SAFETY EVALUATION REPORT

DOCKET NO. 72-1014

HOLTEC INTERNATIONAL

HI-STORM 100 CASK SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1014

AMENDMENT NO. 2

TABLE OF CONTENTS

SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HI-STORM 100 CASK SYSTEM
HOLTEC INTERNATIONAL
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 2

SUMMARY

By letter dated March 4, 2002, Holtec International (Holtec) submitted an amendment application to the U. S. Nuclear Regulatory Commission (NRC), 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 subsequently revised and submitted to the NRC on October 31, 2002. The application, as supplemented; October 31, 2002, (Amendment 2, Revision 1), August 6, 2003, (Amendment 2, Revision 2, Supplement 0), November 14, 2003, (Amendment 2, Revision 2, Supplement 1), February 20, 2004, (Amendment 2, Revision 2, Supplement 2), April 23, 2004, (Amendment 2, Revision 2, Supplement 3), July 22, 2004, (Amendment 2, Revision 2, Supplement 4), August 13, 2004, (Amendment 2, Revision 2, Supplement 5), October 14, 2004, (Amendment 2, Revision 2, Supplement 6), and December 3, 2004, (Amendment 2, Revision 2, Supplement 7), requested 31 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 amendment includes changes to materials used in construction, changes to the types of fuel that can be loaded, changes to shielding and confinement methodologies and assumptions, revisions to various temperature limits, changes in allowable fuel enrichments, and other changes to reflect current staff guidance and use of industry codes. Specifically they include:

1. Add the use of METAMIC® as an alternate neutron poison material and remove specific reference to BORAL® as the neutron poison material.
2. Allow the storage of damaged fuel in the multi purpose canister (MPC) -32 and damaged fuel and damaged fuel debris in the MPC-32F. Additionally, include appropriate values for soluble boron for MPC-32 and MPC-32F based on fuel assembly array/class, intact versus damaged fuel, and initial enrichment.
3. Clarify that use of the aluminum heat conduction elements are permitted under Revision 0 or Amendment 1 to the CoC but are prohibited for use under Amendment 2. Clarify in the CoC reference to HI-STORM 100 and 100S nomenclature, and clarify the differences between the two designs. Delete all information in the CoC pertaining to the authorized contents of each MPC and add a statement defining the suffix to the MPC model number.
4. Revise the CoC to reflect changes in MPC cavity drying, revise the TS to remove the helium leakage test requirement, and relocate the helium backfilling requirements to a new Table 3-2 in the TS.
5. Revise requirements for ensuring MPC cavity bulk helium temperature is less than 200EF (93EC) prior to reflooding instead of existing language that reads “helium gas exit temperature” in the event unloading should be necessary and change the completion

time requirement from 22 hours to “immediately.” Similarly, add new Limiting Conditions for Operation (LCO) to address use of the “Supplemental Cooling System.”

6. Add new Technical Specification (TS) Program 5.7 for radiation protection. Modify associated LCO.
7. Revise the definition of Non-Fuel Hardware to include vibration suppressor inserts and allow for their storage as integral non-fuel hardware that may be stored in an MPC with a fuel assembly.
8. Increase the maximum authorized initial enrichment for Pressurized Water Reactor (PWR) damaged fuel and fuel debris to 5.0 wt.%.
9. Revise burn-up as a function of cooling time and as a function of fuel array/class.
10. Modify associated Completion Times for TS Required Actions to reflect blocked duct accident analysis and Surveillance Requirement acceptance criterion for temperature measurement.
11. Revise Appendix B tables to provide new limits for fuel assembly burnup as a function of decay heat, enrichment, cooling time, and fuel array/class and modify associated completion times.
12. Revise the maximum allowable uranium masses for certain fuel assemblies to be consistent with revised shielding analyses.
13. Revise the maximum allowable burn-up for non-fuel hardware inserts to be consistent with revised shielding analyses.
14. Update American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Edition of record, add new and revised ASME Code alternatives, and incorporate new language into the CoC related to Code alternatives.
15. Revise portions of CoC Appendix B to be consistent with CoC Appendix A and revise affected portions of the CoC to incorporate review guidance contained in Interim Staff Guidance (ISG) -11, Revision 3.
16. Increase off-normal design pressure from 100 psig to 110 psig and increase the normal temperature limit for the overpack lid top plate from 350EF to 450EF.
17. Deletion of Appendices 3.B thru 3.AS of the FSAR and relocate the information to a supporting calculation package.
18. Removal of discussion of three-ducts blocked condition in Chapter 11 of the FSAR.
19. Revised discussion of the Holtec QA program in Chapter 13 of the FSAR to remove redundant information.
20. Revise the CoC to address editorial issues.
21. Modify the language in the CoC Condition 11 to address component certification and use to include changes to the discussion regarding aluminum heat conduction elements.
22. Modification of the MPC drying acceptance criterion to use the Forced Helium Dehydration (FHD) System.
23. Revision of the language pertaining to the list of ASME Code alternatives.
24. Modification of the language in the CoC to include a maximum boron carbide content in METAMIC® to 33.0 wt%.
25. Add language to the CoC incorporating FSAR Section 9.1.5.3 by reference and adding a note to the FSAR Section stating this Section cannot be modified under the provisions of 10 CFR 72.48.
26. Clarification of the manner in which the equation used to determine whether a site may deploy free-standing casks is executed.
27. Modification of FSAR Section numbering specified in the TS.
28. Modification of the design temperatures of the MPC shell, overpack concrete, and Holtite neutron shield material.

29. Modification of FSAR Tables 2.2.6 and 2.2.7 to clarify the Code applicability for the MPC basket and basket angle supports.
30. Addition to the FSAR of Forced Helium Dehydration System Failure and Supplemental Cooling System Power Failure as new off normal events and addition of Supplemental Cooling System failure as a new accident event.
31. Insertion of a new requirement to address degraded cask/pad interface friction for freestanding casks.

This Safety Evaluation Report (SER) documents the review and evaluation of the amended FSAR, supplemental materials, and proposed CoC changes. The 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.

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 non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.1 General Description and Operational Features

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

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. It is made entirely of stainless steel except for the neutron absorbers. 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. Approval of the MPC-32F brings the number of approved MPC designs to eight; 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 have been added and are considered integral non-fuel hardware consisting of zircaloy or stainless steel tubes. Because zircalloy and stainless steel are not new or different materials introduced into the cask environment, there is essentially no change, chemically, to the cask contents. Thus, there are no potentially adverse chemical or galvanic reactions introduced by addition of this component to the MPC contents. The NRC staff finds this addition to be acceptable.

1.1.2 HI-TRAC Transfer Cask

The HI-TRAC transfer cask (TC) 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, however, some calculations were modified to evaluate load and dose changes associated with the other design modifications and addition of contents. 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 was modified. 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; (3) the transfer cask is placed on top of the overpack and the MPC is lowered into the overpack; and (4) the overpack, with the MPC inside, is moved to the storage pad. A loaded HI-TRAC transfer cask can be handled vertically or horizontally. A loaded HI-STORM 100, 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 FSAR contains the non-proprietary drawings for the HI-STORM 100 Cask System, including drawings of the structures, systems, and components important to safety. The drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the entire system. Specific structures, systems, and components are evaluated in Sections 3 through 14 of this SER, 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:

- damaged fuel in the MPC-32 and damaged fuel and damaged fuel debris in the MPC-32F,
- vibration suppressor inserts and allowance for their storage as integral non-fuel hardware that may be stored in an MPC with a fuel assembly,
- higher maximum initial enrichment PWR damaged fuel and fuel debris,
- fuel with new limits for fuel assembly burn-up, and
- non-fuel hardware inserts with higher maximum allowable burn-up.

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 FSAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

F1.2 Drawings for structures, systems, and components important to safety presented in Section 1.5 of the 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.

F1.3 Specifications for the spent fuel to be stored in the dry cask storage system are provided in Section 1.2.3 of the FSAR. Detailed specifications for the spent fuel are presented in Section 2.1 of the FSAR and Appendix B to the CoC.

F1.4 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.

F1.5 The quality assurance (QA) program and implementing procedures are described in Chapter 13 of the FSAR. Specific changes to the QA program and program description are evaluated in Sections 13 of this SER.

F1.6 The staff concludes that the information presented in this Chapter of the 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 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 also identifies the function and governing code of the components. The governing code for the structural design of the MPC, the transfer cask, and the metal components in the overpack is the ASME Code. The governing code for the concrete in the overpack is American Concrete Institute (ACI) 349. Exceptions to these Codes are delineated in the FSAR. Appendix 2.A of the FSAR describes the general design and construction requirements for an ISFSI concrete pad for use with the HI-STORM 100A in high seismic areas. Appendix 2.B describes the design and construction requirements for the Forced Helium Dehydration (FHD) System and Appendix 2.C describes the design and construction requirements for the Supplemental Cooling System (SCS).

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 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 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 FSAR.

2.3 Design Criteria for Safety Protection Systems

The principal design criteria for the MPC and the HI-STORM overpack designs and the TC, are summarized in FSAR Tables 2.0.1, 2.0.2, and 2.0.3, respectively. Design criteria for the ISFSI pads and anchor studs for the HI-STORM 100A are described in Table 2.0.4 and the design criteria for free standing and anchored HI-STORM 100 installation is described in Table 2.0.5. This amendment requested only minor changes to Table 2.0.1 to be consistent with those changes described in greater detail elsewhere in the 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 FSAR.

2.3.1 General

Chapter 2 of the FSAR was modified, per this amendment, to include changes associated with the MPC and overpacks. The major changes include: (1) the option to use alternate neutron poison materials, (2) storage of damaged fuel in the MPC-32 and damaged fuel and damaged fuel debris in the MPC-32F (the applicant declined to incorporate the definition of damaged fuel as detailed in ISG-1, Revision 1, citing that maintenance of the former definition is adequate for storage), (3) changes to helium backfilling requirements, (4) increased off-normal design pressure from 100 psig to 110 psig, (5) increases in the allowable temperatures of the structural materials, new limits for fuel assembly burn-up as a function of cooling time and as a function of fuel array/class, (6) revised soluble boron requirements for the MPC-32/32F based on fuel assembly array/class, intact versus damaged fuel, and initial enrichment, (7) and increase in the maximum authorized initial enrichment for PWR damaged fuel and fuel debris to 5.0 wt.%, and (8) revised maximum allowable uranium masses for certain fuel assemblies.

2.3.2 Structural

The structural analysis is presented in Chapter 3 of the 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 FSAR. The load combinations for which the MPC, transfer cask, and overpack are designed are defined in Section 2.2.7 of the FSAR. Changes made to the structural design criteria per this amendment are described in Section 3 of this SER.

2.3.3 Thermal

The thermal analysis is presented in Chapter 4 of the 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 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 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 ^{10}B 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 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 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 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(b). 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 request addresses a slight increase of 10% in the off-normal internal design pressure, increases in the allowable temperatures of the structural materials and the creation of the eighth type MPC unit: the MPC-32F. No changes were made to the drawings of the various components that have been previously provided in Section 1.5 of the FSAR since no material or design dimensions were revised.

The following structural evaluations were performed by the staff:

- For the MPC-32F, which is for storing some failed fuel and a variant of the already approved MPC-32, the structural changes in the MPC assembly and the resulting modified stresses on the canister shell have been analyzed in the same manner as the MPC-32. The resulting stresses have been demonstrated to provide safety factors against the design limits computed to exceed 1.0 and are acceptable. The impact of increased temperatures on structural material performance has also been evaluated.

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 multi-purpose canister (MPC), the transfer cask (HI-TRAC) and the dry storage overpack/cask (HI-STORM). This structural portion of the SER addresses the MPC and the overpack that were impacted by this amendment.

3.1.1 Structural Design Features

3.1.1.1 Multi-Purpose Dry Storage Canister: MPC-32F

As with the other seven models of MPCs, this multi-purpose dry storage canister is designed and fabricated as all-welded stainless steel cylindrical pressure vessel. The MPCs provide the confinement boundary for the stored spent fuel and the structural integrity of that boundary must be maintained under all design conditions. All canisters used in the HI-STORM 100 Cask System have an identical nominal exterior diameter of 68-3/8 inches with the exterior cylinder heights varying with the MPC model. The largest loaded weight that can be achieved in any of the MPC models is 44.5 tons. Within the canister is an internal assembly known as the “basket” that is designed to accommodate the various types and configurations of spent fuel. The use of different baskets allows accommodation for the different types of spent fuel. The basket assembly is best described as a welded stainless steel multi-celled, egg-crate/ honeycomb type structure with the individual cells accommodating a specific type of spent fuel assembly. The cellular structure is positioned within the circumscribing inside surface of the MPC cylindrical shell. The other major elements of the MPCs besides the canister shell and the basket include the canister baseplate, the canister lid and the closure ring. The configuration and design details allow for a redundant closure system for the canister that can be pressure tested after the final welds are made. The only configuration change made in the creation of the MPC-32F is the thickened section of the cylindrical shell wall at the top of the MPC which is identical to that of the other approved MPC with an “F” in the model designation. This configuration is shown in FSAR Figure 2.1.9 as part of the amendment.

3.1.1.2 Overpacks/Casks

The HI-STORM overpacks/casks provide mechanical protection, thermal cooling, thermal protection, and radiological shielding for the MPC that is contained within the structure. The structure is a vertical cylinder and is fabricated as a blend of carbon steel plate and shell elements along with the concrete fill material. The main structural functions of the overpack are provided by the structural steel while the main radiation shielding function is provided by the mass of plain, unreinforced concrete. Design details and the use of the specific materials provide the necessary characteristics to provide proper thermal performance. The exterior diameter of the HI-STORM 100 series of overpacks/casks is approximately 11 feet. The cylindrical wall thickness of the 100 series is 29-1/2 inches of steel and concrete. No changes in the configuration of the overpack design or the transfer cask (HI-TRAC) design resulted from this amendment.

3.1.2 Structural Design Criteria

The structural design criteria for the HI-STORM 100 Cask System have been evaluated previously and the results of that safety evaluation are summarized in the staff’s SER of CoC 1014 and referenced FSAR and Amendment 1 to CoC 1014 and referenced FSAR. The structural design criteria for the HI-STORM 100 Cask System are summarized in Tables 2.01, 2.02 , 2.03, and 2.2.1 through 2.2.16 of the FSAR. Included in this SER are those elements of the structural design criteria that are modified or added based on the new information provided for in this amendment request including the ASME Code alternatives of Table 2.2.15 that have been expanded to clarify specific code provisions and terminology. Since the approval of Amendment 1 to the HI-STORM 100 Cask System CoC, several one-time alternatives to the ASME Code NDE requirements were submitted to the NRC staff for review. Those alternatives involved specifically identified casks which had deviations from the specified Code inspection requirements. These deviations from the specified construction requirements were compensated for by the performance of additional analyses beyond the normal Code requirements. Those cask-specific alternatives previously found to be acceptable to the NRC staff have been added into the proposed FSAR for completeness.

3.1.2.1 Criteria for Multi-Purpose Dry Storage Canisters

The proposed amendment revises the MPC off-normal internal pressure from 100 psig to 110 psig as noted in Table 2.2.1 of the FSAR. In addition, the amendment reflects an increase in the long term normal condition design temperature limit for the MPC shell from 450EF (232EC) to 500EF (260EC). Table 3.1.17 in the FSAR reflects the revised stress intensity allowables for the MPC shell as revised in Table 2.2.3 for the higher design temperature based on the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, 1995 Edition with Addenda through 1997. No physical changes were necessary to accommodate the revised pressure and temperatures.

3.1.2.2 Criteria for Overpacks/Casks

The proposed amendment for the overpacks indicates a clarification in Table 1.0.3 in the FSAR that the temperature limits for concrete for normal conditions will follow ACI 349, Paragraph A.4.3 and the temperature limits for the off-normal and accident conditions will follow Paragraph A.4.2 of ACI 349. Table 1.D.1 provides the description of what constitutes the computed temperature that is to be compared to the limits derived from American Concrete Institute, “Code Requirements for Nuclear Safety-Related Concrete Structures,” ACI 349-85, Appendix A. No changes in design criteria for the transfer cask were identified in this amendment.

3.1.2.3 Criteria for the ISFSI Pad/Basemat

Alternative conservative static screening criteria are provided in Section 3.4.7.1 of the FSAR for the consideration of the interaction between the basemat and a freestanding cask under a seismic event to produce sliding or to produce overturning of a cask. These screening criteria also allow the evaluation of reduced cask/basemat interface friction that may occur under changing conditions.

3.1.2.4 Individual Loads

The individual loads are identified in Section 2.2 and Table 2.2.13 of the FSAR with additional information provided in Section 3.1.2.1. This amendment does not identify any new individual load sources, but merely identifies some additional values that were considered in the analyses that were used in comparisons to previous analyses to determine the bounding cases. Only those identifiable new load values are discussed in this evaluation because all other individual loads have been discussed in conjunction with the issuance of the original CoC.

The off-normal internal design pressure load for the MPCs increased from a value of 100 psig to 110 psig as noted in Table 1.2.2 of the FSAR amendment with no impact on the physical design of the MPC. The bounding conditions for the design has incorporated higher temperatures, which has had the resulting effect in some cases to decrease the allowable stress intensities, that are based on various temperature increments for the various materials. These have been reflected by using the existing stress intensity versus temperature tables of Section 3.1 of the FSAR (Tables 3.1.6 - 3.1.16) for the various materials.

3.1.2.5 Load Combinations

This amendment does not revise any of the previous loading combinations that were evaluated earlier and now form part of the basis of the CoC. These combinations continue to be identified in Tables 2.2.14, 3.1.1, 3.1.3, 3.1.4 and 3.1.5 of the FSAR.

3.1.2.6 Allowable Stresses or Required Strength

Stress allowables identified for the metallic materials used in the MPCs, the internal baskets, the overpacks and transfer casks remain unchanged by this amendment. These allowable stress limits are based on the ASME Code and provided in Tables 1.2.7, 2.0.4, 2.2.10, 2.2.11 and 2.2.12 with the numerical values listed in Tables 3.1.6 through 3.1.16 of the FSAR as a function of temperature that have not changed as a result of the amendment. Table 3.1.17 of the FSAR amendment reflects the changes in stress intensity allowables based on the higher bounding design temperatures for the MPCs under the various loading combinations.

3.2 Weights and Center of Gravity

Section 3.2 of the FSAR presents the weights and centers of gravity physical data in tabular form listing the weights of various components and models in the HI-STORM 100 Cask System in the various combinations that will be used in spent fuel storage operations. No changes have occurred as a result of this amendment.

3.3 Material Properties

3.3.1 Concrete

Appendix 1.D of the FSAR amendment has been revised to utilize provisions of Section A.4.3 of Appendix A of ACI 349-85 in defining criteria for concrete temperatures based on referenced test data. Two references to previous research and testing are utilized to justify an increase in concrete temperature to 300EF (149EC) under long term conditions. Tests were conducted for a four month period with any significant strength reduction occurring in the first month. For the shielding concrete in the overpack lid, the temperature is taken as the through-thickness average and for the body of the overpack and pedestal concrete the temperature is taken as the maximum local temperature. The referenced tests revealed a potential problem aggregate mineralogy at the 300EF (149EC) temperature for a sustained period of time. The particular aggregate had originally been allowed for use in the concrete as noted in a footnote to Table 1.D.1. The use of that type aggregate has been precluded by elimination of dolomite from the listing in the footnote.

3.3.2 Structural Steels and Bolting Materials

No changes were proposed to structural steels and bolting materials in this amendment.

3.3.3 Non-structural Materials

The applicant has proposed addition of damaged fuel assemblies containing burnable poison rod assemblies (BPRAs) or other control elements to acceptable MPC cask inventory. The possibility that damaged control elements could introduce different materials into the cask environment was assessed. Control elements are composed of zircalloy or stainless steel tubes or solid rods. All the solid rods are composed of either zircalloy or stainless steel, materials which are normally contained within the MPC. Thus no adverse chemical or galvanic reactions occur by the introduction of such solid control elements.

The control elements that are composed of zircalloy or stainless steel tubes contain one of several materials within the tubes. Those materials are: boron carbide, borosilicate glass, silver-indium-cadmium alloy or thorium oxide. The potential for adverse chemical or galvanic reactions between these various compounds and the fuel cladding or the MPC itself was evaluated.

Boron carbide is an extremely stable and generally inert chemical compound. The chemical characteristics of boron carbide show that it is a non-metallic, electrically neutral material. As such, it will not create any galvanic or chemical interaction in either the short-term water environment of cask loading or the long-term helium storage environment inside the MPC.

Borosilicate glass is a stable and inert non-metallic, electrically neutral material that will behave in a similar manner to boron carbide in the MPC environments. Thus no adverse chemical or galvanic reactions would occur.

The silver-indium-cadmium alloy contained in some control elements may oxidize slightly if exposed to water (assuming a perforated zircalloy tube) during the wet loading of fuel into the cask. However, the amount of oxidation or galvanic corrosion that would occur would be severely limited by the short immersion time of the loading process, typically about 4 days. The chemical byproducts of any such corrosion would be in the form of inert metal oxides and a small amount of hydrogen. The metal oxides would not further react in the cask wet or dry environments. The hydrogen would be removed during the dewatering and drying steps of the overall cask loading procedure. Since a hydrogen monitoring or mitigation plan is part of the loading procedures, no problems would arise from the small amount of hydrogen that might be generated.

Thoria, as thorium oxide, is a stable inert metal oxide that will not result in any adverse chemical or galvanic reaction in the wet or dry cask environments.

Based upon these evaluations, the NRC staff finds that the inclusion of the above mentioned BPRAs and/or control elements is acceptable.

Material evaluation of METAMIC® neutron absorber material is presented in Section 6.3.2 of this SER.

3.4 Structural Analysis of HI-STORM 100 Cask System

3.4.1 Normal and Off-Normal Conditions

As noted in Section 2.2 of the FSAR, these conditions include the following situations that influence the structural capability and form the design bases of the system. The normal conditions include the dead weight, handling, pressure, temperature and snow loads that occur routinely. The off-normal conditions include the dead weight, pressure, temperature, partial blockage of air vents, and off-normal handling of the HI-TRAC cask that occur only occasionally. No changes were proposed in this amendment.

3.4.2 Accident Conditions

The accident conditions that govern the design criteria include the following scenarios: handling accident, tipover, fire, partial blockage of MPC basket vent holes, tornado, flood, earthquake, fuel rod rupture, confinement boundary leakage, explosion, lightning, burial under debris, 100 percent blockage of air inlets, and extreme environmental temperature as identified in Section 2.2 of the FSAR. No changes were proposed in this amendment.

3.5 Special Topics

3.5.1 Lifting Devices

No changes were proposed to lifting devices in this amendment.

3.5.2 Differential Thermal Expansion

Resulting differential thermal expansion from the increased temperatures associated with this amendment can be accommodated by the current design.

3.6 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:

F3.1 The FSAR amendment 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 component and fuel material temperatures of the HI-STORM Cask System and HI-TRAC Spent Fuel Transfer 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, system and components important to safety can be assessed under the requirements of 10 CFR 72.236(b). This amendment was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfills the acceptance criteria listed in Section 4 and 11 of NUREG-1536 as well as associated ISG documents.

The following significant changes proposed by the applicant that affect the thermal performance of the HI-STORM 100 Cask System are listed as follows:

1. The thermal analysis is revised to comply with recently issued NRC staff guidance ("Cladding Considerations for the Transportation and Storage of Spent Fuel," ISG-11 Revision 3).

2. The aluminum heat conduction elements (AHCEs), optional under Amendment 1 of CoC 1014, are removed from the design. Removing the AHCEs from the MPC eliminates the constriction of the downcomer flow (see Figure 4.0.1 of the FSAR) and thus further enhances the thermal performance of the MPC.

In addition to the above changes, the staff identified the following major change: for MPCs containing high burnup fuel assemblies, supplemental cooling is required while in the HI-TRAC transfer cask. The amount and type of cooling is left to the end user of the storage cask system. Additional cooling for on-site transfer analysis requires the use of the HI-STORM 100 Cask System thermal methodology described in Chapter 4 of the FSAR.

The staff's conclusions, summarized below, are based on information provided in Amendment 2, as revised, to the HI-STORM 100 Cask System FSAR.

4.1 Spent Fuel Cladding

The applicant adopted certain guidelines of NUREG-1536 and U.S. Nuclear Regulatory Commission, Interim Staff Guidance No. 11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," November 17, 2003, (ISG-11, Revision 3). to demonstrate the safe storage of the material content described in Chapter 2 of the FSAR and the CoC for the HI-STORM 100 Cask System. The applicant proposes to design the HI-STORM 100 Cask System to comply to all the following eight criteria:

1. The fuel cladding temperature at the beginning of dry cask storage should generally be below the anticipated damage-threshold temperatures for the licensed life of the system.

2. The fuel cladding temperature should generally be maintained below 1058EF (570EC) for accidents and off-normal event conditions.

3. The maximum internal pressure of the cask should remain within its design pressures for normal (1% rod rupture), off-normal (10% rod rupture), and accident (100% rod rupture) conditions.

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

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

6. For long term normal and short term operations (defined in Chapter 2 of the FSAR), the maximum commercial spent fuel (CSF) cladding temperature shall be limited to 752EF (400EC).

7. The cask system should be passively cooled except during short-term operations including on-site transfer of the MPC inside HI-TRAC transfer cask.

8. The thermal performance of the cask should be within the allowable design criteria specified in Chapter 2 and 3 of the FSAR for normal, off-normal, and accident conditions.

To ensure explicit compliance with the provisions of ISG-11, Revision 3, a new condition, “short term operation,” corresponding to fuel loading activities is defined in Chapter 2 of the FSAR as those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and on-site handling of a loaded HI-TRAC transfer cask. The short term evolutions that are thermally limiting analyzed in the FSAR are listed below:

1. Vacuum Drying.
2. Loaded MPC in HI-TRAC in the vertical orientation.
3. Loaded MPC in HI-TRAC in the horizontal orientation.

4.2 Cask System Thermal Design

The cask thermal design for the HI-STORM 100 and 100S overpack containing a loaded MPC is presented in Sections 1.2, 2.1, and 4 of the FSAR

4.2.1 Design Criteria

The applicant addressed the HI-STORM and HI-TRAC spent fuel storage and transfer system design criteria developed to meet 10 CFR Part 72 requirements for 20 years of storage of spent nuclear fuel. These design criteria encompass normal, off-normal, and postulated accident conditions. The thermal design criteria for the HI-STORM 100 and 100S overpack with the loaded MPC are given in Section 2.2 of the FSAR.

4.2.2 Design Features

The HI-STORM 100 and 100S Cask Systems consist of a MPC and concrete overpack designed for the dry storage of spent fuel. The MPC is designed for fuel loading, closure, transfer, on-site storage, and off-site transport. It provides for, among other things, the function of passive heat removal for storage and transport for the enclosed spent fuel. The storage cask has a capacity for up to 32 PWR or 68 BWR spent fuel assemblies. The HI-STORM 100 and 100S overpacks are comprised of a metal/concrete composite shell designed for radiation shielding and mechanical protection against natural and manmade phenomenon but which also facilitates cooling via passive natural convective cooling through four vents at the top and bottom of the overpack and the annular passages surrounding the MPC inside the overpack.

The MPC essentially consists of a shell and an internal basket assembly. The MPC is an integrally welded pressure vessel designed to meet ASME Code requirements (see Section 3 of this SER).

In general, the material properties remained unchanged from the original design. The eight MPC models are all engineered as cylindrical prismatic structures with square cross section cavities which contain a varying number of fuel assembly/damaged fuel assembly storage cells. Regardless of the number of cells, the construction of the MPC is fundamentally the same using a honeycomb of cellular elements positioned within a circumscribing cylindrical canister shell. The MPC fuel basket is positioned and supported within the MPC shell by a set of basket supports welded to the inside of the MPC shell. The free volume of the MPC is inerted with pure helium gas under pressure.

The staff verified that methods of heat transfer internal and external to the HI-STORM 100 and 100S Cask Systems are passive excluding the provision for use of the Supplemental Cooling System, when necessary, during transfer operations. Sections 1.4 and 1.5 of the FSAR provide information relative to the materials of construction general arrangement, dimensions of principle structures, and description of all structures, systems, and components important to safety, in sufficient detail to support a finding that the design will satisfy the design bases with an adequate margin, as required by 10 CFR 72.236.

4.3 Thermal Load Specification

The thermal load specifications for an overpack loaded with the MPC are given in Sections 2.2 and 4.4 of the FSAR. FSAR Tables 4.4.20, 4.4.21, 4.4.28, and 4.4.29 list the maximum allowable decay heat load that can be stored. These limits on decay heat loads are based on the calculated maximum cladding temperature limits for normal conditions. It should be noted that the thermal analysis in Chapter 4 of the FSAR has more than a 5% margin in the calculated allowable decay heat for PWR fuel assemblies. Incorporation of all thermal loads into the analytical methods remain unchanged from the original analysis.

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 FSAR. Fuel cladding temperature limits included in Table 4.3.1 of the FSAR are adopted from ISG-11, Revision 3. These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.

Amendment 2 proposed to store damaged fuel assemblies and damaged fuel assembly debris in the MPC32F. The applicant indicated that the internal basket of the MPC-32F is the same as the approved MPC-32, and that the thicker shell at the top of the MPC-32F is similar to the approved designs of the MPC-24F and MPC-68F designs, as such, the existing thermal analyses remain bounding for the MPC-32F design. The staff's structural review of the basket configurations can be found in Section 3 of this SER.

Pacific Northwest National Laboratory (PNNL) has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperature for moderate burnup fuel (MBF) as documented in a PNNL White paper by Lanning and Beyer, "Estimated Maximum Cladding Stresses for Bounding PWR Fuel Rods During Short Term Operations for Dry Cask Storage," 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 FSAR.

4.5 HI-TRAC Thermal Review

The HI-TRAC transfer cask is rugged, heavy walled, cylindrical vessel designed for fuel loading and unloading operations and for movement of the MPC from the loading/unloading area to the storage overpack on the facilities storage pad. The applicant designed the HI-TRAC transfer cask to ensure that fuel integrity is maintained through adequate rejection of decay heat from the spent nuclear fuel. Both the 100 ton HI-TRAC and the 125 ton HI-TRAC designs are provided to house the MPCs. The two HI-TRAC transfer casks are designed identically with the exception of a reduced thickness of lead and water shielding. The 125 ton design has a larger thermal resistance; and, therefore, for normal conditions the 125 ton HI-TRAC thermal analyses bounds the lighter 100 ton design. The staff reviewed changes to operating parameters associated with the HI-TRAC transfer cask and agrees that the existing analyses and results remain bounding and are appropriately conservative. Details of the review of the Supplemental Cooling System (SCS) are described below.

4.5.1 Supplemental Cooling System

Per 10 CFR 72.7, the applicant requested an exemption from 10 CFR 72.236(f). Proposed Amendment 2 includes the use of an active cooling system during on-site transfer operations. The applicant refers to this active cooling as the Supplemental Cooling System (SCS). A general description of the SCS is provided in Appendix 2.C of the FSAR.

Per 10 CFR 72.236(f) requirements, the spent fuel storage cask must be designed to provide adequate heat removal capacity without an active cooling system. Use of the SCS does not satisfy this requirement. However, according to 10 CFR 72.7, "The Commission may, upon application by any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.” The applicant based its request for exemption from the requirements of 72.236(f) on the following features for the SCS:

1. Consistent with its safety function, the SCS is classified as Important to Safety Category B. According to Table 2 of NUREG/CR-6407, a Category B important to safety classification means a failure or malfunction of a category B item has a major impact on safety. Category B items in NUREG/CR-6407 include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a category B item, in conjunction with the failure of an additional item, could result in an unsafe condition.

2. The active components (i.e., electric motors) of the SCS must be connected to redundant power sources to ensure uninterrupted operation.

3. Compared with the storage system licensed life of 20 years and design life of 40 years, the operations involving the SCS occur over a relatively short period of 1 to 3 days. Operations of the SCS will be observed at all times, by virtue of the nature of the on-site loading operations involving the HI-TRAC, so that any interruption of operation will be immediately detected and corrected.

4. The large thermal capacity of the HI-TRAC transfer cask and its contents will suppress rapid temperature changes, so that any short interruption of the SCS operation will not result in large fuel cladding temperature rises.

5. Thermal analysis has demonstrated that, were SCS operation to be interrupted for an extended period of time, the fuel cladding temperatures would remain below short-term allowable limits.

The applicant provided a general description, operation requirements, and design criteria of the equipment used to supply supplemental cooling to the canister during on-site transport in the transfer cask. The applicant adequately justified the availability of the system during normal conditions of transfer. The applicant also provided in the FSAR a description of the thermal model used to perform the analysis of the transfer cask including a description of the need to use a supplemental active cooling system when additional cooling is necessary to comply with the thermal temperature limits. Because several options may exist for providing additional cooling to the transfer cask, the applicant provided a generic description of how the thermal analysis should be applied to demonstrate the adequacy of active cooling methods to keep the material temperatures below allowable limits.

Chapter 8 of the FSAR provides a detailed description of the necessary steps to prepare the MPC for transfer inside the transfer cask for the case when additional supplemental active cooling is necessary.

Also, the applicant provided in the FSAR a description and analysis of the postulated off-normal and accident events by including in the analysis the equipment used to provide supplemental active cooling during transfer of the MPC in the transfer cask. The applicant also provided limiting conditions for operation (LCOs) and appropriate revision to the TS for failure of the supplemental cooling system.

4.5.2 Thermal Evaluation of Short-Term Operations

On-site transport of the MPC generally occurs inside a vertically oriented HI-TRAC allowing natural circulation to occur within the MPC which enhances the heat rejection capabilities of the system. However, some scenarios may involve the on-site transport of the MPC in a horizontally oriented HI-TRAC which place some constraints on the heat rejection capabilities of the system. The applicant provided in the FSAR an evaluation of both scenarios.

In order to comply with ISG-11, Revision 3, allowable temperature limits, for some scenarios it is necessary to provide additional cooling. A set of specifications of this SCS is presented in Appendix 2.C of the FSAR.

The thermal evaluation results for short-term operations remain unchanged. In order to comply with ISG-11, Revision 3, allowable temperature limits, the applicant intends to use the SCS during on-site transfer of an MPC containing high burnup fuel (HBF) since the maximum computed fuel cladding temperature reported in Table 4.5.2 of the FSAR is significantly greater than ISG-11, Revision 3, temperature limit of 752EF (400EC). The applicant left to the end user of the spent fuel storage system the task to analyze the conditions when supplemental cooling is required during on-site transfer operations. The applicant stated that the particular type of augmented cooling is necessarily site-specific and is left to the user to determine, using the thermal methodologies in the HI-STORM 100 Cask System FSAR.

4.6 Conclusion

The staff has reviewed the proposed changes to the material properties, component specifications, and analytical methods used in the thermal evaluation and concludes that they are sufficient to provide the basis for evaluation of the amendment changes against the requirements of 10 CFR Part 72. These changes will not affect previous thermal results; and, therefore, the staff finds these changes acceptable. Applicant's compliance with ISG-11, Revision 3, allowable temperature limits will require 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 end user should perform a thermal analysis, including the SCS, based on the thermal methodology described in the FSAR. In order to use the SCS, the applicant requested an exemption from 10 CFR 72.236(f) (per 10 CFR 72.7). The applicant's reasons and bases for requesting an exemption from 10 CFR 72.236(f) were adequately described, technically justified, and will not endanger life or property or the common defense and security.

Based on the review of the HI-STORM Cask System amendment request, the staff concludes that the applicant adequately described and evaluated the thermal performance of the package and that it meets the applicable regulatory requirements of 10 CFR Part 72.

4.7 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:

F4.1 Chapter 2 of the 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 that is verifiable and reliably 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 An exemption to 10 CFR 72.236(f) allows the applicant to use a supplemental cooling system during short-term operations, including on-site transfer of the MPC in the HI-TRAC transfer cask. The frequency of use of a SCS will not endanger life or property or the common defense and security.

F4.4 The spent fuel cladding is protected against degradation leading to gross ruptures under accident conditions by maintaining cladding temperatures below 1058EF (570EC). 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 verify that there is adequate protection to the public and workers against direct radiation from the cask contents. The review verifies that both changes to the shielding features and contents provide adequate protection against direct radiation to the operating staff and members of the public 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 provided a new methodology to determine the allowable burnup of each loaded spent nuclear fuel (SNF) assembly, as a function of decay heat, cooling time, enrichment, and fuel type. As discussed in Sections 2.1.9 and 5.2.5.3 of the FSAR amendment, the applicant developed a seven-coefficient polynomial equation as listed in Equation 2.1.9.3 and derived coefficients (e.g., A thru G) for several fuel array classes at several different cooling times. The applicant performed unique source term analyses and curve-fitting analyses to derive the coefficients. The new burnup equation and associated coefficients essentially allows a cask user to determine an unlimited combination of fuel parameters (e.g., burnup and cooling time) that result in various decay heats that it could specify for loading. The user may define variable decay heat values for loading based on the limits for uniform loading and thermal methodology for regional loading patterns.

The applicant revised the shielding analysis to incorporate the new burnup equation methodology and to calculate dose rates with larger source terms that bound the possible fuel parameter combinations that are allowed by the equation and maximum decay heat limits for the storage cask system. This type of shielding analysis is outside the scope of the review guidance of the NUREG-1536. However, each subsection in this SER section addresses changes to the revised design and shielding analysis, as it pertains to the new burnup equation method and the bounding dose analyses.

Finally, the applicant provided shielding analyses for the HI-STORM 100S - Version B design configuration. As discussed in Section 5.0 of the FSAR amendment, the analyses were added to aid the staff in the development of the radiation protection program requirements Technical Specification (TS) 5.7 of the CoC. 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, the staff examined and accepted the normal condition dose rates for the Version B design configuration and implemented related dose values in the new TS 5.7 of the CoC (See Section 5.4.5 of this SER).

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The only significant design changes proposed by the applicant is the addition of the MPC-32F canister to store damaged spent nuclear fuel and fuel debris and the use of METAMIC® as an alternate internal neutron absorbing material. The applicant indicated that the internal basket of the MPC-32F is the same as the approved MPC-32 and that the thicker shell at the top of the MPC-32F is similar to the approved designs of the MPC-24EF and MPC-68F. The applicant indicated that the use of METAMIC® does not significantly affect the shielding provided by the MPCs. The applicant did not propose any other significant design changes, as currently described in Revision 2 of the updated HI-STORM 100 Cask System FSAR. Other shielding-related changes to the HI-STORM 100 Cask System involve changes to the allowable contents and new analytical methods.

The staff evaluated shielding design features relevant to the new MPC-32F and use of METAMIC®, and the ability of the current 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 shielding design features associated with the top of the MPC-32F canister is essentially the same as the shielding design features of the approved MPC-24EF and MPC-68F canisters. The staff finds the shielding design features to be acceptable. Based on information provide 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.

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 revised spent fuel and hardware characteristics as described in Section 2.1.9 of the FSAR amendment. Although there are no numerical limits in the regulations for surface dose rates, the dose rates on the surface of the cask system 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 100 mrem/hr on the side and 60 mrem/hr on the top. For both uniform and regional loading configurations, the applicant proposed revisions to the limits for maximum decay heat, maximum burnup, minimum cooling time, and maximum uranium loadings. The applicant also proposed a new burnup equation method to calculate allowable burnups. These limits and the burnup equation method serve as additional source term design criteria, which in turn limits overall dose rates and exposure to the public. Based on these design criteria, the applicant calculated bounding dose rates on the exterior of the HI-STORM 100 Cask System. The applicant calculated bounding dose rates that are less than the proposed design criteria (see Section 5.4 of this SER).

The staff reviewed the design criteria and found it acceptable. The shielding and source term design criteria defined in the FSAR provides 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 a new TS 5.7 to assure compliance with these requirements, with respect to 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 Cask System. 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 regional loading pattern (preferential) as discussed in Section 2.1.9.1 of the FSAR. Both loading patterns are limited by maximum allowable decay heat limits for individual fuel assemblies as specified in Tables 2.1.26 and 2.1.27 of the FSAR. The amendment indicates that the uniform loading pattern continues to provide bounding radial dose rates, as compared to the preferential loading pattern for the various combinations of fuel parameters allowed by the decay heat limits and the new burnup equation method. Therefore, the applicant calculated bounding dose rates in the primary shielding analysis for the uniform loading pattern.

The staff reviewed the use of new 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 uniform loading pattern generally results in bounding dose rates over the preferential loading pattern for various combinations of fuel parameters. The staff notes that each site user must perform an analyses 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.

5.2 Source Specification

The design-basis source specification for bounding calculations are presented in Section 5.2 of the FSAR amendment. 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, and total heat load of 38 kW for the MPC-24 and MPC-32 configuration, and 35.5 kW for the MPC-68 configuration. For the bounding calculations, the applicant used the same design basis fuel types, uranium loading, and source term method previously approved for the HI-STORM 100 Cask System.

Based on the new burnup equation methodology, 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 Tables 2.1.26 and 2.1.27 for uniform and regional loading. For the MPC-32, the applicant calculated bounding source terms for 35 GWD/MTU and 3-year cooling and 75 GWD/MTU and 8-year cooling. For the MPC-24 bounding calculations, the applicant calculated source terms for 47.5 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 40 GWD/MTU and 3-year cooling and 70 GWD/MTU and 6-year cooling. The applicant also increased the allowable Co-60 content of the burnable poison rod assemblies (BPRA) non-fuel hardware. Based on the higher content, the applicant provided new burnups in Table 2.1.25 and used these bounding source terms in the revised shielding analyses. The burnup and cooling times and associated decay heat conservatively bound the allowable decay heats as defined in Section 2.1.9 of the FSAR amendment.

5.2.1 Gamma Source

The new design-basis gamma source terms for MPC-24, MPC-32, and MPC-68 are listed in Tables 5.2.4 through 5.2.13 of the FSAR amendment. The new design-basis source terms for the BPRA hardware is listed in Table 5.2.31. For the bounding source terms and development of the coefficients for the new 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 H-STORM 100. The applicant provided references and indicated that there is uncertainty in the gamma source terms associated with the ORIGEN-S calculations. Based on 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 new design-basis neutron source terms for the MPC-24, MPC-32, and MPC-68 are listed in Tables 5.2.15 through 5.2.17. The applicant used the same axial neutron peaking factors as previously approved. The applicant provided references and indicated that there is uncertainty in the neutron source terms associated with the ORIGEN-S calculations. Based on review of data, the applicant noted that errors in Cm-244 could be between 5 and 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.

5.2.3 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 the SAS2H/ORIGEN-S, and are similar to the calculations preformed 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 revised the maximum fuel uranium loadings for each class listed in Tables 2.1-2 and 2.1-3, and used the individual fuel loading values in developing the coefficients for each array class. The applicant used GNUPLOT to calculate coefficients, including an adjustment to assure all data points are bounded by the fit. The derived coefficients are specified in Table 2.1.29. To demonstrate the curve-fitting technique, the applicant provided graphical representations of the actual data generated with SAS2H/ORIGEN-S along with the fitted-curve for a selected fuel type and selected combinations of enrichment, cooling time, and heat load in Appendix 5.F of the FSAR.

The applicant also indicated that there is uncertainty associated with the ORIGEN-S calculations. The applicant estimated errors in decay heat calculations to be between 1.5% and 5.5% depending on fuel cooling times. Therefore, the applicant applied a 5% decay heat penalty to the derivation of fuel coefficients for BWR array classes. The applicant did not apply a penalty to

PWR array classes because the thermal analysis in Chapter 4 of the FSAR has more than a 5% margin in the calculated allowable decay heat.

5.2.4 Staff Evaluation

The staff reviewed the source term analyses and new burnup equation method in Section 5.2 of the FSAR amendment. 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 analysis of selected bounding burnup and cooling times with SAS-2H. 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 SAS-2H at selected burnup 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 specific HI-STORM 100 Cask System design and specific contents requested in this amendment. The staff's confirmatory analysis of the derivation of burnup equation coefficients indicated the 5% decay heat penalty was incorporated for BWR fuel classes. As discussed in Section 4 of this SER, the staff agrees the allowable decay heats for PWR fuel has a 5% margin. 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 and cooling times and the burnup equation coefficients are incorporated into Appendix B of the CoC. In addition, dose limits based on the shielding analysis is incorporated into criteria for the new TS 5.7

5.3 Shielding Model Specifications

The HI-STORM 100 Cask System shielding and source configuration is described in Section 5.3 and 5.4 of the FSAR amendment. The applicant used the same shielding model specifications, as previously approved for the HI-STORM 100 Cask System, to calculate bounding doses for near-field and off-site dose rates. The shielding model specification with MCNP is similar to the specifications and methods previously approved for the HI-STORM 100 Cask System.

5.3.1 Shielding and Source Configuration

The applicant used the same shielding and source configuration as previously approved for the HI-STORM 100 Cask System. The applicant indicated the source configuration of damaged fuel in the MPC-32 configuration would behave similarly as 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 materials properties as previously approved for the HI-STORM 100 Cask System. 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 material would not significantly affect the shielding ability of the canisters. The applicant also demonstrated that increased localized temperatures in the concrete shield lid, from an increase in heat load for as large as that noted in Section 5.2 of this SER, result in an overall density reduction of approximately 0.4% in localized areas. The applicant stated that this presents a negligible effect on the shielding effectiveness of the concrete shield lid.

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 did not appear to change from that previously approved by NRC. Based on the statements and calculations presented by the applicant, the model remains 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 Section 5.4 and 11 of the FSAR amendment. The applicant indicated it used the same shielding analyses 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 100 overpack designs 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 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. For the various design configurations approved in this licensing action, the maximum surface dose rates for the overpack on the side (mid-height), top (center), and vents are reported as approximately 100 mrem/hr, 8 mrem/hr, and 45 mrem/hr, respectively. For the Version-B design configuration added under 10 CFR 72.48, the maximum surface dose rates for the overpack on the side (mid-height), top (center), and vents are reported as approximately 110 mrem/hr, 20 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 3.1 rem/hr and 1.4 rem/hr, respectively. The dose profiles presented in Figures 5.1.5 through 5.1.11 for the transfer cask further 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 FSAR also indicates that exposures from localized peak dose rate may be mitigated by ALARA practices such as controlling actual locations of personnel, and using temporary shielding during loading and unloading operations.

5.4.2 Occupational Exposures

The applicant estimated higher occupational exposures in Chapter 10 of the FSAR amendment. The exposures were based on estimations from surrounding dose rates calculated in Chapter 5 of the FSAR and the operating procedures referenced in Chapter 8 of the FSAR. 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 FSAR amendment. Based on 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 200 meters to 250 meters. The analyses indicated that the minimum distance at which the annual dose limit is satisfied for a 2x3 cask array, is increased from 300 meters to 400 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 FSAR amendment does not identify an accident that significantly degrades the shielding of the HI-STORM 100 overpack. Loss of water in transfer cask water jacket remains the bounding accident for direct radiation. The applicant estimated, in Section 5.1.2 of the FSAR, that the transfer cask accident dose rates would be approximately 4 mrem/hr at 100 meters. Based on this exposure rate, the accumulated dose at the controlled area boundary would be approximately 3.1 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 Chapter 11 of the FSAR. The applicant estimated in Section 11.2.1.3 of the FSAR amendment that result of loss of water would increase occupational exposures to approximately 15 person-rem.

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 found them acceptable. For the Version B design configuration, added under 10 CFR 72.48, the calculated dose rates are acceptable for implementation into TS 5.7. The dose rates appear consistent with associated differences in the Version B shielding design configuration and the applicant used the same shielding methodology that has been approved by the NRC. 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 characteristic, cask-array configurations, topography, demographics, and distances. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities such as reactor operations. Each general licensee is responsible to verify compliance with 10 CFR 72.104(a) in accordance with 10 CFR 72.212. In addition, a general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public and workers as required, by evaluation and measurements. Because the revised contents result in significantly higher direct radiation dose rates from that previously approved for HI-STORM 100 Cask System, each user may be required to take additional ALARA precautions to minimize doses to personnel, and 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 FSAR amendment. 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. Estimated dose to members of the public at 100 meters and at further distances for a conservative exposure time of 30 days is approximately 40% below the 5 rem accident limit in 10 CFR 72.106(b). The staff notes that the off-site accident dose rate may be less accurate because precise exposure times cannot be predicted. However, the staff notes that the 30-day exposure is very conservative based on realistic considerations, and 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 (including the Version-B configuration) has been added as new TS 5.7 (See section 10.4 of this SER). The decay heat limits are specified in Appendix B of the CoC. The new burnup equation and associated limits for burnup, cooling time, enrichment, and fuel assembly characteristics are also incorporated into Appendix B of the CoC.

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 FSAR amendment 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 72.106 are the responsibility of each general licensee. The HI-STORM 100 Cask System shielding features (as approved by NRC) are designed to satisfy 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 objective of the criticality review is to ensure that the spent fuel will remain subcritical under all credible normal, off-normal, and accident conditions encountered during handling, packaging, transfer, and storage. The objective includes a review of the changes to the criticality design criteria, features and fuel specifications, a verification and review of the configuration and material properties for the HI-STORM 100 Cask System, and a review of the criticality analyses that might include computer programs, benchmark comparisons, and multiplication factors proposed in this amendment request.

The applicant requested several changes to the HI-STORM 100 Cask System design and CoC. Only those changes that may affect the criticality safety of the system are discussed in this section. The staff reviewed proposed Amendment 2 to the HI-STORM 100 Cask System criticality safety analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the HI-STORM 100 Cask System, as revised, meets the following regulatory requirements: 10 CFR 72.124(a), 72.124(b), 72.236(c), and 72.236(g). The staff's review also involved a determination on whether the cask system satisfies the acceptance criteria listed in Section 6 of NUREG-1536.

The following proposed changes were considered for their impact on criticality safety:

a) add damaged fuel to the allowed contents in the HI-STORM 100 MPC-32 when contained in a Damaged Fuel Container (DFC),

b) add the MPC-32F with intact fuel, damaged fuel, and fuel debris as allowable contents for the HI-STORM 100 Cask System,

c) change the specified minimum boron concentration in the water for the MPC-32 and MPC-32F during wet loading and unloading to vary with the initial enrichment and fuel assembly class rather than specifying a single bounding value,

d) increase the allowed maximum initial uranium enrichment for non-intact fuel in HI-STORM 100 MPC-24E for damaged fuel and MPC-24EF for damaged fuel and fuel debris to 5% when being loaded and unloaded in borated water, and

e) add METAMIC® as an alternative to BORAL® for the neutron poison material in the canister basket.

The staff's conclusions, summarized below, are based on information provided in proposed Amendment 2, as supplemented, to the HI-STORM 100 Cask System FSAR.

6.1 Criticality Design Criteria and Features

The design criteria and features of the HI-STORM 100 Cask System are the same as previously approved with the exception of the addition of METAMIC® as an alternative neutron poison material with a designed poison content where 90% of the boron (versus 75% for BORAL®) is credited in the criticality analysis.

The staff reviewed the applicant's model descriptions and assumptions and agrees that they are consistent with the description of the contents given in FSAR Chapters 1 and 2. The staff reviewed the proposed CoC changes to ensure that the fuel specifications important to criticality safety are included.

The applicant proposed a language change in Section 3.2.6 of Appendix B to the CoC to drop the reference to using fuel spacers to assure that the active fuel region does not shift outside the coverage of the poison plated during an accident. While positioning of the fuel is not a highly significant issue for storage, loading fuel without proper spacers would not meet the conditions of a CoC for transportation. Thus, the applicant withdrew the proposed language change and retained the currently approved wording in Section 3.2.6 of Appendix B to the CoC.

6.2 Fuel Specifications

The fuel parameters important to criticality safety were not changed, except for an increase in the allowed initial enrichment and the addition of allowing damaged fuel in the MPC-32 and damaged fuel and fuel debris in the MPC-32F.

During the review, staff questioned whether fuel specifications have only a maximum limit on the fuel pellet outer diameter. A reduced pellet diameter coupled with flooding of the pellet-to-clad gap could cause k_{eff} to increase for some fuel designs. The applicant argued that if the pellet diameter were to decrease, the diameter of the cladding is expected to decrease also because a survey of pellet-to-clad gaps has shown that this gap size varies only slightly among fuel types. Therefore, a smaller pellet diameter is expected to be accompanied by a smaller inside diameter for the cladding. The staff notes that a minimum value is specified for the outer diameter of the cladding, and thus, the cladding thickness would typically increase with decreasing pellet diameter or the fuel would no longer meet the minimum limit on the outer cladding diameter and could not be stored. It is expected that this increase in cladding thickness would provide sufficient moderator displacement to maintain adequate criticality control. Therefore, based on probabilistic considerations, the staff accepts the applicant's argument and concludes that the proposed combination of parameter specifications for the fuel will provide a reasonable assurance of criticality safety.

6.3 Model Specifications

The basic models used for the criticality safety analysis were the same as used in the previous applications with the exception of: 1) the addition of a damaged fuel canister design in the MPC-32, 2) the addition of an MPC-32F with intact, damaged fuel and fuel debris, 3) a more comprehensive analysis of eccentric positioning of the fuel assemblies, 4) an alternate neutron poison model for METAMIC®, and 5) taking credit for 90% of the boron content in the METAMIC®.

6.3.1 Configuration

The applicant requested the addition of a new DFC for the MPC-32/32F. A sketch of the DFC is shown in Figure 2.1.2D of the FSAR. This DFC is used to load damaged fuel and fuel debris into the MPC-32/32F and was modeled in the corners of the canister basket for the analyses of these cases.

The amendment requested approval to load up to eight DFCs into the MPC-32 in the corner locations of the basket as indicated in Figure 6.4.16 of the FSAR. Although the MPC-32F was added as a new canister, which may contain non-intact fuel, the criticality control features remain the same as for the MPC-32. Thus, the model of the basic canister configuration was not changed. Also, the basic approach for modeling damaged fuel and fuel debris was the same as previously used for these fuel conditions in the other MPC versions. For the cases of damaged fuel and fuel debris in the MPC-32/32F, the model consisted of 24 intact fuel assemblies and 8 DFCs in the corner basket locations as shown in Figure 6.4.16 of the FSAR. The modeling approach for fuel in the DFC bounds both damaged fuel and fuel debris.

The applicant modified its analysis of eccentric positioning of the fuel assemblies in the basket cells to compare cases of the fuel assemblies shifted toward the cask center, the fuel assemblies centered in each basket cell, and the fuel assemblies shifted away from the cask center. In all canisters, k_{eff} was lowest when the fuel assemblies are shifted away from the cask center. For the MPC-24E/EF, the k_{eff} for the case of fuel assemblies centered in each basket cell was slightly more reactive that when the assemblies are shifted toward the cask center. However, for the MPC-32/32F, sometimes k_{eff} is highest when the fuel assemblies are centered in the basket cells and sometimes k_{eff} is highest when the fuel assemblies are shifted toward the cask center. Therefore, the applicant's analysis for the MPC-32/32F considered both cases of cell centered and cask centered placement and reported the highest value of k_{eff}.

The model for the METAMIC® as an alternate neutron poison material in the canister basket was modified to reflect the fact that METAMIC® is made of a single sheet of material without cladding in contrast to BORAL® which was modeled with an aluminum cladding sheet on each side. Calculations were made with METAMIC® and BORAL® and the results were compared for any trends or differences. No significant trends or differences were observed.

6.3.2 Material Properties

In the amendment request, the specifications in LCO 3.3.1 of the TS for the boron concentrations in the water during wet loading and unloading of the MPC-32 were changed from a single bounding value to concentrations which vary depending on the initial enrichment and class of the fuel assembly. In addition, boron concentration specifications were added to LCO 3.3.1 in the TS for the MPC-24 E/EF when wet loading or unloading occurs with non-intact fuel assemblies.

METAMIC® was proposed as an alternative neutron poison material because it is expected to show better performance than BORAL® due to the greater uniformity and smaller particle size of its boron content. Therefore, in the criticality safety analysis, the applicant proposed to take credit for 90% of the ^{10}B content in METAMIC® as opposed to 75% credit for BORAL® plates. To justify this change, the applicant will subject the METAMIC® plates to a more extensive and comprehensive program of acceptance tests after fabrication.

METAMIC® is a commercially produced neutron poison that has been previously evaluated and accepted by the NRC for use in dry spent nuclear fuel storage casks . The materials used to fabricate METAMIC® are the same as for BORAL®, although the fabrication details differ substantially. Consequently, no new or different materials are introduced to the cask interior environment by the substitution of one neutron poison for the other.

Among the physical differences, METAMIC® achieves essentially a 100% theoretical density of the powder metallurgy aluminum-boron carbide matrix. This contrasts with the slightly porous nature of BORAL®. This porous property of BORAL® has been suggested as the cause of some hydrogen generation during loading operations. Because the corrosion resisting properties of METAMIC® are similar to or superior to those of BORAL®, there is no adverse change to the previous assessments of potential chemical or galvanic reactions in the cask environment.

The applicant stated a desire to employ METAMIC® with boron carbide contents of up to 33% (for a nominal minimum of 31%). Previously, the staff had approved METAMIC® for use in dry spent nuclear fuel storage casks, however, with boron carbide contents limited to 15%. In order to use a higher concentration of boron carbide, the licensee must, through approved materials evaluation and quality control processes, demonstrate the efficacy of the higher concentration material. The staff finds that the material property change that would result from this increase in boron carbide content would have no significant adverse effect upon the properties of the absorber material.

The applicant's method of demonstrating neutron poison material efficacy has been previously reviewed and accepted by the staff. The NRC staff reviewed these procedures and requirements in conjunction with the proposed addition of METAMIC® to the bill of materials. The staff finds the procedures and methods of testing to be acceptable for use in controlling the production of METAMIC®.

Consequently, the NRC staff finds that the applicant may employ the higher concentration METAMIC® material when it is shown to meet all the specified quality assurance (QA)/quality control (QC) testing requirements and cask design requirements. The applicant's manufacturing and testing requirements are incorporated by reference into Appendix B of the CoC.

6.4 Criticality Analysis

In general, the criticality analysis for all of the changes described above followed the same methodology of the previous applications for the HI-STORM 100 Cask System. Changes in the method that apply to this application are described in Sections 6.1, 6.2 and 6.3 of this SER.

Also, as a result of the more comprehensive analysis of eccentric positioning of the fuel assemblies in their basket cells, the bounding maximum k_{eff} values reported for some of the fuel types were increased slightly from the previous revision to the FSAR. While all values of k_{eff} remained below the acceptance limit of 0.95, the new eccentric positioning analysis did identify configurations with higher values for k_{eff} than previously reported.

6.4.1 Computer Programs

The applicant's use of computer codes and neutron cross sections have not changed from previous submittals and the staff's evaluation is provided in a previous SER.

6.4.2 Multiplication Factor

The applicant's criteria for the number of neutron histories and convergence did not change from the previous submittals and the staff's evaluation is provided in a previous SER.

6.4.3 Benchmark Comparisons

The benchmarking procedures and methods have not changed from previous applications and the staff's evaluation is provided in a previous SER.

6.5 Criticality Evaluation Summary

The staff notes that the methods in the FSAR are consistent with the previous applications except as described above. The applicant explicitly modeled the assemblies using the same computer code and cross section set used in the original application. The applicant used three-dimensional calculation models in its criticality analyses. Sketches of the models are given in the FSAR as discussed above. The models are based on the engineering drawings in the FSAR. The design-basis, off-normal and accident events do not affect the design of the cask from a criticality standpoint. Therefore, the calculational models for the normal, off-normal, and accident conditions are the same.

The staff used the CSAS/KENO-Va codes in the SCALE suite of analytical codes to perform a limited number of confirmatory analyses. These calculations used the 44-group cross-section set in SCALE. The CSAS/KENO code was developed by the Oak Ridge National Laboratory for performing criticality analyses and is appropriate for this particular application and fuel system. The results of the staff's confirmatory calculations were in close agreement with the applicant's results.

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

6.6 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:

F6.1 SSCs important to criticality safety are described in sufficient detail in Chapters 1, 2, and 3 of the FSAR and in the design drawings to enable an evaluation of their effectiveness.

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

F6.3 The criticality design is based on favorable geometry and fixed neutron poisons.

F6.4 The NRC 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 reached on the basis of 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. 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.

7.1 Confinement Design Characteristics

The applicant requested several changes to the HI-STORM 100 Cask System design and CoC. Only those changes that affect the confinement system are discussed in this section. The staff reviewed proposed Amendment 2 of the HI-STORM 100 Cask System confinement analyses to ensure that credible normal, off-normal, and accident conditions have been identified and their potential consequences considered, such that the HI-STORM 100 Cask System continues to meet the regulatory requirements of 10 CFR 72.236.

The HI-STORM 100 Cask System uses a fully welded austenitic stainless steel MPC design to maintain confinement. The confinement boundary on the MPC design include 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 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.

The change requested in this amendment affected the inspection and leak testing of the lid-to-shell structural weld. The applicant applied the criteria described in ISG-15, "Materials Evaluation" and 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" in the amendment request. ISG-15 provides an NRC approved alternative to the ASME Code for the inspection of final closure welds for austenitic materials. The inspection techniques described by ISG-15 will detect any such flaws which could lead to a failure. In addition, ISG-18 states that when the closure welds of austenitic stainless steel canisters are executed in accordance with ISG-15, the staff concludes that no undetected flaws of significant size will exist. Therefore, the staff has reasonable assurance that this inspection demonstrates no credible leakage would occur from the final closure welds of austenitic stainless steel canisters. ISG-18 removes the need for a helium leak test of the lid-to-shell structural weld in accordance with ANSI N14.5.

7.2 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 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 the 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 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, as changed and updated by the applicant in the FSAR with the addition of the helium leak rate testing requirements for the vent and drain port cover plates, as described in section 8.4 of this SER, 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 FSAR presents acceptable operating sequences, guidance, and generic procedures for key operations.

The applicant requested several changes to the operating procedures for the HI-STORM 100 Cask System. Only those changes that affect operating procedures are discussed in this section. The staff reviewed proposed Amendment 2 of 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 FSAR, meet the following regulatory requirements: 10 CFR 72.104(b), 72.122(l), 72.212 (b)(9), 72.234(f), and 72.236(h) and (l) . Proposed Amendment 2 of the FSAR was 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 FSAR, are acceptable to the staff:

a. Revision of requirements for ensuring MPC cavity bulk helium temperature is less than 200EF (93EC) prior to reflooding instead of existing language that reads "helium gas exit temperature" in the event unloading should be necessary .
b. Addition of discussion and revised criteria pertinent to use of the Forced Helium Dehydration (FHD) System and Supplemental Cooling System (SCS)
c. Removal of reference to requirement for performing helium leakage test of the lid-to-shell structural weld in accordance with staff guidance provided in ISG-18.
d. Revision of procedures to address hydrogen generation

The staff's conclusions, summarized below, are based on information provided in Amendment 2, as revised, to the HI-STORM 100 Cask System FSAR.

8.1 Forced Helium Dehydration (FHD) System

The applicant provided an overview of the design, analysis, and testing requirements for the FHD in Chapter 2, Appendix 2.B, of the FSAR. The FHD was developed to remove moisture from a cask loaded with high burnup fuel and as an option for moderate burnup fuel. The FHD was determined to be a more conservative approach to moisture removal and drying than the previously approved vacuum drying system. Using the FHD system for drying operations was determined to be more effective at heat removal and, therefore, providing greater margin to short term peak cladding temperature limits.

Forced helium drying would be used to achieve the 752EF (400EC) cladding temperature limit specified in ISG-11, Revision 3, when the decay heat load would prohibit the use of vacuum drying. Forced helium drying has previously been reviewed by the staff and found to be an acceptable method for drying the cask internals and contents while enabling the maintenance of the cladding temperature limit of 752EF (400EC) or less.

The applicant proposed using the temperature of the gas exiting the FHD system demoisturizer if maintained below 21EF for a minimum of 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain below 22.9EF for greater than 30 minutes as the means to ensure that all liquid water is removed from the MPC during drying operations. Maintaining a dew point below 22.9EF for greater than 30 minutes corresponds to a partial water vapor pressure of 3 torr which is the accepted dryness limit for spent fuel storage casks per NUREG-1536, section 8.V.1. The staff agrees that the proposed Amendment 2 changes to the design, analysis, and testing requirements are 1) adequate to ensure that licensees do not exceed peak cladding temperatures during moisture removal from the HI-STORM 100 Cask System and 2) that the proposed Amendment 2 procedures in Chapter 8 of the FSAR and Design Feature 3.6, “Forced Helium Dehydration System,” of Appendix B to the CoC provide reasonable assurance to ensure that the design implemented by licensees using the HI-STORM 100 Cask System will satisfy the design basis of the cask system.

8.2 Supplemental Cooling System

The applicant provided a general description and requirements for the SCS in 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 FSAR during normal short-term operations. The heat dissipation capacity of the SCS shall be equal to or greater than the minimum necessary to ensure that the peak cladding temperature is below the ISG-11, Revision 3 limit of 752EF (400EC). The applicant left to the end user of the spent fuel storage system the task to analyze the conditions when supplemental cooling is required during on-site transfer operations. The applicant stated that the particular type of augmented cooling is necessarily site-specific and is left to the user to determine, using the thermal methodologies in the HI-STORM 100 Cask System FSAR. The following thermal requirement shall be included in the CoC as a condition for approval of the SCS.

A first time user of a HI-STORM 100 Cask System SCS that uses components or a system that is not essentially identical to components or a system that has been previously tested shall record the coolant temperature measurements and coolant flow rate for inlet and outlet of the annulus between the HI-TRAC and MPC. The user shall also record the MPC operating pressure and decay heat. Subsequently the user shall use this information to validate the thermal methods described in the FSAR which were used to determine the type and amount of supplemental cooling being used by the first-time user of the SCS. A letter report summarizing the analysis and results shall be submitted to the NRC in accordance with 10 CFR 72.4.

8.3 Hydrogen Monitoring During Cask Loading/Unloading Operations

Operational experience at several facilities during 2002 demonstrated the propensity for the licensee’s casks to generate some hydrogen. Most of this hydrogen is evidently produced by the BORAL® neutron absorber. This reaction has been evaluated and found to have no effect on the overall properties or performance of the BORAL® or any other cask material or contents. However, to prevent the accumulation of a flammable amount of hydrogen during the loading or unloading process, especially during lid welding, the applicant has added a hydrogen concentration monitoring procedure to the loading/unloading procedures. Additionally, the FSAR has been revised to reflect the possibility for hydrogen generation, whereas it previously stated that none would occur.

The inclusion of a hydrogen monitoring procedure into the cask loading/unloading procedures brings the Holtec cask procedures into congruence with typical industry practice. The NRC staff finds this inclusion to be acceptable.

8.4 Elimination of Staff Requirement for Helium Leak Testing of the Lid-to-Shell Structural Weld

The applicant proposed to eliminate the previous staff requirement for helium leakage testing of the lid-to-shell structural weld of the MPC in accordance with the revised staff position on helium leakage testing contained in 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", Dated May 2, 2003. However, due to a misinterpretation of the staff intent, the applicant proposed eliminating helium leakage testing of all welds associated with the structural lid, instead of just the lid-to-shell structural weld. The staff found that elimination of helium leakage tests for all of the welds associated with the structural lid to be contrary to the intent of ISG-18.

Acceptable helium leakage test protocols would be either of the following, the first option being continuation of the previously licensed requirement to test every weld associated with the structural lid. The second option would allow elimination of the structural lid-to-shell weld helium leakage test under the guidance of ISG-18.

Option 1: After the MPC structural lid-to-shell weld is completed, the weld is helium leakage tested and hydrostatic tested. If the MPC lid weld is acceptable, the vent and drain port cover plates are welded in place, examined by the liquid penetrant method (root and final), and a leakage rate test is performed. Finally, the MPC closure ring (with no closure ring penetrations) is installed, welded, and inspected by the liquid penetrant method (root and final).

Option 2: Alternatively, the MPC lid to shell weld is completed and hydrostatic tested. If the MPC lid weld is acceptable per the hydro test, the vent and drain port cover plates are welded in place. The vent and drain port cover plate welds are helium leakage rate tested and penetrant tested. No helium leakage rate test of the MPC lid-to-shell weld is required, per the guidance of ISG-18.

Note that for both options, the lid-to-shell structural weld is executed in accordance with the guidance of ISG-15, employing the required multiple penetrant tests. Implementing either Option is acceptable to the NRC staff.

8.5 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:

F8.1 The HI-STORM 100 system can be wet loaded and unloaded. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the 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 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 FSAR with the addition of helium leak rate testing requirements for the vent and drain port cover plates, as described in section 8.4 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 FSAR includes the appropriate acceptance tests and maintenance programs for the HI-STORM 100 system. Only minor changes were made to the acceptance tests and maintenance programs in the applicant's amendment request.

One change to Chapter 9 of the FSAR was reviewed by the staff regarding the use of alternate neutron absorber materials in the HI-STORM 100 Cask System. See Section 6.3.2 of this SER for the staff's evaluation of METAMIC® and review of associated acceptance testing methods. Chapter 9 FSAR language was modified to remove the direct reference to BORAL® as the only approved neutron absorber material.

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 modification made to the acceptance tests and maintenance program for the amendment to the HI-STORM 100 system, with inclusion of helium leak rate testing requirements, 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 review of this section is to ensure that the capability of the current and revised radiation protection design features, design criteria, and the operating procedures, as appropriate, of the HI-STORM 100 system can meet regulatory dose requirements for the revised 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 system are based on the direct radiation dose rates calculated in Chapter 5 of the FSAR and the operating procedures discussed in Chapter 8 of the 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 FSAR.

The proposed change to higher dose rates resulting from revised contents was reviewed to determine if the radiation protection design features for the HI-STORM 100 Cask System, as described in the FSAR, is acceptable to the staff.

The staff's conclusions, summarized below, are based on information provided in Amendment 2, 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, the staff has incorporated TS 5.7 to establish 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 FSAR amendment. 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 it acceptable. Sections 5, 7, and 8 of this SER discuss specific staff evaluations of the design criteria and features for the shielding system, confinement systems, 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 previously approved ALARA assessment for the HI-STORM 100 Cask System and found it acceptable for the higher 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). Because the revised contents result in significantly higher direct radiation dose rates from that previously approved for HI-STORM 100 Cask System, each user may be required to take additional ALARA precautions to minimize doses to personnel, and make additional use of realistic fuel characteristics and distances to demonstrate compliance with public dose limits in 10 CFR Part 20 and Part 72.

10.3 Occupational Exposures

The applicant calculated new occupational exposures in Section 10.3 of the FSAR amendment. The exposures were based on estimations from surrounding direct-radiation dose rates calculated in Chapter 5 and the operating procedures referenced in Chapter 8. Table 10.3.1.b of the FSAR indicates that collective exposure for loading the HI-STORM 100 Cask System with the revised design-basis contents and using the 100-ton HI-TRAC will increase from approximately 2.7 to 5.2 person-rem. Table 10.3.2.b of the FSAR indicates that collective exposure for unloading a single cask will increase from approximately 1.3 to 2.6 person-rem.

The staff reviewed the overall occupational dose estimates and found them acceptable. The occupational dose exposure estimates provide reasonable assurance that occupational limits in 10 CFR Part 20 Subpart C can be achieved. The staff expects actual operating times and personnel exposure rates will vary for each system depending on site-specific operating conditions, including detailed procedures and special measures taken to maintain exposures ALARA. The collective exposures will be distributed among multiple personnel responsible for various tasks. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20, Subpart B. In addition, each licensee will demonstrate compliance with occupational dose limits in 10 CFR Part 20 Subpart C and other site-specific 10 CFR Part 50 license requirements with evaluations and measurements. Staff 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 FSAR amendment. Based on Table 5.4.6 in the 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 200 meters to 250 meters. The analyses indicated that the minimum distance at which the annual dose limit is satisfied for a 2x3 cask array, is increased from 300 meters to 400 meters.

As discussed in Section 7 of this SER, the applicant eliminated the confinement analysis (assuming a hypothetical leak) from the licensing basis because the applicant has demonstrated that the canister designs meet the criteria of ISG-18. Therefore, the applicant indicated leakage from the canister is not credible and doses from effluents are zero. As a result of the change, leak-testing and related surveillance requirements are removed from T.S. 3.1.1 for the MPC. The codes and standards requirements in CoC Appendix B, Section 3.3, require the applicant to satisfy the basic requirements specified in ISG-18.

The staff evaluated the public dose estimates during normal and off-normal conditions and found them acceptable. The primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions is from direct radiation (including skyshine). Leakage from the canister is not credible based on the criteria of ISG-18, and the confinement function is not affected by normal or off-normal conditions. Therefore, public exposure from release of radioactive material is not credible. A discussion of the staff's evaluation and confirmatory analysis of the shielding calculations and confinement analysis are presented in Sections 5 and 7 of this SER, respectively. Section 11 of this SER addresses additional consideration for new off-normal events analyzed in the FSAR.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each general licensee. The general licensee using the HI-STORM 100 Cask System must perform a site-specific evaluation, as required by 10 CFR 72.212(b) to demonstrate compliance with 10 CFR 72.104(a). The actual doses to 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 review of the applicant's shielding analyses, the radiation protection program and features, and generic cask operating procedures, the staff removed TS 3.2.1 and 3.2.3 for the overpack and transfer cask doses rates. The staff has defined criteria for a radiation protection program in new TS 5.7 for the revised HI-STORM 100 Cask System. The criteria include the 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.

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

Chapter 11 of the FSAR amendment 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 results in a dose at the controlled area boundary that is 40% below the regulatory requirements of 10 CFR 72.106(b). Chapter 11 of the FSAR amendment discusses or references the corrective actions for each design-basis accident, as appropriate.

The staff evaluated the public dose estimates from the revised contents for accident conditions and natural phenomena events, and found them acceptable. A discussion of the staff's evaluation of the shielding and confinement analysis 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 are presented in Section 11 of this and previous SERs, as appropriate, for the revised 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 FSAR amendment 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 FSAR amendment 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 FSAR amendment 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:

C Identified all credible accidents
C Provided complete information in the FSAR
C Analyzed the safety performance of the cask system in each review area
C 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 72.122(b)(2), (3), (d), (g), (h)(4), (l), and (l), 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 fulfills the acceptance criteria listed in Section 11 of NUREG-1536.

The following proposed changes were considered for their affect on accident analyses and conclusions;

a. Revision of soluble boron requirements for the MPC-32/32F based on fuel assembly array/class, intact versus damaged fuel, and initial enrichment.
b. Revision to design criteria and use of the FHD system
c. Addition of language to allow use of a SCS.
d. Revision of confinement analysis and methods

The staff's conclusions, summarized below, are based on information provided in Amendment 2, 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 FSAR. The off-normal conditions described in the FSAR include:

C Off-Normal Pressures
C Off-Normal Environmental Temperatures
C Partial Blockage of Air Inlets
C Malfunction of FHD System
C SCS Power Failure

11.1.1 Off-normal pressure

The off-normal design pressure has been increased from 100 psig to 110 psig with no physical changes being required in the structural components as a result of pre-existing margins in excess of those required for adequate safety.

11.1.2 Off-normal environmental temperature

No changes were made in the off-normal environmental temperatures in this amendment that impacted the structural or thermal aspects of the design.

11.1.3 Partial blockage of air inlet

Changes made in this amendment impacting the structural considerations are bounded by other conditions. Thermal analyses, previously approved by the staff, remain bounding.

11.1.4 Dose Limits for Off-Normal Events

Section 11.1 of the FSAR amendment examines the dose consequences for the revised contents and the additional off-normal events that are analyzed with respect to identified off-normal events. The HI-STORM 100 Cask System meets physical design and testing requirements in ISG-18 that assures there is no credible leakage from the MPC, and that there will be no breach of the confinement boundary due to off normal conditions. Therefore, the applicant eliminated the radiation protection evaluation for the leakage of one seal off-normal condition. The applicant indicated that direct radiation conditions continue to be essentially the same as normal conditions analyzed in Chapter 5 and 10 of the FSAR amendment. The applicant indicated that there is no effect on the shielding or confinement system, or consequences to the public from the additional off-normal events added to the FSAR analysis which include failure of the FHD system and SCS power failure.

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 revised 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 evaluate the radiological doses applicable to off-normal events, as appropriate.

11.1.5 Malfunction of FHD System

The applicant stated that the applicable peak cladding temperature limit is 1058EF (570EC) and presented the steady state results in Table 11.1.3 of the FSAR for the different MPC types and maximum design heat load considered in the application. The applicant stated that the HI-STORM 100 Cask System is designed to withstand the FHD failure without an adverse effect on its safety function and therefore, no corrective action is required. The following conditions are assumed for the FHD System malfunction:

1. Steady state maximum temperatures have been reached
2. Design basis heat load
3. Standing column of air in the annulus
4. MPC backfilled with the minimum helium pressure required by the TS

During malfunction of the FHD system, the applicant's results demonstrated that the peak fuel cladding temperatures remain below the applicable limit in the event of a prolonged unavailability of the FHD system.

11.1.6 SCS Power Failure

During a SCS failure, the applicant's results demonstrated that the peak fuel cladding temperatures remain below the applicable limit in the event of a prolonged unavailability of the SCS system. The applicant provided thermal analysis results (Table 4.5.2 of the FSAR), obtained from a steady state calculation of the HI-TRAC transfer cask, demonstrated that without the use of the SCS, maximum peak cladding temperature remains below the acceptable limit for off-normal and accident conditions.

The staff reviewed these events and found them to be bounded by evaluations contained in Chapters 3 and 4 of the 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 FSAR and include:

- HI-TRAC Transfer Cask Handling Accident
- HI-STORM Overpack Handling Accident
- Tip Over
- Fire Accident
- Partial Blockage of MPC Basket vent Holes
- Tornado
- Flood
- Earthquake
- Lightning
- Explosion
- 100% Blockage of Air Inlets
- Burial Under Debris
- Extreme Environmental Temperature
- SCS Failure

Only those changes in the proposed Amendment 2 FSAR that might affect the above postulated accidents are addressed below.

11.2.1 Supplemental Cooling System (SCS) Failure

For a SCS failure under accident conditions, the applicant's results demonstrated that the peak fuel cladding temperatures remain below the applicable limit in the event of a prolonged unavailability of the SCS system.

The staff reviewed this event and found it to be bounded by evaluations contained in Chapter 4 of the FSAR and accepted by the staff in Section 4 of this SER. The staff agrees that all accident-level events and conditions have been identified and all potential safety consequences considered.

11.2.2 Dose Limits for Design-Basis Accidents and Natural Phenomena Events

Section 11.2 of the FSAR amendment examines the dose consequences for the identified design-basis accidents and natural phenomena events for the revised contents and design. The HI-STORM 100 Cask System meets physical design and testing requirements in ISG-18 that assures there is no credible leakage from the MPC, and that there will be no breach of the confinement boundary due to design-basis accidents. As discussed in Section 5.4.4 of this SER, the applicant determined the direct radiation dose to be approximately 3.1 rem for the revised contents and bounding design-basis events (loss of transfer cask water shield) examined in Section 11.2 of the FSAR amendment.

The staff reviewed the design-basis accident analyses with respect to 10 CFR 72.106(b) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual beyond at or beyond the controlled area boundary of 100 meters will not exceed the limits in 10 CFR 72.106(b) for the revised contents and design. Sections 5, 7, and 10 of this SER further evaluate the estimated radiological doses during accident conditions.

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

12.0 CONDITIONS FOR CASK USE —TECHNICAL SPECIFICATIONS

The 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 also reviewed a proposed change to Section 2 of Appendix B to the CoC. The proposed change added new Section 2.3 to Appendix B to allow for NRC approval of case specific alternatives to cask contents in lieu of submittal of an amendment request and subsequent NRC rulemaking to codify the requested (and approved) change. The staff is unable to approve this proposed change as NRC regulations (10 CFR 72.244) prohibit a certificate holder from changing information contained in the CoC (and Appendices) by any means other than an amendment to the CoC. Case specific alternatives to approved cask contents listed in the CoC Appendices may be submitted to the NRC for review via the 10 CFR 72.7 exemption process.

The staff reviewed the proposed change to remove TS 3.1.1 Action Statement C, " Helium leak rate limit not met." As described in Section 8.4 of this SER, removal of this TS requirement is not in agreement with the guidance of ISG-18. Therefore the staff is unable to approve this change. However, because ISG-18 does provide for the elimination of the need to helium leak test the lid-to-shell closure weld the staff finds it acceptable to specify that only the vent and drain port cover plate welds need be helium leak rate tested. Therefore Action Statement C shall read, "MPC helium leak rate test for vent and drain port cover plate welds."

This review did not assess the technical adequacy of the changes which are evaluated in Sections 3 through 11 of this SER.

12.1 Conditions for Use

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

The staff reviewed the proposed CoC and TS changes and, with the addition of the changes to TS 3.1.1 C, 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 system. The TS were revised to reflect test methods used for qualifying neutron absorber materials. Due to the proprietary nature of neutron absorber materials, no nationally recognized standard exists, as in the case for ASME Code materials, to assure the uniform quality, and thus performance, of such materials. Consequently, the staff approved Holtec procedures for testing neutron absorber materials have been incorporated by reference into the TS. This step will assure: 1) that NRC staff accepted tests and test methods are used, and, consequently, 2) uniform performance of these materials will be assured in service. Incorporation of the appropriate methods and procedures is by reference, as stated in the revised TS. This effectively prohibits changes to the incorporated material without prior NRC staff review.

The staff has reviewed the TS and with the addition of the changes to TS 3.1.1 C 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 revision to CoC 1014, Approved Contents and Design Features, to reflect the changes to the contents for the HI-STORM 100 Cask System requested by the amendment. The staff has reviewed the revisions, and with the exception of the proposed addition of new CoC, Appendix B, Section 2.3, as discussed in Section 12.0 of this SER, 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, with the addition of the changes to TS 3.1.1 C, and the Approved Contents and Design Features contained in CoC 1014 for the HI-STORM 100 system, and with the exception of the addition of new language to the CoC, Appendix B, Section 2.3, 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	(intentionally left blank)
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY/SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SPENT FUEL STORAGE CASK (SFSC) Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2	Deleted
3.2.1	Deleted
3.2.2	Deleted
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	(intentionally left blank)
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.

Regulations describing the requirements for a 10 CFR Part 72 Quality Assurance Program are contained in 10 CFR Part 72 Subpart G. It is stated in 10 CFR 72.140(d), in part, that a quality assurance program previously approved by the Commission as satisfying the requirements of Subpart H to 10 CFR Part 71 will be accepted as satisfying the requirements of 10 CFR Part 72 Subpart G, except that a certificate holder using a 10 CFR 72 Subpart H quality assurance program shall also meet the record keeping requirements of 10 CFR 72.174. Also, 10 CFR 72.140(d) states that in filing the description of the quality assurance program required by 10 CFR Part 72 the certificate holder shall notify the NRC of its intent to apply its previously approved 10 CFR Part 71 quality assurance program to ISFSI activities or spent fuel storage cask activities. The notification shall identify the previously approved quality assurance program by date of submittal to the Commission, docket number, and date of Commission approval.

In LAR 1014-2, the applicant, revised the HI-STORM 100 Cask System FSAR, Chapter 13, Quality Assurance, by deleting the 10 CFR Part 72 QA program that had been previously described and by adding reference to the Holtec 10 CFR Part 71 QA program that had been previously submitted and approved by the NRC for 10 CFR Part 71. The NRC approved the Holtec 10 CFR Part 71 QA program in Quality Assurance Program Approval Number 0784, Revision 3, dated September 25, 2001. Holtec notified the NRC of its intent to apply its 10 CFR Part 71 quality assurance program to ISFSI activities or spent fuel storage cask activities in Holtec letter, “USNRC Docket Numbers 72-1008, 1014, and 71-9261, TAC L23082, Holtec International Quality Assurance Program,” dated June 20, 2001. This letter was amended by Holtec letter, “Holtec International QA Program Review,“ dated August 17, 2001. Holtec also specified in the revised FSAR Chapter 13, that the record keeping requirements of 10 CFR 72.174 would be met.

On the basis of the staff's detailed review and evaluation of the QA program described in the HI-STORM 100 Cask System FSAR, submitted by LAR 1014-2, for the HI-STORM 100 Cask System, the staff concluded that the requirements of 10 CFR 72.140(d) were met, and the applicant had described a QA program acceptable to 10 CFR Part 72, in that a quality assurance program previously approved by the Commission as satisfying the requirements of Subpart H to 10 CFR Part 71 was referenced in the FSAR, and the certificate holder committed to meet the record keeping requirements of 10 CFR 72.174 in the FSAR.

Accordingly, the staff concluded that the applicant’s QA program complies with the applicable NRC regulations and industry standards and can be implemented for the HI-STORM 100 Cask System.

Based on the NRC staff's review of the HI-STORM 100 Cask System amendment request, the staff concludes that the HI-STORM 100 Cask System, as amended, meets the acceptance criteria specified, for both intact and damaged fuel, in NUREG-1536.

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 requirement for helium leakage rate testing of the vent and drain port cover plate welds in accordance with ISG-18, and based on the statements and representations contained in the 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 Amendment 2, are acceptable. However, for the purposes of the Amendment 2 review, the staff did not revisit any previously approved methodologies used in the original HI-STORM 100 Cask System application or those reviewed for Amendment 1 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 2 change.

Issued with Certificate of Compliance No. 1014,
on 07 June 2005