HOLTEC INTERNATIONAL

HI-STORM 100 CASK SYSTEM

AMENDMENT NO. 1

SAFETY EVALUATION REPORT

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DOCKET NO. 72-1014
HI-STORM 100 CASK SYSTEM
HOLTEC INTERNATIONAL
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 1

SUMMARY

By letter dated August 31, 2000, as supplemented, Holtec International (Holtec) submitted an amendment application to the U. S. Nuclear Regulatory Commission (NRC), in accordance with 10 CFR Part 72 to modify the HI-STORM 100 Cask System as follows:

- Add four new multipurpose canisters (MPCs). The new MPCs include the MPC-24E, MPC-24EF, MPC-32 for pressurized water reactor (PWR) fuel and the MPC-68FF for boiling water reactor (BWR) fuel.
- Add damaged fuel containers.
- Add the HI-STORM 100S overpack and the 100A and 100SA high-seismic anchored overpacks.
- Allow the storage of high-burnup fuel.
- Revise the thermal analysis to include natural convection heat transfer, revise the helium backfill requirements to allow a helium density measurement to be used, allow a helium drying system, for use with high burn-up fuel, rather than the existing vacuum drying system, and require soluble boron during canister loading for certain higher enriched fuels.
- Relocate the technical specifications for special requirements for the first systems in place and training requirements to the main body of Certificate of Compliance (CoC) 1014.
- Allow the storage of selected non-fuel hardware.

This Safety Evaluation Report (SER) documents the review and evaluation of the Final Safety Analysis Report (FSAR), as amended. The FSAR follows the format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems. This SER uses essentially the same Section-level format, with some differences implemented for clarity and consistency.

The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. The staff's assessment focused only on modifications requested in the amendment and did not reassess previously approved portions of the FSAR.

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 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 and aluminum heat conduction elements. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. The honeycombed basket, which is equipped with Boral neutron absorbers, provides criticality control.

This amendment added four new MPCs to the HI-STORM System.

- The MPC-24E which is designed to store up to 24 PWR fuel assemblies including up to four damaged fuel assemblies, with or without non-fuel hardware.

- The MPC-24EF which is designed to store up to 24 PWR fuel assemblies, with or without non-fuel hardware. In addition, up to four damaged fuel assemblies including up to four stainless steel damaged fuel containers (DFC) containing fuel debris.

- The MPC-32 which is designed to store up to 32 PWR fuel assemblies classified as intact fuel assemblies, with or without non-fuel hardware.

- The MPC-68FF which is designed to store up to 68 BWR fuel assemblies including up to 16 damaged fuel assembles, 8 of which may be fuel debris stored in DFCs. In addition, all fuel loading combinations allowed by the MPC-68F may be stored in this MPC as well.

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 OVERPACKS

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. This amendment adds three new overpacks to the HI-STORM Cask System:

- The HI-STORM 100S is a shorter version of the previously approved HI-STORM 100. To accommodate the height change the location of the air ducts and MPC pedestal height were 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. For the purposes of this SER the discussions involving the HI-STORM 100A are meant to include the anchored versions of both the HI-STORM 100 and 100S unless otherwise specified.

The HI-STORM 100S and 100A overpacks were reviewed for this amendment only to the extent that their designs were not bounded by the staff's previous review of the HI-STORM 100 overpack.

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, backfilled, and leak tested; (3) the transfer cask is placed on top of the overpack and the MPC is lowered into the overpack; and (4) the overpack, with the MPC inside, is moved to the storage pad. A loaded HI-TRAC transfer cask can be handled vertically or horizontally; a loaded HI-STORM 100, 100S, and 100A 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 Chapters 3 through 14 of this SER, as necessary.

1.3 Cask Contents

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

- Storage of spent fuel with a higher initial enrichment in HI-STORM's 24 PWR Multipurpose Canister (MPC-24) made possible by using borated water during loading/unloading.

- Addition of four new fuel assembly array classes (PWR 14x14E and 15x15H; BWR 8x8F and 9x9G) to the approved contents.

- Addition of three new damaged fuel containers (DFC) which include: 1) the Transnuclear DFC currently containing spent fuel from Dresden Unit 1, 2) a Holtec generic PWR DFC, and 3) a Holtec generic BWR DFC.

- Revision of some fuel assembly parameters, for previously approved PWR and BWR assemblies, to ensure all fuel currently intended for storage in the HI-STORM 100 Cask System can be stored safely.

- Storage of non-fuel hardware, including burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), axial power shaping rods (APSRs), control element assemblies (CEAs), wet annular burnable absorbers (WABAs), rod cluster control assemblies (RCCAs), water displacement guide tube plugs and orifice rod assemblies.

- Storage of additional damaged BWR fuel types in the MPC-68.

- Inclusion of one Dresden Unit-1 thoria rod canister containing 18 thoria rods in the MPC-68 and MPC-68F.

- Inclusion of Dresden Unit-1 fuel with one antimony-beryllium source in the assembly in the MPC-68, MPC-68F, and MPC-68FF.

1.4 Evaluation Findings

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 are presented in Section 1.5 of the FSAR. Specific structures, systems, and components are evaluated in Chapters 3 through 14 of this SER, as necessary.

F1.3 Specifications for the additional 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 Certificate of Compliance.

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 program and implementing procedures are described in Chapter 13 of the FSAR were previously reviewed and approved for the CoC 1014 and subsequent NRC inspections and were not reviewed for this amendment.

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

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

Modifications to the 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 NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

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

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. These are evaluated in Chapters 3 through 14 of this SER, as necessary.

2.3 Design Criteria for Safety Protection Systems

The principal design criteria for the various new MPC and the HI-STORM overpack designs and the TC, are summarized in FSAR Tables 2.0.1, and 2.0.2, and 2.0.3, respectively and are essentially the same as the original design. The codes and standards of the design and construction of the system are specified in Section 2.2 of the FSAR. In addition, Tables 2.0.4 and 2.0.5, were added to provide the design requirements for an ISFSI pad for high seismic areas requiring the use of a HI-STORM 100A overpack.

2.3.1 General

Chapter 2 of the FSAR was modified to include design changes associated with the MPC and overpacks. The changes included the designation of an ISFSI pad as important-to-safety if the zero period acceleration (ZPA) at the surface of an ISFSI pad exceeds the threshold for a free-standing HI-STORM installation and requires the use of a HI-STORM 100A overpack.

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.

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. Heat removal, by conduction, radiation, and natural convection, is independent of intervening actions under normal, off-normal, and accident conditions. The thermal design criteria include maintaining fuel cladding integrity and ensuring that temperatures of materials and components important to safety are within the design limits.

2.3.4 Shielding/Confinement/Radiation Protection

The shielding and confinement analyses and the radiation protection capabilities of the HI-STORM 100 Cask System are presented in 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 hydrostatic testing, helium leak testing and weld examinations. Radiation exposure is minimized by the neutron and gamma shields and by operational procedures.

2.3.5 Criticality

The criticality analysis is presented in 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 Boral plates. The continued efficacy of the Boral over a 20-year storage period is assured by the design of the system. Depletion of the ^{10}B in the Boral is negligible because the neutron flux in the MPC over the storage period is low.

2.3.6 Operating Procedures

Generic operating procedures are described in 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

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 Chapters 3 through 14 of this SER.

3.0 STRUCTURAL EVALUATION

This Chapter presents the results of the structural design review of the amendment request for the Holtec International HI-STORM 100 System CoC 1014. The amendment request addresses the addition of the following components to the HI-STORM 100 System: the MPC-24E, MPC-24EF, MPC-32, MPC-68FF, HI-STORM 100S, HI-STORM 100A and HI-STORM 100SA. Minor modifications were made in the calculations for the design of the HI-TRAC transfer cask to reflect an increased temperature gradient which had insignificant impacts on previously calculated safety factors. The review was conducted 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).

The following structural evaluations were reviewed by the staff:

- For the MPC-24E, which is a variant of the already approved MPC-24 for higher enriched fuel, the structural changes in the basket assembly and the resulting modified loads on the canister shell have been analyzed in the same manner as the MPC-24 and the resulting stresses and safety factors computed and provided.

- For the MPC-24EF, which is a variant of the MPC-24E to accommodate damaged fuel or fuel debris, there are no structural changes to evaluate since the MPC-24EF is identical to the MPC-24E except for the thickened top portion of the MPC shell in the lid closure region. The thickened closure lid region modification has been previously evaluated and approved in accordance with 10 CFR Part 72 as part of the HI-STAR 100 System under Certificate of Compliance 1008.

- The MPC-32 is a design to accommodate one-third more PWR spent fuel assemblies than the MPC-24 series. From structural considerations, the MPC-32 is similar to the previously approved MPC-68. The structural analysis methodology and design approach used for the MPC-32 were conducted in the same manner as was done for the previously approved MPC-24 and MPC-68 for the HI-STORM 100 System. The resulting stresses and safety factors have been computed and provided.

- The new MPC-68FF is a variant of the MPC-68F that was previously approved for use under 10 CFR Part 72. The MPC-68FF is designed to allow the storage of BWR fuel debris in a dry storage mode. The variation of the design is limited to the thickening of the MPC shell in the closure lid region, a design detail that has been previously approved as part of the HI-STAR 100 System under Certificate of Compliance 1008.

- The 100S overpack/cask component for the HI-STORM 100 System is designed as a slightly shortened and lighter modification of the standard HI-STORM 100 overpack/cask which was previously approved as part of the original components of the HI-STORM 100 System under Certificate of Compliance 1014. The structural analyses for most design basis conditions for the 100S were bounded by the previously approved analyses. When this was not the case, new analyses were performed using the same methodology that had been used for the standard HI-STORM 100 overpack/cask. The major changes for the 100S relate to the lid and closure region details to the shell body of the cask related to the fact that the upper ventilation duct openings are part of the lid assembly instead of being in the shell body.

- The 100A overpack/cask component for the HI-STORM 100 System is designed for use at independent spent fuel storage installations (ISFSIs) as an anchored storage system where the ISFSI is located in an area considered to be a high seismic area. This is an area where the zero period peak horizontal ground acceleration and the associated zero period vertical acceleration at the top of the supporting base mat/support pad are of such a magnitude as to just prevent incipient horizontal sliding based on the frictional static limit defined in the FSAR. The major differences in the 100A when compared to the 100 is that overpack baseplate is a larger diameter and is penetrated by bolt holes adjacent to the circumference and there are stiffeners between the cylindrical shell, the upper stiffener ring and the baseplate at regular intervals around the overpack.

- The 100SA overpack/cask component for the HI-STORM 100 System is the composite of the 100S and the 100A models that will allow for the use of the shortened overpack/cask in a high seismic region.

- The ISFSI pad/basemat parameters and the subgrade parameters previously required to ensure proper behavior are being revised by the amendment included in this evaluation. The purpose of the revision is to allow for a broader range of usage but still control the design of the ISFSI pad so that it will provide for pad performance under all design basis cask drop events and non-mechanistic tipover events such that the deceleration value imposed on the storage cask system does not exceed the critical design deceleration value for the system.

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

The HI-STORM 100 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 safety evaluation report addresses only the addition of different applications and configurations/models of components in two of those categories : the MPCs and the storage overpacks/casks. The revisions contained in the amendment relative to the structural aspects of the HI-TRAC transfer cask were limited to insignificant changes to the safety factors previously identified.

3.1.1 Structural Design Features

3.1.1.1 Multi-Purpose Dry Storage Canisters: MPC-24E, -24EF, -32, and -68FF

The multi-purpose dry storage canisters are designed and fabricated as all-welded stainless steel cylindrical pressure vessels. 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 in the HI-STORM 100 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.

3.1.1.2 Overpacks/Casks: 100S, 100A, and 100SA

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 height of the 100A is approximately 19 feet 10 inches" while the 100S and 100SA heights are approximately 19 feet 3 inches. The cylindrical wall thickness of the 100 series is 29-1/2 inches of steel and concrete. The base of the 100A overpack/cask is approximately 24 inches of steel and concrete and for the 100S and 100SA the thickness is 17 inches of steel and concrete with more of the total being steel when compared to the 100A. The lid thickness for the 100A is approximately 15-3/4 inches and for the 100S and 100SA the lid thickness is approximately 14 inches. The HI-STORM fabrication is simply described as two steel cylindrical cans open on one end. The smaller can is placed inside the larger can with four radial plates welded the full height of the overpack to structurally integrate the two cylinders. The radial plates are also welded to the baseplate (to the horizontal plates of the air inlet ducts for the 100S) and maintain a space in-between that is then filled with concrete. Welded to the top of each radial plate is a threaded anchor block that can be used for lifting the cask. A closure lid with concrete filling the void spaces between the hollow shell steel portions completes the fabrication of the two main elements of the overpack. The inside diameter of the overpack is several inches larger than the outside of the MPC, creating an annular internal void space. A portion of the space is then taken up by steel channels aligned parallel to the axis of the overpack and attached to the inner shell that serve as guide channels during MPC insertion and removal. In addition, the channels serve as a flexible energy dissipater for MPC movement during any impactive lateral loads.

These channels also allow for air movement and cooling since the overpacks are not closed vessels even though there is a lid. There are four air inlets at the bottom of the overpack that open into the annular volume containing the guide channels and there are also four air outlets at the top of the overpack. These allow for natural draft cooling of the exterior surface of the MPC inside the overpack.

3.1.2 Structural Design Criteria

The structural design criteria for the HI-STORM 100 System have been evaluated previously and the results of that safety evaluation are included with the CoC 1014. The structural design criteria for the HI-STORM 100 System are summarized in Tables 2.01, 2.02 and 2.03 of the FSAR. Included in this safety evaluation report are those elements of the structural design criteria that are modified or added based on the new models of MPCs and overpacks/casks that are provided for in this amendment request.

3.1.2.1 Criteria for Multi-Purpose Dry Storage Canisters

The amendment revises the MPC design accident internal pressure from 125 psig to 200 psig as noted in Table 2.2.1 of the FSAR.

3.1.2.2 Criteria for Overpacks/Casks 100A and 100SA

The structural modifications to the standard HI-STORM 100 made to implement HI-STORM 100S and 100A design changes, meet the same criteria as provided in the HI-STORM 100. For example, the additional steel structural elements added to the HI-STORM 100 to allow for positive anchorage to an ISFSI pad/basemat are designed and fabricated in accordance with the requirements of the ASME Code, Section III, Subsection NF[1] for Class 3 plate and shell components.

3.1.2.3 Criteria for the ISFSI Pad/Basemat

The HI-STORM 100 System can be utilized where the loaded overpacks/casks are free-standing individual structures on the ISFSI pad/basemat, or where the loaded overpacks/casks are structurally anchored to the ISFSI pad/basemat in which case only the HI-STORM 100A or 100SA can be used. In the free-standing condition of the HI-STORM 100 System the ISFSI pad/basemat is considered as not important-to-safety, whereas in the anchored condition the ISFSI pad/basemat is considered important-to-safety. For the anchored condition, all of the anchorage hardware between the 100A or 100SA overpack/cask and the pad/basemat is also considered important-to-safety. Whether or not the overpack/cask must be anchored to the ISFSI pad/basemat depends on whether the zero period acceleration at the surface of the pad/basemat exceeds the limit for a free-standing HI-STORM 100 or 100S.

While the ISFSI pad/basemat for the free-standing overpack/cask (the HI-STORM 100 or 100S) is not classified as important-to-safety, there are necessary design criteria that must be met. This is due to the fact that the behavior of the pad/basemat has a direct influence on the loadings imposed on the overpack/cask under certain conditions. Table 2.0.5 of the FSAR identifies the requirements for the ISFSI pad when the overpack/cask is free-standing and the table contains a comparison to the requirements when the overpacks/casks are to be anchored.

The limitations on the pad/basemat design parameters must be such that the pad has sufficient structural stiffness to meet the strength limits set forth in the specific reinforced concrete code requirements, but at the same time be sufficiently compliant or flexible so that for the controlling or maximum load conditions the pad imposes a maximum deceleration that is less than that allowed on the overpack/cask and the MPC and contents inside.

The bases of this design criterion are the results of an in-depth investigation performed by Lawrence Livermore National Laboratory (LLNL) for the NRC and the information reported in NUREG/CR-6608.[2] The single relevant parameter identified in the LLNL study is that the performance of an ISFSI pad and the surrounding supporting soil media, as a system, not impose a deceleration value under all design basis drop and non-mechanistic tipover events that exceeds the critical design deceleration value of the cask system. This critical design deceleration limit for the HI-STORM 100 System has been previously determined to be 45-g at the top of the basket inside the MPCs. This limit has been evaluated previously and is identified in the existing Certificate of Compliance 1014. The study by LLNL established a set of acceptable ISFSI pad design parameters based on a specific methodology defined in NUREG/CR-6608. In general, based on the LLNL results and using the prescribed methodology, there are a number of combinations of the identified controlling parameters that could constitute an acceptable set so as to maintain the drop or tipover deceleration results that are less than can be tolerated by the cask system. The LLNL analytical work developed one such set of parameters that are identified in Table 2.2.9 of the FSAR as Set A. These define the concrete thickness in the ISFSI pad, the concrete compressive strength, the strength of the reinforcing steel bars which are placed at the top and the bottom of the pad and run in both the length and width directions of the pad. The subgrade effective modulus of elasticity beneath the pad is determined on the basis of prescribed testing and evaluation.

In order to provide users of the HI-STORM 100 System an additional range of applicability for the system, a second set of values of controlling parameters was added as Set B to Table 2.2.9 of the FSAR. These parameter values define a pad which is thinner, with higher concrete strength, supported by a subgrade with a lower modulus of elasticity, than is associated with Set A of Table 2.2.9. If neither of the sets provided in Table 2.2.9 are sufficient for a particular ISFSI site and the specific design of the ISFSI subgrade and pad, the user of the HI-STORM 100 System may base the design for their site on any combination of the design parameters resulting in a structurally sufficient reinforced concrete pad that meets the provisions of ACI 318-95[3] as a minimum, and also limits the deceleration of the cask to less than or equal to 45-g for the design basis drop and tipover events for the HI-STORM System. An additional condition is that the structural analyses for a site-specific set of values for the ISFSI pad design other than those of Table 2.2.9, shall be performed using methodologies consistent with those described in the FSAR, as applicable.

The structural design criteria for the HI-STORM 100A (or 100SA) ISFSI pad/basemat are provided in Appendix 2A of the FSAR. Since this element is necessary for the utilization of this model of the HI-STORM 100 System, the licensee/owner/user must assure that these criteria are met. There are ten (10) distinct general requirements set forth in Appendix 2A and in addition, there are requirements identified in Table 2.0.4 and Table 2.0.5 of the FSAR.

The design of the reinforced concrete of the pad is to meet the requirements of ACI 349-97[4] as noted in Section 2.A.2, Item 1, of Appendix 2A of the FSAR, with factored load combinations as provided in NUREG 1536 and ACI 349-85.[5] Also noted in that section of the FSAR are references to use ACI 360R-92,[6] ACI 302.1R-96,[7] and ACI 224R-90[8] as applicable, in the design and construction of the ISFSI pad. In this context, since these are not codes but serve as guidance, current practice or state of the art documents, it is necessary that the design and construction specifications for the important to safety reinforced concrete pad be developed by the user of the HI-STORM 100A or 100SA so that any necessary provisions from these referenced documents are integrated into the controlling specification. This is considered to be in the scope of the responsibility of the user of the HI-STORM 100A or 100SA since all the requirements of the FSAR for the anchored cask concept are to be met as noted in Section 2.0.4.1 of the FSAR.

For the anchored cask there are several other key elements that must function properly if the cask system is to perform in a predictable and acceptable manner. First, the anchorage provisions of the cask located around the periphery at the flanged and reinforced base change the loading and the resulting stresses on the cask body. The design criteria for this element were addressed in Section 3.1.2.2 above. Second, the design and performance of the anchor studs must be provided for and third, the anchorage or embedment to which the anchor studs are attached must be adequate. Last, the transfer of loads from the embedment to the reinforced concrete pad must be adequate.

The anchor studs are to be designed under the criteria of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF and Appendix F as specified in the FSAR in Section 2.0.4.2.c for the listed loads and load combinations.

The steel embedment and its anchorage to the reinforced concrete pad is described in Appendix 2A, Section 2.A.2, Item 5, to be required to follow the provisions of ACI 349-97, Appendix B, “Steel Embedment,” and the associated Commentary on Appendix B, as applicable. The stated design criterion is that the “anchorage design shall be controlled by the strength criterion for the embedment steel.”

It is noted that the NRC has identified some potential non-conservative conditions that may arise from the direct use of ACI 349-97, Appendix B, in structures other than ISFSI pads. Thus, when a user of the HI-STORM 100A or 100SA prepares a site specific evaluation under 10 CFR 72.212(b)(2) the basis of a specific anchorage embedment and the associated reinforcing steel should consider the NRC concerns regarding edge distance, multiple anchors, etc. Figure 2.A.1 and Table 2.A.1 are presented as providing a typical anchorage arrangement for an anchored HI-STORM 100 System utilizing the ASME Code, Section III, Appendix F, for the design of the anchor studs and the actual concrete embedment designed in accordance with Section III, Subsection NF. For the purposes of this evaluation the details provided are considered to be conceptual details only, with any actual use of such an anchorage embedment being acceptable only after the user has completed the site specific custom engineering noted as being required in Section 2.0.4.4. It is noted that for such embedments exposed to the outdoor environment there will be a need to address corrosion effects over the expected lifespan of an ISFSI. With regard to the limits of usage for the anchored system, Table 2.0.5 defines the maximum seismic input level on the pad at the cask-pad contact surface to be less than or equal to 2.12-g for the zero period horizontal acceleration and less than or equal to 1.5-g for the zero period vertical acceleration. Either value may control the site at which the anchored cask may be used.

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 associated with this amendment, or identifies individual load limits that must be considered by the user and developer of an ISFSI. Only those identifiable new load values are discussed in this evaluation since all other individual loads have been discussed in conjunction with the issuance of the original CoC 1014

The design accident internal pressure for the MPCs increased to a value of 200 psig as noted in Table 2.2.1 of the FSAR.

For the HI-STORM 100A (or 100SA), the maximum overturning moment capacity under accident conditions is defined as 18.7E6 ft-lbs. This limit must not be exceeded by any applied loads resulting from wind, tornado, flowing water from a hydraulic flood condition, or from the impact of any of the spectra of missiles.

Also for the new anchored cask application there are defined permissible values for the zero period horizontal and vertical accelerations at the top surface of the ISFSI pad.

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 1014. 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.

Load combinations for the important-to-safety components that are necessary for the satisfactory performance of the anchored cask configuration that is part of the amendment are provided in Sections 2.0.4.2.b and 2.0.4.2.c of the FSAR for the reinforced concrete ISFSI pad and the steel anchor studs, respectively. The load combinations for the reinforced concrete pad are consistent with those of NUREG-1536.[9] The load combinations for the anchor studs that are governed by the ASME Code, Section III, Division 1, Subsection NF are provided in tabular form in Section 2.0.4.2.c of the FSAR addressing the necessary conditions.

3.1.2.6 Allowable Stresses or Required Strength

Stress allowables 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 3.1.17 of the FSAR. In addition, tabular data in Section 2.0.4.2.c of the FSAR addresses the anchorage stud stresses for the ISFSIs where the anchored casks are utilized.

The user of the anchored cask is responsible for a site specific analysis for the applicability of the HI-STORM 100A (or 100SA) to a particular site. For such cases, the reinforced concrete pad that is important-to-safety since it is for an ISFSI using anchored casks, the allowable stresses and required strength capacity are to be consistent with ACI 349-97 as provided for in

Section 2.0.4.2 and Appendix 2A of the FSAR, however it is noted that a later Code edition may be used provided that a written reconciliation is performed. Certain aspects of the design methodology for embedments for certain specific design configurations that could be designed under that edition of the Code have been identified by the NRC[10] and in sponsored research as potentially non-conservative.[11, 12] Therefore, the owner/user of an ISFSI using the anchored casks will be expected to evaluate the impact of these potential non-conservatisms in the design of the embedded anchorage system in an ISFSI pad using the referenced Code.

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 System in the various combinations that will be used in spent fuel storage operations. Tables 3.2.1 through 3.2.4 have been revised to reflect the new components in the HI-STORM 100 System. Bounding weights previously used encompassed most of the changes resulting from the amendment, however some bounding values for the loaded transfer cask operational values were increased. For example, the 100S overpack lid weight increased by 2500 lbs. (approximately 11 percent) over the 100 overpack lid, but when taken with the respective cask body, the original total bounding value for the 100 overpack is still bounding. For the 125-ton HI-TRAC transfer cask with the transfer lid and a loaded MPC-24 there was a 2,000 lb. increase representing less than a 1 percent increase in the bounding weight.

Table 3.2.3 provides the data relative to the centers of gravity of the HI-STORM 100 System in the various configurations of the various models and components. No significant changes have occurred as a result of the amendment.

3.3 Structural Materials

3.3.1 Concrete

As a result of modifications to the HI-STORM 100 overpack to create the HI-STORM 100S overpack proposed in this amendment, an increase in the density of the plain concrete used to fill the inner volume created between the inner and outer shell walls of the overpack is provided for. Table 1.D.1 has been revised to reflect the requirement that this concrete have a density of at least 155 lbs/cubic foot compared to the 146 lbs/cubic foot used in other volumes within the cask .

3.3.2 Bolting Materials

The amendment provides new information on the materials that can be used as the anchor studs for the proposed 100A (or 100SA) overpack anchorage system. Table 1.2.7 provides a list of acceptable materials that can be used to meet the design parameters also listed in the FSAR but, in Table 2.0.4. These materials represent ASME Boiler and Pressure Vessel Code materials, identified in Section II, Part A (Material Specifications) and Part D (Properties), as bolting materials for ASME Section III, Class 2 and 3 materials. Table 1.2.7 also provides the yield and ultimate strength for each of the listed materials. As a footnote to Table 1.2.7, other bolting materials listed in the ASME Code may be used if they meet the size requirements, the minimum requirements on yield and ultimate strength in accordance with Table 2.0.4, and they are suitable for the environmental conditions.

3.4 Structural Analysis of HI-STORM 100 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.

For the MPC-24E the analysis methods were the same as had been used for the previously approved MPC-24 and MPC-68 associated with CoC 1008 (HI-STAR 100 System), but a separate finite element model was generated with which the analysis was performed. For the purposes of the structural evaluation, the MPC-24EF is no different than the MPC-24E, therefore, a separate structural evaluation was not necessary. The MPC-32 was also analyzed using the same methods used previously for the MPC-24 and MPC-68 and was in fact actually an original part of the HI-STAR 100 System when the system was designed, but was removed from the HI-STAR System because of some non-structural issues. Since the loaded MPC-32 is the heaviest MPC, the design aspects of the MPC that are governed by the weight were and remain the bounding load condition for the family of various MPC models since their original design was based on the MPC-32. The MPC-68FF presents no change regarding the structural evaluation from that completed previously for the MPC-68. There were no changes in the safety factors, which all exceed 1.0, listed in Section 3.4.3.6 of the FSAR from the analyses completed for the lifting of the MPCs. Tables 8.1.1 through 8.1.4 provide the handling weight for the HI-STORM 100 System components. For the design pressures under normal and off-normal conditions there were no changes as shown in Table 2.2.1 for any of the system components.

As shown in Table 2.2.3 of the FSAR, the changes in normal and off-normal design temperatures resulting from the amendment produced a slight decrease in the temperatures used in the structural evaluation. Section 3.4.4.2.1 of the FSAR that addresses the gaps that remain between the overpack or the transfer cask and the MPC, and between the MPC and the fuel basket under the normal condition of the hot environment as analyzed under the methodology of Appendix 3.I was revised under the amendment. The appendices addressing the thermal expansion of the components were revised and included Appendices 3.U, 3.V, 3.W, 3.AF, and 3.AQ. The summary results shown in tabular form in Section 3.4.4.2.1 indicate the previous worst cases have a resulting smaller final clearance, but it is a positive gap for free expansion of the MPC. The MPC-68, including the new MPCs introduced with this amendment, remains as the canister with the maximum through wall thermal gradients and therefore the previous analyses remain as producing the controlling design condition where thermal stress conditions apply. With regard to the changes in the safety factors with respect to the amendment, the MPC shell’s safety factor under the design internal pressure and design operating temperature was reduced from 2.0 to 1.4 as noted in Table 3.4.4 of the FSAR. The staff concludes that this remains an acceptable margin since it is greater than 1.0.

In the case of the new variants for the overpacks, the analysis methods used for the 100S were the same as those used for the basic HI-STORM 100 with revisions to the calculations as necessary to reflect the physical differences in dimensions or configuration. The analytical models for the 100S used in the analyses for the various lifting scenarios that consider dead load and handling loads reflected the changes in the upper portion of the shell, the new heavier 100S lid and the radial ribs and anchor block that are configured differently than in the HI-STORM 100. The details of the analyses and the results are in the amended Appendix 3.D with the summary of the key results included in tabular form in Section 3.4.3.5 of the FSAR which has been revised to include the 100S. All the key safety factors are greater than 1.0. Under normal and off-normal conditions the 100A and the original 100 model will behave the same.

Individual lid lift results were revised on the basis that the new 100S lid is the heaviest and is therefore a bounding case. As shown in revised Appendix 3.AC and summarized in tabular form in Section 3.4.3.7 of the FSAR, the safety factor for the lift of the 100S lid is 2.46 and is 2.73 for the original HI-STORM 100 lid, with both being fully adequate since they are greater than 1.0. Another lid lift load case for the HI-TRACs as summarized in Section 3.4.3.9 of the FSAR resulted in revised safety factors. The staff concludes that this is acceptable.

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.

3.4.2.1 Seismic Event

The same analysis for free-standing casks completed for the HI-STORM 100 with MPC-24 and MPC-68 was performed for the 100S as presented in the amended Section 3.4.7.1 of the FSAR. The combinations of MPC-24E and MPC-32 with the HI-STORM 100 and the HI-STORM 100S were also analyzed as part of the amendment. This showed that the MPC-68 inside the HI-STORM 100 overpack was still the controlling case to assure the “no overturning condition” under a seismic event. The sliding analysis was re-examined for the 100S and found to still be applicable with the same coefficient of friction (0.53) and the same governing static equilibrium conditions to insure against incipient sliding that are represented in equation form in Table 2.2-8 of the FSAR. An additional set of design basis earthquake accelerations was developed as a representative combination with this amendment and also meets the criteria to prevent incipient sliding. This set adds to the three representative combinations originally provided in the FSAR. This combination is represented by zero period accelerations of an equivalent horizontal g-level vectorial sum of each of two orthogonal directions and the corresponding vertical (upward) g-level. The values in the acceptable combination are 0.35g and 0.34g, respectively. The horizontal vector sum is equivalent to 0.25g in each of the orthogonal horizontal directions. Therefore, the design basis earthquake zero period acceleration input at the top of the ISFSI pad for free-standing casks should satisfy the static equilibrium conditions to prevent incipient sliding, which can be satisfied, for example, by the current four representative sets of accelerations, or by a set developed for the site.

The anchored HI-STORM 100 System, HI-STORM 100A, is designed under seismic conditions to have sufficient pretensioning in the anchor studs so as to preclude any sliding between the base plate of the cask and the ISFSI pad top concrete surface. This type of behavior will prevent shear and bending from being induced into the studs. Under seismic loading, the stresses within the pretensioned studs will vary with the location around the circle of studs and as the uplift due to vertical acceleration and the uplift resulting from the seismic overturning moment serve to increase the anchor steel stresses. The repeated cyclic loading from such an event produces conditions that must be checked against fatigue failure criteria. Additionally, the loads that are transmitted to the ISFSI pad must be determined. The basic design for the 100A is founded on the three orthogonal zero period accelerations at the top of the ISFSI pad being taken as 1.5g. The stud size, stud preload, number, spacing and the stud's unbonded or free length were specified. Two analysis methods were used: one quasi-static using a dynamic load factor of 2.0 and the other a dynamic method using Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants,"[13] response spectra as well as a second set of spectra representing a typical western U.S. site. Section 3.4.7.3 of the FSAR provides the details of these two approaches and the methodology that can be used if a site specific analysis is to be performed.

For the selected design conditions with the quasi-static approach, loads in the studs were developed that were then used in a 3D finite element analysis to define the stress conditions in the base of the 100A unit. Design details such as stiffeners and gusset plates and individual welds were then defined based on the computed values.

The dynamic analyses were completed which allowed for the description of the cyclic loading with respect to time for the anchor studs. This allowed an evaluation of the fatigue issue for the anchor studs. For the conditions studied it was concluded that for a single seismic event fitting the Regulatory Guide 1.60 criteria a fatigue failure was very unlikely. It was concluded that the quasi-static approach coupled with a dynamic load factor could reasonably establish the safety factors for the anchored cask. The staff agrees that for the analyzed conditions, a bounding set of ISFSI pad interface loads were developed that could be used by the ISFSI pad designer or the method used could be applied by the ISFSI pad designer for a site specific design.

3.4.2.2 Tornado Wind and Missile Loading

The HI-STORM 100S is a free-standing cask that is shorter than the original 100 model, therefore, lateral loading produces a lesser effect on the shorter unit. As such, the staff agrees that the original design and evaluation for the HI-STORM 100 as provided in Section 3.4.8 of the FSAR remains valid for the 100S for the design conditions addressed in the original design evaluation.

The staff also agrees that the anchored HI-STORM 100A, which has an increased capability for lateral load and the same resistance to missile penetration, the original evaluation provided for the HI-STORM 100 is bounding.

3.4.2.3 Flood Loading

For the HI-STORM 100S the same analysis used for the original HI-STORM 100 demonstrated that the safety factor against sliding is 1.98 compared to 2.09 for the HI-STORM 100, but both are above the design limit of 1.1. The safety factor against overturning of approximately 4.67 for the HI-STORM 100 was only slightly changed for the 100S since the reduced height of the 100S and the lower center of gravity taken together changed the overturning moment from flooding very little. The staff agrees that the analysis and evaluation for the HI-STORM 100 provided in Section 3.4.6 of the FSAR and the comparisons to the 100S, demonstrate that the HI-STORM 100 can be considered as the bounding case.

For the anchored 100A, the lateral effects of flooding are bounded by the lateral seismic loading capability to resist a lateral load of greater than 1.5g which greatly exceeds the lateral flood loading by at least a factor of 10. The resistance to overturning for the 100A exceeds the flood overturning load by greater than ten since the moment arm is greater on the 100A because of the larger base plate. The staff agrees that the HI-STORM 100A is adequate to resist the design flooding.

3.4.2.4 Tipover

The tipover analysis for the HI-STORM 100 provided in Section 3.4.10 and Appendix 3.A of the FSAR for two defined targets with specified physical properties has been completed for the HI-STORM 100 using a dynamic 3D finite element software package. The original FSAR provided only one set of target parameters in order for the imposed deceleration on the cask to be less than 45g. As noted herein in Section 3.1.2.3, the amendment revises the set of influencing parameters and provides target parameters for two sets. The revised calculation methodology can now be used to identify acceptable site conditions or evaluate site conditions if the values of the controlling parameter are outside the range of the two sets that are provided in the amended FSAR. As demonstrated in Section 3.4.10 of the amended FSAR, by the physical parameters the 100S model will have a deceleration value less than (approximately 95 percent of) the deceleration determined previously for the HI-STORM 100. Therefore, the staff agrees that the HI-STORM 100 results can be considered as bounding.

Tipover of the 100A is precluded by the design requirement that it is an anchored cask, but under a non-mechanistic tipover the HI-STORM 100 results would be considered to be bounding.

3.4.2.5 Accidental Drop

Based on the impact pad parameters as discussed in Section 3.1.2.3 herein, the 11-inch height of drop for the two pad parameter sets will limit the cask deceleration to less than 45g for the HI-STORM 100, 100S and 100A for the two sets of parameters provided in the amended FSAR. The staff agrees that the results for the HI-STORM 100 are considered bounding for the HI-STORM 100S and the 100A as addressed in amended Appendices 3.A and 3.M.

3.4.2.6 Fire

No impact on components that are part of the amendment, beyond the considerations made in the original design evaluation for the HI-STORM 100, require to be addressed in Section 3.4.4.2.2 of the FSAR.

3.4.2.7 Explosion

For the HI-STORM 100S the results from the analysis for the original HI-STORM 100 as demonstrated in the original case in Section 3.4.7.2 of the FSAR are bounding based on the smaller height and lesser weight of the 100S cask. The steady state design accident pressure differential of 5 psi for the HI-STORM 100 and the transient accident differential pressure of 10 psi in 1 second can be sustained by the 100S. The staff agrees that the results for the HI-STORM 100 are considered as bounding.

For the anchored cask, HI-STORM 100A, a comparison of the lateral load created by the 5 psi steady state differential pressure and 10 psi differential pressure in the transient state across the cask and the lateral load and overturning moment capacity shows the anchored cask will withstand the design basis explosion.

3.5 Structural Analysis of the HI-TRAC Transfer Cask

3.5.1 Lifting Devices

Revisions to the summary table from the lift analyses of the 125-ton HI-TRAC trunnion region of the shell resulted in small increases in the safety factors shown in Section 3.4.3.3 of the FSAR. The same is true for the 100-ton HI-TRAC in the shell region near the trunnion. A lid lift load case for the HI-TRACs as summarized in Section 3.4.3.9 of the FSAR resulted in revised safety factors, however all remain above 1.0 which is acceptable.

3.5.2 Differential Thermal Expansion

Minor revisions were made in Appendix 3.I due to changes in temperatures, however there was an insignificant change in the resulting dimensional changes.

3.6 Evaluation Findings

Based on the information provided in the amendment to the FSAR, by reference and the supporting documentation, the staff concludes that the HI-STORM 100 System, as amended, meets the acceptance criteria specified in NUREG-1536 and the regulations.

F3.1 The FSAR adequately describes all 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 The SSCs 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 Chapter 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.

3.7 References

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section II and Section III, Division 1, 1995 with Addenda through 1997.

2. NUREG /CR-6608, “Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads,” Lawrence Livermore National Laboratory, Feb 1998.

3. American Concrete Institute, ACI 318-95, “Structural Concrete Building Code.”

4. American Concrete Institute, ACI 349-97, “Code Requirements for Nuclear Safety Related Concrete Structures.”

5. American Concrete Institute, ACI 349-85, “Code Requirements for Nuclear Safety Related Concrete Structures.”

6. American Concrete Institute, ACI 360R-92, “Design of Slabs on Grade.”

7. American Concrete Institute, ACI 302.1R-96, “Guide for Concrete Floor and Slab Construction.”

8. American Concrete Institute, ACI 224R-90, “Control of Cracking in Concrete Structures.”

9. NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” U.S. Nuclear Regulatory Commission, January 1997.

10. NUREG-1503, “Final Safety Evaluation Report for the GE ABWR,” July 1994.

11. NUREG/CR-5434, “Anchor Bolt Behavior and Strength During Earthquakes,” August 1998.

12. NUREG/CR-5563, “A Technical Basis for Revision to Anchorage Criteria,” March 1999.

13. Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” October 1973.

4.0 THERMAL

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.

4.1 Spent Fuel Cladding

A general outline of the thermal criteria and analytical methods are addressed in the HI-STORM 100 FSAR. Design specific calculations are presented in the HI-STORM FSAR appendices.

4.1.1 Methodology for Calculating Maximum Allowable Cladding Temperature Limits

The current revision of NUREG-1536 for storage of spent fuel in dry storage casks essentially limits fuel burnup to 45 GWd/MTU. The NRC has issued, Interim Staff Guidance (ISG) - 15, “Materials Evaluation,”[1] to evaluate applicant’s FSAR for storing spent fuel that have burn up fuel greater than 45 GWd/MTU. ISG-15 requires the applicant to limit the initial cladding temperature during dry storage such that the overall strain due to creep during storage does not reach 1 percent, since failure was conservatively postulated to occur at this creep strain. ISG-15 does not prescribe a mathematical model to compute the creep rate; it is left to the applicant to demonstrate this creep strain limit using their own conservative and supportable methodology. ISG-15 also provides a set of fuel integrity criteria predicated on the extent of corrosion (oxidation) of the fuel cladding to define when high burn up spent fuel should be treated as damaged fuel.

The thermal design criteria for preventing fuel cladding degradation are presented in Chapter 4, Appendix 4A of the FSAR, Report Number HI-2002407, and Appendix H of the applicant’s calculation package. For the HI-STORM 100 dry cask storage system, a creep methodology was developed to calculate cladding temperature limits for PWR and BWR fuels having burnups up to 68,200 MWd/MTU and 59,900 MWd/MTU, respectively, for assumed storage times of up to 40 years. The applicant has proposed an alternate criterion for categorizing spent fuel as damaged by considering a corrosion reserve and accounting for it in stress calculations in lieu of the existing criterion outlined in ISG-15.

The applicant used a creep correlation to calculate the allowable fuel cladding temperature limits for the HI-STORM 100 cask with the maximum permissible accumulated creep strain as 1 percent, which is consistent with the ISG-15 guidance.

4.1.2 Creep Correlation (Creep Model)

The applicant used an empirical creep correlation. The correlation used does not depend on mechanical properties or any operating mechanisms that contribute to creep deformation. The applicant’s correlation was developed from available literature[2] containing creep data (i.e., creep strain versus time) for irradiated Zircaloy-4 cladding materials tested at temperatures and stresses ranging from 300-400 °C and 70-630 MPa, respectively, and test duration up to 8000 hours.

The applicant's correlation is based strictly on a creep strain relationship that includes the spent fuel cladding temperature and hoop stress. The maximum permissible accumulated creep strain is set at 1 percent and has been used to compute the peak cladding temperature (PCT) limits for spent fuel. The creep strain correlation is as follows:

$$\varepsilon = \alpha \exp\left(\frac{-\zeta}{RT}\right)\sinh\left(\gamma\sigma\right)\tau^{\beta}, \qquad (4.1)$$

where ε is the strain at time τ, α, ζ, γ, and β are creep constants with values selected suitably to bound all relevant creep data of irradiated cladding, R is the universal gas constant, T the temperature and σ is the cladding stress. The form of the proposed correlation is consistent with the classical metal creep equations wherein the two principal variables, stress and temperature, respectively, bear an exponential and Arrhenius-type relationship to model creep strain accumulation. The correlation in Equation 4.1 is valid for the primary creep and is used by the applicant to calculate the cumulative creep strains versus time for strains up to 0.5 percent. The onset of secondary creep is assumed to occur when the total primary creep reaches 0.5 percent. For strains greater than 0.5 percent strain, incremental cumulative creep strains are calculated by differentiation of Equation 4.1 and application of the trapezoidal rule. The transition strain value of 0.5 percent from primary to secondary creep is consistent with literature data.[3,4,5]

The staff verified that the applicant's calculation of maximum creep strains bounds all of the data identified in the application as well as other publicly available data.

4.1.3 Hoop Stress for Calculating Maximum Temperature Limits

To determine the stress in the fuel cladding at the beginning of storage, the applicant assumed an initial rod pressure of 2,000 psi and 1,000 psi, respectively, for PWR and BWR fuel. Further, a cladding radius to thickness ratio, W, was assumed by the applicant to calculate the hoop stress in the cladding at the beginning of storage. Using Lame's formula, the applicant computed the product, the initial pressure and W, to obtain an initial stress of 144.7 MPa and 65.5 MPa, for PWR and BWR fuel, respectively. These stresses were used in the creep strain calculations to determine the maximum allowable initial cladding temperature for all fuel types regardless of dimensional differences.

The staff has concluded that the applicant has used adequate margin for the effects of the presence of oxide on the cladding stress calculations. The applicant has included the effects of cladding corrosion (oxidation) by considering two underlying facts, namely, (i) the collateral effect of cladding oxidation on its potential decrease in its resistance to creep deformation, and (ii) the increase in circumferential stress due to loss in cladding wall thickness.

For the PWR fuel cladding, for an internal pressure of 2,000 psi at the start of storage, a radius to thickness ratio of 10.5 was used to determine the cladding stress of 144.7 MPa. Staff has determined that this value corresponds to an oxide thickness of approximately 175 microns, as shown in Figure 4.1.

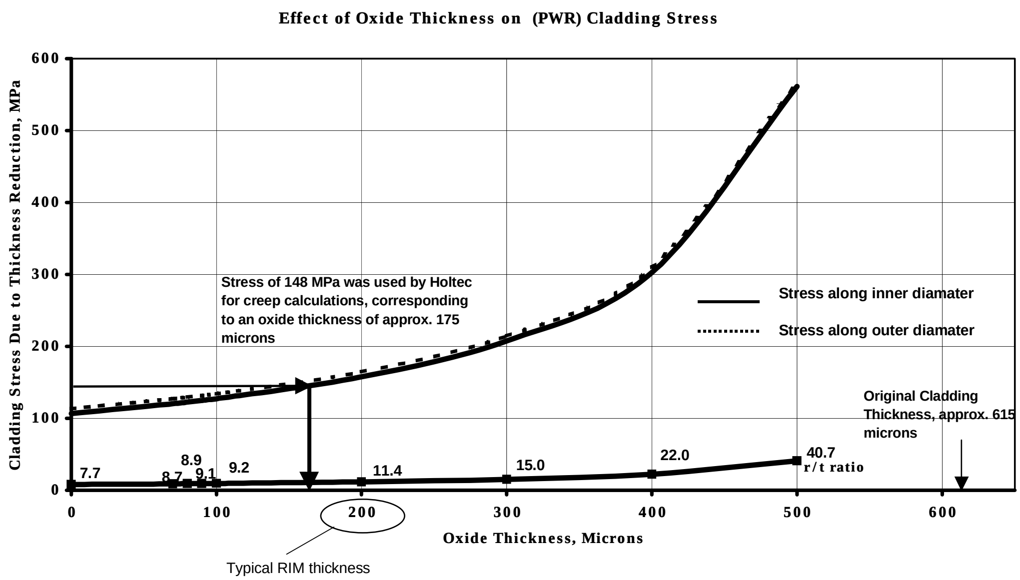

Figure 4.1 The effect of thickness reduction due to oxidation of the PWR cladding on the stress.

Thus, by eliminating the oxide outer layer, the applicant uses a higher stress value for the cladding, as if approximately 175 microns of oxide layer has been removed from the total metal thickness. Considering that the experimentally verified (Figure 4.A.13 in Appendix 4-A of the Holtec SAR) cladding oxide thickness varies between 100 to 140 microns at a burnup of 60 GWd/MTU, the staff has concluded that the applicant's approach has included reasonable conservatism in its analyses.

For the BWR fuel cladding, for an internal pressure of 1,000 psi at the start of storage, a radius to thickness ratio of 9.5 was used to determine the cladding stress of 65.5 MPa. Staff has determined that this value corresponds to an oxide thickness of approximately 80 microns, as shown in Figure 4.2. Thus, by eliminating the oxide outer layer, the applicant uses a higher stress value for the cladding, as if approximately 80 microns of oxide layer has been removed from the total metal thickness. Experimental data have not been provided for cladding oxide thickness as a function of burnup for the BWR fuel cladding. However, considering the less severe potential for oxidation in BWR, when compared to PWR, the staff has concluded that the applicant's approach has included reasonable conservatism in its analyses.

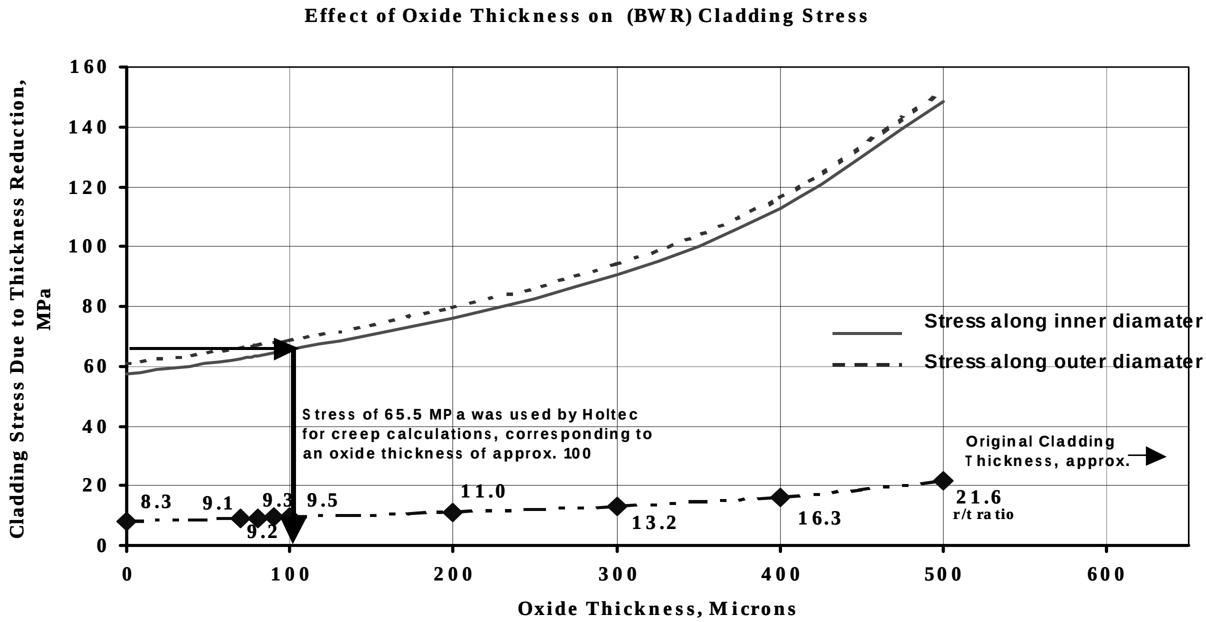

Figure 4.2 The effect of thickness reduction due to oxidation of the BWR cladding on the stress.

To determine the hoop stress with time, the applicant used discrete values of the heat generation rate to determine the cladding temperature (T) and gas temperature (θ). The applicant used the temperature curves for PWR and BWR fuel for the creep correlation from the thermal analysis. The θ curves provide the means to calculate the hoop stress as a function of the time coordinate. The applicant selected a bounding heat attenuation curve (70 GWd/MTU) for PCT calculations, which resulted in a conservative temperature decay relationship.

4.1.4 Corrosion allowance

ISG-15 suggests specific limitations on oxide thickness as a means of ensuring cladding integrity during dry storage. The provisions of ISG-15 related to the oxide limits are listed below.

Fuels having average assembly burnups exceeding 45,000 MWd/MTU can be safely stored if the following acceptance criteria are met:

I. A high burnup fuel assembly containing Zircaloy clad fuel may be treated as intact if both of the following conditions are met:

A1. No more than 1 percent of the rods in an assembly have peak cladding oxide thicknesses greater than 80 micrometers; and

A2. No more than 3 percent of the rods in an assembly have peak cladding oxide thicknesses greater than 70 micrometers.

II. A high burnup fuel assembly should be treated as potentially damaged fuel if either of the following conditions is met:

B1. The fuel assembly does not meet both criteria A1 and A2; or

B2. The fuel assembly contains fuel rods with oxide that has become detached or spalled from the cladding.

The limits on the oxide thickness are intended to assure that there is adequate strength and ductility in the cladding to maintain the fuel in the as-analyzed configuration under all storage conditions.

Rather than using the existing provisions outlined above, the applicant has proposed an alternate criterion for categorizing a spent fuel assembly as intact by considering the concept of a “corrosion reserve”. This criterion considers that a conservative thickness, or amount, of the cladding can be lost to corrosion while still maintaining the structural integrity of the cladding. The applicant calculated the available corrosion reserve, Δ, for each array/class of fuel assemblies to be stored in the HI-STORM cask. The underlying premise for the applicant to use the corrosion reserve concept is that the actual amounts of corrosion that can be lost due to corrosion of a particular fuel assembly array/class will not exceed the limiting value of stress that is used in the Holtec creep correlation and the calculation of the maximum allowable cladding temperature for PWR and BWR fuel. The values of Δ have been calculated using the following equation:

$$\Delta = \left(t_{nom} - \frac{0.5 x d_{nom} - t_{nom}}{W} \right) x 25{,}400$$

where:

Δ = the available corrosion reserve thickness (micrometers), W= the assumed maximum allowable fuel cladding inner radius-to-fuel cladding thickness ratio (10.5 or 9.5), t_{nom}= the nominal, pre-irradiated fuel cladding thickness, and d_{nom}= the nominal, pre-irradiated fuel cladding outer diameter.

Therefore, a high burnup fuel assembly is classified as an intact fuel assembly if the loss of cladding thickness due to fuel cladding oxidation must not increase the fuel cladding inner radius-to-cladding thickness ratio above 10.5 for PWR or 9.5 for BWR in FSAR Tables 4.A.4 and 4.A.5. The applicant has assumed a bounding value of W to be 10.5 for PWR and 9.5 for BWR fuel in its calculation of Δ for the fuel array/class in the HI-STORM. The staff has verified that the applicant’s assumed values of W (i.e., 10.5 and 9.5 for PWR and BWR, respectively) are bounding for all fuel types listed in Table 4.A.5 and 4.A.5 by computing the values of W for the fuel assembly array/class in Table 4.A.4 and 4.A.5 of Appendix 4.A. For example, staff calculated W for all fuel types and found that the values were lower than 10.5 and 9.5 for PWR and BWR fuel, respectively. The lower values of W were obtained by the staff from the fuel assembly arrays/classes provided in the tables. Lower values of W would in turn give lower values of hoop stresses than the bounding values used by the applicant for creep strain calculations. As recalled from above, the bounding values for the stresses were used in the applicant's creep correlation to determine the maximum allowable initial cladding temperature.

The staff acceptable the applicant's alternate criterion and the values given in Table 4.A.4 and 4.A.5 of Appendix 4.A for the available corrosion reserve limits for PWR and BWR cladding, respectively. If a high burnup fuel assembly does not meet the applicant's criterion, as described above, it shall be considered a damaged fuel assembly.

4.1.5 Maximum Allowable Temperature Limits

As described in Section 4.A.7 in the FSAR, the maximum allowable cladding temperature limits are calculated by assuming an initial value of the peak cladding temperature. From the temperature versus heat load curves of Figures 4.A.8 and 4.A.10 for PWR and BWR fuel, respectively, the maximum heat generation rate is determined. Based on the maximum heat generation rate, the internal fuel rod gas temperature is determined from Figures 4.A.8 and 4.A.10 for PWR and BWR fuel, respectively, and the maximum expected hoop stress in the cladding is calculated. With the maximum cladding temperature and stress defined, the applicant calculates the total accumulated creep strains for 40 years, as describe above. If the creep strain is greater than 1 percent, then the initial assumed value of the peak cladding temperature is appropriately adjusted. Using an iterative approach, the calculation of the maximum allowable temperature limit is performed until the total accumulated creep strain in 40 years is equals 1 percent with a tolerance of ± 0.001.

The staff verified that the applicant's temperature limit calculations are reproducible from the information contained in the FSAR for both PWR and BWR fuel for 5, 6, 7, 10, and 15 year old spent fuel. The staff verified that the total cumulative creep strain is less than or equal to 1 percent for all fuel types and storage periods up to 40 years.

Using the method described above, the applicant calculated long-term temperature limits for PWR and BWR fuel assuming 5, 6, 7,10, and 15 year cooling times for normal conditions of storage. To build in additional margin of conservatism in allowable heat load for the MPCs, the maximum allowable temperature limit is further reduced. The temperatures in Table 4-1 are the values calculated in the thermal analysis. These temperature limits are conservatively lower than the maximum allowable temperatures derived from the applicant's creep correlation.

Table 4-1
High Burnup Fuel Allowable Peak Clad Temperature Limits (from Thermal Analysis)

Fuel Age at Initial Loading	PWR Fuel Limit	BWR Fuel Limit
5 years	**359.7°C [679°F]**	**393.2°C [740°F]**
6 years	**348.7°C [660°F]**	**377.9°C [712°F]**
7 years	**335.0°C [635°F]**	**353.7°C [669°F]**
10 years	**327.2°C [621°F]**	**347.9°C [658°F]**
15 years	**321.9°C [611°F]**	**341.1°C [646°F]**

For the off-normal conditions of forced helium dehydration, the temperature limit of the fuel is maintained below the normal peak cladding temperature limits to inhibit major annealing of the cladding.

The applicant established a short-term temperature limit of 570°C (1058°F) for hypothetical accident conditions for Zircaloy clad fuel in accordance with the guidance of NUREG-1536 and PNL-4835.[6] The staff finds this short-term temperature limit acceptable.

4.1.6 Conclusions Related to the Spent Fuel Cladding Integrity

The staff has reviewed the Holtec's methodology and finds it acceptable for calculating maximum allowable cladding temperature limits for HI-STORM. Further, the staff agrees that the assumptions related to calculating cladding hoop stress are bounding because they account for a reduction in wall thickness and an increase in circumferential stress due to cladding oxidation. Likewise, a bounding rod pressure was assumed in the analysis. Holtec's creep methodology is consistent with the goals and objectives of NRC's Standard Review Plan, NUREG-1536, and ISG-15, and meets the requirements of 10 CFR Part 72.

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 storage system 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 pressurized water reactor (PWR) or 68 boiling water reactor (BWR) spent fuel assemblies. The canister design includes two overpack configurations indicated by the HI-STORM 100 and HI-STORM 100S nomenclature. Additionally, a HI-STORM 100A and 100SA overpack have been designed with additional anchorages for use in high seismic areas. Use of these additional anchorages do not affect thermal performance of the overpack or HI-STORM 100 System. 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 3.0 of this SER).

The MPCs share identical exterior dimensions, manufacturing requirements, and handling features but differ in their spent fuel arrangement details from the originally approved HI-STORM 100 system. In general, the material properties remained unchanged from the original design**.** The seven MPC models are all engineered as cylindrical prismatic structures with square cross section cavities which contain a varying number fuel assembly/damaged fuel assembly storage cells, 24, 32 or 68, designated as MPC-24, 24E, 24EF, 32, 68, 68F, and 68FF depending on the spent fuel to be housed. 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. Between the periphery of the basket, the MPC shell, and the basket supports, heat conduction elements are installed. These heat conduction elements are fabricated from thin aluminum alloy and are installed along the full length of the MPC basket except in the drain pipe location, to create a nonstructural thermal connection which facilitates heat transfer from the basket to the shell. The free volume of the MPC is inerted with pure helium gas under pressure.

The MPC-24 is designed to store up to twenty four (24) intact PWR fuel assemblies with or without non-fuel hardware. The MPC-24E is designed to store up to twenty four (24) total PWR fuel assemblies, with or without non-fuel hardware, including up to four (4) damaged PWR fuel assemblies to be stored in designated locations only. The MPC-24EF is designed to store up to 24 total PWR fuel assemblies, with or without non-fuel hardware, including up to four (4) damaged PWR fuel assemblies or fuel classified as fuel debris to be stored in designated locations only. The MPC-32 is designed to store up to 32 total PWR fuel assemblies with or without non-fuel hardware. The MPC-68 is designed to store up to 68 total BWR fuel assemblies including up to 68 damaged Dresden Unit 1 or Humboldt Bay BWR fuel assemblies. Damaged BWR fuel assemblies other than Dresden Unit 1 and Humboldt Bay are limited to 16 storage locations in the MPC-68 with the remainder being intact BWR fuel assemblies up to a total of 68. The MPC-68F is designed to store up to 68 intact or damaged Dresden Unit 1 and Humboldt Bay BWR fuel assemblies. Up to four (4) of the 68 fuel storage locations in the MPC-68F may be Dresden Unit 1 and Humboldt Bay BWR fuel assemblies classified as debris. The MPC-68FF is designed to store up to 68 total BWR fuel assemblies. This may include up to sixteen (16) damaged BWR fuel assemblies from locations other than Dresden Unit 1 or Humboldt Bay. Any number of the 68 fuel assemblies stored in the MPC-68FF may be Dresden Unit 1 or Humboldt Bay BWR fuel assemblies classified as intact fuel or damaged fuel. Up to eight (8) of the 16 BWR damaged fuel assembly storage locations may be filled with BWR fuel classified as fuel debris. All fuel defined as damaged or defined as fuel debris is placed into damaged fuel containers.

The staff verified that all methods of heat transfer internal and external to the HI-STORM 100 and 100S Cask Systems are passive. 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 72.24(c)(3).

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. Tables 4.1 and 4.2 list the maximum allowable decay heat load that can be stored in the MPC-24, MPC-32, and MPC-68 as a function of time following removal of the fuel assemblies from the reactor core (e.g., fuel decay time). These limits on decay heat loads are based on the calculated maximum cladding temperature limits for normal conditions. Incorporation of all thermal loads into the analytical methods remain unchanged from the original analysis.

4.4 Model Specification and Confirmatory Analyses

The applicant, in this amendment, is requesting approval for changes primarily to the following areas which are pertinent to the thermal design characteristics of the HI-STORM and HI-TRAC cask system: 1) to store spent fuel in four (4) new MPC designs and a modified overpack, 2) to store spent fuel with both moderate burn-up (up to 45,000 MWD/MWT and high burn-up greater than 45,000 MWD/MTU) and with a higher peak total heat load, 3) to institute regionalized fuel loading configurations, 4) to store PWR damaged fuel and damaged fuel debris, 5) to take credit for gravity driven internal convection as a means for heat transfer from the spent fuel to the environment, 6) to store non fuel hardware, and 7) to use a forced helium dehydration system for moisture removal.

This amendment seeks approval to store spent fuel in the following four (4) new or modified MPC design configurations; MPC-24E, MPC-24EF, MPC-32, and MPC-68FF. The applicant applied the same methodology used for the original MPC-24 and MPC-68 for development of analytical models for the newly designed baskets. Similarly, because the basic configuration of the MPC-24 and MPC-68 remain unchanged with the exception of DFCs for storage of damaged fuel and damaged fuel debris, the fundamental analytical techniques remain applicable. In the case of the MPC-32 the applicant developed a unique model defining the bounding characteristics of the basket but adhered to the same methodology and assumptions used for the MPC-24 and MPC-68 basket designs. The MPC-32 is similar in design to the MPC-68 (i.e., without flux traps). The staff's structural review of the basket configurations can be found in Section 3 of this SER.

The applicant requested approval to utilize the HI-STORM 100S overpack for use at facilities that have low clearance/profile truck bay (or rail bay) doors. The applicant developed a three dimensional axisymmetric thermal model of the HI-STORM 100 overpack. The applicant provided the staff with calculations that evaluated the HI-STORM 100 and the shorter variation HI-STORM 100S designs. The applicant concluded that the fuel cladding temperatures for the MPC placed in the HI-STORM 100S overpack are bounded by the HI-STORM 100 system thermal model solution. With assistance from the Pacific Northwest National Laboratory, the staff reviewed the applicant's calculations and independently verified using the applicants input parameters, that the results are suitably conservative and applicable to the geometries proposed. The staff also reviewed assumptions pertinent to these calculations and found their use appropriate.

The applicant requested to store high burn-up and moderate burn-up fuels with commensurate higher heat loads. The applicant performed thermal-fluid analyses based on a detailed heat transfer model using a combination of the ANSYS and FLUENT computer codes that conservatively account for all modes of heat transfer in three dimensional space in the MPC and overpack. The original analytic methodology, previously approved by the staff, did not take credit for natural internal circulation. In this amendment the applicant credited the thermosiphon effects and asserted that the effects are intrinsic to the HI-STORM fuel basket design. When crediting the thermosiphon effects the applicant did not apply the "Rayleigh" correlation in the basket/MPC interface region used in the original thermal analyses. Additionally, in the thermosiphon enabled model/analyses, the applicant conservatively ignored the presence of the aluminum conduction elements for heat dissipation in the downcomer spaces. Therefore, the critical element in this new approach is the inclusion of internal convective heat transfer in the analytic models. The heat loads for cooling times and cask configurations for each burn-up are listed in Tables 4.2 and 4.3 of this SER.

Table 4.2
Design Basis Maximum Permissible Heat Load Versus Fuel Age at Loading (Moderate Burnup) for Uniform Fuel Loading

Fuel Age at loading (yrs)	MPC-24E (kW)	MPC-32 (kW)
5	28.17	28.74
6	27.33	27.95
7	25.05	25.79
10	24.53	25.26
15	23.95	24.68

Table 4.3
Design Basis Maximum Permissible Heat Load Versus Fuel Age at Loading (High Burnup) for Uniform Fuel Loading

Fuel Age at Loading (yrs)	MPC-24 (kW)	MPC-24E (kW)	MPC-32 (kW)	MPC-68 (kW)
5	27.12	27.50	28.10	28.19
6	26.09	26.44	27.10	26.81
7	24.74	25.05	25.79	24.71
10	24.02	24.31	25.05	24.18
15	23.50	23.79	24.53	23.60

The staff performed confirmatory calculations to verify the thermal performance of the Holtec HI-STORM 100 cask system and soundness of the Holtec analytical approach. The staff, with assistance from the Pacific Northwest National Laboratory, modeled the Holtec cask using the COBRA-SFS (Spent Fuel Storage) thermal hydraulic computer software. The analyses assumed bounding fuel and solar energy (insolation) conditions and natural circulation (convection) cooling inside the multi-purpose canister (MPC). A summary description of the approaches taken and the results from each effort are provided below.

A 1/8 symmetry section of the Holtec HI-STORM 100 spent fuel storage system was modeled using the COBRA-SFS computer code. The COBRA-SFS code modeled in detail the flow field inside the canister, accounting for conduction, convection, and thermal radiation heat transfer mechanisms. The code has been rigorously validated against full scale experimental data for various cask designs, including ventilated concrete casks similar to the HI-STORM design. In addition to validation of the code to model internal convective flow inside a loaded canister, the code has been validated to model flow in the annulus formed between the MPC and the concrete overpack. The PNNL calculations modeled the heat transfer throughout the cask internals, (fuel assemblies, basket, and flow channels) into and across the annulus, through the vents, through the concrete overpack and to the ambient. The COBRA-SFS simulations included insolation heat input on the cask sides and lid. The ten cases evaluated are listed in Table 4.4 of this SER.

Table 4.4
Summary of COBRA-SFS simulations

Case#	**21.5 kW decay heat (5 ATM Helium)**	**COBRA-SFS Predicted Peak Clad Temperature**
1	Ambient air at 52°F (11.1°C)	534 °F (278.9°C)
2	Ambient air at 60°F (15.5°C)	543 °F (283.9°C)
3	Ambient air at 67°F (19.4°C)	550 °F (287.8°C)
4	Ambient air at 80°F (26.7°C)	563 °F (295.0°C)
5	Ambient air at 125°F (51.7°C)	608 °F (320.0°C)
6	Ambient air at 150°F (65.6°C)	631 °F (332.8°C)
7	Ambient air at 200°F (93.3°C)	680 °F (360.0°C)
Case#	**20.88 kW decay heat (5 ATM Helium)**	**COBRA-SFS Predicted Peak Clad Temperature**
8	Ambient air at 60°F (15.6°C)	531 °F (277.2°C)
9	Ambient air at 250°F (121.1°C)	713 °F (378.3°C)
Case#	**28.74 kW decay heat (5 ATM Helium)**	**COBRA-SFS Predicted Peak Clad Temperature**
10	Ambient air at 80°F (26.7°C)	691.9 °F (366.6°C)

Cases were investigated for a cask loaded with 24 PWR spent fuel assemblies initially with 0.896 Kilowatts (kW) per assembly, for a total of 21.5 kW of decay heat. This total decay heat is actually higher than the design limit of 20.88 kW. Therefore, the calculated PCT for Cases 1-7 are conservatively high. Two additional cases quantified the impact of using the correct decay heat. Case 8 repeated Case 2 with a decay heat load of 20.88 kW and Case 9 repeated Case 7 with the corrected decay heat load. Lastly, Case 10 investigated the thermal response of a cask loaded with the equivalent amount of decay heat (28.74 kW) planned for a 32 assembly basket. To approximate the thermal performance of the 32 assembly design, the decay heat for each assembly in the 24 assembly design was increased uniformly to 1.1975 kW each, for a total of 28.74 kW. In this case a peak clad temperature of 691.9°F was predicted.

These results compared favorably with the results obtained by Holtec. The staff reviewed the assumptions used by Holtec and agrees that input parameters are consistent with design values for the MPC and the HI-STORM overpack. The staff finds that the applicant selected suitably bounding and appropriate boundary conditions. The staff reviewed the results of the validation of the computer code and analytical method used by Holtec in the HI-STORM analyses. The results of Holtec's analytical method, also is in agreement with the staffs analyses, showed good agreement with DOE/EPRI test data and results.[7] The staff therefore has reasonable assurance that this methodology, as applied to calculation of normal, off-normal and HAC temperatures, will yield suitably conservative and accurate results

The applicant is also requesting the storage of two groups of BWR low heat emitting (LHE) fuel assemblies and design basis fuel assemblies in the MPC-68 storage configuration. The LHE fuel are characterized by low burnup, long cooling time and short active length fuels. The applicant asserts that the heat loads for the LHE fuel is considerably smaller than that of the DB group of fuel assemblies. The Dresden-1 (6x6 and 8x8), Quad+, and Humboldt Bay (7x7 and 6x6) fuel assemblies are grouped as LHE fuel. The applicant evaluated this fuel when encased in DFCs. The applicant asserts that due to the fact that the analyses are performed with the fuel encased in DFCs the reduced radiation heat exchange between the fuel assembly and the fuel basket will consequently result in this configuration being bounding for thermal evaluation. The applicant considers the Dresden-1 8x8 LHE fuel assembly to be the most resistive and therefore the bounding assembly for thermal evaluation. The staff reviewed the applicant's methodology and agrees with assumptions made. The temperature results were verified and found to be correctly calculated using the identified inputs, assumptions, and methodology.

This application is seeking approval to store spent fuel in two fuel storage scenarios. The first scenario is designated as uniform loading, whereby every basket cell is assumed to be occupied by fuel producing heat at the maximum rate. The applicant analyzed the storage of moderate burnup and high burnup fuels for this loading scenario. The second scenario is designated as regionalized loading whereby a two region configuration is stipulated. The two regions are defined as an inner region for storing hot fuel and an outer region, enveloping the inner region, for storing low decay heat fuel. Tables 4.5 and 4.6 delineate the regionalized heat load limits. Fuel loading configurations are delineated in Table 2.1.13, "MPC Fuel Loading Regions," of the FSAR. The applicant conservatively asserts in the analysis that the outer region fuel is old fuel (>15 years cooled). The applicant defines the fuel temperature limits for the outer region to be lower than those for the inner region with the outer region temperature bounded by the interface cladding temperature. The inner region cladding temperature limits are dictated by the post core decay time. Whichever limit is reached first is the limit which dictates the inner and outer region cladding temperature limits and ultimately governs the cask maximum heat load. The applicant asserts that regional fuel temperature limits and cask maximum heat loads for the MPC-32 and MPC-68 designs are governed by the interface cladding temperature limits. The staff reviewed the applicant's calculations and agrees with the results. The staff reviewed assumptions pertinent to regionalized loading and determined that they are suitably conservative and applicable to the loading configurations proposed by the applicant.

Table 4.5
Regionalized Loading Outer Region Heat Load Limits

MPC Type	Inner Region Assemblies	Outer Region Assemblies	Outer Region Heat Load (kW)
MPC-24	4	20	18
MPC-24E	4	20	18
MPC-32	12	20	12
MPC-68	32	36	9.9

Table 4.6
Regionalized Loading Inner Region Heat Load Limits (kW)

Fuel Age (years)	MPC-24	MPC-24E	MPC-32	MPC-68
5	5.88	6.16	13.58	16.02
6	5.88	6.16	12.87	14.99
7	5.34	5.58	11.92	13.40
10	4.94	5.16	11.40	12.99
15	4.66	4.86	11.02	12.54

The applicant sought approval to store damaged fuel assemblies and damaged fuel assembly debris. Damaged fuel assemblies are defined by the applicant as fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered fuel debris. Fuel debris may be constituted of ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means as noted above. Damaged fuel assemblies and damaged fuel assembly debris must be stored in damaged fuel containers (DFCs).

For the damaged fuel assemblies, the original analyses remain bounding for those fuel assemblies that remain in the as loaded configuration. Similarly, the design temperature limits remain bounding for fuel that, in an off normal or hypothetical accident conditions, fractures and/or collapses within the damaged fuel can. The damaged fuel assemblies contain fuel types similar to those previously analyzed, the only factor differing being the presence of pinhole leaks, or hairline cracks. The staff agrees that a cask loaded with damaged fuel assemblies will have a heat output that is bounded by the maximum heat load of 28.74kW (MPC-32). Similarly, when the damaged fuel remains in the as loaded configuration, which is commensurately bounded by the physical dimensions of the undamaged fuel, and since the fuel originates from the same reactor core and using given burnup and cooling times, the staff agrees that the axial heat flux profile will be bounded by that of the undamaged fuel.

The staff agrees that the presence of pinhole leaks and hairline cracks will not affect pin-to-pin heat transfer via radiation or conduction/convection. Additionally, the absence of rod fill gas and internal pressure will not affect the computed effective conductivity. The staff agrees that the overall effective thermal conductivity of the undamaged fuel will remain bounding for the damaged fuel assemblies. Analogous to this, due to the absence of fill gasses in the damaged fuel rods/fuel debris there will, therefore, be no introduction of fission gasses to the canister atmosphere from the damaged fuel rods/fuel debris nor will there be the propensity for additional canister pressure due to rod failures beyond that which has already been determined for the undamaged fuel assemblies. The staff concludes that the determinations reached by the applicant are acceptable. Specifically, that the temperature increase due to the damaged fuel can and the peak cladding temperatures remain bounded by the limits determined for the undamaged fuel assemblies.

The applicant sought approval to store PWR non-fuel hardware (BPRAs - burnable poison rod assemblies, TPDs - thimble plug devices, CRAs - control rod assemblies, and APSRs - axial power shaping rods, etc.) in the MPC-24 and MPC-32 design configurations. Appendix B to CoC 1014 delineates fuel loading restrictions. The applicant asserts that there are two effects to be analyzed for the inclusion of this hardware into cask storage. The presence of the non-fuel hardware increases the effective basket conductivity and therefore enhancing heat dissipation and lowering fuel temperatures as well as the gas temperature filling the space between fuel rods. However, the gas volume displaced by the mass of the non-fuel hardware lowers the cavity free volume. These two effects have an opposing influence on the MPC cavity pressure. The applicant’s analysis computed the temperature field with non-fuel hardware excluded and concluded that this configuration provided a conservatively bounding temperature field for the PWR baskets (MPC-24, MPC-24E, MPC-24EF, and MPC-32). The applicant computed the MPC cavity free space based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included. The applicant stated that based on bounding BPRA rod internal pressures, a large hypothetical quantity of helium (7.2 g-moles/BPRA) is available for release into the MPC cavity. The staff reviewed the applicant’s methodology and agrees with assumptions made. The temperature results were verified and found to be correctly calculated using the inputs identified by the applicant, the applicant’s assumptions, and the methodology.

The applicant requested approval to store Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source in the MPC-68 and MPC-68F baskets designs. The storage of this material will be limited to one Thoria Rod Canister in combination with other intact and damaged fuel and damaged fuel debris as delineated in Appendix B to the CoC 1014. Thoria Rod canisters contain 18 fuel rods of long cooled very low fuel burnup which constitute a very small source of decay heat. The staff agrees that their contribution to heat load is small and that thermal implications of casks loaded with Thoria Rod Canisters are bounded. Thoria rod loading, due to the length of cooling time for this material, is stipulated by Technical Specifications, to be positioned toward the basket periphery. The staff concurs that this provides assurance that these fuel rods will be stored in a benign thermal environment, and therefore remain protected during long term storage.

The applicant reanalyzed the thermal consequences of blockage of the concrete overpack air vents. This reanalysis considered blockage of two, three, and all four air inlet vents. The applicant analyzed this accident condition using the thermal load of the MPC-32 canister. The MPC-32 has the highest decay heat load of all the MPC designs. The applicant asserts that the largest temperature rise of the MPC or its contents as a result of blockage of two vents, is 25°F for the MPC shell. The applicant asserts that the largest temperature rise of the MPC or its contents as a result of blockage of three vents is 81°F, for the MPC shell. Regarding 100 percent blockage of the overpack air vents, the applicant concluded no component reaches its short-term temperature limit for the assumed 72-hour event duration. The MPC internal pressure was also calculated and was determined to remain below the design maximum accident pressure of 200 psig. Given the increased heat rejection rate using the thermosiphon enabled model, and the relative position and configuration of the MPC lid and MPC shell, the time to reach the short term concrete temperature limit remains applicable. It should be noted that with increased rod breakage and associated additional rod fill gas in the MPC cavity during hypothetical accident conditions (HAC), heat transfer from the thermosiphon effect will increase adding conservatism to the above results. The staff agrees that the time limits specified and the original Corrective Actions to be taken, per Technical Specification requirements, therefore remain applicable.

4.5 Pressure Analysis

The applicant presented recalculated HI-STORM 100 system MPC calculated pressures for normal, off-normal and accident conditions. The maximum pressure was calculated using free volume of the MPC, ideal gas law, and accounted for backfill helium gas along with a fraction of the stored fuel helium gas and fission product gas. The normal, off-normal, and accident conditions were differentiated by the assumption of the fraction of stored spent fuel, and non fuel hardware, which contribute fill gas and fission gas to the MPC in conjunction with gas temperatures. These fractions are in agreement with NUREG-1536. The resulting pressures are summarized in Tables 4.4.1.14 (for normal conditions of storage) in the FSAR. The calculated maximum pressure for each MPC under all conditions remains below its appropriate design pressure. The staff reviewed the applicant’s calculations and methodology for determining cask free volume and cask pressures during normal off-normal, and accident scenarios. The staff agrees with the conclusions reached by the applicant and the calculated results.

4.6 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 applicant performed reanalyses of onsite transportation activities using the HI-TRAC transfer cask. This analyses used the same methodology performed in the original analyses. In this study the cask is positioned in two orientations, vertically and horizontally. The applicant clarified that for maximum temperatures under transport conditions, when in the vertical orientation, passive heat rejection is via natural convection and radiation to a hot ambient environment (100°F/37.8°C). There is also less metal to metal contact between the internal components, such as the fuel, fuel basket, and MPC shell. For this reason the gap resistance between these parts is higher than in a horizontally oriented HI-TRAC. The applicant asserted that to bound these gap resistances the various parts are postulated to be in a centered configuration. The applicant calculated the peak cladding temperature, taking credit for internal convective heat transfer, under these conditions was computed to be 872°F (466.7°C). Furthermore, the applicant, in order to demonstrate defense in depth, neglected all means of convective heat transfer within the canister and determined that the peak cladding temperature for a bounding fuel load to be 1025°F (551.7°C) which is below the short term limit of 1058°F (570.0°C). The staff reviewed the applicant's calculations and methodology and agrees with the applicants results. A review of the applicant's assumptions pertinent to HI-TRAC fuel transfer operations determined that their use is applicable and appropriate. The staff agrees with the applicant's conclusions regarding temperature limits during HI-TRAC fuel transfer.

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

MPC fuel unloading operations are performed with the MPC inside the HI-TRAC cask. For this operation, a helium cool down system is connected to the MPC via lid access ports. Forced helium cooling of the MPC fuel and MPC is then initiated. The applicant analyzed scenarios where ambient air access is not restricted and where ambient air access is restricted such as in the cask pit area. Cladding temperature rise is to be limited to no more than 100°F under these circumstances as previously approved by the staff. The applicant determined the time limit to reach this 100°F temperature rise is 22 hours in the original analyses. This was postulated, in the original analyses, with a bounding heat load of 22.5 kW, assuming no credit for passive cooling mechanisms. The applicant is requesting approval to increase the bounding heat load to 28.74 kW. In light of this, the applicant recalculated the cladding temperature rise to remain below 100°F during this 22 hour time period but asserted that the rate of cooling is degraded by 90 percent (originally set at 100 percent with no heat rejection to the environment) with 10 percent of decay heat dissipated to the environment. The staff reviewed the new parameters used to determine the relative temperature increase and agrees that the methodology is sound and that the results are appropriately conservative.

4.6.1 MPC Drying Operations

Regarding vacuum drying operations the applicant, using the original methodology, recalculated the peak cladding and basket temperatures using the increased design maximum heat load for each MPC loaded with moderate burnup spent fuel while connected to the Vacuum Drying System (VDS). During vacuum drying operations, the applicant asserts that the short term peak clad temperatures with design basis maximum heat loads, stated in Table 4.7 of this SER, are calculated to be less than 1000°F (537.8°C) for all MPC basket designs. Additionally this is less than the short term maximum fuel clad temperature limit of 1058°F (570.0°C). The staff evaluated the input parameters in conjunction with the applicant's methodology and agrees with the results.

Table 4.7
Peak Cladding Temperature Under Vacuum Conditions
for Design Basis Maximum Heat Loads

MPC	Temperature
MPC-24	950°F (510.0°C)
MPC-24E	926°F (496.7°C)
MPC-32	974°F (523.3°C)
MPC-68	932°F (500.0°C)

For drying canisters containing the high burnup spent fuel, and as an alternative method for drying canisters containing moderate burnup spent fuel, the applicant proposed the use of a forced helium dehydration system wherein the forced recirculation of helium through the MPC cavity is used as a means for moisture removal. Warm dry helium gas is supplied to the MPC drain port and forced through the MPC cavity where it absorbs moisture. The humidified gas is vented and filtered as it is cooled and dried. This process continues until the gas exiting the MPC cavity meets the limit specified in the Technical Specifications. The staff agrees in principle that the process proposed by the applicant is a viable means for moisture removal from inside the cask provided the design criteria delineated in the Technical Specifications and Appendix 2.B of the FSAR are satisfied. Additionally, the staff agrees that the forced flow of helium will preclude exceeding the short term temperature limits and that the evaluation and results for vacuum drying operations bound the temperature limits determined for this new process. Commensurate with this determination are the conclusions reached by the staff regarding cladding creep stress invoked temperature limits addressed in Section 4.1 of this SER.

4.7 Conclusions

The staff has reviewed 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 new MPC basket configurations and fuel load types against the requirements of 10 CFR Part 72. Additionally, the methods used in the thermal evaluation are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package thermal design.

Based on the review of the the HI-STORM spent fuel storage system amendment request to 1) store spent fuel in four (4) new MPC designs and a modified overpack, 2) store spent fuel with both moderate burn-up (up to 45,000 MWD/MWU and high burn-up (greater than 45,000 MWD/MTU) and with a higher peak total heat load, 3) regionalize fuel loading configurations, 4) store PWR damaged fuel and damaged fuel debris, 5) credit for gravity driven internal natural convection heat transfer inside the MPC from the spent fuel to the environment, 6) store non fuel hardware, and 7) to use a forced dehydration system for moisture removal, 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.

4.8 Evaluation Findings

Part 72 of Title 10 of the *Code of Federal Regulations* requires an analysis and evaluation of the dry cask storage system thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years. This section reviewed the performance of the HI-STORM-100 dry cask storage system. The staff concludes that the HI-STORM Dry Cask Storage System design fulfills the following acceptance criteria:

F4.1 Fuel cladding temperature in Table 4-1 of this SER are below the maximum allowable temperatures limits for normal conditions

F4.2 The staff finds, in accordance with 10 CFR 72.122(h), that the spent fuel cladding is protected against degradation leading to gross ruptures by maintaining the cladding temperature for Zircaloy clad below the temperature limits. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.3 The applicant established a short-term temperature limit of 570 ^{o}C (1058 ^{o}F) for off-normal and hypothetical accident condition for Zircaloy clad fuel in accordance with guidance of NUREG-1536 and PNL-4835. The staff finds the short term temperature limits acceptable.

F4.4 The staff finds the forced helium dehydration system acceptable. This system will inhibit major annealing of the cladding.

F4.5 Structures, systems, and components important to safety are described in sufficient detail in Chapters 1 and 2 of the FSAR to enable an evaluation of their thermal effectiveness. These structures, systems, and components important to safety remain within their operating temperature range.

F4.6 The HI-STORM 100 and 100S overpacks, with loaded MPC-24 (including E and EF), MPC-32, or MPC-68FF, are designed with a heat removal capacity that is verifiable and reliable. The cask is designed to provide adequate heat removal capacity without an active cooling systems.

F4.7 The staff concludes that the thermal design described in the FSAR is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the HI-STORM 100 and 100S overpacks with the loaded MPC-24 (including E and EF), MPC-32, or MPC-68FF cask will allow safe handling and storage of spent fuel for a certified life of 20 years.

4.9 References

1. ISG-15, “Materials Evaluation,” January 10, 2001.

2. H. Spilker, et al, “Spent LWR Fuel Dry Storage in large Transport and Storage Casks after Extended Burnup,” Journal of Nuclear Materials, Vol. 250, pp. 63-74, 1997.

3. K.L. Murthy, G.S. Clevinger, and T.P. Papazoglou, “Thermal Creep of Zircaloy-4 Cladding”, 4th International Conference On Structural Mechanics in Reactor Technology, San Francisco, paper C3/4 (1977).

4. Mayuzumi and T. Onchi, “The applicability of the Strain Hardening Rule to Creep Deformation of Zircaloy-4 Fuel Cladding Tube during Dry Storage Conditions”, Journal of Nuclear Materials, 178, pp. 73-79 (1991).

5. K.E. Amin, A.K. Mukherjee, and J.E. Dorn, “A Universal Law for High-temperature Diffusion Controlled Transient Creep,” Journal of Mechanical Physics Solids, 18, pp. 413-419 (1970).

6. A.B. Johnson, Jr., and E.R. Gilbert, Pacific Northwest Laboratory, “Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases,” PNL-4835, September 1983.

7. EPRI NP-5128/PNL-6054, “The TN-24P PWR Spent Fuel Storage Cask: Testing Analysis,” April 1987.

5.0 SHIELDING EVALUATION

The staff reviewed the capability of the HI-STORM 100 Cask System, as modified, to provide adequate protection to the public and workers against direct radiation. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), 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. This amendment was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfills the acceptance criteria listed in Section 5 of NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems.”

5.1 HI-TRAC Dose Rate Acceptance Criteria, TS LCO 3.2.1

The applicant has requested to: (a) increase the HI-TRAC transfer cask dose rate acceptance criteria in TS LCO 3.2.1; (b) delete Figure 3.2.1-1 in Appendix A to CoC 1014, and revise SR 3.2.1.1; and (c) increase the time to perform a written evaluation, from 24 to 48 hours, in the event the average surface dose rate limits are not met.

The NRC staff reviewed the applicant's request to increase the HI-TRAC transfer cask dose rate acceptance criteria in LCO 3.2.1, and concludes that the increased dose rates in the technical specification are consistent with the evaluation findings of the new fuel contents.

The NRC staff reviewed the applicant’s request to delete Figure 3.2.1-1 and revise SR 3.2.1.1. Figure 3.2.1-1 shows the dose rate measurement locations on a HI-TRAC transfer cask. To replace the figure, the applicant added verbiage to SR 3.2.1.1, to direct users to the proper location for taking dose rate measurements necessary to demonstrate compliance with LCO 3.2.1. Application of the new verbiage added to SR 3.2.1.1, along with the deletion of Figure 3.2.1-1, results in the same performance of average surface dose rate measurement determinations as currently approved.

The NRC staff reviewed the applicant's request to increase the time to perform a written evaluation, from 24 to 48 hours, in the event the average surface dose rate limits are not met,. Staff concludes that the written evaluation completion time is acceptable.

5.2 HI-TRAC Contamination Surveys Inside Part 50 Facility, TS LCO 3.2.2

The applicant has requested to add a note to TS LCO 3.2.2 to require contamination surveys to be performed on the HI-TRAC transfer cask, only if the transfer operations occur outside the Part 50 facility. The staff agrees that decontamination activities of the empty transfer cask that occur inside the Part 50 facility, are governed by the user’s Part 50 program. Therefore, the change to Technical Specification LCO 3.2.2 is acceptable to the staff.

5.3 HI-STORM Dose Rate Acceptance Criteria, TS LCO 3.2.3

The applicant has requested to: (a) increase the HI-STORM storage cask dose rate acceptance criteria in TS LCO 3.2.3; (b) delete Figure 3.2.3-1, and revise SR 3.2.3.1; (c) delete the words

“Transport Operations” from LCO 3.2.3, and revise Required Action A.2 to substitute a written evaluation in lieu of an analysis; and (d) increase the time to perform a written evaluation, from 24 to 48 hours, in the event the average surface dose rate limits are not met.

The NRC staff reviewed the applicant's request to increase the HI-STORM storage cask dose rate acceptance criteria in LCO 3.2.3, and conclude that the increased dose rates in the technical specification are consistent with the evaluation findings of the new fuel contents.

The NRC staff reviewed the applicant’s request to delete Figure 3.2.3-1, and revise SR 3.2.3.1. Figure 3.2.3-1 shows the dose rate measurement locations on a HI-STORM storage cask. To replace the figure, the applicant added verbiage to SR 3.2.3.1, to direct users to the proper location for taking dose rate measurements necessary to demonstrate compliance with LCO 3.2.3. Application of the new verbiage added to SR 3.2.3.1, along with the deletion of Figure 3.2.3-1, resulted in the same performance of average surface dose rate measurement determinations as currently approved.

The NRC staff reviewed the applicant’s request to delete the words “Transport Operations” from LCO 3.2.3, and revise Required Action A.2 to substitute a written evaluation in lieu of an analysis. The staff concludes that the written evaluation will ensure that the cask user properly assesses, in writing, any nonconforming conditions.

The NRC staff reviewed the applicant's request to increase the time to perform a written evaluation, from 24 to 48 hours, in the event the average surface dose rate limits are not met. Staff concludes that the written evaluation completion time is acceptable.

5.4 TN/D-1, Generic PWR, and Generic BWR DFC

The HI-STORM system is currently approved to store damaged fuel or fuel debris when the fuel is contained in a Holtec DFC. The applicant requested the addition of a Transnuclear (TN) DFC containing Dresden Unit 1 (D-1) fuel, a Holtec generic PWR DFC, and a Holtec generic BWR DFC to the HI-STORM approved contents. Figure 2.1.2 show the key dimensions of the TN/D-1 DFC, and Figures 2.1.2B, and 2.1.2C show the dimensions of the PWR and BWR DFCs, respectively.

For the TN/D-1 DFC, the applicant justified why the current analysis for the Holtec DFC is bounding. The applicant assumed that the fuel collapsed to a height of 80 inches in the Holtec DFC, to simulate damaged fuel in post-accident conditions or fuel debris. This height was determined by using the inner dimensions of the Holtec DFC. The source per inch was then calculated. Since the inner diameter of the TN/D-1 DFC is smaller than the inner diameter of the Holtec DFC and the fuel is identical, the height of the collapsed fuel in a TN/D-1 will be greater (i.e., for two cylinders with the same volume but different diameters, the cylinder with a smaller diameter will have a greater height). Therefore, the source per inch will be less in the TN/D-1 DFC and the shielding evaluation for the Holtec DFC in Chapter 5 of the FSAR bounds the TN/D-1 DFC.

Based upon the review of the applicant’s analysis, the staff agrees that use of the TN/D-1 DFC is bounded by the current analysis and further evaluation is not required.

For the generic PWR and BWR DFCs, the applicant performed a shielding analysis to show that fuel debris under normal or accident conditions, or damaged fuel in a post accident condition configuration, does not result in a significant increase in the dose rates around the 100-ton HI-TRAC. The applicant concluded that the dose rate evaluation for the 100-ton HI-TRAC bounds the evaluations for the 125-ton HI-TRAC and the HI-STORM.

The applicant used MCNP-4A to create a radial model of debris/post-accident damaged fuel assemblies in the four peripheral damaged fuel locations in the MPC-24E, and the 16 peripheral locations in the MPC-68. The fuel density was doubled, which in turn, doubled the source term. The dose rate was observed to have increased by less than 20 percent at the bottom radial area of the 100-ton HI-TRAC, while the dose rate in the radial midplane area decreased.

The NRC staff found the addition of the PWR and BWR DFCs to the approved contents to be acceptable. The justification for acceptance is: (a) although the increase in dose rate in the 100-ton HI-TRAC was 20 percent at the bottom, the increase in dose rate for the HI-STORM casks would be much smaller due to much greater shielding, (b) the applicant used conservative assumptions in the analysis, (c) the majority of significantly damaged fuel assemblies will have a lower burnup and longer cooling time than assumed in the analysis.

5.5 Storage of La Crosse Fuel in Stainless Steel Channels

The applicant analyzed the storage of La Crosse BWR fuel assemblies in stainless steel channels. La Crosse has thirty-two stainless steel channels, some of which have been in the reactor core for approximately the lifetime of the plant. Since cobalt (Co) activation is significant, the applicant requested to store sixteen of these channels per cask in the innermost locations of the MPC-68, and the MPC-68FF.

The applicant used specific information from the La Crosse BWR power reactor to determine the source term of one stainless steel BWR channel. The applicant assumed that one channel had been irradiated continuously for 6388 days, at a power level of 29.17 MW/MTU, for a burnup of 180,000 MWd/MTU. The resulting source term was multiplied by 16 (since the applicant requested to store 16 channels per cask), and cooled for 13 years. The applicant then added the Co component of the source term of the channel (energy range of 1.0 to 1.5 MeV), to the Co component of the total source term of the design basis stainless steel clad BWR fuel with a 13 year cooling time. Since the steel channels in the active fuel zone at La Crosse are 83 inches in length, the source terms were scaled to 144 inches. This was done so that the Co component of the 13 year cooled La Crosse assemblies, along with the 13 year cooled La Crosse channels, could be compared to the Co component of the design basis stainless steel clad BWR fuel cooled for 10 years, as shown in Table 5.2.8. The applicant showed that the Co component of both source terms are nearly identical. Therefore, no further analysis was necessary.

Based upon the review of the applicant's analysis, the staff agrees that storage of the stainless steel La Crosse fuel channels, which have been cooled for 13 years, is bounded by the current analysis and further evaluation is not required.

5.6 Regionalized Loading

The applicant requested to have regionalized fuel loading as approved contents, shown in Tables 2.1-6 and 2.1-7 of Appendix B to the CoC. The purpose of the change is to allow fuel with higher heat emission rates to be stored in the center of an MPC surrounded by fuel with lower heat emission rate. The center region is described by the applicant as Region 1, and the periphery region is Region 2.

The applicant performed a comparison to determine for each MPC, when and where the regionalized loading would produce higher dose rates than uniform loading. The following table summarizes the applicant's results:

	axial center surface	axial center @ 1 m	radial midplane surface	radial midplane @ 1 m
MPC-32	regionalized, 15% higher	regionalized, 5% higher	uniform	uniform
MPC-24	regionalized, 21% higher	uniform	uniform	uniform
MPC-68	regionalized, 21% higher	regionalized, 5% higher	uniform	uniform

The applicant showed that dose rates for uniform loading bound the dose rates for regionalized loading at the surface and at 1 meter distance from the HI-STORM cask in the radial direction. Regionalized loading patterns reduce the dose rate in the radial direction by shielding the hotter fuel on the inside of the cask with colder fuel on the outside of the cask. In the axial direction, surface dose rates in the center of the cask are higher for regionalized loading, but dissipate at distances as close as 1 meter. Therefore, the applicant did not do full scale regionalized loading patterns in Chapter 5, rather, the applicant assumed uniform loading for all the analyses.

Based upon the review of the applicant's analysis, the staff agrees that uniform loading generally bounds regionalized loading, and that regionalized fuel loading patterns are acceptable for storage. The applicant has shown that in those cases where regionalized loading produces higher dose rates, it is a localized effect on the top or the bottom of the cask. Because these doses will dissipate at the edge of the top of the cask (as depicted in the graphical data in Section 5.1.1 of the applicant's FSAR), and workers minimize their stay at the surface of the top of the cask, there will not be a significant increase in occupational exposure rates. The localized increases in dose rates at the bottom center of the HI-STORM storage cask is an area where workers will not be present.

5.7 Addition of Non-Fuel Hardware

The applicant requested revision to the approved contents to allow storage of PWR fuel assemblies with non-fuel hardware, such as burnable poison rod assemblies (BPRAs), wet annual burnable absorbers (WABAs), thimble plug devices (TPDs), orifice rod assemblies, water displacement guide tube plugs, control rod assemblies (CRAs), axial power shaping rods (APSRs), control element assemblies (CEAs), and rod cluster control assemblies (RCCAs) in the HI-STORM system. The applicant requested unrestricted storage of the BPRAs and TPDs in the MPC-24, MPC-24E, MPC-24EF, and MPC-32. The applicant requested that storage of the CRAs and APSRs be restricted to the innermost four locations in the MPC-24, MPC-24E, MPC-24EF, and the MPC-32. The only significant radiation source is from irradiation of the Co-59 impurities in the stainless steel and Inconel in the non-fuel hardware, which creates Co-60.

The applicant firstly determined the bounding BPRA (including WABAs), and TPD (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs), by analyzing all BPRAs and TPDs found in References 1 and 2 of this document, to determine which produced the highest source term and decay heat for a specific burnup and cooling time. The applicant used the SCALE 4.3, Revision 5, SAS2H module, with ORIGEN-S, to calculate the radiation source term and decay heat load for the BPRAs and TPDs. In the ORIGEN-S calculations, the impurity level was assumed to be 0.8 g of Co-59 per kg of stainless steel, and 4.7 g of Co-59 per kg of Inconel. The applicant determined that the bounding TPD was the Westinghouse 17x17 guide tube plug. The bounding BPRA was determined by combining the higher masses of the Westinghouse 17x17 and the 15x15 into a single, hypothetical BPRA. The total curies of cobalt and the decay heat were then calculated as a function of burnup and cooling time. The BPRA and TPD limits for burnup and cooling time are given in Table 2.1-8 of Appendix B of the CoC.

Secondly, the applicant determined the bounding CRA (including CEAs and RCCAs), and APSR. The design basis CRA was determined to be the Silver-Indium-Cadmium (Ag-In-Cd) CRA, when partially irradiated. The design basis APSR is the gray APSR (versus the black APSR). Since the level activation of CRAs and APSRs can vary significantly, the applicant requested to limit the storage of these non-fuel hardware devises to the innermost four locations per cask. The applicant used the SCALE 4.3, Revision 5, SAS2H module, with ORIGEN-S, to calculate the radiation source term for the CRAs and APSRs. In the ORIGEN-S calculations, the impurity level was assumed to be 0.8 g of Co-59 per kg of stainless steel, and 4.7 g of Co-59 per kg of Inconel. The applicant determined the total curies of Co, and the total source term in the 0.3 to 1.0 MeV energy range (due to the activation of Ag) as a function of burnup and cooling time up to 630,000 MWd/MTU. The scaling factors were applied, as appropriate. The applicant analyzed three configurations (1) CRA and APSR at 10 percent insertion, (2) CRA and APSR fully removed, and (3) an APSR fully inserted. The results are presented in Table 5.2.34 and 5.2.35 of the applicant's FSAR.

The applicant then compared the results of all data, and determined, as presented in Section 5.4.6 of the FSAR, that the BPRAs resulted in the highest dose rate increase on the radial surfaces of the cask, while the APSR resulted in the highest dose rate increase on the bottom of the cask. Since the increase in dose rate at the bottom of the cask does not significantly affect occupational exposures, the additional dose rate from the BPRAs was included in the design basis analysis presented in Section 5.1 of the applicant's FSAR. The occupational dose rates estimates found in Chapter 10 of the FSAR were also revised to include the dose rate contribution from BPRAs. However, the controlled area boundary dose rate analysis was not revised to include the effect of BPRAs because the analysis had been performed with a bounding burnup of 52,500 MWd/MTU, and cooling time of 5 years.

Based upon the review of the applicant's analysis, the staff agrees that design basis BPRA will result in the highest dose rate increase on the radial surface of the cask, and generally bounds the source term produced from other non-fuel hardware. For the staff's confirmatory calculations of the HI-STORM 100S cask, the high burnup fuel, and the new MPC-32, the design basis BPRA was used for the analyses. The staff's results of the confirmatory calculations to determine the additional source term and activity from the design basis BPRA are in close agreement with the applicant's results. The staff used the SCALE 4.4, Revision 6, SAS2H computer code and the accompanying 44-group cross-section library.

The staff concludes that the following non-fuel hardware are acceptable for storage within the MPC-24, MPC-24E, MPC-24EF, and MPC-32, in accordance with the appropriate burnup/cooling times specified in Table 2.1-8 of Appendix B of the CoC:

- BPRAs, WABAs, TPDs, orifice rod assemblies, water displacement guide tube plugs. Permitted for storage in any location in the MPC.

- CRAs, APSRs, CEAs, RCCAs. Permitted for storage in only the innermost four locations of the MPC.

5.8 Addition of Dresden Unit 1 Thoria Rod Canister

The applicant requested the addition of the Dresden Unit 1 Thoria Rod Canister to the HI-STORM approved contents. The Dresden Unit 1 Thoria Rod Canister was requested for storage in the MPC-68 and the MPC-68F. The canister contains up to 18 thoria rods which have a maximum burnup of 16,000 MWD/MTIHM and a minimum cooling time of 18 years. The applicant used the SCALE 4.3, Revision 5, SAS2H module, with ORIGEN-S, to calculate the source terms. The thoria rod source terms, listed in FSAR Tables 5.2.37 and 5.2.38, are bounded by the source terms for the design basis BWR fuel in all neutron groups and in all gamma groups except in the 2.5-3.0 MeV group. To demonstrate that the gamma dose rate from the thoria rods is bounded by the design basis fuel, the applicant calculated the gamma dose rate on the radial surface of a 100 ton HI-TRAC and HI-STORM cask completely filled with the thoria rods. The results were compared to the gamma dose rate of a cask filled with design basis BWR fuel. The gamma dose of the design basis BWR fuel bounded that of the 100 ton HI-TRAC completely filled with thoria rods. However, the HI-STORM cask completely filled with thoria rods had a gamma dose rate 17 percent higher than the design basis BWR fuel.

The staff review found the applicant's analyses acceptable, since there are sufficient conservatisms built into the analysis. Licensees will only be permitted to store one Thoria Rod Canister per MPC-68 or MPC-68F, rather than an entire cask load, as was analyzed. Based upon the review of the applicant's analysis, the staff agrees that the Thoria Rod Canister, with up to 18 thoria rods per canister and the burnup and cooling time limits given above, is acceptable for storage in the MPC-68 or MPC-68F.

5.9 Addition of Antimony-Beryllium (Sb-Be) Source in Dresden Unit 1 Fuel Assemblies

The applicant requested the addition of Dresden Unit 1 fuel assemblies containing an Sb-Be source to the HI-STORM approved contents. The Dresden Unit 1 assemblies containing an Sb-Be source were requested for storage in the MPC-68, MPC-68F and the MPC-68FF.

Beryllium produces neutrons through gamma irradiation, with the antimony (Sb-124) used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.66

MeV. Even though all of the initial Sb-124 has decayed away (18 years cooled), there are two other gamma sources which can exceed the energy threshold, and produce neutrons from the Sb-Be source, (a) gammas from fission product decay of the assemblies, and (b) gammas from Sb-124 being produced from neutron activation in the MPC.

The applicant estimated the gammas emanating from the aforementioned sources by using MCNP to explicitly analyze a 6x6 design basis BWR fuel assembly with a Sb-Be source. The reduction of antimony and beryllium during burnup was conservatively neglected. The neutron source was then calculated. Since the Sb-Be source is only 77.25 inches long, the applicant compared the neutron sources from the 6x6 design basis BWR fuel containing an Sb-Be source to the 7x7 design basis BWR fuel, on a per inch basis. As shown in Table 5.4.18, the 6x6 fuel with the Sb-Be neutron source is bounded by the 7x7 fuel. The applicant also considered the gamma source due to activation of the source's stainless steel cladding, which was shown to be bounded by the design basis fuel.

The staff reviewed the applicant's analysis and agrees that the Dresden Unit 1 fuel assemblies containing an Sb-Be source is bounded by the current shielding analysis for the design basis fuel, and is acceptable for storage in the MPC-68, MPC-68F, and the MPC-68FF.

5.10 MPC-24E

The applicant requested the addition of the MPC-24E basket design to the approved contents. The basket design provides for the storage of higher enriched fuel, and the storage of four damaged PWR fuel assemblies in generic PWR DFCs. The damaged PWR DFC will be stored in the periphery locations of the basket. The MPC-24E has essentially the same shielding properties as the MPC-24. The applicant determined that the shielding analysis of the contents approved for the MPC-24 bounds the contents for the MPC-24E.

Based upon the review of the applicant's analysis, the staff agrees that use of the MPC-24E is bounded by the current shielding analysis and that further evaluation is not required.

5.11 MPC-32

The applicant requested the addition of the MPC-32 canister design to the approved contents. The canister design provides for the storage of 32 intact PWR spent fuel assemblies.

The applicant modeled the new basket design, using MCNP-4A to calculate dose rates, and SCALE 4.3, Revision 5, SAS2H module, with ORIGEN-S to calculate the source terms. The same shielding methodology as used in the original application, is also used for this amendment request (i.e., gamma source terms, as in the original application, were calculated using all gammas with energies in the range of 0.45 to 3.0 MeV, etc.).

The applicant calculated numerous dose rates, broken down into gamma and neutron components all around the storage and transfer casks and at inlet and outlet vents, for surface doses and 1 meter doses. The complete list of those values can be found in Chapter 5 of the FSAR. Only the total dose rate for radial midpoint and axial center, for surface and 1 meter are providede in this SER for illustration.

The total dose rates, in mrem/hr, for an MPC-32 inside a HI-STORM 100S storage cask are:

	axial center surface	axial center @ 1 m	radial midplane surface	radial midplane @ 1 m
Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs - burn: 45 GWd/MTU cool: 5 yrs enrich: 3.6 wt.%	6.16	2.16	45.15	23.44
Fully loaded with design basis stainless steel clad PWR fuel (WE 15x15) - burn: 40 GWd/MTU cool: 9 yrs enrich: 3.4 wt.%	--	--	39.08	19.57

The axial center surface and 1 meter dose rates were not calculated for design basis stainless steel clad fuel because the end fittings are assumed to be the same mass as the end fittings for the Zircaloy clad fuel. Additionally, for the stainless steel fuel, the burnup is lower, and the cooling time is longer when compared to the Zircaloy clad fuel.

The total dose rates for an MPC-32 inside a 100-ton HI-TRAC transfer cask, in mrem/hr, are:

	axial bottom transfer lid center surface	axial bottom transfer lid center @ 1 meter	radial midplane surface	radial midplane @ 1 m
Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs - burn: 32.5 GWd/MTU cool: 5 yrs enrich: 2.9 wt.%	5201.86	2067.33	1212.20	532.82
Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs - burn: 45 GWd/MTU cool: 10 yrs enrich: 3.6 wt.%	3726.61	1416.91	827.28	353.53

The calculated dose rates at the bottom of the TC are high when compared to other locations on the TC. Therefore, licensees need to monitor occupational doses more frequently during operations that expose the TC bottom to workers. Additionally, the applicant provided acceptable occupational dose rates for each iteration of loading and unloading operating procedures. During loading and unloading operations, occupational radiation workers must be in compliance with 10 CFR 20, and must use basic ALARA practices, i.e., minimizing the amount of time the MPC is in the transfer cask, maintaining far distances from the transfer cask, and using auxiliary shielding during operations. Regardless of the high dose rates the applicant calculated in the FSAR, the NRC staff notes that the dose rate limits for the side and top of the transfer cask are only 1500, and 315 mrem/hr, respectively, as reported in TS LCO 3.2.1.

The staff reviewed the applicant's analysis, and found it to be acceptable. The staff also performed independent calculations to confirm the applicant's major conclusions. The staff analyzed an MPC-32 inside a HI-STORM 100S cask using the SCALE 4.4, Revision 6, SAS4 code to calculate dose rates, and the SCALE 4.4, Revision 6, SAS2H module to calculate source terms. The staff's model assumed the cask was fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burned to 45 GWd/MTU, enriched to 3.6 wt.%, and cooled for 5 years. The staff's source terms were in very close agreement with the applicant's source terms. The staff did not explicitly model each fuel assembly, but rather homogenized the fuel, cladding, spacers, basket, etc., as one cylindrical source. Therefore, the staff's dose rates were slightly higher than the applicant's dose rates, but were still in close agreement.

The spent fuel burnup values requested by the applicant for the MPC-32 are lower than that for the MPC-24, resulting in a higher radial dose on the surface of the HI-STORM cask for the MPC-24. Therefore, the MPC-24 was used for the site-boundary evaluation to demonstrate compliance with 10 CFR 72.104.

Based upon the staff's confirmatory calculations, and review of the applicant's analyses, the NRC staff found the addition of the MPC-32 to the approved contents to be acceptable.

5.12 MPC-68FF

The applicant requested the addition of the MPC-68FF canister design to the approved contents. The canister design provides for the storage of sixteen damaged BWR fuel assemblies in generic BWR DFCs, with eight of those being fuel debris. The damaged and debris BWR DFCs will be stored in the periphery locations of the basket. The MPC-68FF also provides for the storage of intact, damaged, and debris MOX fuel, Dresden Unit 1 fuel assemblies containing an Sb-Be source, and La Crosse BWR fuel assemblies in stainless steel channels. The MPC-68FF has essentially the same shielding properties as the MPC-68. The applicant determined that the shielding analysis of the contents approved for the MPC-68 is applicable for the contents for the MPC-68FF.

Based upon the review of the applicant's analysis, the staff agrees that use of the MPC-68FF is bounded by the current shielding analysis and further evaluation is not required.

5.13 MPC-24EF

The applicant requested the addition of the MPC-24EF canister to the approved contents. The canister design provides for the storage of four damaged PWR spent fuel assemblies, or four fuel debris in generic PWR DFCs. The damaged PWR DFC will be stored in the periphery locations of the basket. The MPC-24EF has essentially the same shielding properties as the MPC-24. The applicant determined that the shielding analysis of the contents approved for the MPC-24 bounds the contents for the MPC-24EF.

Based upon the review of the applicant's analysis, the staff agrees that use of the MPC-24EF is bounded by the current shielding analysis and further evaluation is not required.

5.14 Revision of Uranium Masses

The applicant requested an increase in the maximum allowed uranium masses for certain fuel assemblies. Increasing the uranium mass will result in an increase in the neutron and gamma source term, and decay heat for a specified burnup and cooling time. The applicant has conservatively analyzed design basis fuel in Chapter 5 using uranium masses which are higher than what is being requested in Appendix B of the CoC.

The staff agrees that the masses may be increased as requested and further evaluation is not required since these values are below the design basis values used in the shielding analysis.

The design basis decay heat was used to determine the burnup and cooling times, found in Appendix B of the CoC, for the non-design basis fuel assemblies. However, the applicant calculated the burnup and cooling times without consideration for the decay heat from the non-fuel hardware.

The staff concludes that this is acceptable, because the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in the Appendix B of the CoC regardless of the heat source. Additionally, the actual decay heat from the non-fuel hardware is expected to be minimal.

5.15 Revision of Fuel Assembly Parameter Limits

The applicant requested minor changes to certain fuel assembly parameter limits such as cladding thickness and guide tube/water rod thickness. These changes do not impact the shielding analysis.

The staff agrees that the proposed dimensional changes have a negligible impact on the shielding analysis and further evaluation is not required.

5.16 Addition of New Fuel Assembly Array Classes

The applicant requested the addition of four new fuel assembly array classes to the HI-STORM approved contents, the PWR 14x14E and 15x15H, and the BWR 8x8F and 9x9G. These assemblies are very similar to currently approved fuel assemblies and the uranium masses are bounded by the design basis fuel assemblies. The burnup and cooling times are also the same as previously analyzed. These assemblies are bounded by the current shielding analysis for the design basis fuel assemblies.

Based upon the review of the information presented in the application, the staff agrees that these assembly array classes are bounded by the design basis fuel and further evaluation is not required.

5.17 Addition of High Burnup Fuel

The applicant requested the addition of “high burnup” fuel assemblies, which are those burned to greater than 45 GWd/MTU. For those increased burnups, the applicant requested corresponding increased cool times, to maintain dose rates and heat loads at acceptable levels.

The applicant revised all shielding calculations in Chapters 5 and 10 of the FSAR to reflect the increased burnups. As a result of the increased dose rates, the dose rate limits specified in Technical Specifications LCOs 3.2.1, and 3.2.3 have been increased.

The same shielding methodology used in the original application, is also used for this amendment request (i.e., the same computer codes were utilized, etc.). To justify the use of SAS2H for high burnup fuel, the applicant included seven references in Chapter 5 of the FSAR which present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 and 57 GWd/MTU, respectively. The referenced studies provide evidence that there is good agreement between SAS2H and measured data.

The applicant calculated numerous dose rates, broken down into gamma and neutron components all around the storage and transfer casks and at inlet and outlet vents, for surface doses and 1 meter doses. The complete list of those values can be found in Chapter 5 of the FSAR. Only the maximum dose rate, and its location, is provided in this SER for illustration. The applicant re-analyzed the following configurations:

MPC-24 inside a HI-STORM 100 storage cask:

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 52.5 GWd/MTU; cool: 5 yrs; enrichment: 3.9 wt.%. The highest dose rate was found to be 45.77 mrem/hr at the radial surface midpoint.

- Controlled area boundary array calculation, fully loaded with design basis PWR fuel (B&W 15x15), burn: 52.5 GWd/MTU; cool: 5 yrs; enrichment: 3.9 wt.%. The highest dose rate was found to be 23.83 mrem/year at 250 meters for a 2x2 array of casks, for a 8760 hour occupancy. The applicant noted that the actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 5 year cooling time, as specified in the CoC, is lower than the burnup used for this analysis. The dose rate at 100 meters for one cask is 130.0 mrem/year, for a 8760 hour occupancy.

MPC-68 inside a HI-STORM 100S storage cask:

- Fully loaded with design basis BWR fuel (GE 7x7), burn: 47.5 GWd/MTU; cool: 5 yrs; enrichment: 3.2 wt.%. The highest dose rate was found to be 36.80 mrem/hr at the radial surface midpoint.

- The applicant only analyzed the MPC-24 for controlled area boundary calculations because the MPC-24 has a higher allowable burnup fuel than the MPC-68 and the MPC-32. Therefore, the staff agrees for the allowable burnup and cooling times the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 and MPC-32.

MPC-24 inside a 100 ton HI-TRAC transfer cask:

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 42.5 GWd/MTU; cool: 5 yrs; enrichment: 3.4 wt.%. The highest dose rate was found to be 4622.29 mrem/hr at the axial bottom center surface of the pool lid.

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 52.5 GWd/MTU; cool: 10 yrs; enrichment: 3.9 wt.%. The highest dose rate was found to be 3846.35 mrem/hr at the axial bottom center surface of the pool lid.

- Flooded conditions, with empty neutron shield, fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 42.5 GWd/MTU; cool: 5 yrs; enrichment: 3.4 wt.%. The highest dose rate was found to be 1418.98 mrem/hr at the axial bottom center surface of the pool lid.

- Flooded conditions, with full neutron shield, fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 42.5 GWd/MTU; cool: 5 yrs;

enrichment: 3.4 wt.%. The highest dose rate was found to be 1417.69 mrem/hr at the axial bottom center surface of the pool lid.

- Accident conditions, with four DFCs, remainder loaded with design basis PWR fuel (B&W 15x15), burn: 42.5 GWd/MTU; cool: 5 yrs; enrichment: 3.4 wt.%. The highest dose rate was found to be 1123.53 mrem/hr at the radial surface midpoint.

- At 1 meter, accident conditions, with empty neutron shield, fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 57.5 GWd/MTU; cool: 12 yrs; enrichment: 4.2 wt.%. The highest dose rate was found to be 2083.22 mrem/hr at the radial surface midpoint.

MPC-68 inside a 100 ton HI-TRAC transfer cask:

- Fully loaded with design basis BWR fuel (GE 7x7), burn: 40 GWd/MTU; cool: 5 yrs; enrichment: 3.0 wt.%. The highest dose rate was found to be 5260.69 mrem/hr at the axial bottom center surface of the transfer lid.

- Fully loaded with design basis BWR fuel (GE 7x7), burn: 50 GWd/MTU; cool: 10 yrs; enrichment: 3.6 wt.%. The highest dose rate was found to be 3847.94 mrem/hr at the axial bottom center surface of the pool lid.

- Accident conditions, with sixteen DFCs, remainder loaded with design basis BWR fuel (GE 7x7), burn: 40 GWd/MTU; cool: 5 yrs; enrichment: 3.0 wt.%. The highest dose rate was found to be 1322.43 mrem/hr at the radial surface bottom, under the water jacket.

MPC-24 inside a 125 ton HI-TRAC transfer cask:

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 57.5 GWd/MTU; cool: 12 yrs; enrichment: 4.2 wt.%. The highest dose rate was found to be 1247.7 mrem/hr at the axial bottom center surface of the pool lid.

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 42.5 GWd/MTU; cool: 5 yrs; enrichment: 3.4 wt.%. The highest dose rate was found to be 990.61 mrem/hr at the axial bottom center surface of the pool lid.

- At 1 meter, accident conditions, fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 57.5 GWd/MTU; cool: 12 yrs; enrichment: 4.2 wt.%. The highest dose rate was found to be 1317.24 mrem/hr at the radial surface midpoint.

The staff reviewed the applicant's analysis, and found it to be acceptable. The staff also performed independent calculations to confirm the applicant's major conclusions. The staff analyzed the following configurations:

MPC-24 inside a HI-STORM 100 storage cask:

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 52.5 GWd/MTU; cool: 5 yrs; enrichment: 3.9 wt.%.

MPC-68 inside a HI-STORM 100S storage cask:

- Fully loaded with design basis BWR fuel (GE 7x7), burn: 47.5 GWd/MTU; cool: 5 yrs; enrichment: 3.2 wt.%.

MPC-24 inside a 100 ton HI-TRAC transfer cask:

- Fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 42.5 GWd/MTU; cool: 5 yrs; enrichment: 3.4 wt.%.

- Accident conditions, with empty neutron shield, fully loaded with design basis PWR fuel (B&W 15x15), and design basis BPRAs, burn: 57.5 GWd/MTU; cool: 12 yrs; enrichment: 4.2 wt.%.

The staff used the SCALE 4.4, Revision 6, SAS4 module to calculate dose rates, and the SCALE 4.4, Revision 6, SAS2H module to calculate source terms. The staff's source terms for all configurations were in very close agreement with the applicant's source terms. For each configuration, the staff did not explicitly model each fuel assembly, but rather homogenized the fuel, cladding, spacers, basket, etc., as one cylindrical source. Therefore, the staff's dose rates were slightly higher than the applicant's dose rates, but were still in close agreement.

While the dose rates around the casks, and the occupational dose rates to the workers have increased, the applicant has shown that the HI-STORM 100 system is in compliance with 10 CFR 72.104 and 10 CFR 72.106.

5.18 HI-STORM 100S

The applicant requested the addition of an alternate configuration of the currently approved cask design, HI-STORM 100S. The significant differences in the HI-STORM 100, and the HI-STORM 100S are (1) the MPC has been moved closer to the upper and lower air ducts, resulting in a local increase in dose rates at the opening of the ducts; (2) the "inner gamma shield" has been removed from the inside of the inner shell of the cask, and replaced with higher density concrete; and (3) the lid design has been changed by moving the concrete shielding from below the 4 inch thick steel, to above the 4 inch thick steel plate in the top lid. The radial shielding is the same between the HI-STORM 100 and the HI-STORM 100S. The HI-STORM 100S has mandatory gamma shield cross plates which are inserted into the inlet and outlet vent ducts during storage.

The applicant, and the NRC staff, modeled the MPC-32 and MPC-68 inside the HI-STORM 100S, and modeled the MPC-24 inside the HI-STORM 100. The HI-STORM 100 cask was still used for the controlled area boundary dose rate calculations, because the significant difference between the two cask designs is a localized effect at the duct opening, which would not affect the off-site dose.

Based upon the review of the applicant's new modified cask design, the staff finds that the dose rates surrounding HI-STORM 100S, when loaded with design basis fuel, has been adequately analyzed and is acceptable for storage.

5.19 Evaluation Findings

F5.1 Chapter 5 of the FSAR sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F5.2 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.

F5.4 The staff concludes that the design of the radiation protection system of the HI-STORM 100, as modified, 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 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.

5.20 References

1. DOE/RW-0184, "Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Waste Which May Require Long-Term Isolation," US Department of Energy, December 1987.

2. DOE/RW-0184-R1, "Characteristics Database System LWR Assemblies," US Department of Energy, July 1992

6.0 Criticality Evaluation

The applicant requested several changes to the HI-STORM-100 system design and Certificate of Compliance (CoC). Only those changes that may affect the system criticality safety are discussed in this section. The staff reviewed Revision 1 of the HI-STORM 100 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 system, as revised, meets the following regulatory requirements: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g). Revision 1 of the FSAR was also reviewed to determine whether the cask system fulfills the acceptance criteria listed in Section 6 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

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

a. Storage of spent fuel with a higher initial enrichment in HI-STORM's 24 PWR Multipurpose Canister (MPC-24) by using borated water during loading/unloading (proposed changes #5, 23).
b. Addition of four new fuel assembly array classes to the approved contents (proposed change #27).
c. Addition of three new damaged fuel containers (proposed change #10).
d. Addition of four new fuel baskets to the HI-STORM-100 system; the MPC-24E, MPC-24EF, the MPC-32, and the MPC-68FF (proposed changes #5, 19, 20, 21, 22, 23, 29).
e. Revision of certain CoC definitions important to criticality safety (proposed change #9).
f. Revision of some fuel assembly parameters (proposed changes #24, 25, 26).
g. Storage of non-fuel hardware (proposed change #14).
h. Storage of additional damaged BWR fuel types in the MPC-68 (proposed change #15).
i. Inclusion of one Dresden Unit-1 thoria rod canister containing 18 thoria rods in the MPC-68 and MPC-68F (proposed change #16).
j. Inclusion of Dresden Unit-1 fuel with one antimony-beryllium source in the assembly in the MPC-68 and MPC-68F and MPC-68FF (proposed change #17).

The staff's conclusions, summarized below, are based on information provided in amendment number 1, as revised, to the HI-STORM FSAR.

6.1 Storage of Spent Fuel With a Higher Initial Enrichment in the MPC-24

The MPC-24 is currently approved to store intact PWR fuel assemblies with a U-235 enrichment of up to 4.6 weight percent (wt%), depending on the fuel type, as specified in the Technical Specifications (TS). This limitation is due in part to the assumption that fresh water is used during loading/unloading operations. The applicant requested storage of fuel enriched up to 5 wt% U-235 in the MPC-24. The enrichment can be increased by using borated water (minimum 400 ppm boron) during MPC loading/unloading. The applicant performed calculations with the MPC-24 completely loaded with spent fuel with an initial U-235 enrichment of 5 wt%. The applicant's calculations, summarized in Table 6.1.2, show that the MPC-24, loaded with the proposed fuel types, will meet the design criterion of $k_{eff} \leq 0.95$ when using borated water containing a minimum of 400 ppm boron. The proposed CoC allows loading/unloading of spent fuel in the MPC-24 using either borated water or unborated water, depending on the assembly type and the initial enrichment.

A new TS Limiting Condition of Operation (LCO) 3.3.1 was also proposed by the applicant. This LCO requires a minimum boron concentration in the MPC water when loading specific fuel types to ensure that k_{eff} is $\leq$ 0.95. This applies when one or more assemblies to be loaded has an initial enrichment greater than the value in CoC Appendix B Table 2.1-2 for loading without borated water. Two independent verifications of the minimum soluble boron content in the MPC water is required to ensure that the requirements of 10 CFR 72.124(a) is met.

Since borated water was not included in the original application, the code validation was revised to consider the soluble boron. The applicant performed benchmark calculations on selected critical experiments, chosen, as much as possible, to bound the range of boron concentration in the MPCs (300 ppm to 2600 ppm). The experiments chosen had a boron concentration from 0 to 2550 ppm boron. Results show that there are no trends in the bias. The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations.

The staff reviewed the applicant's model descriptions and assumptions and agrees that they are consistent with the description of the cask and contents given in FSAR Chapters 1 and 2. The staff also reviewed the applicant's revision to the benchmark analysis and agrees that the critical experiments chosen for the soluble boron are relevant to the cask design. The staff performed independent confirmatory calculations for the MPC-24. The staff's results are in close agreement with the applicant's results.

6.2 Addition of four new fuel assembly array classes

The applicant requested the addition of four new assembly array classes to the allowable contents; 14x14E, 15x15H, 8x8F, and 9x9G. The 14x14E and 15x15H assembly array classes are to be stored in the MPC-24, MPC-24E, MPC-24EF, and MPC-32. The 8x8F and 9x9G assembly array classes are to be stored in the MPC-68 and MPC-68FF only. The fuel assembly characteristics are given in Tables 6.2.2 and 6.2.1, respectively, of the amendment request. The applicant performed calculations which show that the HI-STORM 100 will meet the design criterion of $k_{eff} \leq 0.95$ when loaded with these new spent fuel types.

The staff reviewed the applicant's assembly 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 staff also performed independent confirmatory calculations for the new assembly classes. The staff's results are in close agreement with the applicant's results.

6.3 Addition of three new damaged fuel containers

The applicant requested the addition of three new damaged fuel containers (DFCs) to the HI-STORM-100 system; the Transnuclear TN/D-1 DFC containing Dresden Unit-1 fuel only, the Holtec generic PWR DFC, and the Holtec generic BWR DFC. Sketches of these DFCs are shown in figures 2.1.2, 2.1.2.B, and 2.1.2.C of the FSAR. These DFCs are used to store damaged fuel or fuel debris. The applicant performed calculations that showed that the HI-STORM 100 will meet the design criterion of $k_{eff} \leq 0.95$ when loaded with these new DFCs as specified in the proposed TS.

Due to the design of the DFCs, the applicant also considered preferential flooding. The 250x250 fine mesh screens used in the DFCs could cause uneven draining of the DFCs with respect to the canister. The applicant's evaluation shows that the HI-STORM 100 will meet the design criterion of $k_{eff} \leq 0.95$ even if there is up to 12 inches of water left in the DFCs after draining of the canister.

6.3.1 TN/D-1 Damaged Fuel Container

The applicant requested storage of the TN/D-1 DFC already loaded with Dresden Unit 1 fuel in the MPC-68, MPC-68F, and MPC-68FF. Refer to FSAR figure 2.1.2 for the DFC dimensions. These are the same contents approved for the Holtec DFC. The TN/D-1 is similar to the previously approved Holtec DFC design, except that it is slightly smaller. The applicant performed analyses which show that the TN/D-1 may be used to store damaged 6x6 and 7x7 BWR assemblies and dispersed fuel powder (68F and 68FF only). The allowed number of DFCs are specified in CoC Appendix B. The results of the analyses are given in Table 6.4.5 of the amendment request and show that the TN/D-1 DFC loaded with the proposed assemblies is less reactive or statistically the same as the Holtec Dresden Unit 1/Humboldt Bay DFC.

6.3.2 Holtec Generic PWR DFC

The applicant requested the use of the generic PWR DFC in the MPC-24E and MPC-24EF. Refer to figure 2.1.2.C for the DFC dimensions. The applicant performed analyses which show that the generic PWR DFC may be used to store damaged PWR fuel and fuel debris (MPC-24EF only) with a maximum enrichment of 4.0 wt% and up to four DFCs per cask. The results of the analysis are given in Section 6.4.4.2 of the amendment request.

6.3.3 Holtec Generic BWR DFC

The applicant requested the use of the generic BWR DFC in the MPC-68, and MPC-68FF only. Refer to figure 2.1.2.B for the DFC dimensions. The applicant performed analyses which show that up to 16 generic BWR DFCs, each containing a damaged fuel assembly or fuel debris (MPC-68FF only) may be loaded into the MPC-68 and MPC-68FF. The other basket cells may not contain a generic BWR DFC. The maximum planar average enrichment of the fuel and the number of allowed DFCs are specified in CoC Appendix B. The results of the analysis are discussed in Section 6.4.4.2 of the amendment request.

6.3.4 Staff Review of the New Damaged Fuel Containers

The staff verified that the amendment request contains figures and tables that are sufficiently detailed to support an in-depth staff evaluation of the DFCs. The staff also reviewed the applicant's analysis and results and has reasonable assurance that the HI-STORM 100 will meet the design criterion of $k_{eff} \leq 0.95$ when using these DFCs .

6.4 Addition of Four New Fuel Basket Designs

The applicant requested the addition of four new basket designs to the HI-STORM-100 system; the MPC-24E, MPC-24EF, MPC-32, and MPC-68FF. Sketches of the criticality models for the new baskets are shown in Sections 6.3 and 6.4 of the amendment request. In addition to the fuel parameters important to criticality safety, the HI-STORM 100 cask system design features relied upon to prevent criticality for these new baskets are the fuel basket geometry, permanent neutron-absorbing Boral panels, and minimum soluble boron concentration in the water used in the MPC.

The applicant performed calculations which show that the HI-STORM 100, when using these new baskets as proposed in the TS, will meet the design criterion of $k_{eff} \leq 0.95$. Additional analyses were not required for the MPC-68FF since analyses were performed for the MPC-68, including storage of fuel debris, and the two baskets are identical. The difference between the MPC-68 and MPC-68FF is the top shell which does not impact the criticality safety analyses.

6.4.1 MPC-24E and MPC-24EF Basket Designs

The MPC-24E and MPC-24EF designs are similar to the MPC-24 basket. The baskets are designed such that the Boral (neutron poison) panels are fixed to the fuel cell walls, as in the MPC-24. The fuel cell storage configuration, flux trap size, and Boral B-10 loading have been modified. These modifications allow storage of higher enriched fuel and up to four damaged PWR fuel assemblies in damaged fuel containers (DFCs). Also, for the MPC-24EF, the top shell in the closure lid region has been thickened which allows storage of fuel debris (in DFCs). The larger lid-to-shell weld and the thickened top shell are for confinement purposes and do not impact criticality safety. These baskets are otherwise identical. For the MPC-24E and MPC-24EF, CoC Appendix B Design Feature 3.2 requires a minimum flux trap size of 0.776 inches for the four DFC cells and 1.076 inches for all other cells, and a minimum ^{10}B content of 0.0372 g/cm^2 in the Boral panels. The MPC-24E and MPC-24EF were analyzed with two different ranges of fuel enrichment. The lower enrichment range is analyzed using unborated water during loading/unloading and the higher enrichment range (up to 5 wt%) is analyzed using borated water containing a minimum of 300 ppm boron. For the soluble boron requirement, LCO 3.3.1 was added to the technical specifications. Refer to Section 6.1 above for further discussion of this LCO.

6.4.2 MPC-32 Basket Design

The MPC-32 is designed such that Boral panels are fixed to the fuel cell walls, as in the MPC-24. However, the MPC-32 does not utilize flux traps, thus it has a higher capacity of 32 intact PWR assemblies. Borated water containing a minimum of 1900 ppm boron must be used during all loading/unloading operations. The MPC-32 was analyzed with two different ranges of fuel enrichment. During loading/unloading operations, the lower enrichment range is analyzed using borated water containing 1900 ppm boron and the higher enrichment range (up to 5 wt%) is analyzed using borated water containing 2600 ppm boron.

The MPC-32 is limited to intact fuel only. CoC Appendix B Design Feature 3.2 requires a minimum fuel cell pitch of 9.158 inches and a minimum ^{10}B content of 0.0372 g/cm^2 in the

Boral panels. For the soluble boron requirement, LCO 3.3.1 was added to the technical specifications. Refer to Section 6.1 above for further discussion of this LCO.

6.4.3 MPC-68FF Basket Design

The MPC-68FF combines the MPC-68 basket with the thickened top shell of the MPC-68F, both previously approved by the staff. Thus, the fuel cell pitch, basket wall thickness, and B-10 loading are identical to the MPC-68. The MPC-68FF is used to store 68 BWR assemblies, including damaged fuel or debris as limited in CoC Appendix B. The Holtec generic BWR DFC (up to a maximum of 16 per cask) may be used in the MPC-68FF which allows storage of a wider range of damaged fuel or fuel debris types than the MPC-68F. The MPC-68F is limited to storage of Dresden Unit 1 or Humboldt Bay damaged fuel or fuel debris only. For the MPC-68FF, the MPC-68 basket, which has a higher boron content in the Boral panels than the MPC-68F, is combined with the MPC-68F top shell, which is required for storage of fuel debris. CoC Appendix B Design Feature 3.2 requires a minimum cell pitch of 6.43 inches and a minimum ^{10}B content of 0.0372 g/cm^2 in the Boral panels which is identical to the MPC-68 requirements.

6.4.4 Staff Review of the New Basket Designs

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

The staff also verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety.

However, the staff notes that the applicant did not always consider the tolerances when modeling the boral plates in the basket. Use of bounding tolerance values is consistent with the SRP, thus the staff disagrees with the applicant’s method of using nominal boral widths in the criticality models discussed above.

The staff performed independent confirmatory calculations using the minimum boral widths. The staff’s results are in close agreement with the applicant’s results. Therefore, the criticality calculations were relatively insensitive to nominal versus bounding tolerance values

6.5 Revision of Definitions Important to Criticality Safety

The definitions of Damaged Fuel Assembly and Intact Fuel Assembly were revised to clarify that any assembly without fuel rods in all of the specified fuel rod locations will either have dummy rods inserted or will be classified as damaged fuel. This is consistent with the criticality safety calculation and is considered an editorial change only.

6.6 Revision of fuel assembly parameter limits for certain fuel assemblies

6.6.1 Increase in Uranium Mass

The applicant requested an increase in the design initial uranium masses for selected fuel assemblies to provide consistency with the shielding analysis. This increases the uranium masses for certain non-design basis fuel assemblies to the mass used as the design basis fuel within each class. The staff concludes that, given the bounding fuel assembly dimensions defined in the current and proposed certificate, these increases in the uranium mass will not increase the overall system reactivity.

6.6.2 Increase in U-235 Enrichment for 6x6B Assembly Array Class

The applicant also requested an increase in the allowable U-235 enrichment from 0.612 to 0.635 wt% for the 6x6B assembly array class. The staff previously reviewed and approved this change during review of the HI-STAR Amendment #1 which uses the MPC-68 and MPC-68F. The MPC-68 and MPC-68F design features important to criticality safety have not been changed. The staff found that there was an increase in reactivity from 0.7611 to 0.7824. Since the MPC-68FF is identical to the MPC-68 except for the top shell, further staff review is not necessary.

6.6.3 Miscellaneous Revisions to Assembly Parameters

The applicant requested revisions to various other assembly parameter limits for selected fuel assemblies. The revised fuel parameters are given in proposed FSAR Tables 6.2.1 and 6.2.2. The applicant explicitly modeled each assembly type and the calculational results show that the HI-STORM-100 will continue to meet the design criterion of $k_{eff} \leq 0.95$.

The staff reviewed the applicant's assembly model descriptions and assumptions and agrees that they are consistent with the description of the contents given in Chapters 1 and 2. The staff reviewed the proposed CoC changes to ensure that the fuel specifications important to criticality safety are included. The staff also performed independent confirmatory calculations, using the various PWR MPCs. The staff's results are in close agreement with the applicant's results.

6.7 Inclusion of Non-Fuel Hardware

The applicant requested approval to store Burnable Poison Rod Assemblies (BPRAs), Thimble Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet annular burnable absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), and Control Element Assemblies (CEAs) in the PWR MPCs. For PWR MPCs using unborated water, the reactivity is lower due to the reduction in the amount of moderator in the MPC. For PWR MPCs using borated water, the reactivity may increase due to the reduction in the amount of poison (B-10) available. To account for this, the applicant performed calculations for the casks using borated water with and without the non-fuel hardware present. The applicant assumed the non-fuel hardware was located in the guide tubes, but the instrument tubes contained borated water. The applicant's calculations show that the PWR MPCs will meet the design criterion of $k_{eff} \leq$ 0.95.

The staff reviewed the applicant's model assumptions and performed independent confirmatory calculations. The staff's results are in close agreement with the applicant, thus the staff has reasonable assurance that the HI-STORM 100 will continue to meet the design criterion of $k_{eff} \leq 0.95$ when storing the above non-fuel hardware as described above.

6.8 Storage of Additional Damaged BWR Fuel Types

The applicant requested approval to use the generic Holtec BWR DFC to store additional damaged BWR fuel assemblies in the MPC-68 and MPC-68FF only, or fuel debris in the MPC-68FF. The allowable contents are discussed in Section 6.3.3 above. The applicant performed calculations that showed that the HI-STORM 100 loaded with up to 16 generic BWR DFCs, with the remainder of the cask loaded with either intact assemblies or damaged Dresden Unit-1 or Humboldt bay fuel, would meet the design criterion of $k_{eff} \leq 0.95$.

The staff reviewed the applicant's model assumptions, analysis, and results and has reasonable assurance that the HI-STORM 100 will continue to meet the design criterion of $k_{eff} \leq 0.95$ when storing these fuel types.

6.9 Inclusion of One Dresden Unit-1 Thoria Rod Canister

The applicant requested approval to store one thoria rod canister within the MPC-68 or MPC-68F only. A sketch of the thoria rod canister is given in figure 2.1.2.A. The thoria rod contents are described in Section 6.4.6 of the amendment request. The applicant modeled the thoria rod canister explicitly and performed an analysis for a cask filled with 68 of these canisters. The applicant calculated a k_{eff} of 0.18.

The staff previously reviewed and approved this canister and its contents for the MPC-68 and MPC-68F during review of the HI-STAR Amendment #1 and further review is not necessary. The MPC-68 and MPC-68F design features important to criticality safety have not been changed. The transfer cask used with HI-STORM, the HI-STORM overpack, and the HI-STAR overpack are constructed of different materials. Thus, the effectiveness of these materials to reflect neutrons was explicitly evaluated during the staff's initial review and licensing of the HI-STORM 100 to determine the effect on the system reactivity. The staff's analysis showed that the difference in the overpack materials and transfer cask materials do not significantly affect the system reactivity.

6.10 Inclusion Dresden Unit-1 Fuel with One Antimony-Beryllium Source

The applicant requested approval to store Dresden Unit 1 fuel assemblies containing one antimony-beryllium neutron source in the assembly lattice in the MPC-68, MPC-68F, and MPC-68FF. The source is located in the water rod location of the assembly. The source displaces water, and will not increase the system reactivity.

The staff previously reviewed and approved these contents for the MPC-68 and MPC-68F during review of the HI-STAR Amendment #1. Also, the MPC-68FF is identical to the MPC-68 except for the top shell, thus further staff review is not necessary. Refer to Section 6.9 above for further justification.

6.11 Criticality Evaluation Summary

The staff did not review or approve the methods used by the applicant in this amendment. The staff notes that the methods were consistent with the previous application, except for not including the dimensional tolerances of the boral panels. 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 calculational models in its criticality analyses. Sketches of the models are given in FSAR as discussed above. The models are based on the engineering drawings in the FSAR and considers the dimensional tolerance values, except as noted above for the boral panels. 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's confirmatory analyses were performed using the CSAS/KENO.Va modules of SCALE developed at Oak Ridge National Laboratory. The code is a standard in the industry for performing criticality analyses. The results of the staff's confirmatory calculations were in close agreement with the applicant's results.

Based on the applicant's criticality evaluation for this amendment, as confirmed by the staff, the staff has reasonable assurance that the HI-STORM system will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

6.12 Evaluation Findings

Based on the NRC staff's review of the HI-STORM 100 amendment request, the staff concludes that the HI-STORM 100, as amended, meets the acceptance criteria specified, for both intact and damaged fuel, in NUREG-1536. In addition, 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 on the design drawings to enable an evaluation of their effectiveness.

F6.2 The HI-STORM 100 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 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 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 review of confinement features and capabilities of the HI-STORM 100 cask ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that otherwise might lead to gross ruptures. This review has been completed to determine if changes to the design and contents of the HI-STORM 100, as described by the applicant in the SAR, are acceptable to the staff.

7.1 Confinement Design Characteristics

The HI-STORM 100 consists of any one of seven fully welded MPC designs. The confinement boundary includes the following: MPC shell, bottom baseplate, MPC lid (including vent and drainport cover plates), MPC closure ring, and associated welds.

All components of the confinement boundary are important to safety, Category A, as specified in the applicant's SAR Table 2.2.6. The MPC confinement boundary is designed and fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practicable.

Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal, and backfilling during MPC loading operations, and for fuel cool-down and MPC flooding during MPC unloading operations. The MPC vent and drain ports are supplied with metal-to-metal seals to prevent degradation from high temperatures and radiation fields. The vent and drain connectors allow the vent and drain ports to be operated as valves and prevent the need to "hot tap" into the penetrations during unloading operations. During fuel loading operations and subsequent MPC closure procedures, the lines used for venting or draining are routed to the plant's spent fuel pool or radioactive waste processing systems. Closure operations are to be performed inside the plant's fuel building.

External contamination control is accomplished by preventing the MPC outer surface from coming into contact with the spent fuel pool water. An inflatable seal is used to seal the annulus between the MPC shell and the HI-TRAC transfer cask. This annulus is filled with uncontaminated water, and the seal is placed at the top of the annulus and inflated. This precludes the entry of spent fuel pool water into the annulus area.

The MPC vent and drain ports are sealed by cover plates which are seal welded to the MPC lid. The applicant does not take credit for the seal provided by the vent and drain ports. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld, providing the redundant closure of the MPC vessel. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld does not result in release of radioactive material to the environment. This redundant closure satisfies the requirements of 10 CFR 72.236(e).

The MPC is designed to provide confinement of all radionuclides under normal, off-normal, and accident conditions. The closure procedure for the MPC includes the following steps: after fuel is loaded and the draining, moisture removal, and helium backfill are completed, the vent and drain port cover plates are welded to the MPC lid. The MPC closure ring is then welded to the MPC lid on the inner diameter of the of the ring and to the MPC shell on the outer diameter.

Confinement boundary welds are performed, inspected, and tested in accordance with applicable requirements of ASME Code Section III, Subsection NB. Weld inspections include a root pass and final surface liquid penetrant inspection, on multiple pass welds. Volumetric examination is also done on the welds. If volumetric examination is not performed, multi-layer liquid penetrant examination must be performed. The welds on the MPC are also helium leak tested. In addition, a hydrostatic test is performed on the MPC lid-to-shell weld to confirm the weld's structural integrity.

The MPC is designed for the storage of specified damaged fuel assemblies and fuel debris in a damaged fuel container (DFC). The applicant evaluated fuel assemblies classified as damaged fuel or fuel debris in order to add these types of fuel to the Approved Contents of the cask.

The stainless steel DFC is a welded square container with a removable lid. The container lid is designed to provide for handling of the canister. Stainless Steel wire mesh screens (250-by-250) in the top and bottom of the container are provided for draining, moisture removal, and helium backfill operations. The DFC provides no pressure retention function.

7.1.1 Changes Requested by the Applicant

Each of the changes requested by the applicant to the original approved HI-STORM 100 design are discussed by the applicant in the SAR section entitled "LAR 1014-1 Summary of Proposed HI-STORM 100 Changes." These changes are summarized below and a brief discussion on the effect of these changes on the confinement review is included. The staff's review focused on the changes that had a significant impact on the original confinement analysis presented by the applicant, and reviewed by the staff in the previous staff SER.

7.1.1.1 Regionalized Loading of Fuel Assemblies

The applicant wishes to load fuel with higher burnup along with lower burnup fuel in a prescribed arrangement, which maintains the cask below its thermal limits. The higher burnup fuel will have an increased radionuclide inventory, due to increased fission products. The applicant accounted for this in their evaluation by calculating a bounding source term for fuel with a burnup of 70,000 MWd/MTU. This will bound any fuel to be loaded in the cask.

7.1.1.2 Storage of the Dresden Unit 1 Thoria Rod Canister in the MPC-68 and MPC-68F

The Thoria Rod Canister (TRC) is designed to hold a maximum of 20 fuel rods arrayed in a 5x4 configuration. The maximum burnup of these rods is < 16,000 MWd/MTIHM. The TRC internals are designed in a honeycomb structure formed from 12 gage stainless steel plates. This honeycomb serves as an additional means of heat removal in the fuel cell space. These rods will be stored in a relatively benign thermal environment and will remain protected during long-term storage.

The confinement analyses for the MPC-68 and MPC-68F have been revised to account for new isotopes associated with the TRC. These isotopes had a negligible effect on the resulting doses, and because only one TRC will be authorized for loading, the applicant did not include these isotopes in the confinement analysis inputs or results. The staff finds this acceptable.

7.1.1.3 Addition of the MPC-24E and the MPC-24EF Designs

The MPC-24E allows for storage of higher enriched fuel than the MPC-24. In order to store higher enriched fuel, the applicant optimized the MPC storage cell layout and has taken credit for soluble boron in the MPC water for PWR fuel assemblies with higher enrichment.

The MPC-24E has a minimum free volume that is less than the MPC-24. This results in an increased concentration of radionuclides due to the smaller dilution volume. The resultant doses from the MPC-24E are presented in the SAR by the applicant, and bound the doses for the MPC-24. The staff finds this acceptable.

The MPC-24EF design will allow storage of a wide range of damaged PWR fuel or fuel debris loaded into DFCs.

7.1.1.4 Addition of the MPC-32 Design

MPC-32 allows for the storage of more PWR assemblies and eliminates the use of flux traps from the design. Credit for soluble boron is taken for storage of assemblies in the MPC-32.

The applicant presented a conservative radionuclide inventory for the MPC-32 for a design basis assembly with a burnup of 70,0000 MWd/MTU at a 5 year cooling time. The staff reviewed the MPC-32 to assure that it would meet the requirements of 10 CFR Part 72, for storage of spent fuel.

7.1.1.5 Addition of the MPC-68FF Design

This design will allow storage of a wide range of damaged BWR fuel or fuel debris loaded into DFCs.

7.1.1.6 Increased Initial Uranium Masses

The applicant has increased the allowable initial uranium masses for selected PWR and BWR fuel assemblies. This is in response to customer requests for this enhancement. Customers have fuel assemblies with higher initial uranium masses than is currently approved for storage in the HI-STORM cask. The proposed change increases the allowed uranium masses for non-design basis fuel assemblies to those used in the analysis for the design basis fuel assembly.

The applicant states that the source term will not change, and that the existing confinement analysis is bounding. The staff agrees with this conclusion.

7.1.1.7 Addition of Four New Fuel Assembly Array/Classes

The applicant is adding new fuel assembly array/classes in order to accommodate the needs of individual customers. The applicant claims that the source terms used for the existing confinement analysis bound those of the new fuel assembly array/classes. The staff agrees with this conclusion.

7.1.1.8 Increase of Per-Assembly Limits on Fuel Burnups, Cooling Time, and Decay Heat

Due to the credit that is given to the thermo-siphon effect (internal convection) the applicant is seeking increased limits on burnup, cooling time, and decay heat.

This will have the effect of increasing the source terms available for release in all MPC configurations. The applicant chose bounding fuel assemblies to account for these increased limits. The applicant also applied the recommendations in ISG-11 for determining source terms available for release in high burnup fuel by increasing the source term available for release from 1.0% to 2.5% for normal conditions and from 10% to 11.5% for off-normal conditions. This is evaluated further in this SER.

7.2 Confinement Monitoring Capability

The MPC is a seal welded pressure vessel. It has no bolted closure or mechanical seals. Because a redundant seal weld closure system is employed, no direct leakage monitoring of the closure is required.

7.3 Nuclides with Potential for Release

The quantity of radioactive nuclides postulated to be released to the environment for the proposed contents of the HI-STORM 100 and the applicable bounding calculation method have been assessed as discussed in NUREG/CR-6487, "Containment Analysis of Type B Packages Used to Ship Various Contents" The release fractions used for calculations by the applicant are consistent with NUREG/CR-6487. A design-basis leakage rate of 5 x 10-6 cm^3/sec is specified in the SAR. This leak rate was derived from an analysis that implements 10 CFR Part 71, Appendix A criteria. The leak rate was increased to 7.5 x 10-6 cm^3/sec (a 50% increase) by the applicant for their calculation of an equivalent break flow diameter for assessing offsite consequences for normal, off-normal, and accident conditions.

The applicant has determined the source term for the types of fuel assemblies to be stored in the different MPC designs. The applicant completed the analysis of spent fuel assemblies with burnups in excess of 45 GWd/MTU, to determine the source term, based on guidance provided by the staff in ISG-5 and ISG-15.

In section 4.1 of this SER, the staff has accepted the methodology presented by the applicant for the fuel cladding temperature limits. This methodology is different from that offered by the staff guidance given in ISG-15. ISG-15 states that fuels having assembly burnups exceeding 45 GWd/MTU can be safely stored if the following criteria are met: No more than 1% of the rods in the assembly have a peak cladding oxide thickness of greater than 80 micrometers, and no more than 3% of the rods in the assembly have peak cladding oxide thicknesses greater than 70 micrometers. ISG-15 also states that in their analysis, the applicant should assume that the source term of 50% of the rods with peak cladding thickness greater than 70 micrometers are available for release from the cask. While the applicant did not use the ISG-15 criteria for their analysis of fuel rod temperature limits, they did use this criteria for their confinement calculations.

The applicant determined the percentage of the source term available for release for normal and off-normal conditions using a formula which took in to account the maximum number of rods with an oxide thickness of greater than 70 micrometers. The release fractions calculated were 2.5 percent for normal, 11.5 percent for off-normal, and 100 percent for accident conditions. The staff reviewed the applicant's release fraction calculations and finds them adequate.

7.4 Confinement Analysis

The applicant then used several different inputs into a spreadsheet to calculate the postulated doses for normal, off normal and accident conditions. Inputs included releasable source term, confinement boundary leakage rate, percentage of nuclides remaining airborne, fraction of volume released, release fraction, radionuclide release rate, atmospheric dispersion factor, Dose conversion factors, occupancy time, and breathing rate. Descriptions of the applicant's assumptions for their analysis are provided in Section 7.2 of the SAR for normal and off-normal conditions and SAR Section 7.3 for accident conditions. A summary of the applicant's assumptions are provided in Table 7.1 of the SER.

Table 7.1
Analytic Assumptions for Calculating Offsite Radiological Consequences

	Normal Operating Conditions	Off-Normal Operating Conditions	Accident Conditions
% Source Term Available for Release	2.5%	11.5%	100%
Breathing Rate	3.3×10^{-4} m^3/sec	3.3×10^{-4} m^3/sec	3.3×10^{-4} m^3/sec
χ/Q	Reduction factor 50	Reduction factor 50	No Reduction Factor
Wind Speed	1 m/sec	1 m/sec	1 m/sec
Dispersion Factor	F-Stability Diffusion	F-Stability Diffusion	F-Stability Diffusion

7.4.1 Normal and Off Normal Conditions

Since the confinement boundary of the HI-STORM 100 is welded and the temperature and pressure of the MPC are within design basis limits, no discernable leakage from the MPC is credible. However, to demonstrate that the HI-STORM 100 Cask System meets the requirements of 10 CFR 72.104(a), the applicant completed an analysis of the dose rates that would be seen at a site area boundary of 100 meters if the cask were to leak at a rate of 7.5×10^{-6} atm-cm^3/sec at reference test conditions as described in Table 7.3.7 of the FSAR.

The only means of pressurizing the cask would be through decay heat, normal external heating (insolation) and release of backfill and fission gas contents from fuel rods into the MPC cavity. The applicant analyzed the pressure in the MPC cavity for normal and off-normal conditions of 1% fuel rod failure and 10% fuel rod failure, respectively, as well as full insolation and design $7.5x10^{-6}$ atm-cm^3/sec (reference conditions), design basis decay heat, and the resulting pressures were below the normal design pressure.

The applicant makes the conservative assumption that 2.5 percent of the fuel inventory is available for release under normal conditions and 11.5 percent is available for release under off-normal conditions. The initial pressure in the MPC for these evaluations was assumed to be 101.4 psi (6.90 atm).

The applicant calculated the annual dose equivalent for the whole body, thyroid, and other critical organs to an individual at the site boundary as a result of the assumed effluent release under normal and off-normal conditions. These doses are listed in Table 7.2 and are compared to regulatory limits. Doses for all the MPC designs were below regulatory limits.

7.4.2 Accident Conditions

The applicant provided an analysis of hypothetical accident conditions to show that the requirements of 10 CFR 72.106(b) are met. The applicant evaluated the consequences of a non-mechanistic ground level breach of the MPC confinement boundary. The design accident pressure of 200 psig, was increased to 225 psig for their analysis. The duration of the accident condition leakage rate was 30 days. Because the applicant considered 100% of the fuel rods failed for the accident analysis, a separate analysis of the consequences of a mis-loading on the accident dose is not necessary.

Table 7.2
Postulated Bounding Doses Compared to Regulatory Limits
(In Units of mrem, unless otherwise noted)

Postulated Doses As a Result of an Assumed Effluent Release[1]					
	PWR		BWR		Regulatory Limit (mrem)
	MPC-24, MPC-24E, MPC-24EF	MPC-32	MPC-68, MPC-68FF	MPC-68F	
10CFR72.104(a) - Normal Conditions					
TEDE	0.114	0.159	0.515	0.087	25
Critical Organ ADE (max)	0.642	0.022	2.95	0.639	75
Skin/Extremity SDE	0.000834	0.00117	0.00385	0.000631	25
10CFR72.104(a) - Off Normal					
TEDE	0.346	0.486	0.735	0.130	25
Critical Organ ADE (max)	2.91	4.08	4.07	0.704	75
Skin/Extremity SDE	0.00252	0.00353	0.00565	0.000836	25
10CFR72.106(b) - Accident					
TEDE	20.9	29.1	39.2	7.29	5 rem
Critical Organ ADE (max)	182	254	216	38.9	50 rem
Skin/Extremity SDE	0.152	0.212	0.303	0.0453	50 rem
Notes: 10. For an individual located at the site boundary 100 meters from the cask.					

The staff conducted a confirmatory audit of the applicant's calculations and confirmed compliance with the requirements of 10 CFR 72.104(a) for normal and off -normal conditions and the requirements of 10 CFR 72.106(b) for accident conditions.

7.5 Supportive Information

Supportive information and documentation provided by the applicant included justification of assumptions and analytical procedures, computer spreadsheets, drawings of the MPC confinement boundary and applicable information from documents referenced in the SAR.

7.6 Evaluation Findings

F7.1 Chapter 7 of the SAR describes confinement structures, systems, and components important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the HI-STORM 100 Cask System adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the staff's relevant temperature considerations.

F7.3 The design of the HI-STORM 100 Cask System provides redundant sealing of the confinement system closure joints using dual welds on the MPC lid and the MPC closure ring.

F7.4 The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. No instrumentation is required to remain operational under accident conditions. Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.

F7.5 The quantity of radioactive nuclides postulated to be released to the environment has been assessed as discussed above. In Section 10 of the SER, the dose from these releases is added to the direct dose to show that the Hi-STORM 100 Cask System satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.6 The confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, described in SAR Chapter 8, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F7.7 The staff concludes that the design of the confinement system of the HI-STORM 100 Cask System, as changed and updated by the applicant in the SAR, is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the 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 analysis, and acceptable engineering practices.

7.7 References

1. NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.

2. Anderson, B.L. et al. *Containment Analysis for Type-B packages Used to Transport Various Contents.* NUREG/CR-6487, UCRL-ID-124822. Lawrence Livermore National Laboratory, November 1996.

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

4. Interim Staff Guidance-11, Revision 1, “Transportation and Storage of Spent Fuel Having Burnups in Excess of 45GWd/MTU”, May 16, 2000.

5. Interim Staff Guidance-15,“Materials Evaluation”, January 10, 2001.

6. Interim Staff Guidance-5, Revision 1, “Normal Off-Normal, and Hypothetical Dose Estimate Calculations”, June 18, 1999.

8.0 OPERATING PROCEDURES

The 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 amendment request made several minor changes to the loading and unloading procedures, based on lessons learned from previous cask loading operations, for the HI-STORM 100 Cask System. All minor changes made to the procedures were reviewed by the staff and found acceptable.

To accommodate the design modifications requested by the amendment, procedures were added in three areas. The use of a forced helium dehydration system was added for the draining and drying the casks loaded with high burnup fuel and instructions were added for anchoring the HI-STORM 100A to the ISFSI pad.

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.

The staff agrees that the design, analysis, and testing requirements contained in Chapter 2 of the FSAR is adequate to ensure that licensees do not exceed peak cladding temperatures during moisture removal from the HI-STORM 100 Cask System. The staff also agrees that the procedures in Chapter 8 of the FSAR and Design Feature 3.6, "Forced Helium Dehydration System," of CoC 1014 provide reasonable assurance to ensure that the design implemented by licensees using the HI-STORM 100 will meet the design basis of the cask system.

8.2 HI-STORM 100A ISFSI Pad Operational Requirements

The applicant provided an overview of the design and construction requirements for the ISFSI pad for use with the HI-STORM 100A overpack in Chapter 2, Appendix 2.A, of the FSAR. The requirements of Appendix 2.A will be implemented when the HI-STORM 100A overpack is used in high seismic areas. The sufficiency of the HI-STORM 100A and pad interaction was evaluated in Section 3 of this SER.

The staff agrees that the Chapter 8 of the FSAR contains sufficient detail, including hardware torque requirements, to ensure that a HI-STORM 100A is properly anchored to the ISFSI pad for it to meet its design basis.

8.3 Evaluation Findings

F8.1 The HI-STORM 100 Cask System can be wet loaded and unloaded. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the FSAR. These procedures were appropriately modified to include the design modifications made in the amendment. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 The staff concludes that the generic procedures and guidance for the operation of the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the FSAR 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 Holtec's FSAR includes the appropriate acceptance tests and maintenance programs for the HI-STORM 100 Cask 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 installation of lead in the HI-TRAC TC. The applicant requested that, as an alternative to pouring the molten lead, the HI-TRAC lead shielding may be installed as pre-cask sections. The staff agrees that precast sections is an acceptable substitute for poured lead provided measures are in place to ensure any neutron streaming from gaps between adjacent sections is bounded by the shielding and dose evaluations in Chapters 5 and 10 of the FSAR.

9.1 Evaluation Findings

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

10.0 RADIATION PROTECTION EVALUATION

This section evaluates the capability of the radiation protection design features, design criteria, and the operating procedures of the HI-STORM 100 Cask System to meet regulatory dose requirements. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d).

Occupational exposures from the HI-STORM 100 Cask System are based on the direct radiation dose rates calculated in Chapter 5 of the FSAR and the operating procedures discussed in Chapter 8 of the FSAR. 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 and the dose rates from design-basis atmospheric releases calculated in Chapter 7 of the FSAR.

10.1 Increase in Occupational Dose Rate and Auxiliary Shielding Equipment

The applicant requested to increase the occupational dose rates, per specific loading and unloading operating procedure, in Chapter 10 of the FSAR. Additionally, the applicant requested to include a note to Table 10.1.2 of the FSAR, to allow the cask system user to determine which auxiliary shielding equipment is needed for their spent fuel loading purpose.

The staff has reviewed the information presented in Chapter 10 of the FSAR, and finds the revised occupational dose rates to be acceptable. The staff also finds the note to Table 10.1.2 of the FSAR to be acceptable.

10.2 Evaluation Findings

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

F10.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

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

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

F10.5 The FSAR adequately evaluates the HI-STORM 100 Cask System and its systems important to safety, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F10.6 The FSAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F10.7 Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.06 are the responsibility of the site licensee. The HI-STORM 100 Cask System is designed to assist in meeting these requirements.

F10.8 The staff concludes that the design of the radiation protection system for the HI-STORM 100 Cask System is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM 100 Cask System will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

11.0 ACCIDENT ANALYSIS EVALUATION

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

- Identified all credible accidents
- Provided complete information in the FSAR
- Analyzed the safety performance of the cask system in each review area
- Fulfilled all applicable regulatory requirements

11.1 Off-Normal and Normal Operations

Off-normal operations are Design Event II as defined by ANSI/ANS 57.9[1]. These events can be expected to occur with moderate frequency or on the order of once per year. The HI-STORM 100 off-normal operations are described in Chapter 11 of the FSAR. In many instances, Chapter 11 of the SAR has not changed from the original application which was previously approved by the staff. Five off-normal operations were considered in the design of the HI-STORM 100:

- Off-Normal Pressures
- Off-Normal Environmental Temperatures
- Leakage of One MPC Seal Weld
- Partial Blockage of Air Inlets
- Off-Normal Handling of HI-TRAC TC

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 integrity from any off-normal event.

11.2 Accident Events and Conditions

Accident events and conditions are Design Event III and IV as defined in Reference 1. They include natural phenomena and human-induced low probability events. The applicant provided analyses to demonstrate design adequacy for the accident-level events discussed below. The HI-STORM 100 postulated accidents are described in Chapter 11 of the FSAR. In many instances, Chapter 11 of the SAR has not changed from the original application which was previously approved by the staff. The staff reviewed these events and found them to be bounded by evaluations contained in Chapters 3 and 4of the FSAR and accepted by the staff in Sections 3 and 4 of this SER. The staff agrees that all accident-level events and conditions have been identified and all potential safety consequences considered.

11.3 Evaluation Findings

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

F11.2 The applicant has evaluated the HI-STORM 100 Cask System to demonstrate that it will reasonably maintain confinement of radioactive material under off-normal and credible accident conditions.

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

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

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

F11.6 The applicant has evaluated off-normal and design-basis accident conditions to demonstrate with reasonable assurance that the HI-STORM 100 Cask System radiation shielding and confinement features are sufficient to meet the requirements in 10 CFR 72.104(a) and 10 CFR 72.106(b).

F11.7 The staff concludes that the accident design criteria for the HI-STORM 100 Cask System are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

12.0 CONDITIONS FOR CASK USE —TECHNICAL SPECIFICATIONS

The purpose of the review is to assess whether the applicant has modified the of CoC 1014 Conditions and TS as 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 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 Cask System were modified to add descriptions of the design changes requested by the amendment. In addition, Conditions for special requirements the first system in place were relocated to the CoC 1014 Condition 9 and the pre-operational testing and training requirements were relocated from TSs 5.1 and 5.2 to the CoC 1014 Conditions 10. The staff reviewed these changes and 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 Technical Specifications, as modified by the amendment, for the HI-STORM 100 Cask System. The staff has reviewed the TS and 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 revised CoC 1014, Approved Contents and Design Features to reflect the additional contents for the HI-STORM 100 Cask System requested by the amendment. The staff has reviewed the revisions and finds that they provide sufficient information to ensure that the additional 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

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

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

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION
1.1	Definitions
1.2	Logical Connectors
1.3	Completion Times
1.4	Frequency
2.0	(intentionally left blank)
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY/SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SPENT FUEL STORAGE CASK (SFSC) Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2	SFSC Radiation Protection
3.2.1	TRANSFER CASK Average Surface Dose Rates
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	OVERPACK Average Surface Dose Rates
3.3	SFSC CRITICALITY CONTROL
3.3.1	Boron Concentration
Table 3-1	MPC Model-Dependent Limits
4.0	(intentionally left blank)
5.0	ADMINISTRATIVE CONTROLS
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Cask Transport Evaluation Program
5.5	Radioactive Effluent Control Program
5.6	Fuel Cladding Oxide Thickness Evaluation Program

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether Holtec has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G. The staff has previously reviewed and accepted the Holtec QA program and verified its implementation by inspection, therefore, no reevaluation of the program was performed for the amendment.

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. Therefore, the staff did not reevaluate this area for the amendment request.

15.0 CONCLUSIONS

15.1 Overall Conclusion

The staff has reviewed the amendment to the Safety Analysis Report for the HI-STORM 100 Cask System. Based on the statements and representations contained in the FSAR as amended, and the conditions given in the Certificate of Compliance 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 in this amendment application for the design modifications to the HI-STORM 100 Cask System are acceptable. However, the staff did not review any methodologies used in the original HI-STORM 100 Cask System application and did not make a determination on the adequacy of the previous methodologies.

Issued with Certificate of Compliance No. 1014,
on July 18, 2002