev. 3, for the thermal operating conditions specified in the amendment request.

4.8 Thermal Evaluation for Short-Term Operations

4.8.1 Vacuum Drying

The applicant developed an axisymmetric FLUENT thermal model of the MPC, employing the MPC in-plane conductivity as an isotropic fuel basket conductivity (i.e., conductivity in the basket radial and axial directions is equal), to determine peak cladding temperature at design basis heat loads. To avoid excessive conservatism in the computed FLUENT solution, partial recognition for higher axial heat dissipation is adopted in the peak cladding calculations.

For total decay heat loads up to and including 20.88 kW for the MPC-24 and 21.52 kW for the MPC-68, vacuum drying of the MPC is performed with the annular gap between the MPC and the HI-TRAC filled with water. The presence of water in this annular gap will maintain the MPC shell temperature approximately equal to the saturation temperature of the annulus water. Thus, the thermal analysis of the MPC during vacuum drying for these conditions is performed with cooling of the MPC shell with water at a bounding maximum temperature of 232°F (111°C).

For higher total decay heat loads in the MPC-24 and MPC-68 or for any decay heat load in an MPC- 24E or MPC-32, vacuum drying of the MPC is performed with the annular gap between the MPC and the HI-TRAC continuously flushed with water. The water movement in this annular gap will maintain the MPC shell temperature at about the temperature of flowing water. Thus, the thermal analysis of the MPC during vacuum drying for these conditions is performed with cooling of the MPC shell with water at a bounding maximum temperature of 125°F (51.6°C).

Vacuum Drying (VD) is permitted for MBF under certain conditions. If these conditions are not met, or if the MPC also contains HBF, then the Forced Helium Dehydration (FHD) system must be used for moisture removal. Table 1 shows the moisture removal limits and requirements determined by the applicant. These limits are specified in Technical Specification (TS) 3.1.1. The applicant calculated peak fuel clad temperatures for MBF during short-term vacuum drying operations with design-basis maximum heat loads to be less than 1058°F (570°C) for the MPC-24 and MPC-24E baskets by about 100°F margin. However, the applicant calculated a small margin for the MPC-68 and MPC-32.

The applicant committed in the proposed FSAR to limit the vacuum drying time to 40 hours for MPC total decay heat loads between 23 and 28.74 kW. It is the applicant's expectation that this limit will account for uncertainties of the 2-D thermal models regarding modeling simplifications and radiation heat transfer model, as identified by the staff.

Table 1 Moisture Removal Limits and Requirements				
Condition	Fuel in MPC	HI-TRAC Annulus Cooling Requirement	MPC Heat Load (kW)	Moisture Removal Method
1	All MBF	Standing Water	MPC-24: ≤20.88 MPC-68: ≤ 21.52	VD*
2	All MBF	Circulating Water	≤28.74**	VD*
3	All MBF	None	>28.74	FHD
4	One or More HBF	None	Any	FHD

*The FHD drying method is also acceptable under the Condition 1 and Condition 2 heat loads, in which case HI-TRAC annulus cooling is not required.

** Vacuum drying for MPC heat loads greater than 23 kW is limited to 40 hours.

MBF Moderate Burnup Fuel
HBF High Burnup Fuel
VD Vacuum Drying
FHD Forced Helium Dehydration

4.8.2 Normal Onsite Transport in a Vertical Orientation

A 2-D axisymmetric FLUENT thermal model of an MPC inside a HI-TRAC transfer cask was developed to evaluate temperature distributions for onsite transport conditions. A bounding steady state analysis of the HI-TRAC transfer cask has been performed using the hottest MPC (MPC-32), the highest design-basis decay heat load for which Supplemental Cooling System (SCS) is not required (Table 2.1.6 of the proposed FSAR), and design-basis insolation levels. While the duration of onsite transport may be short enough to preclude the MPC and HI-TRAC from obtaining a steady-state, a steady-state analysis is conservative

The maximum computed temperatures listed in Table 4.5.4 of the proposed FSAR are based on the HI-TRAC cask at maximum heat load that can be handled in HI-TRAC without needing the Supplemental Cooling System (see Table 2 below), passively rejecting heat by natural convection and radiation to a hot ambient environment at 100°F (37.7°C) in still air in a vertical orientation

For HBF, however, the maximum computed fuel cladding temperature reported in Table 4.5.4 is significantly greater than the temperature limit of 752°F (400°C) for HBF. Consequently, it is necessary to utilize the SCS described in Appendix 2.C of the proposed FSAR during onsite transfer of an MPC containing HBF emplaced in a HI-TRAC transfer cask. The exact design and operation of the SCS are necessarily site-specific. The design is required to satisfy the specifications and operational requirements of Appendix 2.C of the proposed FSAR to ensure compliance with ISG-11, Rev. 3 temperature limits.

Table 2 Requirements and Limits for HI-TRAC Transport in a Vertical Orientation			
Condition	Fuel in MPC	MPC Heat Load (kW)	SCS Required
1*	All MBF	≤ 28.74	NO
2	All MBF	> 28.74	YES
3	One or More HBF	Any	YES

* The highest temperatures are reached under this un-assisted cooling threshold heat load scenario. Under other conditions the mandatory use of the Supplemental Cooling System, sized to extract 36.9 kW from the MPC, will lower the fuel temperatures significantly assuring ISG 11, Rev. 3 compliance with thermal cycling and maximum temperature limits, with large margins.

MBF Moderate Burnup Fuel
HBF High Burnup Fuel
SCS Supplemental Cooling System

The staff reviewed the applicant's analyses and calculated results and finds the applicant's thermal evaluation for short-term operations acceptable because the maximum calculated temperatures remain below the allowable limit specified in ISG-11, Rev. 3, for the thermal operating conditions specified in the amendment request. The use of the SCS, required only for HBF in Amendment 2 to HI-STORM 100 Cask System CoC, has been extended to MBF for MPC heat loads larger than 28.74 kW. Use of SCS during normal onsite transport in a vertical orientation was approved previously (Safety Evaluation Report of Amendment 2 to the HI-STORM 100 Cask System CoC 1014).

4.9 Off-Normal and Accident Events

Per NUREG-1536, the applicant evaluated the HI-STORM 100 Cask System for the effects of off-normal and accident events. The design basis off-normal and accident events are defined in Chapter 2 of the proposed FSAR. For each event, the applicant discussed and evaluated the cause of the event, means of detection, consequences, and corrective actions in Chapter 11 of the proposed FSAR. To support the Chapter 11 evaluations, thermal analyses of limiting off-normal and accident events are provided for the following events:

Off-Normal Events:

1. Off-Normal Pressure
2. Off-Norma Environmental Temperature
3. Partial Blockage of Air Inlet Vents

Accident Events

1. Fire Accident
2. Jacket Water Loss
3. Extreme Environmental Temperature
4. Full Blockage of Air Inlet Vents
5. Burial Under Debris

The applicant's changes to the thermal performance of HI-STORM 100 Cask System are provided in Section 4.1 (Description of Proposed Changes) of this SER. Specifically, an increase in the MPC heat load and operating pressure will directly affect the thermal performance of the storage system during off-normal and accident events. The applicant confirmed that all thermal results were still below applicable allowable limits for the increased MPC heat load and operating pressure. The staff reviewed all the modeling approaches and assumptions for the off-normal and accident events, in accordance with NUREG-1536, and finds the thermal evaluation for off-normal and accident events acceptable.

4.10 Staff's Confirmatory Analysis of HI-STORM 100 Cask System

4.10.1 Normal Conditions of Storage

The staff reviewed the applicant's models and calculation options to determine the adequacy of HI-STORM 100 Cask System proposed changes. Additionally, the staff performed some confirmatory analyses using the FLUENT finite volume CFD code, as an independent evaluation of the thermal analysis and modeling options presented in the applicant's proposed FSAR.

Specifically, the staff's evaluation focused on applicant's modeling assumptions and calculation options that have the greatest impact on the calculated results. Some of the modeling options the staff considered were the use of a porous media to represent the fuel basket and fuel compartments as homogenized regions characterized only by a viscous and inertial resistance coefficients. Also, the staff independently performed confirmatory analysis and validation to confirm the applicant's assumption on the flow regime used to characterize the air flow through the annular gap between the MPC and the concrete cask. In order to perform confirmatory analyses, the staff modeled in detail the MPC-68 using FLUENT CFD code. The staff also performed some scoping calculation to confirm the adequacy of the effective thermal conductivity model proposed by the applicant in the FSAR.

Based on the applicant's thermal analysis, and own confirmatory analysis the staff concluded the following regarding applicant's modeling options:

Use of the porous media approach to represent the fuel compartments and fuel basket is adequate, provided the porous media parameters to characterize the flow are carefully implemented and calculated based on explicit three-dimensional (3-D) flow characterization of the bounding fuel assembly geometry. The staff developed 3-D CFD models to represent the fuel assembly bounding geometry and used two approaches to calculate the flow resistance parameters: pressure drop method and shear stress method. Both methods were applied for sections without flow area changes (i.e., no contractions or expansions). Both approaches are related and should lead to the same values.

The air flow in the inlet and outlet vents and annular gap between the MPC and the concrete inner shell is expected to be in the transitional regime. It is therefore necessary to specify an appropriate turbulence model for the air flow in order to obtain accurate predictions of local velocities and temperatures in the air stream, and local wall temperatures on the surfaces of the annulus and inlet/outlet vent structures. Based on applicant's calculation and staff's validation efforts (by modeling VSC-17 using FLUENT and comparing to VSC-17 measured data), the staff concluded that the flow in the air annular gap is found to be in the transitional region of turbulence. Based on the applicant's thermal evaluation and staff's confirmatory analysis supported by FLUENT code validation using VSC-17 measured data, only turbulence models that are capable of dealing with a transitional flow regime should be used to perform the thermal analysis of ventilated storage casks as HI-STORM 100 Cask System design.

The staff's confirmatory calculation of effective thermal conductivity resulted in values that were on the same order of magnitude as compared to the applicant calculated values. Therefore, the staff found the applicant's effective thermal conductivity calculation approach acceptable for this specific design.

4.10.2 Short-Term Operations

The applicant submitted a 3-D FLUENT model on January 31, 2007 to calculate vacuum drying temperatures, using the Discrete Transfer Radiation Model (DTRM) theory to represent radiation heat transfer. The staff adjusted this 3-D FLUENT to represent radiation heat transfer using Discrete Ordinates (DO). The staff performed a transient analysis to calculate a time limit for the vacuum drying process. The staff's transient calculation using the applicant's 3-D model has the following characteristics:

I. Radiation heat transfer is based on DO radiation method.

ii. The entire outer surface of the MPC shell is assumed to be at 125°F (51.6°C).

iii. A total decay heat of 28.74 kW is assumed for the analysis (see Condition 2 of Table 1)

iv. An initial temperature of 212°F (100°C) is assumed for MPC internals and contents (spent fuel).

Based on the transient calculation and assumptions described above, the staff determined that 40 hours is an appropriate vacuum drying time limit to assure short-term peak cladding temperature limits are not exceeded, as specified in TS 3.1.1. The vacuum drying time limit applies only to MBF for heat loads in the range of 23 kW to 28.74 kW, as shown on Table 1. For MBF with heat loads greater than 28.74 kW, the Forced Helium Dehydration System (FHD) shall be used. The FHD system is also the mandated drying method if one or more HBF assemblies are loaded in any MPC type. If the vacuum drying time limit is not met, the MPC cavity shall be backfilled with helium to a pressure of at least 0.5 atm or more before the vacuum drying time can be restarted again. The staff performed a steady state calculation with the MPC cavity filled with helium and calculated a peak cladding temperature below the allowable limit. Backfilling the MPC with a helium environment will provide a better heat transfer environment and will assure ISG-11, Rev. 3, "Cladding Considerations for Transportation and Storage of Spent Fuel," short-term peak cladding temperature limits are not exceeded.

4.11 Conclusion

The applicant adequately described and justified the proposed changes to the CoC in the License Amendment Request. These changes considerably affect previous thermal results but the staff finds these changes acceptable. Applicant's compliance with ISG-11, Rev. 3, allowable temperature limits during normal onsite transport in a vertical orientation requires the use of a supplemental cooling system. The addition and type of supplemental cooling system is left to the end user of the storage system. The applicant stated in the proposed FSAR that the end user shall perform a thermal analysis, including the SCS, based on the thermal methodology described in the proposed FSAR.

4.12 Evaluation Findings

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

F4.2 The HI-STORM 100 Cask System is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. Except during short-term operations, the cask is designed to provide adequate heat removal capacity without active cooling systems.

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

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

F4.5 The staff finds that the thermal design of the HI-STORM 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 thermal design provides reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5.0 SHIELDING EVALUATION

The objective of the shielding review is to ensure that there is adequate protection to the public and workers against direct radiation from the cask contents. The review seeks to ensure that changes to the shielding features and contents provide adequate protection to the operating staff and members of the public against direct radiation and that direct radiation exposures can satisfy regulatory requirements during normal operating, off-normal, and design-basis accident conditions. The objective includes review of changes to the shielding design description, radiation source definition, shielding model specification and shielding analyses for the HI-STORM 100 Cask System proposed by this amendment request.

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, 10 CFR 72.106(b), 10 CFR 72.212, and 10 CFR 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 of this SER.

The applicant proposed modifications to the burnup and cooling time combinations to reflect the increase in allowable heat load; however, the maximum permissible burnups of 65 and 68.2 GWD/MTU for BWR assemblies and PWR assemblies, respectively, are unchanged. The increase in heat load and resulting decrease in the margin in the allowable decay heat (in the thermal analysis) also necessitated a revision in the coefficients used to determine allowable PWR parameters to include a 5% decay heat penalty. This amendment also includes proposed increases in the number of allowable storage locations for Axial Power Shaping Rods (APSRs) in the MPC-32/32F and for Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), and Control Element Assemblies (CEAs) in the MPC-24, MPC-24E/24EF, and MPC-32/32F. Additional proposed changes include modification of the regionalized loading scheme to allow higher heat load assemblies to be stored either in the inner or in the outer basket regions, expansion of the MPC-24 and MPC-24E/24EF inner region from the four innermost basket locations to the twelve innermost locations and a reduction in the minimum overpack concrete density to 140 lb/ft3.

Finally, the applicant performed the shielding analyses with the HI-STORM 100S Version B design configuration. The Version B design configuration was implemented by the applicant under the change authority of 10 CFR 72.48. The NRC has not approved any aspect of this design configuration in this licensing action. However, during its review of a previous amendment, the staff examined and accepted the dose rates for the Version B design configuration and implemented dose values in Technical Specification (TS) 5.7 of the Certificate of Compliance (CoC) based upon the normal condition dose rates for the Version B.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The applicant proposed a change to the minimum density of the overpack concrete to permit a wider range of concrete densities for flexibility in implementing the HI-STORM Cask System at various sites, particularly to allow use at sites with restrictive weight handling limitations. The change reduces this density from 146 lb/ft3 and 155 lb/ft3 to 140 lb/ft3. The applicant also proposed that the densities of the overpack lid, pedestal and body all have this same minimum. This density reduction will result in higher dose rates from the overpack. As described in

Section 5.3.2 of this SER, the applicant's analysis does include the effects of this reduced density in the dose analysis. The applicant states, on proposed FSAR pages 5.3-7 and 8, that the density may be increased to 200 lb/ft3, which may be beneficial to some site users based on on-site and off-site "As Low As Reasonably Achievable" (ALARA) considerations. While the staff finds the applicant's analysis with the reduced concrete density acceptable, ALARA criteria (both for occupational and public dose) should be considered by the site user, such that overpacks with the maximum concrete density possible are used unless significant site circumstances, precluding use of overpacks with the higher density concrete, exist that cannot be reasonably modified to allow their use. While the staff finds the applicant's analysis with the reduced concrete density acceptable, the site user should appropriately consider ALARA principles (both for occupational and public dose) to determine the concrete density for overpacks to be used at its site such that overpacks with the maximum concrete density supported by site conditions, including any reasonable modifications, be used.

Besides the reduction in concrete density, no significant changes to the shielding design features were proposed in the current amendment. The staff evaluated the ability of the HI-STORM 100 Cask System design to shield the revised contents. The staff did not review any potential shielding design changes that may have been incorporated under the change authority of 10 CFR 72.48. The staff finds the shielding design features to be acceptable. Based on information provided by the applicant, the staff has reasonable assurance that the shielding design features of the HI-STORM 100 Cask System can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72 for the revised contents.

5.1.2 Shielding and Source Term Design Criteria

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). The applicant analyzed the HI-STORM 100 Cask System with spent fuel and hardware having the proposed characteristics described in Section 2.1.9 of the Proposed FSAR. Although there are no numerical limits in the regulations for cask system surface dose rates, these dose rates serve as design criteria to assure there is sufficient shielding to meet radiological limits in accordance with 72.236(d). Because the revised contents result in higher radiation source terms, the applicant proposed maximum surface dose rate criteria for the storage overpack to be 300 mrem/hr on the side, 175 mrem/hr at the openings of the air vents, and 60 mrem/hr on the top. Based on these design criteria, the applicant calculated bounding dose rates on the exterior of the HI-STORM 100 Cask System. These calculated dose rates are less than the proposed design criteria (see Section 5.4 of this SER).

The staff reviewed the design criteria and found them acceptable. The shielding and source term design criteria defined in the proposed FSAR provide reasonable assurance that the HI-STORM 100 Cask System can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72. Each user will be required to protect personnel from the increased dose rates in accordance with ALARA principles and the regulations of 10 CFR Part 20. A radiation protection program is defined in TS 5.7 to assure compliance with these requirements, with respect to the new contents proposed by the applicant. Dose rate limits based on the bounding shielding analysis are also incorporated into TS 5.7 for the side and top of the HI-STORM 100 overpack. Limits related to maximum decay heat, maximum burnup, minimum cooling time, maximum uranium loadings, and the burnup equation coefficients are incorporated into Appendix B of the CoC.

5.1.3 Preferential Loading Criteria

The HI-STORM 100 Cask System is designed to store fuel in either a uniform loading pattern or a regional loading pattern (preferential), as discussed in Section 2.1.9.1 of the proposed FSAR. The limitations for both loading patterns are specified in Section 2 of Appendix B to the CoC. Section 2.1.3 of Appendix B limits regional loading to only assembly classes with zirconium-based cladding (Zr-clad). There are also some Zr-clad BWR assembly classes which cannot be loaded in a regional pattern. The maximum decay heat limits for these Zr-clad and for the stainless steel-clad assembly classes are given explicitly in Table 2.1-1 of Appendix B. For those Zr-clad assemblies that may be loaded in either a uniform or a regional pattern, the maximum decay heat limits are specified in Sections 2.4.1 and 2.4.2 of Appendix B.

The currently approved regional loading pattern restricts assemblies of higher heat load to the inner region of the MPC basket. The amendment proposes to modify the preferential pattern to allow assemblies of higher heat loads to be placed in either the outer region or the inner region of the MPC. The applicant relies upon radial dose rates calculated for uniform loading patterns to provide bounding dose rates in the primary shielding analysis, instead of calculating dose rates for specific regional loading patterns. The applicant provided information that indicates that the burnup and cooling time combinations used for the uniform loading patterns evaluated in the shielding analysis bound the combinations that are possible with the regional loading patterns. The staff reviewed the use of preferential loading and uniform loading specifications for the fuel categories with respect to shielding. Based on the statements provided by the applicant, the staff has reasonable assurance that the analyzed uniform loading patterns generally result in dose rates that bound those from the proposed preferential loading patterns for the various combinations of fuel parameters. The staff notes that each site user must perform an analysis under 10 CFR 72.212 to verify dose limits and will have to consider the specific loading pattern that will be used within each cask.

The proposed regional loading pattern will also affect the shape of the dose profile around the transfer cask and storage overpack. Using uniform loading patterns, the applicant has demonstrated that there is a significant reduction in the dose rate as the detector moves radially from the center of the cask top and base toward the edges of the cask. However, a regional loading pattern that places the assemblies with the higher heat load in the outer region will diminish the amount of dose rate reduction actually encountered in a loaded cask as the detector moves radially outward from the center of the cask base or top. In other words, dose rates may, depending on the magnitude of the differences between the two regions, remain relatively constant across more of the cask top, or base, before a significant decrease is observed. In extreme cases, the dose rates may even be higher near the MPC edge than at the center of the cask. In determining the specific loading pattern to be used within each cask, the site user should consider the potential effect on personnel doses during operations and implement appropriate ALARA precautions.

The applicant also proposed to expand the size of the inner region for the MPC-24 and the MPC-24E/EF to the twelve innermost basket cells; the currently approved inner region for these MPCs is the innermost four basket cells. This proposed change results in an increase in the inner region’s contribution to the radial neutron dose and gamma dose. Also, the proposed change creates areas where the outer region does not shield radiation from the inner region, resulting in hot spots in the radial dose profile of these MPCs with regionalized loading. The applicant addresses these issues on proposed FSAR pages 5.4-3 and 4. The staff has reviewed this information and finds that the shielding analysis in the amendment is bounding for this proposed change. The staff does note, however, that while the applicant addresses the azimuthal dose rate variations, the discussion is only qualitative. With regionalized loading, the dose rates variations may be significant. A quantitative analysis is necessary to determine the significance of those variations. Users of the HI-STORM 100 Cask System that load these MPCs under this proposed amendment should therefore properly analyze for (in their 10 CFR 72.212 analyses) azimuthal variations in the radial dose rates and account for higher dose rate areas in their operations, implementing further ALARA precautions as necessary.

5.2 Source Specification

The design-basis source specifications for bounding calculations are presented in Section 5.2 of the proposed FSAR. The applicant calculated new design-basis source terms at higher burnups and lower cooling times to represent bounding source terms allowed by the burnup equation method. For the bounding calculations, the applicant used the same design basis fuel types, uranium loadings, and source term method previously approved for the HI-STORM 100 Cask System. Based on the burnup equation method, the PWR and BWR fuel may have combinations of burnups up to 68.2 GWD/MTU and 65 GWD/MTU, respectively, and cooling times as low as 3 years. The exact combinations of parameters are limited by the maximum decay heats specified in Sections 2.4.1 and 2.4.2 of Appendix B to the CoC for uniform and regional loading. For the MPC-32[1], the applicant calculated bounding source terms for 45 GWD/MTU and 3-year cooling and 69 GWD/MTU and 5-year cooling. For the MPC-24 bounding calculations, the applicant calculated source terms for 60 GWD/MTU and 3-year cooling and 75 GWD/MTU and 5-year cooling. For the MPC-68 bounding calculations, the applicant calculated source terms for 50 GWD/MTU and 3-year cooling. The burnups and cooling times and associated decay heats conservatively bound the allowable decay heats defined in Sections 2.4.1 and 2.4.2 of Appendix B to the CoC, as proposed in the amendment.

These new design-basis source terms also bound the source terms calculated for those assembly classes for which the burnup equation method does not apply. The staff notes that the PWR gamma source in the energy range of 1.0 to 1.5 MeV is greater for stainless steel-clad PWR assemblies than the PWR design-basis source terms in this energy range (but by less than a factor of 2). However, the gamma source of the new PWR design-basis source terms is larger than the source from the stainless steel-clad PWR assemblies for all the other energy ranges (by a factor of 5 or more for most of the other energy ranges). Therefore, in general, the new design-basis sources calculated in this amendment are bounding for the source terms for all the spent fuel contents, as proposed in this amendment.

5.2.1 Gamma Source

The design-basis gamma source terms for the MPC-24, MPC-32, and MPC-68 are listed in Tables 5.2.4 through 5.2.13 and Table 5.2.22 of the proposed FSAR. The design-basis source terms for the proposed non-fuel hardware contents (e.g., Burnable Poison Rod Assemblies – BPRAs, Axial Power Shaping Rods – APSRs, etc.) are listed in Tables 5.2.31, 5.2.34 and 5.2.35. For the bounding source terms and development of the coefficients for the burnup equation method, the applicant used the same neutron flux scaling factors, cobalt impurities, elemental compositions, and axial gamma peaking factors as previously approved for the HI-STORM 100 Cask System. The applicant indicated that there is uncertainty in the gamma source terms associated with the ORIGEN-S calculations and provided references which

[1]**Error! Main Document Only.**Since the MPC-32 and 32F are essentially identical from a shielding perspective, analyses for the MPC-32 apply to the MPC-32F. This is likewise true for the MPC-24, 24E, and 24EF as well as the MPC-68, 68F, and 68FF.

discuss that uncertainty. Based on a review of data, the applicant noted that errors in Cs-134 and Eu-154 (significant gamma contributors) could range from 2 to 20%. The applicant indicated that the uncertainty is off-set by the conservatisms in the source term and shielding calculations in the rest of the shielding analysis.

5.2.2 Neutron Source

The design-basis neutron source terms for the MPC-24, MPC-32, and MPC-68 are listed in Tables 5.2.15 through 5.2.20 and Table 5.2.23 of the proposed FSAR. The applicant used the same axial neutron peaking factors as previously approved. The applicant indicated that there is uncertainty in the neutron source terms associated with the ORIGEN-S calculations and provided references which discuss that uncertainty. The method used to determine the neutron source will result in a localized under-prediction for assemblies which have axial blankets using natural uranium. The applicant provided justification that this under-prediction is a localized effect and does not significantly increase the total assembly neutron source. The staff reviewed this justification and has found it to be acceptable.

5.2.3 Burnup Equation Coefficients

The applicant used a curve-fitting method to develop a seven-coefficient equation and associated coefficients. The decay heat calculations were performed with SAS2H/ORIGEN-S, and are similar to the calculations performed for the design-basis source terms. The applicant did not include thermal contributions from non-fuel hardware in deriving the coefficients. The applicant indicated that the user will be required to verify each fuel assembly conforms with established thermal limits and account for non-fuel hardware as necessary. The coefficients were developed by fitting ORIGEN-S calculated data for specific cooling times ranging from 3 to 20 years for each fuel assembly class. ORIGEN-S calculations were performed for enrichments ranging from 0.7 wt% to 5.0 wt% U-235. The applicant used GNUPLOT to calculate the coefficients, including an adjustment to assure all data points are bounded by the fit. The derived coefficients are specified in proposed Tables 2.1.28 and 2.1.29. The applicant also indicated that there is uncertainty associated with the ORIGEN-S calculations. The applicant estimated that errors in the decay heat calculations were between 1.5% and 5.5%, depending on fuel cooling times. Therefore, the applicant applied a 5% decay heat penalty to the derivation of the coefficients for both the PWR and BWR array classes.

5.2.4 Staff Evaluation

The staff reviewed the source term analyses and burnup equation method in Section 5.2 of the proposed FSAR. The staff has reasonable assurance that the design basis bounding gamma and neutron source terms for the revised HI-STORM 100 Cask System shielding analyses are acceptable. The staff performed confirmatory analyses of selected bounding burnups and cooling times with SAS2. The staff also has reasonable assurance that the coefficients generated by the applicant for the burnup equation are acceptable. The graphical representations presented by the applicant demonstrate that the curve-fitting technique is acceptable. The staff performed confirmatory calculations for selected coefficients for selected fuel classes to test the validity of the equation and coefficients. The staff calculated decay heat source terms with SAS2 at selected burnups and cooling times, and compared them to the values associated with the burnup equation method. The calculated decay heats for selected combinations were in general agreement with burnups and associated thermal values applied in the new burnup equation method.

The staff also agrees that the applicant has reasonably addressed thermal uncertainties in high burnup fuel with respect to the HI-STORM 100 Cask System design and contents requested in this amendment. The staff's confirmatory analysis of the derivation of the burnup equation coefficients indicated the 5% decay heat penalty was incorporated for both the PWR and BWR fuel classes. The applicant did not explicitly quantify the dose impact of conservatisms in the shielding analysis versus the uncertainties in the neutron and gamma source terms in the bounding analysis. However, the staff has reasonable assurance that there are conservatisms in the source term analyses because the applicant applied extreme burnups (e.g., > 65 GWD/MTU) that exceed the maximum burnups specified for loading, and that exceed the maximum decay heat allowed for loading. The staff notes that this does not constitute automatic approval of these extreme burnup source terms as acceptable fuel for future designs. Each user should also consider accounting for any uncertainties in its 10 CFR 72.212 dose analyses. Limits such as burnup, cooling times and decay heats and the burnup equation coefficients are incorporated into Appendix B of the CoC. In addition, dose limits based on the shielding analysis are incorporated into criteria for TS 5.7.

5.3 Shielding Model Specifications

The HI-STORM 100 Cask System shielding and source configuration is described in Sections 5.3 and 5.4 of the proposed FSAR. The applicant used shielding model specifications similar to those previously approved for the HI-STORM 100 Cask System to calculate bounding near-field and off-site dose rates. The shielding models used in this amendment include models for the 100S Version B overpack. Staff's evaluation of the use of the 100S Version B overpack to provide bounding dose rates for the HI-STORM 100 Cask System is discussed in Section 5.4.5 of this SER. The shielding model specifications and methods used with MCNP are similar to those previously approved for the HI-STORM 100 Cask System.

5.3.1 Shielding and Source Configuration

The applicant used a shielding and source configuration similar to the configuration previously approved for the HI-STORM 100 Cask System, accounting for differences in the modeled overpack and changes in the contents. The applicant indicated the source configuration of damaged fuel in the MPC-32 configuration would behave similar to damaged fuel configurations already analyzed and approved for the MPC-24 and MPC-68. Therefore, the applicant concluded the shielding performance of the MPC-32 would not be affected by the damaged fuel.

5.3.2 Material Properties

The applicant used the same material properties as previously approved for the HI-STORM 100 Cask System, with the exception of the overpack concrete (see below). The applicant noted that neutron absorbing panels in the basket have been represented as BORAL®. The applicant indicated that the use of METAMIC® as an alternate neutron absorbing material would not significantly affect the shielding ability of the canisters.

The applicant reduced the minimum allowable concrete density to 140 lb/ft3 for the overpack body, lid and pedestal (for those overpacks with concrete in the pedestal). In the currently approved FSAR, the pedestal and lid concrete have a minimum density of 146 lb/ft3 and the overpack body has either a 146 lb/ft3 or a 155 lb/ft3 minimum density, depending upon the presence of an inner shield shell. This new minimum density was used to determine all of the dose rates provided in the proposed FSAR. Section 5.3.2 of the proposed FSAR notes that the concrete density can be increased up to 200 lb/ft3 at the request of the user to improve the shielding characteristics of the system and address potential ALARA considerations. As stated in Section 5.1.1 of this SER, ALARA criteria (both for occupational and public dose) should be considered by the site user such that overpacks with the maximum concrete density are used unless significant site circumstances preclude the use of overpacks with the higher density concrete. The site user should appropriately consider ALARA principles (both for occupational and public dose) to determine the concrete density for overpacks to be used at its site such that overpacks with the maximum concrete density supported by site conditions, including any reasonable modifications, are used.

5.3.3 Staff Evaluation

The staff evaluated the shielding models and found them acceptable. The shielding model, shielding and source configuration, and material properties are similar to those previously approved by NRC, with the stated exceptions. Based on the statements and calculations presented by the applicant, the staff finds the model, as modified in the proposed FSAR, is valid for the revised contents and design changes.

5.4 Shielding Analyses

The applicant presented dose rates for the new bounding source terms for both normal conditions and accident conditions in proposed FSAR Sections 5.4 and 11. The applicant indicated it used the same shielding analysis techniques as previously approved for the HI-STORM 100 Cask System.

5.4.1 Normal Conditions

The applicant presented new bounding dose rates for various locations surrounding the HI-STORM 100S Version B overpack (added under 10 CFR 72.48) and the HI-TRAC transfer cask designs. The peak dose rates at different locations of the overpack and transfer casks vary, based on the specific MPC configuration and the range of bounding burnup and cooling times used in the analyses. There is not a single MPC configuration or burnup and cooling time combination that results in bounding dose rates at all exterior locations of every design. The maximum overpack surface dose rates at the side (mid-height), top (center), and vents are reported as approximately 273 mrem/hr, 27 mrem/hr, and 130 mrem/hr, respectively. The maximum surface dose rates for the side and top of the 100-ton HI-TRAC transfer cask are reported as 3.8 rem/hr and 1.1 rem/hr, respectively.

The applicant presented transfer cask dose profiles in Figures 5.1.5 through 5.1.11 of the amendment to show that the dose rates significantly decrease from peak locations to the edges of the top, bottom, and sides of the cask. Chapter 10 of the proposed FSAR indicates that exposures from localized peak dose rates may be mitigated by ALARA practices such as controlling actual locations of personnel and using temporary shielding during loading and unloading operations. It is important to note that these figures are based upon calculations with a uniform fuel loading pattern and no non-fuel hardware in the MPC. As discussed in Section 5.1.3 of this SER, the proposed regional loading pattern that would allow assemblies of higher heat loads to be loaded in the outer basket region will affect the dose profile across the top and base of the cask, keeping dose rates higher across more of the cask top and base. Also, the addition of non-fuel hardware may affect the axial dose rate profile. For example, the amendment proposes to allow an increased number of CRAs (twelve versus the currently approved four) to be loaded into the PWR MPCs. CRAs are used in reactors such that neutron activation occurs predominantly in the lower areas of the assembly; therefore, the CRAs' contribute mainly to dose rates around the lower axial areas of the cask, resulting in the dose rates not declining as much from the axial peak as would otherwise occur. This effect would be most noticeable in places around a cask loaded with a MPC-24, 24E, or 24EF in which some basket locations containing CRAs are not completely shielded by basket locations not containing CRAs. Thus, the site user should properly consider the actual spent fuel and non-fuel hardware loading configuration to ensure the implementation of all necessary ALARA precautions.

5.4.2 Occupational Exposures

The applicant estimated higher occupational exposures in Chapter 10 of the proposed FSAR. The exposures were based on estimations from surrounding dose rates calculated in proposed FSAR Chapter 5 and the operating procedures referenced in Chapter 8. The staff found the occupational exposures to be acceptable, as discussed in Section 10 of this SER.

5.4.3 Off-site Dose Calculations

The applicant estimated offsite dose rates at the site boundary for a single cask and an example 2x3 cask array in Section 5.4 of the proposed FSAR. Based on proposed Table 5.4.6, the analyses indicated that the minimum distance at which the annual dose limit of 25 mrem is satisfied for a single cask (assuming design-basis fuel and full occupancy) is increased from 300 meters to 350 meters. The analyses indicated that the minimum distance at which the annual dose limit is satisfied for a 2x3 cask array, is increased from 400 meters to 450 meters. Off-site dose calculations for both direct radiation and releases are further evaluated in Section 10.4 of this SER.

5.4.4 Accident Conditions

Chapter 11 of the proposed FSAR does not identify an accident that significantly degrades the shielding of the HI-STORM 100 overpack; however, the tip-over accident changes the shielding geometry, the cask base being now exposed. The applicant calculated the dose rate at 100 meters for this scenario (see proposed FSAR Section 11.2). Still, loss of water in the transfer cask water jacket remains the bounding accident for direct radiation. The applicant estimated, in proposed FSAR Section 5.1.2, that the transfer cask accident dose rates would be approximately 3 mrem/hr at 100 meters. Based on this exposure rate, the accumulated dose at the controlled area boundary would be approximately 2.2 rem, assuming a 30-day occupancy. The applicant concluded that the 5 rem limit in 10 CFR 72.106(b) would not be exceeded for the most severe design-basis shielding accident identified in proposed FSAR Chapter 11.

5.4.5 Staff Evaluation

Section 10 of this SER evaluates the overall dose (i.e., direct radiation and hypothetical radionuclide release) from the HI-STORM 100 Cask System. The staff reviewed the dose calculations for normal operations and finds them acceptable. The dose calculations are based upon the non-fuel hardware being BPRAs. Staff notes that the applicant had previously proposed allowing CRAs to be loaded in all locations in the PWR MPC baskets. However, the applicant modified its proposal to only allow loading of CRAs in the inner twelve locations of the PWR MPC baskets. The applicant made this modification due to concerns that dose rates from the initially proposed CRA loading configuration were significantly higher around the lower areas of the cask and that analyses (e.g., occupational dose, accident condition doses, dose-to-distance analyses for cask arrays and a single cask) with BPRAs were not adequately bounding. Staff reviewed the dose rate results and finds that analyses based upon BPRAs are acceptable and adequately bounding for casks loaded with CRAs in the new proposed configuration. Staff does note, however, that the modified proposal for loading CRAs results in azimuthal variations (hot spots) in the radial dose rates around the MPC-24, 24E and 24EF, similar to the proposed expansion of the inner basket region in these MPCs, though this phenomenon is confined to the lower areas around the cask for CRAs. Therefore, the user's analysis should include azimuthal dose rate variations arising from CRAs if loading this non-fuel hardware in these MPCs, similar to that done for regionalized loading (see Section 5.1.3 of this SER).

Dose rates were calculated for the 100S Version B overpack to provide bounding dose rates for the HI-STORM 100 Cask System overpacks loaded with the allowable contents, as proposed in this amendment. The Version B overpack was added to the HI-STORM 100 Cask System under 10 CFR 72.48, and its analyses were added in a previous amendment to aid staff in developing TS 5.7. In its review of that amendment, staff examined the dose rates for the Version B and found that they appear consistent with the associated differences in the Version B design versus the other overpack designs. Staff also noted that the applicant used the same shielding methodology that had been used for prior amendments approved by the NRC. A review of the near-field and off-site dose rates from the HI-STORM 100 Cask System overpacks indicated that the dose rates from the Version B are higher than those from the other HI-STORM 100 Cask System overpacks, with the exception of the near-field dose rate at the outlet vent. While the Version B near-field dose rate at the outlet vent is less than the bounding outlet vent dose rate (obtained from the 100S overpack), the difference between these dose rates is not large. Furthermore, the general licensee is required by TS 5.7 to establish an outlet vent dose rate limit and, in conjunction with 10 CFR 20.1101(b) and 10 CFR 72.104(b), to keep occupational doses ALARA. Thus, based upon the foregoing findings, staff finds the dose rates from the Version B overpack calculations acceptable for use in the shielding analysis for this amendment.

The staff has reasonable assurance that compliance with 10 CFR Part 20 and 10 CFR 72.104(a) from direct radiation can be achieved by general licensees. The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and distances. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities, such as reactor operations. Each general licensee is responsible for verifying compliance with 10 CFR 72.104(a) in accordance with 10 CFR 72.212. In addition, a general licensee will also have an established radiation protection program, as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public and workers, as required, by evaluation and measurements. Because the revised contents result in direct radiation dose rates that are significantly higher than those previously approved for the HI-STORM 100 Cask System, each user may be required to take additional ALARA precautions to minimize doses to personnel and to make additional use of realistic fuel characteristics and distances to demonstrate compliance with public dose limits in 10 CFR Part 20 and Part 72.

The staff reviewed the accident dose analysis and found it acceptable for the specific design and contents requested in the proposed FSAR. The staff has reasonable assurance that the direct radiation from the HI-STORM 100 Cask System satisfies 10 CFR 72.106(b) at or beyond a controlled boundary of 100 meters from the design-basis accidents. The estimated dose to members of the public at 100 meters and at further distances for a conservative exposure time of 30 days is approximately 50% below the 5 rem accident limit in 10 CFR 72.106(b). The staff notes that the off-site accident dose may be less accurate because precise exposure times cannot be predicted. However, the staff notes that the 30-day exposure is conservative based on realistic considerations and the fact that direct radiation is relatively easy to mitigate within a reasonable amount of time.

As discussed in Section 10.4 of this SER, general criteria for a radiation protection program that is tailored to the dose rates from the HI-STORM 100 Cask System are provided in TS 5.7 (See Section 10.4 of this SER). The limits for fuel assembly decay heat, burnup, cooling time, enrichment, and other characteristics are specified in Appendix B of the CoC. The burnup equation, equation coefficients and associated limits as well as parameters for regional and uniform loading are also incorporated into Appendix B of the CoC for those contents for which they are applicable.

5.5 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F5.1 The proposed FSAR sufficiently describes shielding design features and design criteria for the structures, systems, and components important to safety.

F5.2 Radiation shielding features of the HI-STORM 100 Cask System 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 each general licensee. The HI-STORM 100 Cask System shielding features (as approved by NRC) are designed to assist in meeting these requirements.

F5.4 The staff concludes that the design of the radiation protection system of the HI-STORM 100 Cask System can be operated 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, the appropriate regulatory guides, applicable codes and standards, the applicant’s analyses, the staff’s confirmatory analyses, and acceptable engineering practices.

6.0 CRITICALITY EVALUATION

The purpose of the criticality review is to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered for the proposed changes to the HI-STORM 100 Cask System such that the cask system continues to meet the criticality requirements of 10 CFR Part 72. These requirements include: 10 CFR 72.124(a), 72.124(b), 72.236(c), and 72.236(g). The proposed FSAR was also reviewed to determine whether the cask system, as modified by this amendment, fulfills the acceptance criteria listed in Section 6 of NUREG-1536.

The amendment proposes the following fuel rod and assembly specification and cask loading condition modifications for the 16x16A assembly array/class relevant to criticality:

1. Increased maximum fuel rod clad internal diameter (ID) from 0.3320 inches to 0.3350 inches.

2. Decreased minimum guide/instrument tube thickness from 0.0400 inches to 0.0350 inches.

3. Increased minimum soluble boron concentration for intact fuel assemblies from 1300 to 1400 ppmb (fuel enrichment up to 4.1 wt % Uranium-235) and from 1900 to 2000 ppmb (fuel enrichment up to 5.0 wt % Uranium-235).

These modifications were initially to support the proposed addition of the 16x16 CE System 80-type assembly to the approved contents. Upon staff review, however, it was determined that the System 80-type assembly does not physically fit in the HI-STORM 100 Cask System. Therefore, the proposal to add the System 80-type assembly to the approved contents was withdrawn by the applicant. However, the proposed changes to the criticality analysis were retained since these changes bound the current fuel rod and assembly specifications and cask loading conditions for the approved 16x16A assembly array/class contents. Other changes proposed in the amendment application do not effect the criticality analysis and are therefore not discussed in this section of the SER.

The amendment application provides supporting analyses that use the method that was previously reviewed by the staff for analyzing the HI-STORM 100 Cask System. The applicant's evaluation and the staff's confirmatory review of the requested changes are described below.

6.1 Criticality Design Criteria and Features

The design criterion for criticality safety is that the effective neutron multiplication factor, keff, 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.

In a previous SER, the staff analyzed the design features of the HI-STORM 100 Cask System relied upon to prevent criticality and also verified that these features would not be adversely affected by design-basis off-normal and postulated accident events. The staff reviewed the changes proposed in this amendment and finds that the changes do not affect the staff's previous finding. The staff also reviewed the applicant's model descriptions and assumptions in proposed Chapter 6 of the proposed FSAR and finds that they are consistent with the description of the cask system, as modified in the amendment application. The staff reviewed the proposed changes to the CoC to ensure that the fuel specifications important to criticality safety are included and consistent with the amendment request. The staff verified that the proposed FSAR contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

6.2 Fuel Specification

The fuel rod and assembly specifications that define the HI-STORM 100 Cask System's approved contents are listed in Tables 2.1-1 through 2.1-3 in Section 2, Appendix B to the CoC. The amendment proposes changes to the fuel specifications for the 16x16A assembly array/class, for which supporting analyses are provided in proposed Chapter 6 of the FSAR. Only analyses involving this assembly class were reviewed in this SER since no changes to the specifications affecting criticality for the other assembly arrays/classes are proposed. Staff reviewed the analysis results for all the assembly classes and finds that, for the MPC-24, 24E and 24EF, changes in system reactivity associated with the proposed changes to the 16x16A assembly specifications are both bounded by the most reactive approved contents and subcritical.

For the MPC-32 and 32F, different levels of soluble boron are required for criticality control. These levels depend upon the assembly class and enrichment. The amendment proposes an increase in the minimum soluble boron levels specified in Limiting Conditions for Operation (LCO) 3.3.1 of the TS for loading intact 16x16A array/class assemblies to maintain system subcriticality. Staff reviewed these changes and finds that, with the proposed increase in soluble boron levels, these MPCs loaded with intact assemblies meeting the proposed 16x16A assembly class specifications are adequately subcritical. No changes were proposed to the minimum soluble boron levels for loading damaged 16x16A fuel assemblies into these MPCs. Staff reviewed the MPC-32 and 32F loaded with 16x16A assemblies meeting the proposed specifications, damaged assemblies being loaded in all allowable locations, and finds that the system reactivity remains adequately subcritical for these conditions as well.

6.3 Model Specification

6.3.1 Configuration

The analysis in the amendment uses the same model configurations that were previously reviewed and found acceptable by staff. These configurations are based on the engineering drawings in Section 1.5, with consideration of worst-case tolerances, and include the assumptions of fresh fuel isotopics, 75% credit for the 10B loading in the BORAL® panels (90% credit in METAMIC® panels), and un-borated water in the fuel rod gap. These configurations also account for the effects that design-basis off-normal and accident events have on system reactivity. Based on the applicant's evaluation and staff's independent analysis, the staff finds that the applicant's considerations of the most reactive moderating conditions apply to the cask system as modified by this amendment.

6.3.2 Material Properties

The compositions and densities of the materials considered in the computational model are unchanged from those in prior analyses of the HI-STORM 100 Cask System, with the exception of the increases in the minimum soluble boron content in a MPC-32/32F loaded with intact 16x16A array/class fuel assemblies (see Section 6.3 of this SER). This change is reflected in the applicant's model.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant used the same computer codes and methods for processing neutron cross-sections and calculating system keff as were used for previous submittals, and staff's evaluation is provided in a previous SER. MCNP4a was used for keff calculations while NITAWL-II was used in conjunction with CELLDAN for cross-section processing.The applicant used MCNP4a for the keff calculations and NITAWL-KENO5a to perform independent verification calculations.

The staff used the CSAS/KENO V.a code in the SCALE suite of analytical codes to perform confirmatory analyses. The CSAS/KENO code is a multi-group Monte Carlo code developed by Oak Ridge National Laboratory for performing criticality analyses and is appropriate for this particular application and fuel system. The staff calculations used the 238-group ENDF-B/V cross-section libraries. CENTRM, which computes continuous energy neutron spectra to provide highly accurate angular fluxes and flux moments, was used, in part, to confirm that the use of NITAWL is appropriate in this case. The staff finds that the codes and cross-section libraries used by the applicant are appropriate for this particular application and fuel system.

The staff's computational model used the information provided in the technical drawings and proposed Chapter 6 of the FSAR. Fuel parameter values were taken from the tables in proposed Section 6.2 of the FSAR. Staff performed additional calculations with changes to the16x16A fuel assembly in the MPC-24E and MPC-32 and all permissible damaged fuel locations occupied by assemblies modeled in the most reactive conditions possible given a maximum allowable assembly weight of 1720 lb. These additional calculations are described in Section 6.5.2 of this SER.

6.4.2 Multiplication Factor

Results of the applicant's criticality analyses show that the keff (including biases and uncertainties) in the HI-STORM 100 Cask System will remain below 0.95 for all fuel loadings. The results of the applicant's criticality calculations for the bounding assemblies are given in proposed Tables 6.1.1 through 6.1.12 of the FSAR. Only the array/class 16x16A fuel assemblies were analyzed for this application.

The staff performed confirmatory calculations to analyze a cask loaded with intact 16x16A assemblies that meet the proposed specifications. The staff also analyzed the reactivity of a loaded cask with all permissible damaged fuel locations occupied by arrays of bare fuel pins of various pitches. All non-fuel hardware was removed and replaced with as much fissile material as possible to investigate possible effects on cask reactivity. Given a design basis weight of 1455 lb, excluding non-fuel hardware and the damaged fuel can, for a 16x16A fuel assembly and a maximum permissible weight of 1720 lb in a fuel basket location, an additional 265 lb of enriched uranium oxide was introduced to each damaged assembly by increasing the density of the bare fuel pins used to characterize damaged fuel. The results of the applicant's calculations are within acceptable agreement of the staff's results for the proposed 16x16A assembly contents. Staff's results were all below the acceptance limit.

Based on its review of the applicant's criticality evaluation and staff's independent calculations, the staff finds reasonable assurance that the HI-STORM 100 Cask System will remain sub-critical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant used the benchmark analysis that was previously reviewed by the staff. Staff reviewed the proposed changes to the approved contents and the associated changes to the criticality analysis and finds that the benchmark analysis appropriately covers the modified criticality analysis and is thus an acceptable and conservative method for determining the computational bias.

6.5 Supplemental Information

The specifications of the spent fuel assembly classes that can be loaded into the HI-STORM 100 Cask System are listed in Section 2.0 of Appendix B to the CoC. All supportive information has been provided in the proposed FSAR, primarily in Chapters 1, 2, and 6.

6.6 Evaluation Findings

Based on its review of the information provided in the proposed FSAR and the staff's own confirmatory analysis of design changes, the staff finds 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 proposed FSAR to enable an evaluation of their effectiveness.

F6.2 The HI-STORM 100 Cask System is designed to be sub-critical under all credible conditions.

F6.3 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.

F6.4 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 is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.0 CONFINEMENT EVALUATION

The objective of the confinement review of the HI-STORM 100 Cask System is to ensure 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 otherwise might lead to gross ruptures. These requirements include 10 CFR 72.236(l). The objective includes review of changes to the confinement design characteristics and confinement analyses for the HI-STORM 100 Cask System, proposed by this amendment request. The applicant did not request any significant changes to the confinement system. The applicant made an editorial change to Chapter 7 that does not affect either the confinement evaluation or the TS.

The confinement boundary on the MPC design includes the following: MPC Shell, bottom baseplate, MPC lids (including vent and drain port cover plates), MPC closure ring, and associated welds. Penetrations to the confinement boundary consist of two penetrations, the MPC vent and drain ports. All components of the confinement boundary are important to safety, Category A, as specified in the applicant's proposed FSAR Table 2.2.6. The MPC confinement boundary is designed, fabricated, inspected, and tested in accordance with ASME Code, Section III, Subsection NB. NRC approved alternatives to the ASME Code are identified in Table 3-1 of the TS. The confinement system design did not change in this amendment request.

7.1 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F7.1 Chapter 7 of the proposed FSAR 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 any 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. Because the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.

F7.5 The confinement system has been evaluated by analysis. Based on successful completion of specified testing and examination procedures, described in proposed FSAR Chapters 7, 8 and 9, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F7.6 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 analysis, the staff's confirmatory review, and acceptable engineering practices.

8.0 OPERATING PROCEDURES

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

Only those changes that affect operating procedures are discussed in this section. The staff reviewed the proposed amendment to the HI-STORM 100 Cask System FSAR and proposed changes to the CoC and TS, to ensure the changes in the operating procedures described in Chapter 8 of the proposed FSAR, meet the following regulatory requirements: 10 CFR 72.104(b), 72.212 (b)(9), 72.234(f), and 72.236(h) and (i). The proposed changes to the FSAR were also reviewed to determine whether the cask system fulfills the acceptance criteria listed in Section 8 of NUREG-1536.

The following changes were reviewed to determine if changes to the operating procedures to accommodate design modifications for the HI-STORM 100 Cask System, as described in the proposed FSAR, are acceptable to the staff:

a. Addition of a new decay heat regionalized scheme.
b. A change in the MPC-32 fuel storage locations for fuel with APSRs and a change in the MPC-24, MPC-24E and MPC-32 fuel storage locations for fuel with CRAs, RCCAs, and CEAs.
c. Addition of a threshold heat load below which operation of the SCS would not be required.

The staff's conclusions, summarized below, are based on information provided in the proposed amendment, as revised, to the HI-STORM 100 Cask System FSAR.

8.1 Decay Heat Regionalized Fuel Loading

In its review of the shielding analysis, the staff noted that the proposed changes to the contents result in significantly higher dose rates than those resulting from the currently approved contents. The staff also noted that proposed changes to the regional loading patterns in the MPCs will also affect the dose rate profile around the cask. Expansion of the inner region for regional loading in the MPC-24 and MPC24E/24EF will result in hot spots in the radial dose profile. Regional loading patterns that place assemblies of higher heat loads in the outer region will result in dose rates remaining higher across a greater area of the cask base and top. Therefore, each user may be required to take additional ALARA precautions to minimize doses to personnel.

8.2 PWR Control Components' Storage Locations

In its review of the shielding analysis, the staff noted that the proposed changes to the allowable loading configurations of CRA and APSR non-fuel hardware contents result in significantly higher casks dose rates than those resulting from their currently approved loading configurations. These dose rate increases are primarily confined to the base of the cask, an area that will only be a concern during horizontal transport of the transfer cask, when the cask base is easily accessible. The staff does note that, while in general the overall impact of the proposed changes on dose rates around the cask is small, the proposed configuration of CRAs in the MPC-24 and the MPC- 24E/24EF will result in hot spots in the radial dose profile around the lower areas of the cask due to some basket locations containing CRAs not being completely shielded by basket locations in which CRAs may not be loaded. Therefore, each user may be required to take additional ALARA precautions to minimize doses to personnel.

8.3 Vacuum Drying Method

The applicant added a Caution note to warn the user that there is a vacuum drying time limit for MPCs above a threshold heat load. The vacuum drying time limit is based on the thermal evaluation presented in Chapter 4 of the proposed FSAR. Technical Specification (TS) 3.1.1 specifies that for MPCs having a total decay heat between 23 and 28.74 kW, vacuum drying is subject to a time limit of 40 hours. As it has been stated in Chapter 4 of this SER, this limit will account for uncertainties of the 2-D thermal models regarding modeling simplifications and radiation heat transfer model, as identified by the staff. Also, during vacuum drying operations, the annulus between the MPC and the HI-TRAC must either be maintained full of water (lower heat MPC-24/24E) or must have a continuous water flow maintained through the annulus between the MPC and the HI-TRAC to keep the water temperature at the outlet of the annulus below 125°F (51.6°C). The applicant stated that the water temperature must be verified via measurement. This requirement has also been specified in TS 3.1.1.

The applicant also stated that If MPC vacuum drying acceptance criteria are not met during allowable time, the MPC cavity should be backfilled with helium to a pressure equal or larger than 0.5 atm and vacuum drying time should be reset per TS 3.1.1. As it has been stated in Chapter 4 of this SER, Backfilling the MPC with a helium environment will provide a better heat transfer environment and will assure ISG-11, Rev. 3 short-term peak cladding temperature limits are not exceeded.

8.4 Supplemental Cooling System

The applicant provided a general description and requirements for the SCS in proposed Appendix 2.C of the FSAR. The SCS is utilized, as necessary, to maintain the peak fuel cladding temperature below the limit set forth in Chapter 2 of the proposed FSAR during normal short-term operations. Based on the thermal evaluation provided in Chapter 4 of the proposed FSAR for HI-TRAC transport in a vertical orientation, the applicant stated that in order to assure the peak cladding temperature for Moderate Burnup Fuel (MBF), the use of the Supplemental Coolant System (SCS) is required for an MPC heat load larger than 28.74 kW and if one or more High Burnup Fuel Assemblies are loaded (HBF). The SCS should be sized to extract 36.9 kW from the MPC.

8.5 Protection of Fuel From Oxidation During Cask Loading/Unloading Operations

The NRC staff identified a potential issue regarding exposure of spent fuel to air during cask loading/unloading operations. To ensure consistency with information and text approved in Amendment 3 to the HI-STORM 100 Cask System CoC effective, May 29, 2007, the staff reviewed the information submitted in support of this proposed amendment. If fuel with a cladding perforation is exposed to air, the possibility exists that significant oxidation of the fuel pellets could occur. If the fuel pellets oxidize sufficiently, the cladding can be ruptured.
The applicant was queried about details of the operational procedure to ensure that the cladding (and hence, fuel pellets, if small cladding breaches exist) was not exposed to air during cask loading/unloading operations. If exposure to air did occur, the applicant had the alternative option of showing that the time and temperature conditions were insufficient to cause significant fuel pellet oxidation.

The applicant confirmed that procedural and mechanical barriers have been in place for all Holtec fuel canisters to prevent exposure of the heated fuel cladding to air during these operations. Additionally, a new note was added to proposed FSAR Section 8.1 to caution the user against exposing the fuel assemblies to atmosphere to prevent oxidation and potential fuel damage. The water lowering process (a part of the loading operation) is currently controlled by using a fixed length dip tube to draw water from the MPC and limit the minimum water level that can be achieved. This has proved to be a reliable barrier against inadvertent lowering of water level to below the top of the fuel rods. Final water removal is accomplished by blowing down the water in a welded canister using inert gas.

A new note has also been added to the proposed FSAR Section 8.3 (MPC unloading operations) to caution the user against exposing the fuel assemblies to atmosphere to prevent oxidation and potential fuel damage. The environment within the MPC is filled with helium during storage and any pre-cooling of the fuel prior to introducing water is accomplished by circulating helium through the MPC.

The staff finds that the applicant has taken adequate steps to protect the fuel pellets from excessive oxidation leading to gross cladding breaches.

8.6 Helium Leak Test of Lid Welds

To ensure consistency with information and text approved in Amendment 3 to the HI-STORM 100 Cask System CoC effective, May 29, 2007, the staff reviewed the following information submitted in support of this proposed amendment. Using the staff guidance of ISG-18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," the applicant has exercised the option to eliminate the helium leak test normally required of the structural-lid-to-shell weld. Elimination of this test is based upon meeting all the criteria of the ISG, which the applicant has demonstrated. For the remaining welds in the confinement boundary, a helium leak test in accordance with the "leak tight" criterial of ANSI N14.5-1997 is applied, in accordance with staff guidance.

That staff guidance, as clarified during conversations with the applicant, states that any weld that is part of the confinement boundary must be helium leak tested. An exemption from the helium leak test is granted under the guidance if (among other requirements): 1) the weld is a multi-pass weld consisting of three or more distinct weld layers, and, 2) a flaw tolerance analysis for that weld has been performed per the guidance of ISG-15, and, 3) the weld is liquid penetrant (PT) examined: 1) after the root pass, and, 2) after each time a weld deposit depth is applied that does not exceed the dimension of the flaw tolerance analysis, and, 3) the final pass is completed, for a minimum of three different PT examinations. Also, the weld metal must not have been deposited when helium could potentially pressurize the root of the weld (no credit for isolating the weld root from the canister helium fill gas is allowed for any shut-off devices under the weld root).
Any welds that are part of the required redundant closure are not required to be helium leak tested. For the Holtec design this means the lid cover ring welds. The Code required weld examinations are sufficient for these secondary containment welds. The staff finds that the applicant has conformed with the guidance and intent of ISG-18.

8.7 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

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 proposed FSAR. These procedures were appropriately modified to include the design modifications made in the amendment. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 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 proposed FSAR with the addition of helium leak rate testing requirements for the vent and drain port cover plates, as described in Section 8.5 of this SER, 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.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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

Under the guidance of ISG-18, the helium leak rate test of the lid-to-shell structural weld is no longer required. The helium leak rate testing of the vent and drain port cover plates must still be conducted in accordance with ANSI N-14.5.

The staff agrees that the changes are appropriate. Additional discussion of the staff's review regarding helium leak rate testing can be found in Section 8 of this SER, and discussion of the neutron absorber materials can be found in Section 6 of this SER.

9.1 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F9.1 The staff concludes that the modifications made to the acceptance tests and maintenance program for the amendment to the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied.

10.0 RADIATION PROTECTION EVALUATION

The objective of the review of this section is to ensure that the capability of the current and the revised radiation protection design features, design criteria, and operating procedures, as appropriate, of the HI-STORM 100 Cask System can meet regulatory dose requirements for the proposed contents. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d).

Calculated occupational exposures from the HI-STORM 100 Cask System are based on the direct radiation dose rates calculated in Chapter 5 of the proposed FSAR and the operating procedures discussed in Chapter 8 of the proposed FSAR. Calculated doses to individuals beyond the controlled area boundary (members of the public) are determined from the direct radiation (including skyshine) dose rates calculated in Chapter 5 of the proposed FSAR.

Changes resulting from the proposed contents were reviewed to determine if the radiation protection design features for the HI-STORM 100 Cask System, as described in the proposed FSAR, are acceptable to the staff.

The staff's conclusions, summarized below, are based on information provided in the proposed amendment, as revised, to the HI-STORM 100 Cask System FSAR.

10.1 Radiation Protection Design Criteria and Design Features

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. As required by 10 CFR Part 20 and 10 CFR 72.212, each general licensee is responsible for demonstrating site-specific compliance with these requirements. In addition, TS 5.7 establishes direct radiation dose rate limits and other radiation protection criteria for the cask system. These criteria are based on bounding dose rate values, which are used to determine occupational and off-site exposures and other design-specific factors important in the radiation protection system. The radiation protection design features are referenced in Chapter 10 of the proposed FSAR. The radiation protection design features of the HI-STORM 100 Cask System are the same as the radiation protection design features previously approved.

The staff reviewed the design criteria and found them acceptable. Sections 5, 7, and 8 of this SER discuss specific staff evaluations of the design criteria and features for the shielding system, confinement system, and operating procedures, as appropriate. Section 11 of this SER discusses staff evaluations of the capability of the shielding and confinement features during off-normal and accident conditions, as appropriate.

10.2 ALARA

The ALARA objectives, procedures, practices, and policies were not changed and have been previously approved. Each site licensee will apply its additional site-specific ALARA objectives, policies, procedures, and practices for members of the public and personnel.

The staff considered the previously approved ALARA assessment for the HI-STORM 100 Cask System and found it acceptable for the described dose rates. Section 8 of this SER discusses the staff's evaluation of the operating procedures with respect to ALARA principles and practices, as appropriate. Operational ALARA objectives, policies, procedures, and practices are the responsibility of the site licensee, as required by 10 CFR Part 20 and 10 CFR 72.104(b).

The staff does note that certain aspects of the proposed changes in this amendment require particular attention with regard to implementation of ALARA principles. One is the proposed reduction in the minimum overpack concrete density. While staff finds that, by maintaining the overpack as a thick walled structure, the design continues to meet the design considerations posed by the applicant in Section 10.1.2 of the proposed FSAR, application of ALARA principles also involves consideration of the properties, and hence the effectiveness, of the materials comprising that structure, as discussed in Section 5.1.1 of this SER. Also, the revised contents result in significantly higher direct radiation dose rates from that previously approved for the HI-STORM 100 Cask System; therefore, each user may be required to take additional ALARA precautions to minimize doses to personnel and to make additional use of realistic fuel characteristics and distances to demonstrate compliance with public dose limits in 10 CFR Part 20 and 10 CFR Part 72. In addition to the contents, proposed changes to the configurations of the contents, such as regional loading with fuel of higher heat loads in the outer basket region, expansion of the inner region of some MPCs, and loading CRAs in more basket locations, will affect dose rates and/or dose rate profiles (as discussed in Section 5 of this SER), requiring consideration in the site user's determination of needed ALARA precautions.

10.3 Occupational Exposures

The staff reviewed the overall occupational dose estimates and found them acceptable. The occupational dose exposure estimates provide reasonable assurance that occupational limits in 10 CFR Part 20, Subpart C can be achieved. The staff expects actual operating times and personnel exposure rates will vary for each system, depending on site-specific operating conditions, including detailed procedures and special measures taken to maintain exposures ALARA. The collective exposures will be distributed among multiple personnel responsible for various tasks. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20, Subpart B. In addition, each licensee will demonstrate compliance with occupational dose limits in 10 CFR Part 20, Subpart C and other site-specific 10 CFR Part 50 license requirements by means of evaluations and measurements. Staff's evaluation of the operating procedures is presented in Section 8 of this SER.

10.4 Public Exposures From Normal and Off-Normal Conditions

The applicant estimated offsite direct radiation dose rates at the site boundary for a single cask and an example 2x3 cask array in Section 5.4 of the proposed FSAR. Based on Table 5.4.6 in the proposed FSAR, the analyses indicated that the minimum distance at which the annual dose limit of 25 mrem is satisfied for a single cask (assuming design-basis fuel and full occupancy) is increased from 300 meters to 350 meters. The analyses indicated that the minimum distance at which the annual dose limit is satisfied for a 2x3 cask array is increased from 400 meters to 450 meters.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each general licensee. The general licensee using the HI-STORM 100 Cask System must perform a written site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance with 10 CFR 72.104(a). The actual doses to an individual beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, distances, and use of engineered features (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 10 CFR 72.104(a) is the responsibility of each general licensee. The NRC may inspect the site-specific use of the HI-STORM 100 Cask System for compliance with radiological requirements.

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

Based on its shielding analyses, the applicant proposed modifications to the dose rate criteria established as part of the HI-STORM 100 Cask System radiation protection program in TS 5.7. The criteria include requirements for the cask user to (1) establish cask-specific surface dose limits based on its 10 CFR 72.212 analyses, (2) assure maximum surface dose rates are below values based on the bounding shielding calculations for the top and side of the overpack, (3) measure dose rates at specific locations on the cask, and (4) implement specific corrective actions if measured doses during operations exceed the limits. The applicant proposed an increase in the maximum overpack surface dose rates to account for the new bounding shielding calculations in the proposed amendment. The staff has reviewed the proposed new maximum surface dose rates, and finds them acceptable based upon the provided bounding shielding analysis.

The applicant also proposed that the requirement for the site user to establish a cask-specific dose rate limit for the top of the transfer cask and to make measurements to ensure compliance with that limit be removed from TS 5.7. The staff reviewed this proposed change and the applicant's justification but does not find the change acceptable. The applicant stated that measurements on the transfer cask are merely precautionary and that measurements on the storage overpack alone are sufficient to ensure the purpose of TS 5.7 is achieved and, therefore, the measurements on the transfer cask top have no added value while increasing personnel dose. The staff however views measurements on the transfer cask as more than precautionary. The measurements provide an early means of detecting whether a mis-load or other problem has occurred, indicated by an exceeded limit. Detection at this stage of operations provides the opportunity for appropriate corrective action to be taken prior to the MPC being placed in the storage overpack and at the ISFSI. With regard to personnel dose and ALARA, 10 CFR 72.104(b) requires the establishment of operational restrictions to meet as low as reasonably achievable objectives for direct radiation levels associated with ISFSI operations. Development of appropriate controls, surveillances and programs in a cask system's TS is an integral and necessary part of these operational restrictions. Additionally, ALARA requirements are met when a necessary surveillance is performed in a manner that minimizes occupational dose, not by removal of the surveillance. Having a TS limit for the cask top will provide the licensee with the information necessary to perform a thorough ALARA evaluation for anticipated cask work to minimize personnel exposure. Also, it is staff's expectation that radiation protection personnel perform numerous measurements around the transfer cask; therefore, the few measurements required on the transfer cask top should fit in as a part of the measurements being performed. Thus, the requirements for establishing a dose limit on the transfer cask top and performing the associated measurements are being retained in TS 5.7.

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

Chapter 11 of the proposed FSAR presents direct radiation dose rates for accident conditions and natural phenomena events to individuals beyond the controlled area. The confinement function of the canister is not affected by design-basis accidents or natural phenomena events. Therefore, there is no credible release of contents. As discussed in Section 5.4.4 of this SER, the accident direct-radiation dose analysis indicates the worst case shielding consequences result in a dose at the controlled area boundary that is 50% below the regulatory limits in 10 CFR 72.106(b). Chapter 11 of the proposed FSAR discusses or references the corrective actions for each design-basis accident, as appropriate.

The staff evaluated the public dose estimates from the proposed contents for accident conditions and natural phenomena events and finds them acceptable. Discussions of the staff's evaluations of the shielding and confinement analyses for the relevant design-basis accidents are presented in Sections 5 and 7 of this SER, respectively. A discussion of the staff's evaluation of the accident conditions and recovery actions is presented in Section 11 of this and previous SERs, as appropriate, for the proposed contents. The staff has reasonable assurance that the effects of direct radiation from bounding design-basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

10.6 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F10.1 The proposed FSAR sufficiently describes the 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 continues to be designed to provide redundant sealing of the confinement system.

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

F10.5 The proposed FSAR 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 proposed FSAR sufficiently describes the means for controlling and limiting occupational exposures for the revised contents 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 72.106 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 of 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, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

11.0 ACCIDENT ANALYSIS EVALUATION

The objective of the accident analysis review 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 proposed FSAR
3. Analyzed the safety performance of the cask system in each review area
4. Fulfilled all applicable regulatory requirements

The objective includes review of changes to the applicant's description and conclusions regarding the cause of an event, detection of an event, summary of event consequences and regulatory compliance, and corrective course(s) of action.

The regulatory requirements applicable to accident analysis changes proposed by this amendment include 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.124(a), 10 CFR 72.236(c), (d), and (l), and 10 CFR 72.212(b). This amendment was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfill the acceptance criteria listed in Section 11 of NUREG-1536.

The following proposed changes were considered for their effect on the accident analyses and conclusions:

A. An increase in the design basis maximum decay heat loads and addition of a new decay heat regionalized scheme.
B. An increase in the maximum fuel assembly weight for BWR fuel in the Multi-Purpose Canister (MPC) -68 from 700 to 730 lb.
C. Changes to the assembly characteristics of 16x16 PWR assemblies to be qualified for the HI-STORM system which include an increase in maximum fuel assembly weight for PWR fuel for assemblies that do not require upper and lower fuel spacers, a change in the Fuel Rod Clad ID and a change in minimum Guide/Instrument Tube Thickness, and a change in minimum soluble boron concentration for Array Class 16x16A for all intact fuel assemblies.
D. A change in the MPC-32 fuel storage locations for fuel with Axial Power Shaping Rod assemblies (APSRs) and a change in the MPC-24, MPC-24E and MPC-32 fuel storage locations for fuel with Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), and Control Element Assemblies (CEAs).
E. Elimination of the restriction that fuel debris can only be loaded into the MPC-24EF, MPC-32F, MPC-68F, and MPC-68FF canisters.
F. Addition of a threshold heat load below which operation of the Supplemental Cooling System (SCS) would not be required and modification of the design criteria to simplify the system.

The staff's conclusions, summarized below, are based on information provided in the amendment application, as revised, to the HI-STORM 100 Cask System FSAR.

11.1 Off-Normal and Normal Operations

Off-normal operations are Design Event II as defined by ANSI/ANS 57.9. These events can be expected to occur with moderate frequency or on the order of once per year. The HI-STORM 100 Cask System off-normal operations are described in Chapter 11 of the proposed FSAR. The off-normal conditions described in the proposed FSAR include:

1. Off-Normal Pressures
2. Off-Normal Environmental Temperatures
3. Leakage of One MPC Seal Weld
4. Partial Blockage of Air Inlets
5. Off Normal Handling of HI-TRAC Transfer Cask
6. Malfunction of FHD System
7. SCS Power Failure
8. Off-Normal Loads
9. Off Normal Wind

11.1.1 Staff Review of Off-Normal and Normal Operations

The staff reviewed the consequences of postulated off-normal events with respect to 10 CFR 72.104(a) dose limits, and found them acceptable. The radiation consequences from off-normal events are essentially the same as for normal conditions of operation for the proposed contents and design. 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 this SER further examine the radiological doses applicable to off-normal events, as appropriate.

The staff reviewed these events and found them to be bounded by evaluations contained in Chapters 3 and 4 of the proposed FSAR and accepted by the staff in Sections 3 and 4 of this SER. The staff agrees that there is no adverse impact on the HI-STORM 100 Cask System integrity from any off-normal event.

11.2 Accident Events and Conditions

Accident events and conditions are classified as Design Event III and IV. They include natural phenomena and human-induced low probability events. The applicant provided proposed changes to the analyses to demonstrate design adequacy for the accident-level events discussed below. The HI-STORM 100 Cask System postulated accidents are described in Chapter 11 of the proposed FSAR and include:

1. HI-TRAC Transfer Cask Handling Accident
2. HI-STORM 100 Overpack Handling Accidents
3. Tip Over
4. Fire Accident
5. Partial Blockage of MPC Basket Vent Holes
6. Tornado
7. Flood
8. Earthquake
9. 100% Fuel Rod Rupture
10. Confinement Boundary Leakage

11. Lightning
12. Explosion
13. 100% Blockage of Air Inlets
14. Burial Under Debris
15. Extreme Environmental Temperature
16. SCS Failure

Only those changes in the proposed amendment that might affect the above postulated accidents are addressed below.

11.2.1 Staff Review of Accident Events and Conditions

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 at or beyond the controlled area boundary of 100 meters will not exceed the limits in 10 CFR 72.106(b) for the proposed contents and design. Sections 5, 7, and 10 of this SER further examine the estimated radiological doses during accident conditions.

The staff reviewed these events and found them to be bounded by evaluations contained in Chapters 3 and 4 of the proposed FSAR and accepted by the staff in Sections 3 and 4 of this SER. The staff agrees that there is no adverse impact on the HI-STORM 100 Cask System integrity from any accident condition.

11.3 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

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 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.

F11.3 A design-basis accident or a natural phenomena event will not prevent the ready retrieval of spent fuel for further processing or disposal.

F11.4 The spent fuel will be maintained in a subcritical condition under accident conditions.

F11.5 Because instrumentation and control systems are not required, no instruments or control systems are required to remain operational under accident conditions.

F11.6 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).

F11.7 The staff concludes that the accident design criteria for the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72 and that 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 objective of this review is to assess whether the applicant has proposed modifications to CoC 1014 Conditions and Appendix A to the CoC (Technical Specifications) and if the changes are appropriate to accommodate the design modifications requested by the amendment. This review focused on evaluating whether the Conditions and TS had been revised to ensure that all safety limits and regulations were met.

The staff reviewed the proposed changes to Section 12.2.10 of the FSAR. This section provides new examples for use of the burnup equation developed by the applicant for identification of fuel that meets the acceptance criteria, specified in Appendix B to the CoC, for loading into the MPC. The examples address the regionalized loading scheme. The staff has reviewed the proposed examples and finds them acceptable and which is further described in Section 5 of this SER.

12.1 Conditions for Use

The CoC 1014 Conditions for Use of the HI-STORM 100 Cask System were modified to add descriptions of the design changes requested by the amendment. The proposed changes are delineated in the Summary section of this SER.

The staff reviewed the proposed CoC and TS changes and, with the addition of the staff changes to Appendix A TS 3.1.1 and Table 3-1, finds that they are appropriate for the modifications made to the HI-STORM 100 Cask System.

12.2 Technical Specifications

Table 12-1 lists the TS, as modified by the amendment, for the HI-STORM 100 Cask System. The TS were revised to address those items modified as described in the Summary section of this SER. Regarding the definition of damaged fuel, the definition currently states:

> Damaged fuel assemblies are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy rods, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage is considered fuel debris.

The staff was concerned with the terminology regarding the “expected” geometric rearrangement of fuel or gross failure of the cladding. Specifically that expected failures may not ensure that appropriate analyses or reviews are being performed by the user to properly identify damaged fuel assemblies. The definition, is as written, a performance based approach that depends upon the design characteristics of the spent fuel storage cask (or transportation system) and the regulatory requirements for fuel performance which are unique to the storage (or transportation) regime. With this approach, the designer identifies the requirements that the fuel must meet, both from the regulations, and the unique design characteristics of their specific storage (or transport) cask. As such the definition has been revised to state:

Damaged fuel assemblies are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy rods, missing structural components such as grid spacers, assemblies whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

The staff has reviewed the TS and, with the addition of the changes to Appendix A TS 3.3.1 and Table 3-1 finds that they are appropriate for the modifications made to the HI-STORM 100 Cask System.

12.3 Approved Contents and Design Features

The applicant proposed revisions to CoC 1014, Appendix B, Approved Contents and Design Features, to reflect the changes to the contents for the HI-STORM 100 Cask System requested in the amendment. The staff has reviewed the revisions and finds that they provide sufficient information to ensure that all the contents to be stored in the HI-STORM 100 Cask System meet the design basis evaluated by the staff in Sections 3 through 11 of this SER.

12.4 Evaluation Findings

Based on the NRC staff's review of information provided in the HI-STORM 100 Cask System amendment request, the staff finds the following:

F.12.1 The staff concludes that the proposed Conditions for Use, the TS, and the Approved Contents and Design Features contained in CoC 1014 for the HI-STORM 100 Cask System have been revised to provide reasonable assurance that the requirements of 10 CFR Part 72 have been satisfied. The TS provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

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	Note Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
3.0	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
3.1.4	Deleted
3.2	SFSC RADIATION PROTECTION
3.2.1	Deleted
3.2.2	Transfer Cask Surface Contamination
3.2.3	Deleted
3.3	SFSC CRITICALITY CONTROL
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINISTRATIVE CONTROLS
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Radioactive Effluent Control Program
5.5	Cask Transport Evaluation Program
5.6	Deleted
5.7	Radiation Protection Program
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements

13.0 QUALITY ASSURANCE EVALUATION

The purpose of this review and evaluation is to determine whether the applicant has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G as applicable to the changes proposed in the amendment request.

The changes requested by the applicant associated with the HI-STORM 100 Cask System have not altered the staff's previous assessment of the QA program. Therefore, the staff did not reevaluate this area for the amendment request.

14.0 DECOMMISSIONING

The modifications requested by the applicant have not altered the staff's previous assessment of decommissioning considerations associated with the HI-STORM 100 Cask System. Therefore, the staff did not reevaluate this area for the amendment request.

15.0 CONCLUSIONS

15.1 Overall Conclusion

The staff has reviewed the proposed changes to the Final Safety Analysis Report for the HI-STORM 100 Cask System. With addition of the staff requirements with respect to loading operations, and based on the statements and representations contained in the proposed FSAR as amended, and the conditions given in the CoC as amended, the staff concludes that the HI-STORM 100 Cask System meets the requirements of 10 CFR Part 72.

15.2 Conclusions Regarding Analytical Methods

The staff determined that, unless otherwise noted in this SER, all analytical methods used by the applicant, that provide the basis for design modifications and the addition to the list of approved cask contents for the HI-STORM 100 Cask System proposed in the amendment request, are acceptable. However, for the purposes of the amendment request review, the staff did not revisit any previously approved methodologies used in the original HI-STORM 100 Cask System application or those reviewed for Amendments 1, 2, and 3 and did not make any new determination on the adequacy of those methodologies unless the methodology was used as the basis for a proposed amendment change.

Issued with Certificate of Compliance No. 1014,
on July 14, 2008