UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HOLTEC INTERNATIONAL
HI-STORM 100
MULTIPURPOSE CANISTER STORAGE SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 15

SUMMARY

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (staff) review and evaluation of Holtec International's (hereafter Holtec or applicant) request to amend Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Multipurpose Canister (MPC) Storage System (hereafter HI-STORM 100 System). Holtec International submitted its request by letter dated March 20, 2019 (Holtec, 2019a), and supplemented it on September 16, 2019 (Holtec, 2019b), April 28, 2020 (Holtec, 2020a), May 15, 2020 (Holtec, 2020b), June 12, 2020 (Holtec, 2020c), June 22, 2020 (Holtec, 2020d), July 30, 2020 (Holtec, 2020e), August 14, 2020 (Holtec, 2020f), September 1, 2020 (Holtec, 2020g), and September 25, 2020 (Holtec, 2020h).

Holtec proposed the following changes:

1. Add a new version of a transfer cask, HI-TRAC MS (maximum shielded), which includes an option for variable weight of the lead and water jacket and cooling passages to the bottom lid. HI-TRAC MS is to be used with all MPCs approved for use in Amendment Nos. 0 through 14 to the HI-STORM 100 System and the newly proposed MPC-32M, MPC-32 Version 1, and MPC-68 Version 1.
2. Include MPC-32M for storage in the HI-STORM 100 System.
3. Include MPC-32 Version 1 and MPC-68 Version 1 for storage in the HI-STORM 100 System.
4. Add the new overpack, HI-STORM 100S Version E (hereafter Version E). The requested permissible MPCs include all MPCs approved for use in Amendment Nos. 0 through 14 to the HI-STORM 100 System and the newly proposed MPC-32M, MPC-32 Version 1, and MPC-68 Version 1.
5. Add three additional boiling water reactor (BWR) fuel types to the approved content for MPC-68M: 10x10I, 10x10J, and 11x11A.
6. Lower the allowed ambient temperature from 80°F to 70°F for Version E.
7. Add HI-DRIP and dry ice jacket (DIJ) ancillary system as additional cooling when the MPC is loaded in the HI-TRAC transfer cask.
8. Allow for partial gadolinium credit for BWR fuel assemblies types 10x10 and 11x11 assembly classes in MPC-68M.
9. Remove dose rate and consequence analysis from accident analysis for non-mechanistic tipover in Final Safety Analysis Report (FSAR) Section 11.2.3.

10. Include allowance for canisters currently loaded under earlier amendments which had different helium leak test requirements.
11. Update Drawing No. 7195 for the MPC-68M by removing dimensions which are not used in the safety analysis.
12. Include DIJ as an optional alternate cooling method for short-term operation of the loaded HI-TRAC.

While reviewing HI-STORM 100 proposed change #9, the staff identified that, in the initial application, Holtec had considered tipover as a part of the design criteria for accident conditions of storage. Holtec subsequently supplemented its application to provide the dose rate and consequence analysis and revised FSAR Section 11.2.3 accordingly (Holtec, 2020f). Holtec did not pursue proposed change #9 in this response, accordingly, the staff did not review proposed change #9 in the review of this amendment request.

This revised CoC, when codified through rulemaking, will be denoted as Amendment No. 15 to CoC No. 1014.

This SER documents the staff's review and evaluation of the proposed amendment. The staff followed the guidance in NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," July 2010 (NRC, 2010a) and NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities," April 2020 (NRC, 2020a). The staff's evaluation is based on a review of Holtec's application and supplemental information to determine whether it meets the applicable requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," for dry storage of spent nuclear fuel. The staff's evaluation focused only on modifications requested in the proposed amendment and did not reassess previous revisions of the FSAR or previous amendments to the CoC.

1.0 GENERAL INFORMATION EVALUATION

The staff reviews the FSAR Chapter 1, "General Description," to ensure that the applicant has provided a non-proprietary description, or overview, in its documentation for the spent fuel storage system that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

In FSAR Chapter 1, the applicant added the descriptions of the HI-DRIP cooling system and DIJ cooling system. The staff determined that the proposed description in general information is adequate for the NRC staff to conduct its evaluation as documented in the following sections of this SER. Therefore, it satisfies the requirements for the general description under 10 CFR Part 72.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

The staff evaluates the principal design criteria related to structures, systems, and components (SSCs) important to safety (ITS) to ensure that the principal design criteria comply with the relevant general criteria established in the requirements in 10 CFR Part 72.

The applicant revised FSAR Chapter 2, "Principal Design Criteria," to add information on the helium leak test requirement for Amendment No. 7 and prior amendments; three new BWR fuel types; the HI-DRIP cooling system and DIJ cooling system; and the partial gadolinium credit.

Based on a review that considered the applicable regulations, regulatory guides, codes and standards, and accepted engineering practices, the staff determined the proposed principal design criteria are acceptable as documented in the following sections of this SER.

3.0 STRUCTURAL EVALUATION

The objective of staff's structural evaluation is to verify that the applicant has performed adequate structural analyses to demonstrate that the system, as proposed, is acceptable under normal and off-normal operations, accident conditions, and natural phenomena events. In conducting this evaluation, the staff focused its review on whether the system will maintain confinement, subcriticality, shielding, and retrievability of the fuel, as applicable, under credible loads of normal and off-normal conditions, accident conditions, and natural phenomena events.

The staff evaluated the following proposed changes that are applicable to the structural review:

- Proposed Change #1: Add a new version of transfer cask, HI-TRAC MS.
- Proposed Change #2: Include MPC-32M.
- Proposed Change #3: Include MPC-32 Version 1 and MPC-68 Version 1.
- Proposed Change #4: Add HI-STORM 100S Version E overpack.
- Proposed Change #5: Add three additional BWR fuel types for MPC-68M: 10X10I, 10X10J, and11X11A.
- Proposed Change #11: Update Drawing No. 7195 for the MPC-68M by removing dimensions which are not used in the safety analysis.

3.1 Proposed Change #1 – A New Version of Transfer Cask, HI-TRAC MS

3.1.1 Description of HI-TRAC MS

The HI-TRAC MS is a ventilated overpack with a multi-buttressed steel shell structure containing an MPC. The HI-TRAC MS design includes a detachable neutron shield cylinder feature that users can remove when moving the MPC and HI-TRAC into and out of the spent fuel pool. This feature maximizes the amount of gamma shielding relative to the crane capacity because the water in the MPC provides neutron shielding. Holtec referred to Drawing No. 11381 in FSAR Section 1.II.5 for details of the HI-TRAC MS. The staff reviewed the aforementioned information and drawings and finds that the applicant provided sufficient information to identify the dimensions, ITS components, material descriptions and design criteria for the HI-TRAC MS.

Holtec provided heights, weights, and centers of gravity for the HI-TRAC MS, as well as for various other components, in FSAR Section 3.II.2 and Tables 3.II.2.1, 3.II.2.3, 3.II.2.5, and 3.II.2.6. The staff finds that the applicant considered the varying thickness of the lead shield and outer diameter of the water jacket in its weight calculations for the HI-TRAC MS. In addition, the staff finds that the applicant considered loaded, including the heaviest loaded configuration, and unloaded HI-TRAC MS configurations for the center of gravity calculations. Overall, the staff finds that the applicant provided sufficient descriptive information to characterize the structure of the HI-TRAC MS.

3.1.2 Design Criteria for HI-TRAC MS

Holtec summarized the HI-TRAC MS design criteria in FSAR Section 2.II.0.3 and discussed governing mechanical loadings germane to the HI-TRAC MS in FSAR Section 2.II.2.1 and Table 2.II.2.1. The applicant discussed the principal design criteria pertinent to the HI-TRAC MS in FSAR Section 2.II.2, which included American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) service limits for normal, off normal, and accident conditions, as well as lifting and handling safety criteria. The staff reviewed the aforementioned information and drawings and finds that the applicant provided sufficient information to identify the design criteria for the HI-TRAC MS. Further, the staff finds that the design criteria are consistent with the guidance of NUREG-2215 (NRC, 2020a) and are acceptable.

3.1.3 Structural Analysis

FSAR Section 2.2 provides a complete list of the governing loads relevant to the safety analysis of the HI-TRAC MS. The discussion below addresses these loads.

3.1.3.1 Lifting and Handling Analysis

Holtec analyzed the strength of the threaded anchor locations (TALs) of the HI-TRAC MS located in the top flange. The applicant summarized the results of these analyses in FSAR Table 3.II.4.2. The staff reviewed the aforementioned table and accompanying calculation package, Holtec proprietary Report HI-2188402, and noted that the applicant conservatively used bounding weights in the stress analyses. In addition, the staff noted that the calculated stresses were compared with the allowable stresses provided in the Table 3.II.4.2 and the calculated factors of safety are greater than 1.0 in all cases. The staff finds that the strength analysis and allowable stresses of the HI-TRAC MS TALs are consistent with the guidance in Regulatory Guide (RG) 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask” (NRC, 1989); NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36” (NRC, 1980); and NUREG-2215. Therefore, the staff finds the approach for normal handling conditions of the HI-TRAC MS acceptable.

The applicant performed stress analyses of the HI-TRAC MS body for lifting operations. The calculation involved elastic plate theory and determined the stress level in the HI-TRAC MS water jacket and on the loaded water jacket welds under the combined effects of internal pressure caused by heating; hydrostatic effects; and dynamic effects during lifting and transportation. The applicant calculated the following stresses, among others, using bounding lifted weights as specified in FSAR Table 3.II.2.6:

- The shear stress in the welds between the top flange and the inner and outer shells.
- The primary membrane stress in the inner and outer shells.
- The tensile stress in the bottom lid bolts.
- The primary bending stress in the bottom lid.

The applicant summarized the results of the stress analysis in FSAR Table 3.II.4.4. The staff reviewed the aforementioned tables and accompanying calculation package, Holtec proprietary Report HI-2188402, and noted that the applicant conservatively used bounding weights in the stress analyses, and therefore the staff found this methodology acceptable. In addition, the applicant compared calculated stresses with allowable stresses per ASME Code, Section III

Division I Level A stress limits for NF Class 3 plate and shell structures. The staff noted that the calculated factors of safety are greater than 1.0 in all cases, consistent with the applicable ASME code. Therefore, the staff concludes that the stresses on the HI-TRAC MS body under normal and off-normal handling conditions are acceptable.

3.1.3.2 Tornadic Wind and Tornado-Propelled Missile Acting Synergistically (Load Case M-1 from FSAR Table 2.II.2.1)

The applicant performed overturning analyses of the HI-TRAC MS under large missile impact and tornado winds. This loading is described in FSAR Section 2.2.3.5 and the loading data are provided in FSAR Tables 2.2.4 and 2.2.5. The acceptance criteria are provided in FSAR Section 2.II.2.2.

The applicant's HI-TRAC MS overturning analyses uses a similar analytical methodology to that used in previous amendments. The staff finds that the applicant's approach is consistent with the guidance in NUREG-2215, which satisfies 10 CFR 72.236, and is therefore acceptable. FSAR Section 3.II.4.4.2 and Table 3.II.4.9 summarize the overturning analysis approach and results for the HI-TRAC MS under the large missile impact and tornado wind. The staff reviewed the aforementioned sections and accompanying calculation package, Holtec proprietary Report HI-2188390, and found that the results show that the HI-TRAC MS remains in a vertical upright position (i.e., no overturning) in the aftermath of a large missile impact and tornado winds. The calculated cask angular rotations were compared with the allowable cask angular rotations, as shown in Table 3.II.4.9, and the calculated factors of safety were greater than 1.0. Therefore, the staff concludes that the design and analysis of the HI-TRAC MS are acceptable.

The applicant performed penetration analyses of the HI-TRAC MS for the impact of small and intermediate missiles to determine the extent to which they would penetrate the HI-TRAC MS and cause potential damage to the MPC enclosure vessel. The results of this analysis, found in FSAR Table 3.II.4.11, shows that the depth of penetration of the small missile is less than 1.5 inches on the side and 9 inches on the lid, which corresponds to the thinnest section of material on the exterior surface of the HI-TRAC MS. Therefore, both the small and intermediate missiles will dent, but not penetrate, the HI-TRAC MS. Therefore, there will be no impairment to the confinement boundary due to small and intermediate missile (i.e., tornado-borne missiles) strikes. Furthermore, since the HI-TRAC MS inner shells are not compromised by the missile strike, there will be no permanent deformation of the inner shells, therefore, assuring retrievability of the MPC.

The staff reviewed FSAR Table 3.II.4.11 and the accompanying calculation package, Holtec proprietary Report HI-2188390, and noted that the applicant used conservative assumptions and that all factors of safety are greater than 1.0, which indicates adequate protection against missiles. The staff reviewed the penetration analysis due to impacts from small and intermediate missiles impact and determined that confinement boundary of the MPC will not be breached and retrievability is maintained. The staff finds that the analysis and results are consistent with guidance in NUREG-2215 and is acceptable.

3.1.3.3 Free Fall of Loaded Cask During Handling (Load Cases M-2 And M-3 from FSAR Table 2.II.2.1)

The applicant stated in FSAR Section 2.II.2.7 that transfer operations for the HI-TRAC MS rely on drop postulated exempt (single-failure proof) features that meet the criteria in NUREG-0612 and American National Standards Institute (ANSI) N14.6, “Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More,” which preclude an uncontrolled lowering of the load. Therefore, the drop and non-mechanistic tipover postulated accident conditions were not considered. The staff noted that this approach is similar to that used in other Holtec systems, such as the HI-STORM Flood/Wind (FW) MPC Storage System (Holtec, 2017) that was approved by the staff. The staff compared the analysis against guidance in NUREG-2215 and finds that, consistent with the guidance, the drop accidents do not need to be considered because the HI-TRAC MS relies on drop postulated exempt features equipment. Therefore, not considering drop and non-mechanistic tipover postulated accident conditions is acceptable.

In summary, the applicant performed structural analyses for the HI-TRAC MS to demonstrate the structural integrity of the system. The staff reviewed the analyses and found that the structural design bases, acceptance criteria, loading conditions, and methodology used for the analyses of the HI-TRAC MS is consistent with guidance in NUREG-2215, similar to previous amendments approved by the staff, and is acceptable.

3.2 Proposed Change #2 – Include MPC-32M

MPC-32M consists of two principal components, namely the enclosure vessel Version 1 and the fuel basket. Enclosure vessel Version 1 is common to all the MPCs introduced in this amendment (MPC-32M, MPC-32 Version 1, and MPC-68 Version 1).

3.2.1 Description of MPC-32M

The applicant described the main design attributes of the MPC-32M in FSAR Section 1.II.2.2 and Table 1.II.2.3. Further, FSAR Section 1.II.5 Drawings No. 11425 R0 and No. 11572R0 provide additional details about the MPC-32M and the Enclosure Vessel Version 1, respectively.

The enclosure vessel Version 1 is made of Alloy X and has a baseplate thickness ¼-inch thicker than the enclosure vessel for previously certified designs for the MPC-32, MPC-68, and MPC-68M. The applicant stated that the increased baseplate thickness is to account for higher design internal pressures for the enclosure vessel Version 1 versus the previously approved MPC designs. FSAR Table 1.II.2.3 shows that the enclosure vessel Version 1 has a design internal pressure of 110 psig under normal conditions whereas previously approved designs (FSAR Table 2.2.1) have a design internal pressure of 100 psig.

The applicant stated that the MPC-32M shares similar Metamic^{TM}-HT fuel basket design features with the MPC-68M which was approved by the staff in Amendment No. 8 (NRC, 2011d). The applicant provided heights, weights, and centers of gravity for the loaded, including the heaviest loaded configuration, and unloaded HI-STORM cask for various components, including the MPC, in FSAR Section 3.II.2 and Tables 3.II.2.1, 3.II.2.2, 3.II.2.5, and 3.II.2.6. The staff reviewed the aforementioned information and drawings and finds that the applicant provided sufficient information to identify the dimensions, ITS components, and material descriptions for the MPC-32M. The staff also finds that the applicant considered bounding weights of the loaded MPC containing “reference [spent nuclear fuel (SNF)]” with and without water. Overall, the staff finds that the applicant provided sufficient information to characterize the structure of the MPC-32M.

3.2.2 Design Criteria for MPC-32M

The applicant summarized the MPC-32M design criteria in FSAR Section 2.II.0.1 and discussed governing mechanical loadings germane to the MPC-32M in FSAR Section 2.II.2.1 and Table 2.II.2.1.

The applicant stated that the enclosure vessel Version 1 is made of Alloy X and has a baseplate thickness ¼-inch thicker than the previously approved designs for the MPC-32, MPC-68, and MPC-68M. The applicant stated that the enclosure vessel is subject to Level D stress intensity limits per Subsection NG of the ASME Code, Section III.

The fuel basket, made of Metamic^{TM}-HT, is subject to the requirements specified in FSAR Section 2.II.2.2. Structural limits on the Metamic^{TM}-HT panel are specified as a lateral deformation limit of its walls under accident conditions of loading (credible and non-mechanistic) (FSAR Table 2.II.2.4). The applicant noted that the design criteria for the fuel basket has been accepted by the staff for other Holtec MPCs, such as the MPC-68M, and in other Holtec systems, such as the HI-STORM FW.

The staff reviewed the aforementioned information and finds that the applicant provided sufficient information to identify the design criteria for the MPC-32M, which are consistent with guidance from NUREG-2215 and are acceptable.

3.2.3 Structural Analysis

3.2.3.1 Lifting and Handling Analysis

The lifting and handling analysis is applicable to all MPCs introduced in this amendment, namely the MPC-32M, MPC-32 Version 1, and MPC-68 Version 1. FSAR Section 3.II.4.3.2 discusses the stress analyses for the MPC under lifting and handling loads.

The applicant performed strength analyses of the TALs for the MPC. FSAR Table 3.II.4.2 provides the resulting factors of safety for the TALs in the MPC's top lid. The staff reviewed the aforementioned table and accompanying calculation package, Holtec proprietary Report HI-2188402, and noted the applicant conservatively used bounding weights and pressure in the analyses. In addition, the staff found that calculated stresses were compared with the allowable stresses and the calculated factors of safety are greater than 1.0 in all cases. The staff finds that the strength evaluation of the MPC and the allowable stresses used by the applicant are consistent with guidance from RG 3.61, NUREG-0612, and NUREG-2215. Therefore, the staff finds the approach acceptable for normal handling conditions of the MPC.

The applicant performed stress analyses of the MPC enclosure vessel Version 1 body for lifting operations using ANSYS software. The applicant summarized the resulting stress intensity values in FSAR Table 3.II.4.3. The staff reviewed the aforementioned table and accompanying calculation package, Holtec proprietary Report HI-2188402, and noted the applicant conservatively used bounding weights in the stress analyses. In addition, the staff found that calculated stresses were compared with the allowable stress limits for NB Class of the ASME Code. The staff finds the allowable stresses from the ASME code Subsection NB acceptable. The staff noted the calculated factors of safety are greater than 1.0 in all cases which indicate that the stresses in the MPC enclosure vessel Version 1 body are acceptable under normal and off-normal handling conditions. The staff finds that the analysis and results are consistent with the guidance in NUREG-2215 and are acceptable.

3.2.3.2 Design Internal Pressure Analysis

FSAR Section 3.II.4.4.1 discusses the stresses incurred on the enclosure vessel for the design internal pressure for normal, off-normal, and accident conditions as specified in FSAR Table 1.II.2.3. In FSAR Section 11.II.1.1, the applicant stated that the off-normal pressure of the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the cavity under normal storage. In addition, per FSAR Section 11.2.9.2, the accident condition is dictated by the 100% fuel rod rupture accident. FSAR Tables 3.II.4.7 and 3.II.4.8 show that the maximum calculated stress intensities in the enclosure vessel are bounded by the allowable stresses obtained from Subsection NB of the ASME Code. The staff reviewed the approach and finds that the calculated factors of safety are greater than 1.0 in all cases which indicate that the stresses in the MPC-32M enclosure vessel Version 1 body are acceptable under normal, off normal, and accident design pressures. The staff finds that the analysis and results are consistent with guidance in NUREG-2215 and are acceptable.

3.2.3.3 Other Mechanical Loadings

The qualification of the MPCs under loadings that act on the loaded HI-STORM overpack or the HI-TRAC overpack is evaluated along with its host overpack. Since all MPCs proposed in this amendment share the same enclosure vessel and the loads used by the applicant are bounding, the following sections are also applicable to the MPC-32M:

- Tornadic wind and tornado-propelled missiles acting synergistically (load case M-1 from FSAR Table 2.II.2.1) – Discussed in SER Sections 3.1 and 3.4.
- Free fall of loaded cask during handling (load cases M-2 and M-3 from FSAR Table 2.II.2.1) – Discussed in SER Sections 3.1 and 3.4.
- Non-mechanistic tip-over (load case M-4 from FSAR Table 2.II.2.1) – Discussed in SER Section 3.4.

The applicant performed structural analyses for the MPC-32M to demonstrate the structural integrity of the system. The staff reviewed the analyses and found that the structural design criteria, acceptance criteria, loading conditions, and methodology used for the analyses of the MPC-32M are consistent with the guidance in NUREG-2215, and are acceptable.

3.3 Proposed Change #3 – Inclusion of MPC-32 Version 1 and MPC-68 Version 1

The MPC-32 Version 1 and MPC-68 Version 1 consist of two principal components, namely the enclosure vessel Version 1 and the fuel basket. Enclosure vessel Version 1 is common to all the MPCs introduced in this amendment (MPC-32M, MPC-32 Version 1, and MPC-68 Version 1).

3.3.1 Description of MPC-32 Version 1 and MPC-68 Version 1

The applicant described the main design attributes and criteria of the MPC-32 Version 1 and MPC-68 Version 1 in FSAR Section 1.II.2.2 and Table 1.II.2.3. Further, as described in FSAR Section 1.II.5, Drawings No. 11574R0 and No. 11578R0 provide additional details on the MPC-32 Version 1 and MPC-68 Version 1.

The enclosure vessel Version 1 is made of Alloy X and has a baseplate thickness ¼-inch thicker than the previously approved designs for the MPC-32, MPC-68, and MPC-68M. Similarly, the fuel baskets for the Version 1 MPCs are made of Alloy X. The applicant stated that increased baseplate thickness is to account for higher design internal pressures and accommodate a higher load-bearing capacity for the enclosure vessel Version 1 versus the previously approved MPC designs. The applicant provided heights, weights, and centers of gravity for the MPC-32 Version 1 and the MPC-68 Version 1, and various other components, in FSAR Section 3.II.2 and Tables 3.II.2.1, 3.II.2.2, 3.II.2.5, and 3.II.2.6. The staff reviewed the aforementioned information and drawings and finds that the applicant provided sufficient information to identify the dimensions, ITS components, and material descriptions for the MPC-32 Version 1 and MPC-68 Version 1. The staff finds that the applicant considered bounding weights of the loaded MPCs containing “reference SNF” with and without water.

Overall, the staff finds that the applicant provided sufficient information to characterize the structure of the MPC-32 Version 1 and MPC-68 Version 1.

3.3.2 Design Criteria for MPC-32 Version 1 and MPC-68 Version 1

The applicant summarized the MPC-32 Version 1 and MPC-68 Version 1 design criteria in FSAR Section 2.II.0.1 and discussed governing mechanical loadings germane to the MPCs in FSAR Section 2.II.2.1 and Table 2.II.2.1. FSAR Section 2.II.2 discusses the principal design criteria pertinent to the MPCs. The applicant stated that both the MPC fuel basket and the enclosure vessel Version 1 are made of Alloy X material and follow the stress intensity limits criteria per Subsection NG of the ASME Code, Section III. The staff reviewed the aforementioned information and finds that the design criteria are consistent with the guidance in NUREG-2215, and are acceptable.

3.3.3 Structural Analysis

FSAR Section 2.2 provides a complete list of the governing loads relevant to the safety analysis of the HI-STORM 100 System containing the aforementioned MPC-32 Version 1 and MPC-68 Version 1. As previously noted, all MPCs introduced in this amendment have the same enclosure vessel Version 1 design. Since the handling and internal pressure analyses for these MPCs depend only on the enclosure vessel design, the discussion and findings related to these variables in SER Section 3.2 are also applicable to the MPC-32 Version 1 and MPC-68 Version 1. Thus, the discussion presented here will only address governing loading conditions germane to the MPC-32 Version 1 and MPC-68 Version 1.

3.3.3.1 Non-Mechanistic Tipover (Load Case M-4 from FSAR Table 2.II.2.1)

The applicant performed a comparative assessment between the MPC-32 Version 1 and MPC68 Version 1 Alloy X baskets in the calculation package, Holtec proprietary Report HI-2188448. The applicant stated that the MPC-32 Version 1 basket design is bounding in terms of both load and capacity, therefore a bounding tipover analysis was performed on MPC-32 Version 1 to encompass both Version 1 MPCs. The staff notes that this approach is reasonable since the MPC-32 Version 1 basket’s fuel weight is higher than the MPC-68 Version 1 basket’s fuel weight and has longer fuel cell spans both of which promote additional deformation. Therefore, this approach is bounding.

The applicant performed a LS-DYNA tipover analysis for the MPC-32 Version 1 basket only and considered two different orientations. These orientations represent the worst loading scenario the basket can experience during a drop event. FSAR Figures 3.II.4.28 and 3.II.4.29 show the two limiting basket orientations. The applicant calculated the induced stresses in the basket panels and the weld between basket panels and compared them against allowable values as specified in FSAR Table 2.II.2.3a. The staff noted the calculated factors of safety are greater than 1.0 in all cases which indicates that the stresses on the fuel basket panels and associated welds are acceptable under accident conditions. The staff finds that the tipover analysis for MPC-32 Version 1 and the MPC-68 Version 1 MPCs is consistent with the guidance from NUREG-2215 and is acceptable.

3.3.3.2 Other Mechanical Loadings

The applicant's qualification of the MPC-32 Version 1 and MPC-68 Version 1 under loadings that act on the loaded HI-STORM overpack or the HI-TRAC overpack is evaluated along with its host overpack. Since all MPCs proposed in this amendment share the same enclosure vessel and the loading conditions used by the applicant are bounding, the following sections are also applicable to the MPC-32 Version 1 and MPC-68 Version 1:

- Lifting and handling analysis – discussed in SER Section 3.2
- Design internal pressure analysis – discussed in SER Section 3.2
- Tornadic wind and tornado-propelled missiles acting synergistically (load case M-1 from FSAR Table 2.II.2.1) – discussed in SER Sections 3.1 and 3.4
- Free fall of loaded cask during handling (load cases M-2 and M-3 from FSAR Table 2.II.2.1) – discussed in SER Sections 3.1 and 3.4

The applicant performed structural analyses for the MPC-32 Version 1 and MPC-68 Version 1 to demonstrate the structural integrity of the system. The staff reviewed the analyses and found that the description, design bases, acceptance criteria, loading conditions, and methodology used for the analyses of the MPC-32 Version 1 and MPC-68 Version 1 are consistent with the guidance in NUREG-2215 and are acceptable.

3.4 Proposed Change #4 – Add HI-STORM 100S Version E

3.4.1 Description of HI-STORM 100S Version E

The Version E overpack is an enhanced version of the HI-STORM 100S Version B that has been in use since the mid-2000s. In Section 2.II.0.2, the applicant stated that most design characteristics for the HI-STORM 100S Version E overpack remain applicable, namely similar large metal mass, heavy weight, and a thick structural top lid. Per FSAR Section 1.II.5, Drawing No. 11371 R0 contains details about Version E. The applicant provided the heights, weights, and centers of gravity for various components, including Version E, in FSAR Section 3.II.2 and Tables 3.II.2.1, 3.II.2.4, 3.II.2.5, and 3.II.2.6.

The staff reviewed the aforementioned information and drawings and finds that the applicant provided sufficient information to identify the dimensions, ITS components, and material descriptions for the HI-STORM 100S, Version E. The staff finds that the applicant considered loaded (containing the heaviest MPC) and unloaded Version E configurations for the center of gravity calculations, including with and without the lid. Overall, the staff finds that the applicant provided sufficient information to characterize the structure for the Version E overpack.

3.4.2 Design Criteria and Load Description for HI-STORM 100S Version E

The applicant summarized the Version E design criteria in FSAR Section 2.II.0.2 and discussed governing mechanical loadings germane to the Version E in FSAR Section 2.II.2.1 and Table 2.II.2.1. The applicant discussed the principal design criteria pertinent to the Version E in the FSAR Section 2.II.2 which include ASME service limits for normal, off normal, and accident conditions and lifting and handling safety criteria. The applicant also discussed design criteria related to sliding and overturning due to natural phenomena like the design basis earthquake and flooding. The staff reviewed the aforementioned information and drawings and finds that the design criteria for the Version E are consistent with the guidance of NUREG-2215 and are acceptable.

3.4.3 Structural Analysis

FSAR Section 2.2 provides a list of the loadings germane to the safety analysis of the HI-STORM 100 System. In previous amendments, the applicant determined and documented in the FSAR that non-governing loadings (such as lightning, high wind, and snow) were not required to be considered. For the Version E overpack, because of the design characteristics of the system (such as a large metal mass, heavy weight, and a thick structural top lid), the applicant also determined that non-governing loadings are not required to be considered. Since the main characteristics of the HI-STORM have remained similar between versions, namely large metal mass, heavy weight, and a thick structural top lid, the staff finds that this assessment is acceptable.

3.4.3.1 Lifting and Handling Analysis

The applicant analyzed the TALs for the Version E containing the heaviest MPC and summarized the results in FSAR Table 3.II.4.2. The staff reviewed the aforementioned table and accompanying calculation package, Holtec proprietary Report HI-2188402. The staff noted that the applicant conservatively used bounding weights in the stress analyses. In addition, the staff found that calculated stresses were compared with the allowable stresses and the calculated factors of safety are greater than 1.0 in all cases. The staff finds that the strength evaluation of the Version E threaded anchor locations for normal handling conditions is consistent with the guidance in RG 3.61, NUREG-0612, and NUREG-2215, and is acceptable.

The applicant performed stress analyses of the Version E body for lifting operations. The calculation used ANSYS software and the results are summarized in FSAR Table 3.II.4.5. The staff reviewed the aforementioned table and accompanying calculation package, Holtec proprietary Report HI-2188402. The staff noted that the applicant conservatively used bounding weights in the stress analyses. In addition, the staff found that calculated stresses were compared with the allowable stresses per the ASME Code, Level A stress limits for NF Class 3 plate and shell structures. The staff noted the calculated factors of safety are greater than 1.0 in all cases which indicates that the stresses of the Version E body under normal-handling conditions are acceptable. Therefore, staff finds stress analyses of the Version E body for lifting operations are consistent with the guidance in NUREG-2215 and are acceptable.

3.4.3.2 Tornadic Wind and Tornado-Propelled Missile Acting Synergistically (Load Case M-1 from FSAR Table 2.II.2.1)

The applicant performed overturning analyses of the Version E under large missile impact and tornado wind. This loading is described in FSAR Section 2.2.3.5 and the loading data is provided in FSAR Tables 2.2.4 and 2.2.5. The acceptance criteria are provided in FSAR Section 2.II.2.2(a).

The applicant used a similar analytical approach from previous amendments for the overturning analyses of the Version E overpack. FSAR Section 3.II.4.4.2 and Table 3.II.4.9 summarize the overturning analysis approach and results for the HI-STOM 100 Version ES under the large missile impact and tornado wind. The staff reviewed the aforementioned sections and accompanying calculation package, Holtec proprietary Report HI-2188390, and noted that the calculated rotations were compared with allowable rotations, and the calculated factors of safety were greater than 1.0 indicating that the design and analysis of the Version E are adequate. These safety factors of greater than 1.0 also indicate that the Version E would remain in a vertical upright position (i.e., no overturning) in the aftermath of large missile impact and tornado wind. The staff finds that the applicant's approach is consistent with the guidance from NUREG-2215 and is acceptable.

The applicant performed penetration analyses of the Version E under the small and intermediate missiles impact to determine the extent to which they will penetrate the Version E overpack and cause potential damage to the MPC enclosure vessel. When analyzing the Version E, the applicant used a similar analytical approach used in previous amendments. FSAR Table 3.II.4.11 shows the analysis results. Based on the results, the depth of penetration of the small missile is less than the thinnest section of material on the exterior surface of the Version E. Therefore, the small missile will dent, but not penetrate, the casks. For the intermediate missile, the analysis results show that the intermediate missile will not penetrate the lead surrounding the Version E inner shell. Therefore, there will be no impairment to the confinement boundary due to the small and intermediate missiles (i.e., tornado-borne missiles) strikes. Furthermore, since the Version E inner shells are not compromised by the missile strike, there will be no permanent deformation of the inner shells, and thereby retrievability of the MPC will be assured. The staff reviewed FSAR Table 3.II.4.11 and the accompanying calculation package, Holtec proprietary Report HI-2188390, and noted that the applicant used conservative assumptions and that all factors of safety are greater than 1.0 which indicates adequate protection against missiles. The staff reviewed the penetration analysis due to impacts from small and intermediate missiles impact and determined that the confinement boundary of the MPC will not be breached and retrievability is maintained. The staff finds that the analysis and results are consistent with the guidance in NUREG-2215 and are acceptable.

The applicant calculated sliding of the Version E due to the impact of a large missile and tornado wind. The maximum sliding calculated is presented in FSAR Table 3.II.4.10 and was compared against the expected inter-cask gap and the distance to the pad's edge at an independent spent fuel storage installations (ISFSI) pad to ensure the casks will not slide against each other or off the pad. The staff reviewed FSAR Table 3.II.4.10 and the accompanying calculation package, Holtec proprietary Report HI-2188390, and noted that the applicant used conservative assumptions and that all factors of safety are greater than 1.0 which means that calculated stresses are below allowable stresses, which indicates an adequate safety margin. Overall, the staff finds that the approach is consistent with the guidance from NUREG-2215 and the results are, therefore, acceptable.

3.4.3.3 Free Fall of Loaded Cask During Handling (Load Cases M-1 and M-2 from FSAR Table 2.II.2.1)

The applicant stated in FSAR Section 2.II.2.7 that transfer operations for the Version E rely on drop-postulated exempt features or single failure proof devices that meet the criteria in NUREG-0612 and ANSI N14.6 which preclude an uncontrolled lowering of the load. Therefore, the drop postulated accident conditions were not analyzed. The staff noted that this approach is similar to other Holtec systems, such as the HI-STORM FW (Holtec, 2017) approved by the staff. The staff reviewed the licensee's approach in this application and was guided by the precedent established in HI-STORM FW and guidance in NUREG 2215. Based on this review, the staff finds that, consistent with applicable guidance, the drop accidents do not need to be considered because the Version E relies on drop-postulated exempt features or single failure proof equipment. Therefore, the applicant's approach is acceptable.

3.4.3.4 Non-Mechanistic Tipover (Load Case M-4 from FSAR Table 2.II.2.1)

The applicant performed a LS-DYNA finite element analysis to simulate the postulated tipover event of Version E overpack with loaded MPC containing Metamic^{TM}-HT (MPC-32M) basket. FSAR Section 3.II.4.4.2 discusses this analysis. The staff noted that many aspects of the LS-DYNA model are similar to the finite element model constructed for the HI-STORM FW System (Holtec, 2017), for ISFSI pad and subgrade material properties and overall cask characteristics. The applicant stated that, a nonlinear material model with strain rate effect was used to model the responses of all Version E cask structural members based on the true stress-strain curves of the corresponding materials. FSAR Figures 3.II.4.8 to 3.II.4.21 depict the finite-element tipover analysis model and results.

Based on this analysis, the applicant concluded that the tipover analysis results complied with the design criteria as established in FSAR Section 2.II.2.2 as follows: (1) no ovalization of the MPC which would preclude removal, (2) limited deformation of the fuel basket panels per FSAR Table 2.II.2.4, (3) no loss of shielding, and (4) no breaching of confinement boundary.

The staff reviewed FSAR Section 3.II.4.4.2 and accompanying calculation package, Holtec proprietary Report HI-2188448, and noted that the applicant used an analytical approach to develop the true stress-strain curves for the MPC-32M Metamic^{TM}-HT fuel basket for the LS-DYNA input. The staff recognizes that this approach has been used in other Holtec systems, such as the HI-STORM FW, and has been found to be adequate on a case-by-case basis. However, in a meeting summary (NRC, 2018), the staff said that the power law equation that describes the flow curve of metal in the plastic deformation region needed to be based on material testing. Specifically, the constants "K" and "n" in the power law equation needed to be determined from the mean test values obtained from material testing rather than analytically.

In this amendment, the applicant did not use the mean test values from material testing. However, the staff notes that the applicant has postulated conservative assumptions in its LS-DYNA model that augmented the model responses providing additional margin and assurance in the final results. Assumptions include:

- Subgrade was modeled using elastic properties.
- The strain rate of the Metamic^{TM}-HT fuel basket was not considered.
- No structural credit was given to basket corner welds.

In addition, the applicant provided other analyses, as discussed in this section of the SER, that provide additional assurance against a tipover event, such as:

- reliance on drop-postulated exempt features or single failure proof equipment during transfer operations for the Version E, and
- conservative assumptions in the calculation of overturning potential against various accident situations, such as the design basis earthquake, flooding, and tornado missiles.

Finally, the staff notes that the true-stress-strain curve is only applicable to the fuel basket. All other components (MPC enclosure vessel, cask, and lid) meet their respective stress intensity criteria. Based on the above conservative assumptions, the staff determined that use of the analytical approach to develop the true stress-strain curves for the MPC-32M Metamic^{TM}-HT fuel basket for the LS-DYNA input is applicable in this case only. Given the aforementioned, the staff finds that the Version E and MPC-32M will continue to satisfy the design criteria after this postulated event.

3.4.3.5 Load Case M-5, Design Basis Earthquake (DBE)

The applicant performed sliding and overturning analyses of the Version E under DBE conditions. This analysis is discussed in FSAR Section 3.II.4.4.2. The applicant provided acceptance criteria against the DBE in FSAR Section 2.II.2.2 which consists of complying with a simple static equilibrium model (for a low intensity DBE) or by a dynamic time history analysis (for other than a low intensity DBE). The applicant stated that if the latter is selected, the general licensees shall perform a 10 CFR 72.212 review to demonstrate compliance. The applicant calculated acceptable horizontal and vertical g-levels for both cases. The staff notes that this approach is consistent with previous analyses performed by the applicant. The staff reviewed FSAR Section 2.II.2.2and Section 3.II.4.4.2 and finds that the applicant used conservative assumptions and that the approach is consistent with the guidance in NUREG-2215 and is acceptable.

3.4.3.6 Moving Floodwater

The applicant determined the maximum floodwater velocity that a loaded Version E overpack on the ISFSI pad can withstand before tipping over or sliding. The kinematic stability analysis consists of writing static equilibrium equations for tipping (i.e., overturning) and sliding. The applicant incorporated a minimum factor of safety of 1.1 against overturning and sliding per guidance in ANSI/ANS 57.9, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)." The resulting velocities are discussed in FSAR Section 3.II.4.4.2. Like all other natural phenomena, the staff notes that general licensees need to ensure that SSCs that are ITS can withstand the effects of natural phenomena, including flooding, in accordance with 10 CFR 72.122(b)(2)(i). The staff finds that the applicant's approach and results are consistent with the guidance in ANSI/ANS 57.9 and NUREG-2215 and is acceptable.

3.5 Proposed Change #5 – Add Three Additional BWR Fuel Types as Allowed Content for MPC-68M: 10x10I, 10x10J and 11x11A

The licensee proposes to add three additional fuel types as allowed content for MPC-68M. In response to request for additional information (RAI) 3-4, in a letter dated April 28, 2020 (Holtec, 2020a), the applicant stated that the maximum allowable weight limits and overall dimensions for the BWR fuel assemblies will not be different than what is described in Tables 2.1.5 and 2.1.22 of the FSAR. The staff reviewed the aforementioned tables and finds that the maximum allowable weight limits for BWR fuel assemblies bounds the weight of the proposed fuel assemblies. Similarly, the staff finds that no changes are being made to the fuel assembly length and width limits provided in FSAR Table 2.1.22t. Therefore, the staff finds that the addition of the three new BWR fuel types causes no change to the center of gravity of the system. Therefore, the addition of 10x10I, 10x10J, and 11x11A as approved contents for MPC-68M is acceptable.

3.6 Proposed Change #11 – Update Drawing No. 7195 for the MPC-68M by Removing Dimensions Which are Not Used in The Safety Analysis

The applicant proposed changing Drawing No. 7195 by removing dimensions for the shims and tolerances on the thickness of the basket panels. The staff reviewed the changes to Drawing No. 7195. The staff notes that the nominal value of the basket shim material's yield strength is a "critical characteristic" that is specified in FSAR Table 3.III.3. The staff also notes that the basket shims, as stated in FSAR Section 3.III.4.4.3.1, do not withstand any tensile loads and are located in a confined space which would prevent their uncontrolled deformation under load. Therefore, the staff finds that removing dimensions and tolerances for these components provides flexibility for the applicant while not significantly impacting the structural function of the system and, therefore, is acceptable.

3.7 Evaluation Findings

The staff concludes that the structural properties of the SSCs of the HI-STORM 100 System discussed in Section 3 of this SER comply with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the structural properties provides reasonable assurance that the HI-STORM 100 System will allow safe storage of SNF. This finding is based on a review that considered the regulations itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. The following findings are made:

F3.1 SNF handling, packaging, transfer, and storage systems are designed to ensure subcriticality. The margins of safety of these systems are adequate for the nature of the immediate environment under accident conditions, and therefore meet the requirements in 10 CFR 72.124(a).

F3.2 SSCs ITS are designed to provide favorable geometry or permanently fixed neutron-absorbing materials, and therefore meet the requirements in 10 CFR 72.124(b).

F3.3 The design bases and design criteria are provided for SSCs ITS that meet the requirements in 10 CFR 72.236(b).

F3.4 The applicant has met the requirements of 10 CFR 72.236(c), for the structural evaluation of items relied on for maintaining subcritical conditions. The structural design and fabrication of the dry storage system includes structural margins of safety for those SSCs important to nuclear criticality safety. The applicant has demonstrated adequate structural safety for the handling, packaging, transfer, and storage under normal, off-normal, and accident conditions.

F3.5 The SNF storage cask is designed to store the SNF safely for the term proposed in the application, and therefore meets the requirements in 10 CFR 72.236(g).

F3.6 The SNF storage cask and its systems that are ITS have been evaluated by appropriate test or other acceptable means and have demonstrated that they will reasonably maintain confinement or radioactive material under normal, off-normal, and credible accident conditions, and therefore meet the requirements in 10 CFR 72.236(l).

4.0 THERMAL EVALUATION

The objective of the thermal review is to ensure that the cask components and fuel material temperatures of the HI-STORM 100 cask system will remain within the allowable values under normal, off-normal, and accident conditions. It includes confirmation that the fuel cladding temperature will be maintained below specified limits throughout the storage period to protect the cladding against degradation that could lead to gross ruptures. This portion of the review also confirms that the cask thermal design has been evaluated using acceptable analytical techniques and/or testing methods.

The review was conducted to evaluate whether requirements for the approval and fabrication of spent fuel storage casks designs described in 10 CFR 72.236 are met. The review also evaluated whether the applicant identified the unique characteristics of the spent fuel to be stored, as required by 10 CFR 72.236(a), and provided the design basis and the design criteria for the SSCs ITS, in accordance with the requirements of 10 CFR 72.236(b). The review also assessed the provided design bases and design criteria. This application was also reviewed to determine whether the HI-STORM 100 System design fulfills the acceptance criteria listed in Sections 2, 4, and 12 of NUREG-1536, as well as associated Interim Staff Guidance (ISG), which provides one way of satisfying the relevant regulatory requirements.

The staff reviewed the information provided by the applicant and evaluated the following proposed changes that are applicable to the thermal review:

- Proposed Change #1 – Add a new version of a transfer cask, HI-TRAC MS.
- Proposed Change #2 – Include MPC-32M.
- Proposed Change #3 – Include MPC-32 Version 1 and MPC-68 Version 1.
- Proposed Change #4 – Add HI-STORM 100S Version E overpack.
- Proposed Change #5 – Add three additional BWR fuel types, 10x10I, 10x10J, and 11x11A, to the approved contents in CoC 1014, Appendix B for MPC-68M only.
- Proposed Change #6 – Lower ambient temperature from 80°F to 70°F for HI-STORM 100S Version E.
- Proposed Changes #7 and #12 – Add HI-DRIP and DIJ ancillary systems.
- Proposed Change #11 – Update Drawing No. 7195 for the MPC-68M to remove dimensions on the drawing that are not used in the safety analysis.

4.1 Heat Removal Systems

4.1.1 General Description

The applicant provided a general description of the HI-TRAC MS transfer cask in the FSAR. The HI-TRAC MS (maximum shielded) transfer cask is similar to the HI-TRAC 100 and HI-TRAC 125 transfer cask models described in this FSAR and previously certified for use. The HI-TRAC MS transfer cask contains a natural ventilation feature by adding an air inlet on the HI-TRAC MS's bottom lid. The HI-TRAC MS licensing drawing shows the natural ventilation passages that enable ventilation of air to passively reject the MPC's decay heat.

The applicant provided a general description of the MPC-32M, MPC-32 Version 1, and MPC-68 Version 1 canisters in the FSAR. The MPC-32M contains a Metamic^{TM}-HT egg-crate fuel basket structure with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. The MPC-32 Version 1 and MPC-68 Version 1 are enhanced versions of the MPC-32 and MPC-68 canisters licensed in the FSAR. These MPCs include a larger storage cell opening compared to previous versions. The MPCs are backfilled with helium to the design-basis pressures specified in the FSAR to provide an inert environment for long-term storage of spent fuel.

The applicant provided a general description of the HI-STORM 100S Version E overpack in the FSAR. The Version E overpack is very similar to its predecessor Version B. Version E is a buttressed dual-shell structure with shielding concrete placed in the inter-shell annulus. The inlet duct system of Version E is comprised of circumferentially separated ducts. The outlet flow is designed to be radially symmetric at the cask body to lid interface. The flow areas of both the inlet and outlet flow passages are enlarged to facilitate increased ventilation flow. The FSAR states the overpack can be used for unsheltered and sheltered configurations. FSAR Section 4.II.4.7 describes the applicant’s approach to perform the thermal evaluation of the HI-STORM 100S Version E for a sheltered configuration.

The applicant provided a general description of the HI-DRIP and DIJ auxiliary cooling systems in the FSAR. The HI-DRIP auxiliary cooling system is designed to prevent the water in the loaded MPC from boiling during the interval after it has been lifted out of the fuel pool and prior to the evacuation of water and backfill of helium. The FSAR states that if the HI-DRIP auxiliary cooling system were to fail, the time-to-boil (TTB) would be monitored. However, the FSAR does not describe the thermal condition of the canister at the time of failure. In response to a RAI, the applicant explained that in case of the HI-DRIP failure, a new TTB (using the existing approach described in FSAR Section 4.5.2) is established by measuring the MPC water temperature to determine the initial condition. For a high heat emitting MPC, an engineered DIJ auxiliary cooling system surrounding all or a portion of the water jacket serves as a passive means to augment heat removal from the system. The applicant’s calculations that show how long the DIJ would provide cooling are included in Appendix N of Holtec proprietary thermal calculation package Report HI-2043317.

4.1.2 Design Criteria

The design criteria common to the MPC models introduced in this amendment are provided in FSAR Table 1.II.2.3. The design criteria for the MPCs provided in FSAR Table 2.0.1 remain applicable for the MPCs introduced in this amendment because FSAR Table 2.0.1 provides all other design criteria that are used in the thermal evaluations performed by the applicant in this amendment request.

4.1.3 Design Features

Thermal design features of the different components are provided throughout the FSAR but particularly in FSAR Chapter 4 II. The application provided a summarized description of the heat transfer characteristics and fabrication materials of different components. The drawings provided in FSAR Section 1.II.5 includes nominal dimensions and components which are important to the thermal evaluation. Drawing No. 7195 shown in FSAR Section 1.II.5 for the MPC-68M was revised to remove dimensions on the drawing that are not used in the safety analysis. Dimensions that were removed from this drawing that are used to develop the MPC-68M thermal model, can be found in the calculation packages provided by the applicant.

The staff reviewed the applicant's general description, design criteria, and design features of the HI-STORM 100 storage system. Based on the information provided in the application, the staff concludes that the description of the decay heat removal system is acceptable as the description is consistent with guidance provided in NUREG-1536 which satisfies the requirements of 10 CFR 72.236(b), 10 CFR 72.236(f), 10 CFR 72.236(g), and 10 CFR 72.236(h).

4.2 Thermal Model

The applicant used the ANSYS FLUENT computer-based analysis program to evaluate the thermal performance of the HI-STORM 100S Version E overpack and MPC-32M in a loaded configuration. As discussed in the next section, this is the bounding case. ANSYS FLUENT is a finite volume computational fluid dynamics program with capabilities to predict fluid flow and heat transfer phenomena in two and three dimensions. FSAR Section 4.II.4.1 provides a general description of the Version E overpack thermal model. Most of the important components are explicitly included in a three-dimensional (3-D) model. The components include the fuel basket, helium inside the cavity, air, basket shims, canister walls, and overpack. FSAR Section 4.II.5.2 provides a general description of the HI-TRAC MS transfer cask thermal model. In the HI-TRAC MS, heat is removed by the convection of ventilation air flowing in the MPC/cask annulus and conducted across multiple concentric layers consisting of the air gap, the HI-TRAC MS inner shell, the lead shielding, the HI-TRAC MS outer shell, the water jacket space, and the jacket shell. The staff reviewed the thermal model and determined that most of the important components relevant to the heat transfer characteristics are included in a 3-D model. The applicant assumed 70°F for the ambient temperature. The staff finds that this temperature is nonconservative for sites where the ambient temperature would be higher (including seasonal variations). In this case, a general licensee would need to demonstrate that the cask thermal analysis that assumes 70°F bounds the site ambient temperature. Otherwise, general licensee site-specific analysis would need to be performed based on the site characteristics (using a maximum normal ambient temperature) to provide bounding results.

The staff reviewed the applicant's description of the HI-STORM 100S Version E overpack and the HI-TRAC MS thermal models. Based on the information provided in the application regarding the thermal model, the staff determined that the application is consistent with guidance provided in Section 4.5.4, "Analytical Methods, Models, and Calculations," of NUREG-1536. Therefore, the staff concludes that the description of the thermal model is acceptable as the description is consistent with NUREG-1536 which satisfies the requirements of 10 CFR 72.236(b), 10 CFR 72.236(f), 10 CFR 72.236(g), and 10 CFR 72.236(h).

4.3 Thermal Evaluation for Normal Conditions of Storage

The applicant used the 3-D thermal model of the Version E overpack, described in the FSAR, to determine the temperature distributions under long-term normal storage conditions (including conditions of low-speed wind, unsheltered, and sheltered configurations). The applicant performed screening calculations to determine which MPC type and loading pattern would result in the highest peak cladding temperature. The peak cladding temperatures for MPC-32 Version 1 and MPC-68 Version 1 are bounded by the MPC-32 shown in FSAR Table 4.4.6 and MPC-68M shown in FSAR Table 4.III.3a, respectively. Consequently, no additional thermal evaluations are necessary for these MPCs as previous evaluated results shown in FSAR Table

4.4.6 and Table 4.III.3a show that predicted maximum temperatures for MPC-32 and MPC-68M remain bounding. FSAR Table 4.II.4.1 shows computed temperatures for all heat loading patterns considered in the analyses. Discrete heat loading Pattern A results in the highest peak cladding temperature for MPC-32M. Therefore, the applicant adopted this loading pattern to perform all thermal evaluations. The results reported in the FSAR show the maximum fuel temperature obtained under damaged fuel isolators (DFI) and damaged fuel containers (DFC) storage are essentially the same as or bounded by the temperatures computed under intact or undamaged fuel storage (FSAR Table 4.II.4.1). In this context, essentially the same or bounded means the maximum temperatures are equal to or slightly smaller than predicted temperatures for undamaged fuel. Thus, additional thermal evaluations are not needed for DFI and DFC. The addition of the BWR fuel types (10x10I, 10x10J, and 11x11A) to the approved contents for the MPC-68M only is bounded by the design basis fuel analyzed in the FSAR. Therefore, no additional thermal evaluations are needed for these fuel types. Finally, the applicant's predicted temperature results under normal storage (including low-speed wind conditions, unsheltered, and sheltered configurations) are all below applicable allowable limits, as specified in the FSAR. Consequently, the staff finds the thermal evaluations for these configurations acceptable.

The applicant calculated the maximum gas pressure in the bounding MPC-32M for a postulated release of fission product gases from fuel rods into the MPC free space. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. Based on fission gases release fractions (as provided in NUREG-1536 criteria), the rods' net free volume and initial fill gas pressure, maximum gas pressures with 1% (normal), 10% (off-normal), and 100% (accident condition) rod rupture are given in FSAR Table 4.II.4.3. The maximum computed gas pressures reported in FSAR Table 4.II.4.3 are all below the MPC internal design pressures for normal, off-normal, and accident conditions as specified in the FSAR.

The staff reviewed the applicant's thermal evaluation of the HI-STORM 100S Version E overpack during normal conditions of storage for the addition of new heat load patterns and bounding heat load. Based on the information provided in the application regarding the thermal model and evaluation, the staff determined that the application is consistent with guidance provided in Section 4.5.4 of NUREG-1536.

Therefore, the staff concludes that thermal model and evaluation are acceptable as the evaluation is consistent with NUREG-1536 which satisfies the requirements of 10 CFR 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.4 Thermal Evaluation for Short-Term Operations

4.4.1 Vacuum Drying

The applicant's methodology for performing cyclic vacuum drying for all MPCs is summarized in FSAR Section 4.II.5.3. The applicant also provided a calculation using MPC-32M at design maximum heat load and a summary of the results (maximum temperature, environment, time, etc.) in the FSAR. All temperatures are below the allowable limit defined by the applicant in the FSAR. The fuel cladding temperature limit set by the applicant in the FSAR to perform cyclic vacuum drying is lower than the allowable fuel cladding temperature limit provided in NUREG-1536. NUREG-1536 limits are applicable to this amendment. Therefore, the staff finds the thermal evaluation during vacuum drying acceptable.

The staff reviewed the applicant's thermal evaluation of the HI-STORM 100 System during vacuum drying operations. Based on the information provided in the application regarding the thermal analysis model and evaluation, the staff determined that the application is consistent with guidance provided in Section 4.5.4 of NUREG-1536.

4.4.2 Onsite Transfer

The applicant's calculated temperature and pressure results for the HI-TRAC MS and HI-TRAC transfer casks are summarized in FSAR Tables 4.II.5.2 and 4.II.5.3, respectively. The maximum temperatures for onsite transfer of previously certified MPC-32 in the HI-TRAC MS transfer cask are provided in FSAR Table 4.II.5.4. All predicted temperatures and pressures remain below applicable limits described in the FSAR. Consequently, the staff finds the thermal evaluation for onsite transfer acceptable.

The staff reviewed the applicant's thermal evaluation of the HI-STORM 100 System for short-term operations. Based on the information provided in the application regarding the thermal analysis model and evaluation, the staff determined that the application is consistent with guidance provided in Section 4.5.4 of NUREG-1536. Therefore, the staff concludes that the thermal evaluation for short-term operations is acceptable as the evaluation is consistent with NUREG-1536 which satisfies the requirements of 10 CFR 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.5 Off-Normal and Accident Events

4.5.1 Off-Normal Events

The applicant performed the thermal evaluation of the HI-STORM 100 Version E loaded with the MPC-32M for off-normal conditions. The applicant stated that the MPC-32 Version 1 and MPC-68 Version 1 canisters introduced in this amendment are bounded by previous certifications, as stated in FSAR Section 4.II.4.3. The evaluation considered multiple off-normal events including off-normal pressure, off-normal ambient temperature, partial blockage of air vents, and malfunction of forced helium dehydrator. All predicted temperatures and pressure remain below the applicable limits provided in the FSAR.

4.5.2 Accident Events

The applicant performed the thermal evaluation of the HI-STORM 100 Version E loaded with MPC-32M for accident conditions. The accident events that were considered in the evaluation include fire, extreme ambient temperature, 100% blockage of air inlet vents, burial under debris, most adverse flood water level at an ISFSI, 100% fuel rods rupture, 30-day 100% vent blockage, and loss of ventilation under sheltered configuration. All reported predicted values provided in the FSAR during accident conditions remain below the applicable limit provided in the FSAR.

The applicant performed the thermal evaluation of the HI-TRAC MS for accident conditions. The accident events considered in the evaluation include fire and blockage of HI-TRAC MS vents. All reported predicted values provided in the FSAR during accident conditions remain below the applicable limit provided in the FSAR.

The staff reviewed the applicant's thermal evaluation of accident events. Based on the information provided in the application regarding the thermal analysis model and evaluation, the staff determined that the application is consistent with guidance provided in Section 4.4.4 of NUREG-1536. Therefore, the staff concludes that the thermal evaluation for off-normal and accident events is acceptable as the evaluation is consistent with NUREG-1536 which satisfies the requirements of 10 CFR 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.6 Confirmatory Analyses

The staff reviewed the applicant’s thermal models used in the analyses, checked the code input in the calculation packages, and confirmed that the applicant used the proper material properties and boundary conditions. The staff verified that the applicant’s selected code models and assumptions were adequate for the flow and heat transfer characteristics prevailing in the HI-STORM 100S Version E geometry and analyzed conditions for the bounding configuration, as demonstrated by the applicant.

The staff also consulted engineering drawings to verify that adequate geometry dimensions were translated to the analysis models. The staff reviewed the material properties presented in the FSAR to verify that they were appropriately referenced and applied. The staff assured that the applicant performed appropriate sensitivity analysis calculations to obtain mesh-independent results that would provide bounding predictions for all conditions analyzed in the application. Additionally, the staff performed an audit review of bounding configurations, as described in the FSAR. Specifically, the staff reviewed and audited the applicant’s thermal analysis models corresponding to normal conditions of storage (including unsheltered and sheltered configurations, low-speed wind effect for unsheltered configuration, and on-site transfer of the HI-TRAC MS). The staff verified that adequate modeling options and model domains were used in order to capture important heat transfer characteristics for the different configurations. The staff also verified that the analysis results were consistent with the information provided in the FSAR.

Based on this review, the staff finds the applicant’s bounding configuration for HI-STORM 100S Version E acceptable because the thermal model and analysis results are consistent with the information provided in the FSAR and all predicted thermal results are below allowable limits described in the FSAR.

4.7 Evaluation Findings

F4.1 Chapter 2 of the FSAR describes SSCs ITS to enable an evaluation of their thermal effectiveness, in accordance with 10 CFR 72.236(f) and 72.236(h). Cask SSCs ITS remain within their operating temperature ranges, in accordance with 10 CFR 72.236(a) and 72.236(b).

F4.2 The HI-STORM 100S storage and transfer system is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems, as required by 10 CFR 72.236(f).

F4.3 The spent fuel cladding is protected against degradation leading to gross ruptures under long-term storage by maintaining cladding temperatures below 752°F (400°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal, in accordance with 10 CFR 72.236(g), 72.236(l), and 72.236(m).

F4.4 The spent fuel cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining cladding temperatures below 1,058 °F (570°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal, in accordance with 10 CFR 72.236(g), 72.236(l), and 72.236(m).

F4.5 The staff finds that the thermal design of the HI-STORM 100 storage system is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5.0 CONFINEMENT EVALUATION

The objective of the confinement review is to ensure that radiological releases from the storage system to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

The staff reviewed the information provided by the applicant and evaluated the following proposed changes that are applicable to the confinement review:

- Proposed Change #2 – Include MPC-32M.
- Proposed Change #3 – Include MPC-32 Version 1 and MPC-68 Version 1.
- Proposed Change #10 – Allow different helium leak test requirements for earlier amendments.
- Proposed Change #11 – Remove dimensions from MPC-68M drawing.
- New Appendices C and D to the CoC. (Proposed Changes #1 and #4 for the addition of HI-TRAC MS and HI-STORM 100S Version E, respectively.)

5.1 Proposed Change #2 - Include MPC-32M

The applicant's safety evaluation to include the MPC-32M fuel basket when used with the MPC enclosure vessel Version 1, that is also the confinement boundary, is in FSAR Supplement 7.II. The applicant described in the response to NRC's RAI 3-3 (Holtec, 2020a, Attachment 2), that the MPC enclosure vessel Version 1 baseplate is slightly thickened to increase its pressure and load bearing capacity when used with the MPC-32M fuel basket. The staff finds this to be acceptable based on the evaluation in Section 3.2 of this SER.

The staff also confirmed the MPC confinement boundary stress intensity limits in FSAR Table 2.II.2.3b have not changed when compared to the previously approved values in FSAR Table 2.2.10. Therefore, given that the MPC confinement boundary stress intensity limits have not changed and the materials of construction for the enclosure vessel are the same, the staff finds the MPC confinement boundary stress intensity limits to be acceptable.

The applicant included in FSAR Table 2.II.2.10 a list of the MPC enclosure vessel Version 1 confinement components. The staff verified the confinement boundary components for the MPC enclosure vessel Version 1 are (1) consistent with the applicant's confinement boundary definition in the glossary of the FSAR; (2) completely described on the licensing Drawing No.

11572R0, including the redundant closure ring and closure ring welds; and (3) described as ITS. FSAR Table 2.II.2.10 referenced Table 2.2.6, which classified the confinement components as ITS Category A, and the staff finds it to be acceptable. The staff concludes that the description of the confinement boundary is acceptable as the description is consistent with the guidance provided in NUREG-1536 which satisfies the requirements of 10 CFR 72.236(d). Based on the applicant's use of redundant closure ring welds, the staff finds that a confinement boundary closure monitoring system is not required, which is consistent with the guidance provided in NUREG-1536, and satisfies the requirements of 10 CFR 72.236(e) for redundant sealing of the confinement system.

The applicant described the use of and limitations for use of DFC (FSAR Figure 2.II.1-4) and DFI (FSAR Figure 2.II.1-5) in FSAR Table 1.II.2.3 and described the fuel assembly characteristics for damaged fuel and fuel debris in FSAR Section 2.II.1.5 for use with the MPC-32M fuel basket. The staff confirmed the use of DFC and DFI was previously approved in the HI-STORM 100 initial issuance (NRC, 2000) and in the HI-STORM 100 Amendment No. 14 (NRC, 2019a), respectively, and therefore the staff finds their use to be acceptable with the MPC-32M fuel basket because the DFC minimizes the dispersal of gross particulates and DFI prevents the potential migration of fissile material, and the applicant specified the fuel assembly characteristics for damaged fuel and fuel debris in FSAR Section 2.II.1.5 for use with the MPC-32M fuel basket. Due to the inclusion of damaged fuel or fuel debris, the staff also confirmed the entire confinement boundary is leakage rate tested to the ANSI N14.5, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," leaktight (1×10^{-7} reference cubic centimeter per second (ref-cm^3/s) of air) acceptance criterion as described by the applicant in FSAR Section 9.1.3. As the confinement boundary is leakage rate tested to the ANSI N14.5 leaktight acceptance criterion, the confinement boundary does not need a confinement release analysis. The staff also verified the helium leakage rate testing in FSAR Section 9.1.3 and Table 9.1.1 for the entire confinement boundary has not changed except as described and addressed in Section 5.3 of this SER for HI-STORM 100 System Amendment Nos. 2 through 7; and therefore, the staff finds the helium leakage rate testing to be acceptable.

The applicant evaluated normal, off-normal, and accident conditions pressures and reported the results in FSAR Table 4.II.4.3. Staff verified the results are below the allowable design pressures in FSAR Table 7.II.1.1. As they are, the staff determined the normal, off-normal, and accident conditions pressures to be acceptable. The applicant also evaluated the normal storage conditions confinement boundary component temperatures and reported the results in FSAR Table 4.II.4.2. The staff also verified the results are below the allowable limits in FSAR, Revisions 17 and 18, Table 2.2.3 (Holtec, 2019a) and Table 1.A.6 (Holtec, 2019c), and therefore the staff determined the normal storage conditions confinement boundary component temperatures to be acceptable.

The applicant described in FSAR Table 1.II.2.3 that the canister fill gas is high purity helium, specifically with a purity of greater than or equal to 99.995% as described in CoC Appendix C, Table 3-2. The staff finds the use of helium as an inert gas to reduce the potential for fuel oxidization and subsequent cladding failure to be acceptable as it is consistent with the guidance provided in NUREG-1536, which satisfies the requirements of 10 CFR 72.236(a). The applicant provided the minimum and maximum initial fill gas quantities in FSAR Table 4.II.3.1; the staff confirmed that the initial fill gas pressures established in Table 4.II.3.1 were consistent with the values in CoC Appendix C, Table 3-2. As they are consistent, the staff finds those values acceptable.

The applicant described the canister drying criteria in FSAR Table 1.II.2.3 and stated the criteria have remained unchanged for all FSAR revisions; therefore, the staff finds the canister drying criteria to provide a nonreactive helium gas environment to be acceptable since the MPC designs are similar.

The staff reviewed FSAR Table 7.II.1.4 and found it to be the same as FSAR Table 7.1.4 (Holtec, 2019c), except that in FSAR Table 7.II.1.4 the MPC is no longer designed to withstand 45 g deceleration loadings and the cask system is no longer analyzed to verify that decelerations due to credible drops and non-mechanistic tipovers will be less than 45 g's. As described in the applicant's response to RAI 3-5 (Holtec, 2020a, Attachment 2), the 45 g deceleration limit specified in FSAR Table 3.1.2 does not apply to the non-mechanistic tipover of the Version E. Therefore, the staff has determined the confinement boundary continues to meet the guidance in ISG-18, "The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage," which was incorporated into the standard review plan NUREG-1536 (NRC, 2010a). The staff also provided additional evaluation of the non-mechanistic tipover in Section 3.4 of this SER.

The applicant states that the main body of FSAR Chapter 7 remains fully applicable for the HI-STORM 100 System using an MPC-32M. The staff finds this to be acceptable because the confinement boundary components have not changed, other than the slightly thickened baseplate as addressed earlier in Section 5.1 of this SER, and the leakage rate testing and acceptance criteria have also not changed. Based on the staff's review of the changes in Supplement 7.II of the FSAR and the above considerations, the staff finds the addition of MPC-32M to be acceptable.

5.2 Proposed Change #3 - Include MPC-32 Version 1 and MPC-68 Version 1

The applicant's safety evaluation to include the MPC-32 Version 1 and MPC-68 Version 1 fuel baskets when used with the MPC enclosure vessel Version 1, that is also the confinement boundary, is in FSAR Supplement 7.II. The applicant described in its response to RAI 3-3 that the MPC enclosure vessel Version 1 baseplate is slightly thickened to increase its pressure and load bearing capacity when used with the MPC-32 Version 1 and MPC-68 Version 1 fuel baskets. The staff finds this to be acceptable based on the evaluation in Section 3.3 of this SER.

The staff also confirmed the MPC confinement boundary stress intensity limits in FSAR Table 2.II.2.3b have not changed when compared to the previously approved values in FSAR Table 2.2.10. Therefore, given that the MPC confinement boundary stress intensity limits have not changed and the materials of construction for the enclosure vessel are the same, the staff finds the MPC confinement boundary stress intensity limits to be acceptable.

The applicant included in FSAR Table 2.II.2.10 a list of the MPC enclosure vessel Version 1 confinement components. The staff verified the confinement boundary components for the MPC enclosure vessel Version 1 are (1) consistent with the applicant's confinement boundary definition in the glossary of the FSAR; (2) completely described on the licensing Drawing No. 11572, including the redundant closure ring and closure ring welds; and (3) described as ITS. FSAR Table 2.II.2.10 referenced Table 2.2.6, which classified the confinement components as ITS Category A, and the staff finds to be acceptable. The staff concludes that the description of the confinement boundary is acceptable as the description is consistent with the guidance provided in NUREG-1536, which satisfies the requirements of 10 CFR 72.236(d). Based on the applicant's use of redundant closure ring welds, the staff finds that a confinement boundary closure monitoring system is not required, which is consistent with the guidance provided in NUREG-1536, which satisfies the requirements of 10 CFR 72.236(e), for redundant sealing of the confinement system.

The applicant described the use of and limitations for use of DFC (FSAR Figure 2.II.1-4) and DFI (FSAR Figure 2.II.1-5) in FSAR Table 1.II.2.3. The applicant stated in Table 1.II.2.3 that DFIs are not permitted in the MPC-32 Version 1 or MPC-68 Version 1. The applicant also described the fuel assembly characteristics in FSAR Sections 2.II.1.2 and 2.II.1.3 for the MPC-32 Version 1 and MPC-68 Version 1, respectively. The staff confirmed the use of DFC was previously approved in the HI-STORM 100 initial issuance, and therefore the staff finds its use to be acceptable with the MPC-32 Version 1 and MPC-68 Version 1 fuel baskets because the DFC minimizes the dispersal of gross particulates, and the applicant specified the fuel assembly characteristics for damaged fuel and fuel debris in FSAR Sections 2.II.1.2 and 2.II.1.3 for use with the MPC-32 Version 1 and MPC-68 Version 1 fuel baskets, respectively. Due to the inclusion of damaged fuel or fuel debris, the staff also confirmed the entire confinement boundary is leakage rate tested to the ANSI N14.5 leaktight (1×10^{-7} ref-cm^3/s of air) acceptance criterion as described by the applicant in FSAR Section 9.1.3. As the confinement boundary is leakage rate tested to the ANSI N14.5 leaktight acceptance criterion, the confinement boundary does not need a confinement release analysis. The staff also verified the helium leakage rate testing in FSAR Section 9.1.3 and Table 9.1.1 for the entire confinement boundary has not changed except as described and addressed in Section 5.3 of this SER for HI-STORM 100 System Amendment Nos. 2 through 7; and therefore, the staff finds the helium leakage rate testing to be acceptable.

The applicant evaluated the normal, off-normal, and accident conditions pressures and reported the results in FSAR Table 4.II.4.6. The staff verified the results are below the allowable design pressures in FSAR Table 7.II.1.1. As they are, the staff determined the normal, off-normal, and accident conditions pressures to be acceptable. The applicant also evaluated the normal storage conditions confinement boundary component temperatures and reported the results in FSAR Table 4.II.4.5. The staff also verified the results are below the allowable limits in FSAR Table 2.2.3 (Holtec, 2019a) and Table 1.A.6 (Holtec, 2019c), and therefore the staff determined the normal storage conditions confinement boundary component temperatures to be acceptable.

The applicant described in FSAR Table 1.II.2.3 that the canister fill gas is high purity helium, specifically with a purity of greater than or equal to 99.995% as described in CoC Appendix C Table 3-2. The staff finds the use of helium as an inert gas to reduce the potential for fuel oxidization and subsequent cladding failure to be acceptable as it is consistent with the guidance provided in NUREG-1536, which satisfies the requirements of 10 CFR 72.236(a). The applicant provided the minimum and maximum initial fill gas quantities in FSAR Table 4.II.3.2; the staff confirmed that the initial fill gas pressures established in Table 4.II.3.2 were consistent with the values in CoC Appendix C, Table 3-2. As they are consistent, the staff finds those values acceptable.

The applicant described the canister drying criteria in FSAR Table 1.II.2.3 and stated that the criteria have remained unchanged for all FSAR revisions. Since the new MPCs are similar to previous versions, the staff finds the canister drying criteria to provide a nonreactive helium gas environment to be acceptable.

The staff reviewed FSAR Table 7.II.1.4, which compares the Holtec MPC design with ISG-18 guidance for storage. Staff found it to be the same as FSAR Table 7.1.4 (Holtec, 2019c), except that in Table 7.II.1.4 the MPC is no longer designed to withstand 45 g deceleration loadings and the cask system is no longer analyzed to verify that decelerations due to credible drops and non-mechanistic tipovers will be less than 45 g's. As described in the applicant's response to RAI 3-5 (Holtec, 2020a, Attachment 2), the 45 g deceleration limit specified in FSAR Table 3.1.2 does not apply to the non-mechanistic tipover of the HI-STORM 100S Version E. Therefore, the staff has determined the confinement boundary continues to meet the ISG-18 guidance which was incorporated into the standard review plan NUREG-1536 (NRC, 2010a). The staff also provided additional evaluation of the non-mechanistic tipover in Section 3.4 of this SER.

The applicant stated that the main body of FSAR Chapter 7 remains fully applicable for the HI-STORM 100 System using an MPC-32 Version 1 or MPC-68 Version 1 fuel basket. The staff finds this to be acceptable because the confinement boundary components have not changed, other than the slightly thickened baseplate as addressed earlier in Section 5.2 of this SER, and the leakage rate testing and acceptance criteria have also not changed. Based on the staff's review of the changes in FSAR Supplement 7.II and the above considerations, the staff finds the addition of MPC-32 Version 1 and MPC-68 Version 1 to be acceptable.

5.3 Proposed Change #10 – Allow Different Helium Leak Test Requirements for Earlier Amendments

Proposed change #10 affects certain casks loaded under Amendment Nos. 2 through 7 and could allow licensees with those affected casks to upgrade to Amendment No. 15. Casks loaded to Amendment Nos. 2 through 7 used different helium leakage rate test requirements. The applicant described the fabrication helium leak test in HI-STORM 100 System, Amendment No. 15, CoC Appendix A, Section 5.8, "Fabrication Helium Leak Test," and the field helium leak test in CoC Appendix A limiting condition for operation (LCO) 3.1.1, both in accordance with the ANSI N14.5 leaktight acceptance criteria. The fabrication helium leak test and the field helium leak test had been described in CoC Condition 3 in Amendment No. 7 (Holtec, 2009a) and CoC Appendix A, LCO 3.1.1 in Amendment No. 7 (Holtec, 2009b) and is also described in the FSAR Section 9.1.3 and Table 9.1.1. The helium leak testing language was further clarified to add helium leak testing on the lid base metal and cover plate base metal to CoC Condition 3 in Amendment No. 8 (Holtec, 2012a). With proposed change #10, licensees who loaded casks under Amendment Nos. 2 through 7 may be able to upgrade to Amendment No. 15 after following the 10 CFR 72.212 requirements.

5.3.1 ANSI N14.5 Helium Leak Test

As stated in CoC Appendix A, Section 5.8 and LCO 3.1.1, Holtec uses ANSI N14.5 leak testing during fabrication and in the field to certify leak-tightness of the confinement boundary; this test demonstrates that leakage is not considered credible during normal and accident conditions. Leak testing has been and continues to be an acceptable way to meet the confinement requirements of 10 CFR Part 72. In the SER for Amendment No. 8 (NRC, 2011d), the staff determined that if using ANSI N14.5 as the basis for leaktight certification, the entire confinement boundary including the confinement welds and base material (i.e., the lid, shell, baseplate, and vent and drain port covers) would be required to be helium leak tested.

5.3.2 Fabrication Helium Leak Test

For the proposed change #10, the applicant moved the description of fabrication helium leak test from the previously approved CoC Condition 3 to CoC Appendix A, Section 5.8. This is acceptable because the CoC Condition 3 and technical specifications (TS) (Appendix A, Section 5.8) include the same test requirements.

In Amendment No. 15, the applicant described in CoC Appendix A, Section 5.8 that casks initially loaded to Amendment Nos. 2 through 7 before July 1, 2009, do not meet the fabrication helium leak test requirements in CoC Appendix A, Section 5.8 after the completion of the MPC shell to baseplate welding. The applicant continued to describe that these casks may still be upgraded to Amendment No. 15. As part of its review of proposed change #10, the staff reviewed Holtec's response to EA-09-190 (Holtec, 2009c). In its response under the section that described the corrective actions taken in response to the violation and corrective steps taken to reinstate helium leak rate testing, Holtec stated that, "Leakage testing has been reinstated at the manufacturing facility for all MPCs currently being fabricated or in storage. Leakage tested on newly fabricated MPCs was reinstated on July 1, 2009." Therefore, the staff finds the use of the date of July 1, 2009, to be acceptable in CoC Appendix A, Section 5.8 for Amendment No. 15.

NRC letter, "Response to Holtec International (Holtec) Reply to EA-09-190" (NRC, 2010b), described the need for additional site-specific information to ensure that the loaded casks would continue to meet the regulatory requirements. The NRC conversation record, "Follow-up Actions Regarding NRC Enforcement Discretion Letter No. EA-09-190" (NRC, 2009a), described the follow-up actions to the NRC Enforcement Discretion Letter No. EA-09-190 (NRC, 2009b). These follow-up actions included the need for site-specific information to be provided by the affected 10 CFR Part 50 general licensees related to their determinations that the affected MPC's could remain in service. The staff requested the following information to assist in reviewing the corrective actions:

1. Verification that the decay heat in each cask is below 21 kilowatts (kW), or if it is above 21 kW, the length of time that would elapse before the spent fuel heat load would decay below 21 kW;
2. Verification that the site radiological monitoring programs had not detected adverse effluent conditions associated with the casks and that all measured site radiological parameters were within the limits provided in 10 CFR 72.104; and
3. Verification that the sites dispositioned the deficiency through their non-conforming and corrective action process to reinstate leakage rate testing and determined that continued use of the casks was found to be acceptable.

As part of its review of proposed change #10, the staff reviewed the affected general licensees' responses to items 1 through 3 in the conversation record (SNC, 2010; Exelon, 2010; TVA, 2010; PSEG, 2010; EN, 2010; Entergy, 2010). The staff also reviewed the NRC responses to the respective general licensees (NRC, 2010c, 2010d, 2010e, 2011a, 2011b, 2011c), where the NRC determined, in each of the NRC response letters, that continued use of the loaded MPCs at the licensees' sites was acceptable. At that time, the staff's conclusion for each general licensee's continued use of loaded MPCs was based on the verification of the decay heat in each cask being below 21 kW, no adverse effluent conditions associated with the use of the affected MPCs, and all measured site radiological parameters being within the limits provided in 10 CFR 72.104, and appropriate disposition of the deficiency. The staff determined that these factors continue to provide reasonable assurance that the continued use of MPCs loaded under Amendment Nos. 2 through 7 are safe. Therefore, based on the staff's review of the previously described documents, the staff finds that CoC No. 1014, HI-STORM 100 System casks fabricated and loaded to Amendment Nos. 2 through 7 before July 1, 2009, that had different helium leak test requirements, and do not meet the fabrication helium leak test required after the completion of the MPC shell to baseplate welding continue to be safe. Accordingly, the staff concludes that these casks may be upgraded to Amendment No. 15 need not perform additional helium leak tests.

5.3.3 Fabrication Leak Test for the Lid Base Metal

For the proposed change #10, the applicant also stated in CoC Appendix A, Section 5.8, that casks initially loaded to Amendment Nos. 2 through 7 before July 1, 2009, must meet the fabrication helium leak test requirement for the lid base metal, of the amendment to which the casks were initially loaded. The staff reviewed the SER for CoC No. 1014, HI-STORM 100 System, Amendment No. 8 (NRC, 2011d) and confirmed that the description of leakage rate testing for the lid base metal was added to CoC Condition 3 of CoC No. 1014, HI-STORM 100 System in Amendment No. 8 (Holtec, 2012a). The absence of leakage rate testing for the lid base material was not, however, evaluated as part of the NRC enforcement activities related to EA-09-190. The staff has determined, based on its review of the NRC conversation record (NRC, 2009a) and the NRC's response to Holtec's Reply to EA-09-190 (NRC, 2010b), that the information requested by staff as part of the NRC conversation record as mentioned in Section 5.3.2 of this SER and the associated site-specific responses nonetheless continues to be supportive of a safety finding. Specifically, the staff concludes that the CoC No. 1014, HI-STORM 100 System initially loaded to Amendment Nos. 2 through 7 can be upgraded without performing leakage rate testing on the base metal for the lid because:

1. the decay heat for each loaded cask is below 21 kW,
2. site radiological monitoring programs had not detected adverse effluent conditions and all measured site radiological parameters were within the limits provided in 10 CFR 72.104, and
3. general licensees' sites dispositioned the deficiency through their non-conforming and corrective action process to reinstate leakage rate testing.

The staff finds that while the casks initially loaded to CoC No. 1014, HI-STORM 100 System, Amendment Nos. 2 through 7 did not have the lid base metal leakage rate tested, the casks loaded to Amendment Nos. 2 through 7 were in compliance with CoC No. 1014, HI-STORM 100 System, CoC Condition 3 effective at the time, and continue to be safe. Accordingly, the staff concludes that these casks may be upgraded to Amendment No. 15 need not perform additional helium leak tests. .

5.3.4 Field Helium Leak Test

For the proposed change #10, the applicant also stated in CoC Appendix A, LCO 3.1.1, Note 2, that the MPC helium leak rate limit for cover plate base metal listed in Condition D and SR 3.1.1.3 is not applicable to casks that were initially loaded to Amendments Nos. 2 through 7. The leakage rate testing for the vent and drain port cover plate base metal was also added as part of CoC No. 1014, HI-STORM 100 System, Amendment No. 8, Condition 3 (Holltec, 2012a), and to Appendix A LCO 3.1.1 and surveillance requirement 3.1.1.3 of the CoC No. 1014, HI-STORM 100 System, Amendment No. 8 (Holtec, 2012b); however, the absence of leakage rate testing for the vent and drain port cover plate base material was not evaluated as part of the NRC enforcement activities related to EA-09-190. The staff has determined based on its review of the NRC conversation record (NRC, 2009a) and the NRC's response to Holtec's Reply to EA-09-190 (NRC, 2010b), that the information requested by staff as part of the NRC conversation record as mentioned in Section 5.3.2 of this SER and the associated site-specific responses continue to be supportive of a finding that these casks may be used safely.

Specifically, the staff concludes that the CoC No. 1014, HI-STORM 100 System initially loaded to Amendment Nos. 2 through 7 can be upgraded without performing leakage rate testing on the vent and drain port cover plate base metal because:

1. the decay heat for each loaded cask is below 21 kW,
2. site radiological monitoring programs had not detected adverse effluent conditions and all measured site radiological parameters were within the limits provided in 10 CFR 72.104, and
3. general licensees' sites dispositioned the deficiency through their non-conforming and corrective action process to reinstate leakage rate testing.

The staff finds that while the casks initially loaded to CoC No. 1014, HI-STORM 100 System, Amendment Nos. 2 through 7 might not have had the vent and drain port cover plate base metal leakage rate tested based on the applicant's language in CoC Appendix A, LCO 3.1.1, Note 2, the casks initially loaded to Amendment Nos. 2 through 7 were in compliance with CoC No. 1014, HI-STORM 100 System CoC Condition 3, and to Appendix A LCO 3.1.1 and surveillance requirement 3.1.1.3 effective at the time, and continue to be safe. Accordingly, the staff concludes that these casks may be upgraded to Amendment No. 15 need not perform additional helium leak tests.

In Amendment No. 15, the applicant described the field helium leak test of the vent and drain port welds in CoC Appendix A, LCO 3.1.1 and in FSAR Section 9.1.3 and Table 9.1.1, including the helium leak test of the vent and drain port cover plate base metal. The staff finds, based on the review of the HI-STORM 100 CoC No. 1014, Amendment No. 15, Appendix A, LCO 3.1.1, and FSAR Section 9.1.3 and Table 9.1.1, consistent with ANSI N14.5, that the helium leak test of the vent and drain port welds includes the helium leak test of the vent and drain port cover plate base metal, and therefore is acceptable.

5.3.4 Conclusion

The staff finds it acceptable that the CoC No. 1014, HI-STORM 100 Systems initially loaded to Amendment Nos. 2 through 7 before July 1, 2009: (1) do not meet the fabrication helium leak test required after the completion of the MPC shell to baseplate welding, and (2) must meet the fabrication helium leak test requirement for the lid base metal, of the amendment to which they were initially loaded. This conclusion is based on the above discussion. Casks initially loaded to Amendment Nos. 2 through 7 on or after July 1, 2009, and casks loaded to Amendment No. 8 forward shall have a fabrication helium leak test at completion of welding the MPC shell to the baseplate, which includes a fabrication helium leak test for the lid base metal, as described in CoC No. 1014, HI-STORM 100 System, Amendment No. 15, Appendix A, Section 5.8.

In Amendment No. 15, the applicant described the field helium leak test of the vent and drain port covers in Appendix A, LCO 3.1.1, and FSAR Section 9.1.3 and Table 9.1.1, consistent with ANSI N14.5; therefore, based on the discussion in SER Section 5.3.3 on the field helium leak test the staff finds this description acceptable. Holtec uses ANSI N14.5 leak testing during fabrication and in the field to certify leak-tightness of the confinement boundary; therefore, leakage is not considered credible during normal and accident conditions.

In this amendment request, the applicant did not make any change to the confinement boundary leakage rate tests acceptance criterion which must still be performed in accordance with the ANSI N14.5 leaktight acceptance criterion as described in CoC Appendix A, Section 5.8 and LCO 3.1.1, as well as in FSAR Section 9.1.3 and Table 9.1.1, and the staff continues to find that description acceptable. The staff finds that, for the casks affected by proposed change #10, there is reasonable assurance of adequate protection for confinement and that licensees may follow 10 CFR 72.212 requirements to upgrade these casks to Amendment No. 15.

5.4 Proposed Change #11 – Remove Dimension from MPC-68M Drawing

The applicant stated that Drawing No. 7195 for the MPC-68M was updated to remove dimensions that are not needed for safety analyses. The staff reviewed Drawing No. 7195 and confirmed that the components on the drawing were not confinement boundary components; therefore, confinement was not impacted by this change.

5.5 New Appendices C and D to the CoC

The Appendices C and D of CoC No. 1014 are new to Amendment No. 15 for HI-STORM 100S Version E cask and HI-TRAC MS. The staff reviewed Appendices C and D of the TS with respect to confinement. Based on the staff's review of the TS, the staff found that the CoC includes: (1) leak testing of the vent and drain port cover plates base metal and confinement welds to the ANSI N14.5 leaktight acceptance criteria in Appendix C, Section 5.4; (2) the fabrication helium leak test to the ANSI N14.5 leaktight acceptance criteria in Appendix C, Section 5.4; and (3) vacuum drying criteria that were consistent with FSAR Table 1.II.2.3. Therefore, the staff finds this to be acceptable.

5.6 Findings

F5.1 On the basis of review of the proposed changes described in Section 5.0 of this SER and the submitted documents, the staff concludes that the proposed changes have no negative impact on the confinement system and that the HI-STORM 100 System continues to meet the confinement requirements of 10 CFR Part 72.

6.0 SHIELDING EVALUATION

The objective of the shielding review is to ensure that the design features relied on for shielding provide adequate protection against direct radiation from the HI-STORM 100 System contents. In reviewing the proposed changes to the HI-STORM 100 System's shielding capability, the staff followed the guidance in Chapter 6 of NUREG-2215 which documents the basis for the staff's approval for the proposed changes with respect to shielding and meeting the applicable dose limits in 10 CFR Part 72. The applicant's proposed changes and the staff's evaluation follows.

6.1 Proposed Change #1 – Addition of HI-TRAC MS

The applicant proposed a new transfer cask called the "HI-TRAC MS" which includes an option for a variable thickness for the lead and water jacket and adds cooling passages to the bottom lid. The variations in lead and neutron shield water jacket are designed to accommodate general licensees with lower crane capacities and ensure that the dose rates remain as low as reasonably achievable (ALARA) by ensuring the contents are loaded to meet the dose rate limit in TS 5.3.4.c to Appendix C to the certificate of compliance. The HI-TRAC MS is for use with all previously approved MPCs as well as the MPC-32M, MPC-32 Version 1, and MPC-68 Version 1 that are part of this amendment request (see proposed change #3 discussed in Section 6.3 of this SER). The HI-TRAC MS is only for use with the HI-STORM 100S Version E.

6.1.1 Shielding Design Description

The staff reviewed Drawing No. 11381 R0 of the HI-TRAC MS and found that it is sufficiently detailed to support a review. The drawing contains nominal dimensions. The staff found this acceptable as the applicant modeled nominal dimensions within its shielding evaluations. The main shielding features of the HI-TRAC MS are the lead shield for gamma shielding and the water jacket for neutron shielding that are part of the multi-shell cylindrical cask body on the radial side. The bottom lid is removable to facilitate loading and unloading operations and is made from steel. The steel MPC enclosure lid provides the shielding at the top of the transfer cask. The water jacket and the lead shielding are both variable in thickness with the minimum allowable thickness defined in FSAR Table 3.II.2.3. The minimum dimensions are not specified in the drawing, however there are flag notes referencing the appropriate sections of the FSAR where these dimensions are defined.

6.1.2 Radiation Source Definition

As specified in CoC Appendix D, Section 2.1, all of the currently authorized MPCs are allowed to be used within the HI-TRAC MS as well as the proposed MPCs within this amendment. Therefore licensees can use the following MPCs within the HI-TRAC MS: MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F, MPC-68FF, and the MPC-68M; and the proposed MPC-32M, MPC-68 Version 1, and the MPC-32 Version 1.

The allowable fuel for the HI-TRAC MS includes all of the spent fuel assemblies, including damaged and undamaged assemblies along with fuel debris and non-fuel hardware, that are allowed within the above-mentioned MPCs.

CoC Appendix D, Table 2.1-1 has the allowable contents for MPC-32M. The allowable contents for the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F, MPC-68FF, MPC-68M, MPC--68 Version 1, and MPC-32 Version 1, are listed in CoC Appendix B, Table 2.1-1. CoC Appendix D Sections 2.1.d and 2.1.e state that the Appendix B tables for the MPC-68 and MPC-32 can be used for the MPC-68 Version 1 and the MPC-32M Version 1, respectively. The MPC-32M, MPC-68 Version 1, and MPC-32 Version 1 were added as part of this amendment (see Sections 6.2 and 6.3 of this SER).

As shown in FSAR Table 5.II.4.3, the MPC-32M and its allowable contents give the highest dose rate. Consequently, the MPC-32M was used to perform all dose rate calculations for the HI-TRAC MS. The staff found this appropriate and acceptable. Because the shielding on the HI-TRAC MS is variable to account for different crane capacities and loadings, the applicant used several demonstration evaluations with various content and shielding configurations to calculate dose rates around the HI-TRAC MS. The HI-TRAC MS includes a technical specification dose rate limit. As demonstrated by FSAR Table 5.II.4.1, calculations using minimum shielding thicknesses and bounding burnup, enrichment, and cooling time combinations from spent fuel and allowable hardware, as described in FSAR Section 5.II.2, would not meet this dose rate limit. Therefore, as discussed in FSAR Section 5.II.4.1, as a more realistic representation of the HI-TRAC MS, considering its configurable shielding options and operational limitations, the applicant also performed calculations using thicker shielding sufficient to match the dose rate limit and a higher cooling time for evaluations with the minimum thickness of the shielding. To be clear, however, the TS dose rate limits bind general licensees. General licensees may not load the casks with a shielding thickness such that they fail to meet that limit.

The MPC-32M employs a new strategy for defining allowable contents. Section 6.2.2 of this SER discusses how the bounding source term was determined for both normal and accident conditions.

6.1.2.1 Accident Conditions

The accident condition for the HI-TRAC MS is the loss of the neutron shield. As the shielding characteristics change under the accident condition, the maximum dose comes from fuel with different burnup/cooling time characteristics than the normal condition dose rates. Consequently, the applicant used a different source definition for accident conditions than for normal conditions. For the loss of the neutron shield for the HI-TRAC MS, the applicant used the same procedure discussed in Section 6.2.2.1 of this SER to determine the source term that gives the maximum dose. The fuel characteristics the applicant determined to be bounding at 100 meters for the loss of neutron shield condition are shown in Table C.4.7 of Holtec's proprietary Report HI-2188253, Revision 5. As discussed in Section 6.2.2.1 of this SER, the staff found the procedure to determine contents for the loss of neutron shield used by the applicant reasonable, and based on the values of burnup, enrichment and cooling time in Table C.4.7, the staff finds that it has reasonable assurance that this produces a source term that maximizes neutron dose rates.

6.1.3 Shielding Model Specification

The applicant analyzed the normal condition with the dry MPC cavity and accident conditions for the loss of the neutron shield. The HI-TRAC MS contains a variable thickness of the lead and neutron shield. For the normal condition calculations, the applicant performed calculations with the bounding minimum allowable lead and water jacket thicknesses as specified in FSAR Table 3.II.2.3 as well as other thicknesses to demonstrate more realistic configurations.

6.1.3.1 Configuration of the Shielding and Source

The applicant stated in FSAR Section 5.II.3.1 that it used nominal dimensions within the shielding models with the exception of the HI-TRAC MS variable gamma and neutron shield. The variable gamma and neutron shields were modeled using a range of dimensions specified in FSAR Table 5.II.3.1, including minimum dimensions. Using these minimum dimensions is conservative as it provides the minimum amount of shielding. The applicant also used adjusted dimensions to provide more realistic dose rates that would meet TS dose rate limits. Therefore, the staff found the modeling of the gamma and neutron shields acceptable because dose rate limits would be met based on conservative and realistic modeling demonstrated by the applicant. The applicant provided the tolerances of the dimensions in the proprietary Attachment 1 to its April 28, 2020, RAI response (Holtec, 2020a) and the staff found that the tolerances are sufficiently small that nominal dimensions are sufficient to represent the other HI-TRAC MS components within the shielding evaluation.

Under hypothetical accident conditions where the neutron shield is assumed to be lost and replaced with void, the applicant modeled the minimum lead shielding thickness. This is discussed in FSAR Section 5.II.1.2. The staff found it conservative because it simulates more dose than a person would receive and is therefore acceptable.

The staff has reasonable assurance that all components important for shielding are appropriately represented based on its independent calculations which were developed by Oak

Ridge National Laboratory (ORNL) using design information, and the results show relative agreement with the applicant's results (see Section 6.1.5 of this SER).

6.1.3.2 Material Properties

The applicant documents the density and composition of the materials used to model the HI-TRAC MS in FSAR Table 5.II.3.2. While the applicant did not specify the composition and density of the gamma and neutron shield assumed in this table, these are elemental materials (lead and water). As the values used within Section A.3.1 of Report HI-2188253, Revision 5 are consistent with open literature (PNNL-15870, Revision 1, "Compendium of Material Composition Data for Radiation Transport Modeling" [McConn, 2011]), the staff found the values used in the calculation acceptable.

Section A.2.3 of Report HI-2188253, Revision 5 states that the carbon steel component material properties are from the HI-STORM FW FSAR. The staff reviewed the composition and density of carbon steel in this document and found that they are also consistent with standard values from open literature, PNNL-15870, Revision 1 (McConn, 2011), and are acceptable for this application.

6.1.4 Shielding Analyses

6.1.4.1 Computer Codes

The applicant performed the shielding analyses using an updated version of MCNP, MCNP5 Version 1.51 using default cross sections. Other previously approved analyses within the HI-STORM 100 FSAR used MCNP-4A (NRC, 2000). The staff found that using a more recent version of this code is acceptable.

The applicant described its variance reduction technique in Section A.2.6 of Report HI-2188253, Revision 5. The staff performed independent calculations using a different code and variance reduction technique as discussed in Section 6.1.5 of this SER, the staff found that it has reasonable assurance that the applicant's variance reduction technique as implemented is acceptable based on the staff's calculations having general agreement with the applicant's.

6.1.4.2 Flux-to-Dose-Rate Conversion Factors

The applicant stated in Section 5.4 of the main part of the SAR that it used the flux-to-dose-rate conversion factors, the 1977 ANS/ANSI-6.1.1 standard, "Neutron And Gamma-Ray Fluence-To-Dose Factors." Section 5.II.4.1 of the SAR, discussing the evaluations performed for the HI-TRAC MS as well as the HI-STORM 100S Version E, states that the same methodology used in Section 5.4 of the SAR is used for dose rate calculations. The staff found the flux-to-dose-rate conversion factors used by the applicant acceptable because it is consistent with applicable guidance in NUREG-2215.

6.1.4.3 Dose Rates

The applicant discussed its tally specification in Section A.2.7 of Report HI-2188253, Revision 5. The staff reviewed this information and based on its judgment found that the tally bins are reasonably small enough to represent a maximum dose rate. The staff also performed independent calculations, discussed in Section 6.1.5 of this SER, to support its finding that the applicant's modeling of the HI-STORM 100 System, including tally specification, is reasonable.

6.1.4.3.1 Normal Conditions

The applicant presented its design basis dose rate calculations for normal conditions for the HI-TRAC MS in FSAR Table 5.II.1.3. These are considered design basis as the shielding was adjusted to a greater thickness that results in a dose rate that meets the CoC Appendix C Section 5.3.4.c TS dose rate limit of 4 Rem/hr at the side of HI-TRAC MS. The staff reviewed the applicant's results. Based on that review, the staff found the dose rates reported in FSAR Table 5.II.1.3 to be appropriately representative of the design under normal conditions and therefore acceptable.

FSAR Table 5.II.4.1 shows that, with bounding content and minimum shielding, the dose rates at the side are much higher than the TS limit. Therefore, this combination would never be allowed to be loaded and is an unrealistic representation of the HI-TRAC MS. For the system to meet the TS limits, (1) the shielding on the side must be increased, (2) contents that give lower dose than the bounding content must be loaded, or (3) some combination thereof. The applicant has performed dose rate calculations for these two scenarios (i.e., increased side shielding and content management) plus other loading configurations, including reference shielding thickness and an annulus filled with water to provide dose rate estimates around the HI-TRAC MS. This approach considers (1) the allowable content, (2) range of shielding thickness for the transfer cask, (3) technical specification dose rate limits, and (4) operational conditions. Since the transfer cask is used for short-term loading operations, meaning licensee employees will be working near the cask, the staff also uses occupational dose limits in 10 CFR 20.1201 to determine if the system meets 10 CFR 72.236(d) which requires that the cask vendor ensures that the requirements of 10 CFR 72.104(b) and (c) are met. Consideration for all these factors (e.g. content management, applicant's calculations using a range of shielding thicknesses, and technical specification dose rate limits) provides the staff with reasonable assurance that the applicant has demonstrated that the HI-TRAC MS meets regulatory requirements.

6.1.4.3.2 Accident Conditions

Under accident conditions, the maximum dose rate from FSAR Table 5.II.1.4 shows that, if a 30-day duration of an accident is assumed, the dose at 100 meters from the HI-TRAC MS meets the requirement in 10 CFR 72.106. For this simulation, the applicant assumed a bounding content and minimum shielding, based on the above discussion, this is a very conservative representation.

The staff found that the applicant demonstrated that the HI-TRAC MS will meet the accident condition dose rates requirement in 10 CFR 72.106. Based on this, and staff's finding that the HI-TRAC MS meets the requirements of 10 CFR 72.104, staff finds it meets the requirement in 10 CFR 72.236(d).

6.1.5 Independent Calculations

The staff performed independent calculations to confirm the source term for the MPC-32M as discussed in Section 6.2.5 of this SER. For the shielding calculations, the staff used an ORNL-developed model of the HI-TRAC MS using design information found in the drawings and the FSAR.

The ORNL model employed the Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) code Version 4.0 which is available through the Radiation Safety Information Computational Center at ORNL. UNF-ST&DARDS is a comprehensive integrated data and analysis tool developed for the U.S. Department of Energy, Office of Nuclear Energy, Office of Spent Fuel and Waste Disposition with support from the NRC. UNF-ST&DARDS simplifies and automates performance of spent fuel analyses. UNF-ST&DARDS uses the SCALE 6.2.3 code system which employs ORIGAMI for source term evaluations and Monaco/MAVRIC for the dose rate calculations.

6.1.5.1. Normal Condition

The staff also performed independent calculations to confirm the dose rates from the HI-TRAC MS. The staff modeled the same source term as the applicant did for the MPC-32M for the HI-TRAC MS, as documented in Table C.4.4 of Report HI-2188253, Revision 5. Although the applicant uses different burnup, enrichment, and cooling time combinations for the discrete loading zones to maximize dose rates at various locations around the HI-TRAC MS, the staff only calculated the dose using the loading determined by the applicant to maximize dose at the side, top, and bottom of the HI-TRAC MS.

The staff used the Westinghouse 17x17 assembly type in its calculations. The pressurized water reactor (PWR) design basis fuel assembly for the HI-STORM 100 is the B&W 15x15 as documented in FSAR Table 5.II.2.1. The staff, however, chose the Westinghouse 17x17 based on its availability within UNF-ST&DARDS at the time when the staff performed these calculations. As documented in NUREG/CR-6716, “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks” (NRC, 2001), and discussed in Sections 6.2.2.1.1 and 6.5 of this SER, with regard to different assemblies, the mass of the assemblies makes the greatest difference in dose. Given that the Westinghouse 17x17 and the B&W 15x15 assembly types have nearly identical masses, the Westinghouse 17x17 assembly type should result in a similar dose to the B&W 15x15 assembly type. The staff added the source from burnable poison rod assemblies in every location as is allowed in the MPC-32M basket (discussed in Section 6.2.2.3 of this SER).

The results of the staff’s calculations were much lower than that of Holtec’s which caused ORNL to redefine the tally structure. ORNL found that due to the highly heterogeneous loading pattern, the tally cell size needs to be sufficiently small to capture the dose rate peaks around the cask associated with the assemblies with higher source terms. With the new tally specification, ORNL found that the results were in overall agreement with the dose rates reported in the FSAR Table 5.II.1.6. The staff found that the agreement between ORNL’s confirmatory results provides reasonable assurance that the applicant has appropriately modeled the HI-TRAC MS as well as demonstrates that the applicant’s tally specification is appropriate to capture these localized peaks in dose rate.

6.1.5.2 Accident Condition

The staff confirmed the accident condition dose rates using UNF-ST&DARDS for the loss of the neutron shield. For this calculation, the staff used the design basis B&W 15x15 fuel assembly as well as the associated design basis depletion information from FSAR Table 5.2.1. The staff used the same fuel loading specifications for the discrete loading from Table C.4.7 of Report HI-2188253, Revision 5 that the applicant determined to be bounding. The staff also used fuel characteristics from a regionalized loading that the staff concluded would give high accident dose rates for this event (regionalized and discrete loading patterns are discussed in Section

6.2.2.1.4 of this SER). The staff included the Cobalt-60 (Co-60) source from burnable poison rod assemblies (BPRAs) in every location (discussed in Section 6.2.2.3 of this SER). The staff's results from both calculations (regionalized and discrete loadings) agree with the applicant's result in FSAR Table 5.II.1.4. Based on this, the staff concludes there is reasonable assurance the applicant has appropriately modeled the HI-TRAC MS under accident conditions, and that it meets the dose limit in 10 CFR 72.106 at 100 meters under accident conditions.

6.1.6 Conclusion

Based on the above considerations, the staff found that with the addition of the HI-TRAC MS, the HI-STORM 100 System design continues to meet 10 CFR Part 72 regulations pertaining to shielding in 10 CFR 72.236(d).

6.2 Proposed Change #2 – Inclusion of MPC-32M

The applicant requested the inclusion of the MPC-32M. This new basket, made entirely of Metamic^{TM}-HT, is for storing PWR SNF. In this amendment, the applicant also requested for the MPC-32M to be included in the HI-STORM 100S Version E Overpack and all transfer casks approved in all previous amendments for the HI-STORM 100, including the new HI-TRAC MS which is a part of this amendment.

6.2.1 Shielding Design Description

Although the MPC-32M basket will provide some shielding, the basket is made of Metamic^{TM}-HT which is mostly aluminum and does not provide significant shielding.

The MPC-32M will be used with MPC enclosure vessel Version 1. The staff reviewed the design of the MPC-32M basket in Drawing No.11425R0 and the design of the enclosure vessel Version 1 in Drawing No. 11572R0 and determined they are sufficiently detailed to perform a review of this component. The applicant provided tolerances for the MPC-32M as well as for the enclosure vessel Version 1 in the proprietary Attachment 1 to its April 28, 2020, RAI response (Holtec, 2020a). The enclosure vessel lid is the only shielding when the basket is used in the HI-TRAC transfer casks. The staff found the thickness of this shielding component appropriately specified in terms of a minimum thickness on Drawing No. 11572R0. The sides and the bottom of the enclosure vessel Version 1 provide far less shielding than the HI-TRAC transfer casks or the HI-STORM 100S Version E overpacks that will contain the MPC-32M. Therefore, the staff found it acceptable for these parts of the enclosure vessel Version 1 to be specified as nominal dimensions. Overall the staff found that the licensing drawings of the MPC-32M were appropriate for specifying the design specifications important for safety related to shielding.

6.2.2 Radiation Source Definition

The allowable contents for the MPC-32M are shown in CoC Appendix D, Table 2.1-1, Section V.A. This includes uranium dioxide (UO_2) assemblies with or without non-fuel hardware and damaged fuel and fuel debris with or without non-fuel hardware. The PWR fuel assembly characteristics are shown in CoC Appendix D, Table 2.1-2. The applicant defines the allowable contents, which form the basis for the source term definition, as required by 10 CFR 72.236(a) using the procedure discussed in the following subsections.

6.2.2.1 Zircalloy Clad Spent Nuclear Fuel Source Term

6.2.2.1.1 Design Basis Assembly

The applicant performed the dose rate calculations for the zircaloy clad-fuel using the B&W 15x15 assembly type, which has been used in the FSAR calculations supporting the other previously approved parts of the HI-STORM 100 System (HI-STORM 100 FSAR Section 5.1). The description of the design basis fuel is in FSAR Table 5.2.1. As discussed below, mass of heavy metal (uranium) is the most important assembly characteristic for determining the limiting assembly in terms of source term. Given that, to calculate maximum dose rates, an applicant would use the assembly with the highest uranium mass to perform the shielding evaluations. The staff calculated the UO_2 mass of each of the assemblies in CoC Appendix D, Table 2.1-2 based on the listed characteristics. The staff found that a few assemblies had higher uranium masses than that of the design basis fuel assembly in FSAR Table 5.2.1. This includes the 15x15D, 15x15E, 15x15F and 15x15G. The largest difference was about 4% higher than that of the design basis assembly.

ORNL performed a sensitivity study on the effect of different properties, including uranium mass, on dose rates in NUREG/CR-6716 (NRC, 2001). This report states that uranium mass is of intermediate importance to evaluating dose rates. Section 3.4.2.3 of NUREG/CR-6716 discusses the results of the sensitivity study which increased the uranium mass by 10%. The study found a 6% increase in dose rates from neutron radiation and 2% increase in gamma dose rates. Therefore, the staff found that the small differences in mass, less than about 4%, from non-design basis assemblies would only have a small impact on system dose rates and found the use of the B&W 15x15 as defined in FSAR Table 5.2.1 appropriate for this evaluation.

6.2.2.1.2 Fuel Hardware

The applicant showed the Co-60 contribution from the activation of hardware associated with the fuel assembly in Table 5.II.2.3 (for the design basis assembly) for an example burnup and cooling time. The Co-60 source assumed in Table 5.II.2.3 is about 10% lower than the values assumed in FSAR Table 5.2.11 used for the MPC-32 and MPC-24 baskets for the same burnup and cooling time. This decrease in Co-60 would have a non-conservative effect on dose rates. That said, the staff found the change in Co-60 to be relatively small. Further, the original calculation discussed in FSAR Section 5.2.1 uses conservative Co-59 impurities. This means reducing the amount of Co-60 is likely more realistic. Therefore, the staff found that the decrease in Co-60 source is still reasonable.

In addition, using UNF-ST&DARDS, discussed in Section 6.2.5 of this SER, the staff independently evaluated the Co-60 source term based on a conservative Co-60 impurity, mass of the hardware, and a flux scaling factor. Although the staff did not compare its resultant Co-60 values directly to the applicant's (this is impractical given the amount of Co-60 from fuel hardware is based on depletion characteristics and there are a wide range used for all the various calculations), the staff's calculations are in overall agreement with the applicant's. This provides the staff with reasonable assurance that the source term values are comparable, or if there is a discrepancy, that it does not make an impactful difference in the overall result of the calculation, or that there are other conservatisms within the applicant's evaluation that compensates and makes it acceptable as part of the overall calculation methodology.

6.2.2.1.3 Fuel Specifications (Burnup, Enrichment, and Cooling Time)

The applicant proposed new fuel qualification requirements that are applicable to the MPC-32M. For the MPC-32M, the applicant proposed a new method for selecting the enrichment used in dose evaluations. Holtec discussed this method for selecting the enrichment to be used in the dose evaluations in Holtec's proprietary Report HI-2188480. This new method is similar to the method previously approved in HI-STORM Flood/Wind (FW) MPC Storage System Amendment No. 5 for the MPC-37 and MPC-89 (NRC, 2020b).

The applicant's proposed method specified the required minimum cooling time as a function of burnup as stated in CoC Appendix D, Table 2.1-4. The coefficients for the correlation are defined in the Table 2.1-4 for each reference decay heat within the loading patterns in CoC Appendix D, Figures 2.4-1 and 2.4-2. The staff verified that the reference decay heats encompass the allowable decay heats for each loading zone as defined by the loading patterns in CoC Appendix D, Figures 2.4-1 and 2.4-2. The decay heat is only used as a reference for the loading pattern to show which correlations are used for which location.

Cooling times for any burnup values computed by the correlation in CoC Appendix D, Table 2.1-4, that are below the minimum allowable cooling times from CoC Appendix D, Table 2.1-1, are set to the value in Table 2.1-1 since the cooling times in Table 2.1-1 are the minimum allowable cooling times.

Similar to the methodology that is used for HI-STORM FW MPC Storage System Amendment No. 5, this amendment does not use enrichment as a controlling parameter for fuel specification. Although enrichment is important in determining the source term, the applicant has instead used a conservative value of enrichment for each analyzed burnup based on data collected from all unique discharged fuels on U.S. Department of Energy's Nuclear Fuel Data Survey Forms RW-859 and GC-859. These forms are used to collect information on SNF for all commercial reactors. Form RW-859 is for all SNF before 2002. Form GC-859 includes assemblies up to 2013 and includes the data from Form RW-859 minus any SNF assemblies that were shipped to away-from-reactor facilities.

The applicant then determined the enrichment that would bound 99% of the discharged fuel population within 5 GWd/MTU burnup bins (e.g., 0-5 GWd/MTU, 5-10 GWd/MTU, 10-15 GWd/MTU, etc.) as shown in Table 7.4 of Report HI-2188480. The applicant used the enrichments it derived with this method to calculate the dose as shown in FSAR Table 5.II.2.4 for MPC-32M.

For the higher burnup assemblies, the source term is more sensitive to changes in enrichment than it is for lower burned assemblies based on Figures 11, 12, and 13 from NUREG/CR-6716 (NRC, 2001). Thus enrichment at high burnup levels (> 45,000 MWd/MTU) could have a significant impact on the source term, especially on dose from neutron radiation.

Based on Figures 7-1 and 7-2 of Report HI-2188480, there are fewer data for assemblies with burnup over 55,000 MWd/MTU, causing the minimum enrichment assumed within the evaluations to be potentially less certain for these much higher burnup assemblies. The staff recognizes fewer data in over 55,000 MWd/MTU range as a potential non-conservatism of this approach; however, the staff found it acceptable for this amendment based on the following: (1) the current data supports the values chosen by Holtec, (2) changes in enrichment at higher burnups affects neutron dose more than gamma and the dose tables supplied by Holtec indicates that the system is more effective at shielding neutrons than gammas, and (3) the applicant performed a sensitivity study in Section B.5.2 of Report HI-2188253 Revision 5 showing that using a lower enrichment within particular regions shows a relatively small difference in overall dose.

6.2.2.1.4 Loading Patterns

The MPC-32M can be loaded in uniform, regionalized, or discrete loading patterns. In a uniform loading, all cells have the same heat load limit; whereas in a regionalized or discrete loading pattern, the head load limits vary for different cells or regions. As described below, the applicant uses the discrete loading pattern as the design basis loading for the MPC-32M.

The maximum decay heat for the uniform loading pattern is specified in CoC Appendix D, Section 2.4.1, as 38 kW. The applicant used the burnup and cooling time resulting from the correlation in CoC Appendix D, Table 2.1-4, associated with the 1.25 kW decay heat as well as the minimum enrichment in FSAR Table 5.II.2.4 to represent the source for the uniform loading.

The maximum allowable decay heat per storage location for regionalized loading is determined using the set of equations in CoC Appendix D, Section 2.4.1. For the regionalized loading, the applicant modeled the most extreme loading patterns for the two zones that allow for the highest decay heat for a zone and would maximize dose (as correlations that define the radiological parameters are based on decay heat values). The burnup/enrichment/cooling times selected from the correlations for these loading patterns are shown in FSAR Table 5.II.2.5.

The staff performed independent calculations, discussed in Section 6.4.5 of this SER, using a selection of fuel specifications from the regionalized loading pattern, and compared the results to a couple of loading patterns that use fuel specifications from the discrete loading pattern at the controlled area boundary. The staff did not systematically determine limiting source characteristics with respect to burnup/cooling time combinations, and used its judgment to model allowable fuel specifications in one case that produced a high neutron radiation and another with high gamma radiation. The staff's calculations show that the regionalized loading pattern produces a higher dose rate than the discrete loading pattern used by the applicant at 100 meters, but at 300 meters and beyond, the two are roughly equivalent.

Even though the staff's calculations show that it may be non-conservative to use the discrete loading pattern as the design basis at 100 meters, the staff found the use of the discrete loading pattern acceptable for design basis of the MPC-32M for this amendment for HI-TRAC MS and HI-STORM 100S Version E because (1) as discussed in Section 6.4.4.2 of this SER, the applicant has demonstrated that the system needs at least 300 meters to meet the annual dose limit in 10 CFR 72.104 and at this distance the dose rate from regionalized and discrete loadings are statistically equivalent; and (2) the highly heterogeneous discrete loading pattern creates higher localized dose rate peaks in the vicinity of the cask that are important for occupational exposures considerations and justifying TS dose rate limits around the transfer cask and overpack.

The discrete loading patterns are shown in CoC Appendix D, Figures 2.4-1 and 2.4-2, and provide the maximum allowable heat load per fuel storage location. In accordance with CoC Appendix D, Section 2.4.4, the general licensees may have to adjust cell heat load limits, quadrant heat load limits, and total heat load limit to allow for fuel with a longer or shorter active length.

The discrete loading pattern has five different regions as indicated in CoC Appendix D, Figures 2.4-1 and 2.4-2. As shown in FSAR Figure 5.II.2-1, the applicant combined the regions within the dose rate evaluation so that two of the zones are represented by a higher decay heat. This is conservative as cells restricted to a lower allowable decay heat are assumed to have a higher decay heat and source term. This also makes the calculation simple as fewer combinations are needed for analysis.

Since there are infinite burnup/enrichment/cooling time combinations that would be allowed from the correlations in CoC Appendix D, Table 2.1-4, the applicant chose a limited number of points along the correlation to analyze. The applicant shows the burnup/enrichment/cooling time combinations it used for each reference decay heat in FSAR Tables 5.II.2.5. The staff verified that these burnup/enrichment/cooling time combinations are appropriate for the burnup/cooling time correlations defined in CoC Appendix D, Table 2.1-4 and the minimum enrichments as specified in FSAR Table 5.II.2.4 and demonstrate a reasonable sampling of bounding points represented by the correlations.

The applicant performed calculations that encompass all of the possible combinations to determine the limiting combination for each zone. The combinations it determined to be the most limiting are reported in HI-2188253, Revision 5. The staff found these loading tables acceptable.

6.2.2.1.5 Variable Assembly Height

CoC Appendix D, Section 2.4.4 allows reference decay heat loads to be increased when storing a longer assembly. Although decay heat and source term are not directly correlated, reference decay heats are used to select the appropriate loading correlation for burnup and cooling time from CoC Appendix D, Table 2.1-4. Based on the allowable fuel to be loaded from CoC Appendix D, Table 2.1-2, the longest allowable fuel assembly is 150 inches as compared to the design basis assembly of 144 inches. Using the equations in CoC Appendix D, Section 2.4.4, this would result in an allowable decay heat increase of 2%. Based on the decay heat loading patterns in CoC Appendix D, Figures 2.4-1 and 2.4-2, an increase in 2% of the allowable decay heat values would only cause different burnup/cooling time correlations from Table 2.1-4 to be selected than were analyzed within the shielding evaluation in the FSAR for the four 0.825kW assemblies in Figure 2.4-1. Despite this, the staff found the adjustment to allowable decay heat for variable height assemblies in CoC Appendix D, Section 2.4.4 acceptable because doses are not expected to increase impactfully for the following reasons:

- These 0.85 kW assemblies are relatively low in decay heat as compared to the other allowable assemblies.
- These four assemblies, although near the exterior, would still receive shielding by the surrounding assemblies.
- The applicant analyzed the loading patterns in a conservative way which would likely compensate for this potential non-conservatism, for example:
    - As discussed earlier in this paragraph, the 5-zone loading pattern was combined into 3 by representing 2 zones as higher decay heat than would be allowed; and
    - Figure 2.4-1, where the 0.85 kW assemblies would be loaded, is likely not the limiting loading pattern based on the 3.26 kW assemblies allowed in the periphery in Figure 2.4-2.

6.2.2.1.6 Stainless Steel Replacement Rods

The applicant includes fuel assemblies with stainless steel replacement rods in a category of fuel assemblies called “repaired/reconstituted fuel assembly” within the definitions in CoC Appendix C. The repaired/reconstituted fuel assembly category is included as part of the definition of an “undamaged fuel assembly” in CoC Appendix C, meaning that these fuel assemblies are allowed in all MPC-32M locations.

Stainless steel replacement rods replace fuel rods within assemblies that may be damaged or leaking so that the fuel assembly can continue to be used in later cycles. Because the Co-59 impurities within the stainless steel can become activated to Co-60 from the high neutron flux within the core, stainless steel replacement rods can be an additional source of Co-60 which is especially of concern at low cooling times.

The applicant did not provide an analysis of the impact that the replacement rods will have on dose rates with the MPC-32M around the transfer cask or overpack. The applicant states that it did an evaluation in FSAR Section 5.4.10 with four irradiated stainless steel rods within every assembly for the MPC-32 and that the results show that the dose rates increase at the top and the sides of the HI-STORM 100S Version B by about 10%.

There is no limit on the number of stainless steel replacement rods that can be present in any number of assemblies; however, based on the staff’s engineering judgment and experience, four replacement rods per assembly is a reasonable assumption. Although the MPC-32 and the HI-STORM 100S Version B may not represent the exact shielding behavior of MPC-32M, the transfer casks (HI-TRAC MS), or storage overpack (HI-STORM 100S Version E), the staff found the inclusion of the stainless steel replacement rods acceptable for the following reasons:

(1) Based on the results of calculations in Table B.2.3 of Report HI-2188253, Revision 5, for the same source, the Version E has better shielding performance at 1 meter. This comparison includes some sources with low cooling time. This gives the staff reasonable assurance that when it comes to comparing the difference in dose due to stainless steel replacement rods for the Version E and Version B, the Version E should perform better at larger distances than the Version B and would likely not have a significant impact on doses beyond the controlled area boundary.

(2) While the MPC-32M should have overall worse shielding performance than the MPC-32 because stainless steel components are replaced with aluminum which has less ability to attenuate gammas, the basket is not considered a major shielding component. Therefore, it is the staff’s judgment that this difference will not have a major effect to doses outside (e.g. 1 meter from the source, or beyond controlled area boundary) of the system.

(3) An increase in source from Co-60 in the stainless steel rods would likely not be significant. This is for several reasons. First, these rods replace irradiated fuel and they typically have less exposure in comparison with the rest of the fuel rods in the same fuel assembly. This means that there will be less time for Co-59 to be activated into Co-60. Second, higher burnup assemblies would have more Co-60 produced within the replacement rods, because they will undergo more irradiation, permitting neutrons to activate more Co-59 atoms into Co-60. Higher burnup assemblies, however, would also have longer cooling times associated with them per the loading correlations in CoC Appendix D, Table 2.1-4. The half life of Co-60 is relatively short (roughly 5 years) compared to the longer-lived fission products and actinides and is more likely to decay with longer cooling times. Third, with the replacement rods, there

would be some loss of self-shielding. The loss of the irradiated fuel, which typically produces much stronger source terms than the replacement rods, would compensate for this.

(4) CoC Appendix C requires, as part of the definition of the "repaired/reconstituted fuel assembly," that the source from the stainless steel replacement rods be considered in site-specific dose evaluations, in which case, TS limits on the surface of the HI-TRAC MS and HI-STORM 100S (where the MPC-32M is allowed to be stored) cannot be exceeded.

6.2.2.2 Accident Source Term

There are two accident conditions that result in an increase in dose at the controlled area boundary: (1) the draining of the neutron shield water jacket for the transfer cask and (2) the non-mechanistic tipover which exposes the bottom of the cask. The HI-TRAC MS accident source term for the loss of neutron shield is discussed in Section 6.1.2.1 of this SER. The HI-STORM 100S Version E accident source term for the non-mechanistic tipover is discussed in Section 6.4.2 of this SER.

6.2.2.3 Non-Fuel Hardware Source Term

Non-fuel hardware is allowed to be stored within the MPC-32M based on CoC Appendix D, Table 2.1-1, Section V.A.1. According to FSAR Table 2.II.1.1, fuel assemblies containing burnable poison rod assemblies (BPRAs), thimble plug devices, wet annular burnable absorbers, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts, with or without instrument tube tie rods, may be stored in any fuel storage location. Fuel assemblies containing control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), control element assemblies (CEAs), or axial power shaping rods (APSRs) are restricted to interior locations of the basket. There is one neutron source assembly permitted per canister and it is restricted to a central location. The burnup and cooling time requirements for non-fuel hardware are in CoC Appendix D, Table 2.1-3.

The applicant calculated the Co-60 contribution from the hardware and shows the activity in curies in FSAR Table 5.II.2.6 for BPRAs at 1-year cooling time. The applicant shows the gamma spectra and source term and the Co-60 from the Inconel hardware for the CRAs at a 2-year cooling time in FSAR Table 5.II.2.7. The applicant shows the Co-60 source term for APSRs at a 2-year cooling time in FSAR Table 5.II.2.8. Report HI-2188253, Revision 5 states that the Co-60 source was calculated using the methods in the proprietary report HI-951322, Revision 27, "HI-STAR 100 Shielding Design and Analysis for Transport and Storage." The staff reviewed this information and found the codes and method used to determine the Co-60 activity for the non-fuel hardware to be reasonable, as they use industry standard codes and the inputs are reasonable, and acceptable for this application.

The staff verified that the burnup and cooling time requirements for non-fuel hardware for the MPC-32M shown in CoC Appendix D, Table 2.1-3 are consistent with the assumptions used to activate the components as stated in FSAR Section 5.II.2.4. Based on the half-life of Co-60 and the decrease in cooling (decay) time from previously approved Co-60 values established in FSAR Tables 5.2.31 and 2.1.25, the staff found the amount of increase in Co-60 acceptable for this application. Based on the applicant's statements and analysis assumptions as well as previously approved Co-60 levels and source terms, the staff found that these source terms are reasonable for simulating hardware that is to be stored within the MPC-32M.

The applicant also discussed the choice to use the BPRA as the bounding non-fuel hardware device. The applicant based this decision on the BPRAs showing a larger contribution to dose at a distance rather than increases near the cask, as well as restricted loading for CRAs, RCCSs, CEAs, and APSRs to interior locations. Therefore, the staff found the use of the BPRA source term to represent all non-fuel hardware, as discussed in FSAR Sections 5.1.1 and 5.4.6, acceptable.

6.2.2.4 Stainless Steel Clad Spent Nuclear Fuel Source Term

As shown in CoC Appendix D, Table 2.1-1 Section V.A.2.c.i, Stainless steel-clad fuel has a longer minimum cooling time and lower maximum burnup than zirconium-based-clad fuel. Stainless steel cladding contains additional gamma source from the Co-60 activation product within the cladding as compared to the zircalloy cladding. However, because the stainless steel SNF has a significantly lower maximum burnup and longer minimum cooling time than the zircalloy SNF, the staff found that the doses from the MPC-32M loaded with the stainless steel clad fuel would be bounded by the doses calculated for the zircalloy fuel.

6.2.2.5 Depletion Codes

The applicant used the TRITON/ORIGAMI modules of ORNL's SCALE 6.2.1 code system to calculate the source term for the MPC-32. For previous calculation/modeling approvals in FSAR Section 5.1 (NRC, 2000), the applicant used the depletion codes SAS2H and ORIGEN-S for the source terms. For the Amendment No. 15 calculations, however, the applicant updated the depletion code, TRITON/ORIGAMI, for the source terms because these updated codes are more recent and contain more detailed physics. The staff has found the updated codes to be more appropriate for simulating depletion especially for high burnup fuel as these codes have more detailed physics models.

6.2.3 Shielding Model Specification

The NRC staff and ORNL performed independent calculations using design information from the FSAR, as discussed in Sections 6.1.5 and 6.2.5 of this SER. These calculations used the MPC-32M, and the staff and ORNL were able to reproduce the applicant's results giving the staff reasonable assurance that the applicant modeled the MPC-32M appropriately.

6.2.3.1 Configuration of the Shielding and Source

The applicant modeled the top of the enclosure vessel using the minimum thickness from Drawing No. 11572R0. This is acceptable to the staff because when the MPC-32 is in the transfer cask configuration, the lid is the only shielding for the system. Consequently, this maximizes the potential dose and is, therefore, conservative.

The applicant states in FSAR Section 5.II.3.1 that it modeled the MPC-32M base plate using the minimum nominal thickness and that there is an option for increased thickness. The staff found that this is conservative and acceptable because modeling less shielding increases the dose rate and, other than the tipover event (discussed in Section 6.4.4.2.2 of this SER), the bottom of the cask is not exposed, therefore the modeling at the bottom of the cask is less significant.

The applicant states in FSAR Section 5.II.3.1 that it used nominal dimensions to model all other MPC-32M features. The staff found this acceptable considering that other features provide some shielding although are not primarily used for shielding and that modeling a nominal dimension is reasonable.

6.2.3.2 Damaged Fuel

According to CoC Appendix D, Section 2.1, licensees may load damaged fuel assemblies and fuel debris within the MPC-32M. CoC Appendix D, Table 2.1-1, Section V.B allows for up to 16 damaged fuel assemblies per the storage location specified in CoC Appendix D, Table 2.4-1.

The new loading patterns contain damaged fuel. With respect to damaged fuel modeling, the applicant assumed that the damaged fuel is the same as intact fuel. The applicant justified this assumption by using a comparison documented in FSAR Section 5.II.4.3. This comparison shows that dose rates increase with the inclusion of the damaged fuel model (in comparison to modeling with intact fuel). The largest increase is at the top of the cask because the damaged fuel was modeled as having twice the density and twice the source term with a flat burnup distribution. This caused the largest increase at the top due to the normally bottom-peaked power distribution. The staff found this modeling to be conservative because if fuel collapses it would collapse to the bottom of the cask, which makes this model more appropriate for representing the effect of damaged fuel near the bottom of the cask. The applicant's calculations show an increase in dose rate that is still significant at the bottom side of the cask. It is the staff's judgment, however, that the amount of increase is small enough that the staff has reasonable assurance that the system meets regulatory annual dose limits with damaged fuel being represented by intact fuel. The staff based its judgment on it being unlikely for all of the damaged fuel assemblies to fail in the worst possible way. The dose rate increases from damaged fuel should be less than what is reported in the FSAR Tables 5.II.4.8 and 5.II.4.9, and accordingly the staff found that the applicant's analysis does not underrepresent doses from potentially damaged fuel in an impactful way. Accordingly, the staff found the applicant's analysis will not impact dose in a meaningful way.

6.2.3.3 Material Properties

The MPC-32M basket is made of Metamic^{TM}-HT. The applicant showed the material density and composition of the Metamic^{TM}-HT used for the MPC-32M basket model in FSAR Table 5.II.3.2. The staff verified that this is consistent with the description of this material in Holtec's proprietary report HI-2084122, "Metamic^{TM}-HT Qualification Sourcebook," Revision 14.

The enclosure vessel is made of Alloy X. Alloy X does not have a specific composition and could therefore represent a variety of stainless steels. In the shielding calculations, Alloy X is represented as carbon steel. The staff found this acceptable as the iron is the main source of shielding within steel components and the density difference between the carbon steel and the stainless steels represented as Alloy X would have a negligible effect on dose. The staff verified that the description of carbon steel is consistent with publicly available literature.

6.2.4 Shielding Analyses

6.2.4.1 Computer Codes

The applicant performed the shielding analyses using an updated version of MCNP, MCNP5 Version 1.51 using default cross sections. Other analyses within the HI-STORM 100 FSAR use MCNP-4A. The staff found that using a more recent version of this code is acceptable.

6.2.4.2 Dose Rates

The applicant evaluated the MPC-32M dose rates when it is within the HI-TRAC MS transfer cask and HI-STORM 100S Version E. This SER discusses these evaluations in Sections 6.1.4.3 and 6.4.4.2, respectively.

While the MPC-32M is only allowed to be stored in the Version E overpack, the applicant stated in its summary of changes (Holtec, 2019a, Attachment 1) that the MPC-32M was to be used with any transfer cask. The applicant did not provide an evaluation of the MPC-32M in any other transfer cask other than the HI-TRAC MS. Staff reviewed the evaluation of the MPC-32M for use in the HI-TRAC MS transfer cask and found that use acceptable in Section 6.1 of this SER. Additionally, in the staff's judgment, the analyses presented for the HI-TRAC MS with the MPC-32M would reasonably represent the other transfer casks. This is because:

(1) There is a transfer cask dose rate limit in CoC Appendix C, "Technical Specifications for the HI-STORM 100S Version E Cask." Since the MPC-32M is only allowed within the Version E overpack, Appendix C is used whenever the MPC-32M is loaded in the Version E overpack, which means that there will be a technical specification dose rate limit when using the MPC-32M even if another transfer cask (other than HI-TRAC MS) is used. There is no transfer cask dose limit at the side of cask in CoC Appendix A for the other transfer casks when used with other HI-STORM 100 overpacks.

(2) Although there may be differences in shielding characteristics between the HI-TRAC MS and the other transfer casks, the HI-TRAC MS with minimum dimensions should be the limiting transfer cask based on the thicknesses of the shielding. Even if it is not the limiting transfer cask, the HI-TRAC MS is similar enough in design to the other transfer casks to be reasonably representative. Other transfer casks used to load the HI-STORM 100S Version E will be limited by the same TS dose rate limit used for the HI-TRAC MS.

Therefore, the staff found the MPC-32M acceptable to be loaded in the other approved HI-TRAC transfer casks associated with the HI-STORM 100 system.

6.2.5 Independent Calculations

The staff used the ORIGAMI code from the SCALE 6.2.3 code package to verify the spent fuel source term for the new loading patterns and fuel qualification strategy. The staff verified the spent fuel gamma and neutron source terms in FSAR Tables 5.II.2.1 and 5.II.2.2 for the PWR source terms as the MPC-32M is for PWR fuel only. When performing these calculations, as input, the staff used the burnup and cooling time as stated in FSAR Tables 5.II.2.1 and 5.II.2.2 as well as the specific power and fuel assembly mass from FSAR Table 5.2.1. The staff used the enrichment from FSAR Table 5.II.2.4. Although these source terms are representative and do not encompass all source terms that would be used within all calculations associated with the MPC-32M, the staff found that confirming these provide reasonable assurance that the applicant's calculation process results in reasonable source terms.

Since the MPC-32M is the limiting basket for the HI-TRAC MS and the HI-STORM 100S Version E, the staff performed calculations of the dose rate from these systems using the MPC-32M in these evaluations discussed in Sections 6.1.5 and 6.4.5 of this SER.

6.2.6 Conclusion

Based on the above considerations, the staff found that with the addition of the MPC-32M, the HI-STORM 100 dry storage system design continues to meet 10 CFR Part 72 regulations pertaining to shielding in 10 CFR 72.236(a) and (d).

6.3 Proposed Change #3 – Inclusion of MPC-32 Version 1 and MPC-68 Version 1

The applicant requested the inclusion of the MPC-32 Version 1 and MPC-68 Version 1. The Version 1 MPCs are slightly modified versions of the previously approved MPC-32 (NRC, 2002) and MPC-68 (NRC, 2000). The MPC-32 Version 1 and MPC-68 Version 1 have larger cell openings than the MPC-32 and MPC-68 to improve the ability to store DFCs. In Amendment No. 15, the applicant is seeking approval of the Version 1 baskets with the HI-STORM-100 overpack and the HI-STORM 100S Version E overpack. The allowable contents are the same as for the previously approved MPC-32 and MPC-68.

The Version 1 baskets have increased baseplate and basket shim thicknesses and slightly reduced wall thicknesses relative to the MPC-32 and MPC-68, that may affect the shielding of the contents. The applicant stated in FSAR Section 5.II.1 that it did not perform any shielding evaluations because the design changes would only have a small effect on dose rates.

The staff reviewed FSAR Drawing No. 11574 Revision 0 for the MPC-32 Version 1 and Drawing No. 11578 Revision 0 for the MPC-68 Version 1 and compared these to Drawing No. 3927 Revision 19 for the MPC-32 and Drawing No. 3928 Revision 19 for the MPC-68. The staff found that the decrease in basket wall thickness is small and, therefore, should not have a significant effect on dose rates. The applicant states that the increased basket shims would provide additional shielding, however the staff finds that the shims are made of aluminum and may not compensate for the reduced amount of steel within the baskets as compared to the MPC-32 and MPC-68. Still, the staff concludes that the outermost assemblies contribute the most to the dose rate and the added aluminum would provide some additional shielding to the outer assemblies.

The staff compared the drawings of the Version 1 MPC enclosure vessel in Drawing No. 11572 Revision 0 with that of the MPC enclosure vessel in Drawing No. 3923 Revision 36 and found that the Version 1 enclosure vessel has as much or more shielding (at the bottom) than the MPC enclosure vessel. Therefore, the staff concludes that the inclusion of MPC-32 Version 1 and MPC-68 Version 1 would not impact the HI-STORM 100 system’s capability of meeting regulatory dose requirements.

6.4 Proposed Change #4 – Addition of HI-STORM 100S Version E Overpack

The applicant requested the inclusion of the HI-STORM 100S Version E overpack. The HI-STORM 100S Version E is a variant of the HI-STORM 100S Version B and is for use with all MPCs in the HI-STORM 100 system approved before the date of Amendment No. 15’s approval. The applicant also requested that the new MPC-32M, MPC-32 Version 1, and MPC-68 Version 1 be approved for use in the HI-STORM 100S Version E overpack.

6.4.1 Shielding Design Description

The HI-STORM 100S Version E cask is similar to the HI-STORM 100S Version B, however, the Version E inlet and outlet vent systems have a new design and the shielding concrete has increased density.

6.4.2 Radiation Source Definition

Licensees may store all previously approved canisters and their allowable contents in the HI-STORM 100S Version E. The previously approved MPCs are the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F, MPC-68FF, and MPC-68M. Licensees may also store the Amendment No. 15's proposed canisters—MPC-32M, MPC-32 Version 1, and MPC-68 Version 1—in the HI-STORM 100S Version E.

The applicant has performed calculations demonstrating that the MPC-32M gives the highest dose rate of any of the canisters and showed the results in FSAR Table 5.II.4.4. It therefore used the MPC-32M in its modeling of the HI-STORM 100S Version E. The staff reviewed Table 5.II.4.4 and found it acceptable to use the MPC-32M as the design basis canister because it has the largest source term and provides the bounding dose rates. The applicant derived the normal condition source term for the HI-STORM 100S Version E using the procedures discussed for the MPC-32M in Section 6.2.2 of this SER, which the staff found acceptable.

The accident condition for the HI-STORM 100S Version E is the non-mechanistic tipover. The applicant used the same procedure discussed for the MPC-32M in Section 6.2.2 of this SER to determine the loading pattern that gave the highest dose rates for this configuration. The applicant provided this information in Table B.5.2 of Report HI-2188253 Revision 5.

The staff's understanding is that the applicant modeled a more realistic representation of the non-fuel hardware by incorporating BPRAs than the configuration given by the procedure in section 6.2.2, since it is unlikely that all non-fuel hardware (such as RCCAs and APSRs) would be loaded to the maximum extent. The applicant provided stability evaluations for the HI-STORM Version E in Sections 3.II.4.4.2, which demonstrate the low likelihood of tipover and that the cask will remain upright under the seismic and tornado conditions, as discussed in Section 3.4.3.5 of this SER. Therefore, based on the staff's findings in Section 3.4.5.3 of this SER and the four bullet points discussed in Section 6.4.4.2.2 of this SER, in the staff's judgment, this accident has a low likelihood of occurring and therefore, the staff finds the applicant's modeling of non-fuel hardware acceptable.

6.4.3 Shielding Model Specification

6.4.3.1 Configuration of Source and Shielding

ORNL performed independent calculations using design information from the FSAR, as discussed in Section 6.2.5 of this SER. ORNL was able to reproduce the applicant's results under normal conditions which gives the staff reasonable assurance that the applicant modeled the HI-STORM 100S Version E appropriately. Therefore, the staff has reasonable assurance that the applicant appropriately modeled the HI-STORM 100S Version E.

6.4.3.2 Material Properties

Section A.2.3 of proprietary report HI-2188253 Revision 5 states that the concrete properties for the HI-STORM 100S Version E are from the HI-STORM FW FSAR (Holtec, 2017). The staff reviewed the composition of the concrete in the HI-STORM FW FSAR (Holtec, 2017) and found that it is consistent with compositions that can be found in open literature (McConn, 2011). Therefore, the staff found it to be acceptable for this application. The staff verified that the concrete density used in the shielding evaluations as specified in FSAR Table 5.II.3.2 and Section A.2.2.3 of HI-2188253 Revision 5 is consistent with or lower than the value specified in FSAR Table 1.II.2.4. Therefore, the staff found the applicant's modeling of the concrete for the HI-STORM 100S Version E acceptable.

6.4.4 Shielding Analyses

The applicant presented the surface dose rates for the HI-STORM 100S Version E overpack with the MPC-32M under normal conditions in FSAR Table 5.II.1.1 and at 1 meter in Table 5.II.1.2. The applicant calculated the annual dose rate from a single cask and an array of casks for various sizes and set distances in FSAR Table 5.II.1.3. The staff reviewed this table and found that it shows that the cask system would meet the annual dose limit in 10 CFR 72.104(a).

6.4.4.1 Computer Codes

The applicant performed the shielding analyses using an updated version of MCNP, MCNP5 Version 1.51 using default cross sections. The applicant used MCNP-4A for other earlier analyses in the HI-STORM 100 FSAR. The staff found that using a more recent version of the code is acceptable.

6.4.4.2 Dose Rates

6.4.4.2.1 Normal Condition

The applicant presented the results of the dose rate evaluations for the HI-STORM 100S Version E in FSAR Table 5.II.1.1. This table shows that the maximum dose rate at the side of the cask is 159.67 mrem/hr. CoC Appendix C, Section 5.3.4b limits the dose rate at this location to 200 mrem/hr. Although this is higher than the calculated dose rate, the staff has reasonable assurance, based on the calculations presented in the FSAR, that as long as the cask is loaded in accordance with the fuel qualification requirements in the CoC Appendix C, the cask would meet site boundary dose rates calculated in FSAR Table 5.II.4.5. The dose rate limit in CoC Appendix C for Version E overpack at the side of the cask is conservative as compared to the dose rate of 300 mrem/hr for other overpacks in the HI-STORM 100 system as specified in CoC Appendix A, Section 5.7.4b. Therefore, the staff found the established dose rate limit in the technical specifications acceptable.

FSAR Table 5.II.1.1 shows that the maximum dose rate at the top of the cask is 12.83 mrem/hr. CoC Appendix C, Section 5.3.4a limits the dose rate at this location to 20 mrem/hr. Although this is higher than the calculated dose rate, the staff has reasonable assurance, based on the calculations presented in the FSAR, that as long as the cask is loaded in accordance with the fuel qualification requirements in the CoC Appendix C, the cask would meet site boundary dose rates calculated in FSAR Table 5.II.4.5. The dose rate limit in CoC Appendix C for Version E overpack at the top of the cask is conservative as compared to the dose rate of 30 mrem/hr for other overpacks in the HI-STORM 100 system as specified in CoC Appendix A, Section 5.7.4a.

Therefore, the staff found the established dose rate limit in the technical specifications acceptable.

6.4.4.2.2 Accident Condition

The applicant provided the dose rates calculated from the tipover accident in FSAR Section 11.II.2.3.3. The results of the applicant's calculations show that the HI-STORM 100S Version E will exceed the regulatory dose limit in 10 CFR 72.106 in 11 days. Section 9.5.4.3 of NUREG-2215 states that "a bounding exposure duration assumes that an individual is also present at the controlled area boundary for 30 days" (this is repeated in Section 10B.5.3.2 of NUREG-2215). The staff understands the applicant's results to mean that licensees will either perform event recovery within 11 days of exposure or reduce dose rates to levels that meet 10 CFR 72.106 within 30 days of exposure consistent with the guidance in NUREG-2215.

FSAR Section 11.II.2.3.3 does not discuss how the cask will be returned to its normal operating state within 11 days. It states, however, that, based on the dose rates, the use of temporary shielding will be developed and implemented to ensure the dose rate is ALARA during recovery options. In addition, the staff found the 11-day exposure time acceptable because:

- The applicant provided an analysis that demonstrates that the cask would remain stable for design basis accident conditions in FSAR Section 3.II.4.4.2. This analysis is discussed in Sections 3.4.3.2, 3.4.3.5 and 3.4.3.6 of this SER.
- NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," (NRC, 2007) concludes that the probability of tipover is near zero. Although this NUREG employs a different cask, the staff found that there are enough similarities to provide insights that the HI-STORM 100S Version E is also very unlikely to tip over.
- The applicant assumed a very conservative distance to the controlled area boundary of 100 meters. This is the minimum distance as required by 10 CFR 72.106(b) but the actual distance is likely to be much further because the HI-STORM 100S Version E requires 300 meters to meet normal condition dose rates for a single overpack and 500 meters for a 2x5 array (see Section 11.4.2 of this SER).
- The general licensee would utilize temporary shielding to maintain doses ALARA.

Given the low likelihood of a tipover event (as discussed in Section 6.4.2), as judged by the staff as part of this review, the staff found the 11-day exposure time acceptable.

Consequently, the staff found that the HI-STORM 100S Version E meets the regulation in 10 CFR 72.236(d) because it meets 10 CFR 72.106 after 11 days of exposure and, as discussed in Section 11.4 of this SER, the staff also found that it meets 10 CFR 72.104.

6.4.5 Independent Calculations

6.4.5.1 Normal Condition

ORNL performed independent calculations of the HI-STORM 100S Version E with the MPC-32M and the discrete loading pattern to confirm local (surface and 1 meter) dose rates for the HI-STORM 100S Version E. ORNL used SCALE/MAVRIC within UNF-ST&DARDS for its calculations. ORNL used the B&W 15x15 assembly and did not model the non-fuel hardware. It did this to decrease the amount of time needed for the calculation to converge and because contribution from these components was relatively low. Due to the highly heterogeneous discrete loading pattern, the models required relatively small tally sizes which significantly increased run time for these models. ORNL used the loading patterns from Table B.3.5 of proprietary report HI-2188252 Revision 5 that maximized dose for the top and 1 meter from the top air gap, and ORNL's results agreed relatively well with the applicant's.

The staff used UNF-ST&DARDS to calculate the site area boundary dose using the discrete loading pattern and fuel characteristics from Table B.4.2b of report HI-2188253 Revision 5 for the HI-STORM 100S Version E for both a single cask and a 2x3 array of casks, which is consistent with the example in FSAR Section 5.4.3. The staff also used the equations in CoC Appendix D, Section 2.4.1 for the regionalized loading to represent an extreme regionalized loading to compare to the discrete loading as discussed in Section 6.2.2.1.4 of this SER. The staff used the B&W 15x15 assembly type with fuel mass and specific power from FSAR Table 5.2.1 and included the BPRAs in every location. For the calculations performed using a regionalized loading, the staff did not systematically determine the highest burnup/cooling time combinations, but it chose source terms from the allowable content correlations that it determined represent bounding conditions from a neutron/gamma contribution perspective.

The results of the staff's calculations showed that at 100 meters the regionalized loading pattern gives higher dose rates than the discrete loading pattern, but at 300 meters and beyond, which is the distance needed to meet the controlled area boundary annual dose limits in 10 CFR 72.104(a) for a single cask, the regionalized and the discrete loading patterns, used by the applicant as the design basis, gave statistically equivalent results.

When using the same fuel loading as the applicant, the staff's results agreed well with those in FSAR Tables 5.II.4.5 and 5.II.4.6 for a single cask and a 2x3 array, respectively. Both ORNL and NRC's calculations provide additional assurance that the applicant appropriately modeled the system as specified in the FSAR and that the design meets the controlled area boundary dose in 10 CFR 72.104, at 300 meters for a single cask and 400 meters for a 2x3 array.

6.4.5.2 Accident Condition

To confirm the reported dose rates, the staff modeled the non-mechanistic tipover event using UNF-ST&DARDS with several different discrete loading patterns including the loading pattern identified by the applicant in Table B.5.2 of report HI-2188253 Revision 5 to be the limiting loading pattern. The staff found the same loading pattern the applicant identified resulted in the highest dose rate at 100 meters. This gives the staff assurance that the applicant's procedure is capable of identifying the loading that produces the highest dose.

The staff modeled a case with this same loading pattern that includes the BPRAs from FSAR Tables 2.II.1.1 and 5.II.2.6, to the maximum amount allowed by FSAR Table 2.II.2.8, as well as APSRs from FSAR Tables 2.II.1.1 and 5.II.2.8, to the maximum extent allowed by FSAR Table 2.II.2.8. As stated in FSAR Section 5.II.4.2, the BPRA is used as the design basis non-fuel hardware in the dose rate calculations for the HI-STORM 100S Version E. FSAR Section 5.II.4.2 also states that the dose rate out of the bottom of the overpack is substantial due to the RCCAs and APSRs and that the dose rates occur in an area which is not normally occupied by people. Therefore, the applicant concluded, they do not pose a risk from an operations perspective. However, the staff found that this would not be the case for the tipover event.

The staff's calculations show that the most limiting configuration is the one with the APSRs, which have a significant effect on dose rates at the bottom of the HI-STORM 100S Version E.

The staff's calculations show that the accident dose limit in 10 CFR 72.106 will be exceeded in about 6.5 days at 100 meters.

The staff's calculation with BPRAs instead of APSRs is much closer to the applicant's result in FSAR Section 11.II.2.3.b. Given the differences between the staff and applicant's calculated dose rates, the applicant likely did not model the APSRs as the bounding non-fuel hardware. However, as discussed in Section 6.4.2 of this SER, the NRC staff found this acceptable due to the low likelihood of tipover.

6.4.6 Conclusion

Based on the above considerations, the staff finds that with the addition of the HI-STORM 100S Version E overpack, the HI-STORM 100 dry storage system design meets 10 CFR Part 72 regulations pertaining to shielding in 10 CFR 72.236(a) and (d).

6.5 Proposed Change #5 – Three Additional BWR Fuel Types: 10x10I, 10x10J, and 11x11A

The applicant requested to add three additional BWR fuel types, 10x10I, 10x10J, and 11x11A as the approved content for the MPC-68M basket. The applicant uses the GE 7x7 as the design basis assembly in all of its evaluations of the MPC-68M. The design basis fuel properties for the GE 7x7 are shown in FSAR Table 5.2.1. The staff compared the fuel masses for the 10x10I, 10x10J, and 11x11A in CoC Appendix B Table 2.1-3 to the design basis GE 7x7's fuel mass. The 10x10J has about a 7% higher mass and the 11x11A is about 2% higher than the GE 7x7. The mass of the 10x10I is lower than the GE 7x7 and is therefore bounded.

ORNL performed a sensitivity study on the effect of uranium mass on dose rates and documented the results in NUREG/CR-6716 (NRC, 2001). The report states that uranium mass is of intermediate importance to evaluating dose rates. Section 3.4.2.3 of NUREG/CR-6716 discusses the results of the sensitivity study where a 10% increase in the uranium mass results in a 3% increase in overall dose rates. The casks used in the sensitivity study are based on a concrete cask design that is likely applicable to the HI-STORM 100 storage overpacks which is also based on concrete, and a steel and resin design that is more applicable to the HI-TRAC transfer casks. Therefore, the sensitivity study's results can be used to provide insights to the HI-STORM 100 storage system and the HI-TRAC transfer casks.

NUREG/CR-6716 concluded that for the systems studied fuel mass affects doses from neutron radiation more than from gamma radiation, and the change in overall dose was so small—i.e. 3% dose increase from a 10% fuel mass increase—because gamma radiation contributed to the studied system's dose more than neutron radiation. Gamma and neutron contributions to dose are not consistent at every location around the HI-TRAC MS, but overall gamma contribution from dose is overall higher than neutron. Based on FSAR Table 5.II.1.3 for the MPC-68M within the HI-TRAC MS, locations that tend to have a larger neutron contribution to the dose also have lower doses in general.

From FSAR Table 5.II.1.4, gamma and neutron radiation contribute approximately equally to dose rate from the design basis, loss of neutron shield accident for the HI-TRAC MS. Therefore, the increased mass from the new assemblies would have a larger effect on the loss of neutron shield accident condition dose rate than on normal condition dose rates. NUREG/CR-6716, Section 3.4.2.3, states that, for the conditions studied, a 10% increase in mass has a 6% increase in the neutron dose rate. This increase is small relative to the margin between the highest calculated dose from the loss of neutron shield accident, shown in FSAR Table 5.II.1.4, and the regulatory limit. Consequently, the staff found that the inclusion of the new fuel types would not prevent the system from meeting regulatory dose requirements in 10 CFR 72.104 and 10 CFR 72.106 as well as occupational dose requirements in 10 CFR 20.1201.

As discussed above, the GE 7x7 assembly type bounds the 10x10I fuel type and is appropriate to represent the 10x10J and 11x11A fuel types in this amendment. CoC Appendix B, Table 2.4-4 lists the correlation coefficients specifying allowable burnup, enrichment, and cooling time for the bounding GE 7x7 assembly type. Given that the GE 7x7 assembly type is bounding or representative, the staff verified that the applicant edited Table 2.4-4 to make it clear it applied to the 10x10I, 10x10J, and 11x11A fuel types.

Based on the above considerations, the staff found that with the addition of the 10x10I, 10x10J, and 11x11A fuel types, the HI-STORM 100 dry storage system design meets 10 CFR Part 72 regulations pertaining to shielding in 10 CFR 72.236(a) and (d).

6.6 Proposed Changes #7 & #12 – Addition of HI-DRIP and DIJ Ancillary System

The applicant proposed to add HI-DRIP and a DIJ ancillary system that would prevent the water with the MPC from boiling during loading/unloading operations while the loaded MPC is in the HI-TRAC transfer cask. Although water in the MPC is important for shielding the contents during loading operations, the staff's assessment on the system's ability to keep the water from boiling is not discussed here as the system itself does not serve a shielding function.

6.7 Proposed Change #11 – Remove Dimensions from MPC-68M Drawing

The applicant proposed to update Drawing No. 7195 for the MPC-68M to remove dimensions that are not used in the safety analysis from the drawing.

The staff reviewed the changes to Drawing No. 7195 and found that the applicant removed dimensions for the shims and tolerances on the thickness of the basket panels. The shims and the Metamic^{TM} basket provide some shielding, however these are aluminum shims and the main component in the Metamic^{TM} is aluminum and would not contribute significantly to shielding as most of the shielding is performed by gamma and neutron shields of the transfer cask or storage overpack. Therefore, the staff found that removing dimensions and tolerances for these components provides flexibility for the applicant while not significantly impacting the shielding function of the system.

6.8 Evaluation Findings

F6.1 The FSAR provides specifications of the spent fuel contents to be stored in the HI-STORM 100 System in sufficient detail to adequately define the allowed contents and allow evaluation of the dry storage system shielding design for the proposed contents. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed contents' specifications. Thus, the applicant has satisfied the requirements in 10 CFR 72.236(a).

F6.2 The FSAR describes the SSCs important to safety that are relied on for shielding in sufficient detail to allow evaluation of their effectiveness for the proposed term of storage. Thus, the applicant satisfies the requirements in 10 CFR 72.236(b) and 10 CFR 72.236(g).

F6.3 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of the HI-STORM 100 System's compliance with 10 CFR 72.236(d). This evaluation is described in the radiation protection review in SER Chapter 11. Therefore, the applicant satisfies the requirements in 72.236(d).

F6.4 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of consideration of ALARA in the HI-STORM 100 System's design and evaluation of occupational doses. This evaluation is described in the radiation protection review in SER Chapter 11.

Based upon its review, the staff has reasonable assurance that the design of the shielding system of the HI-STORM 100 system, Amendment No. 15 complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system design provides reasonable assurance that the HI-STORM 100 system, Amendment No. 15 will allow safe storage of spent fuel in accordance with 10 CFR 72.236(d). This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, and the statements and representations in the application.

7.0 CRITICALITY EVALUATION

The objective of the criticality evaluation is to verify that the spent fuel contents remain subcritical under the normal, off-normal, and accident conditions of handling, packaging, transfer, and storage. The applicable regulatory requirements are those in 10 CFR 72.24(c)(3), 10 CFR 72.24(d), 10 CFR 72.124, 10 CFR 72.236(c), and 10 CFR 72.236(g).

The staff reviewed the information provided in the amendment request to determine whether the HI-STORM 100 system continues to fulfill the acceptance criteria listed in Section 7 of NUREG-1536 which is one acceptable way of meeting the above-cited regulatory requirements.

The staff reviewed the information provided by the applicant and evaluated the following proposed changes that are applicable to criticality review:

- Proposed Change #2 – Include MPC-32M for storage in HI-STORM 100 System.
- Proposed Change #3 – Include MPC-32 Version 1 and MPC-68 Version 1 for storage in HI-STORM 100 System.
- Proposed Change #5 – Add three additional BWR fuel types to the approved content for MPC-68M: 10x10I, 10x10J, and 11x11A.
- Proposed Change #8 – Allow for partial gadolinium (Gd_2O_3) credit for BWR fuel assembly types 10x10 and 11x11 assembly classes in MPC-68M.
- Proposed Change #11 – Update Drawing 7195 for the MPC-68M by removing dimensions which are not used in the safety analysis.

7.1 Proposed Change #2 – Include MPC-32M

7.1.1 Fuel Specification

For the MPC-32M canister, all of the PWR fuel assembly classes acceptable for storage in the standard MPC-32 canister are allowed, except for the 14x14E fuel assembly class (as shown in Table 2.II.1.8 of the SAR). The fuel parameters for all allowable PWR fuel assembly classes are shown in FSAR Table 2.II.1.8, and the minimum soluble boron requirements for fuel enriched up to 4.0 and 5.0 weight percent ^{235}U are shown for each allowable configuration of undamaged and damaged/failed fuel in FSAR Tables 2.II.1.9 through 2.II.1.11. The applicant proposes using linear interpolation between soluble boron levels for 4.0 and 5.0 weight percent ^{235}U to determine the required soluble boron level for fuel with enrichments between these two values. This approach was previously reviewed by the staff and approved for the MPC-32 canister (NRC, 2002). The staff finds linear interpolation to determine the required soluble boron level between fuel enrichments of 4.0 and 5.0 weight percent (wt. %) ^{235}U acceptable. The staff has reasonable assurance that system k-effective (k_{eff}) for enrichments between 4.0 and 5.0 wt. % and linearly interpolated minimum soluble boron concentrations is statistically similar (i.e., within two times the Monte Carlo calculation uncertainty (2σ)) to system k_{eff} calculated at 4.0 and 5.0 wt. %.

The applicant stated that the MPC-32M has eight allowable configurations of undamaged and damaged or failed fuel in basket positions with DFIs or DFCs, as shown in FSAR Figure 6.II.1-1. These configurations are:

- Configuration 1: Undamaged fuel in all basket locations
- Configuration 2: Damaged fuel in up to 12 peripheral basket locations with DFIs, and undamaged fuel in remaining locations
- Configuration 3: Damaged fuel in up to 16 peripheral basket locations with DFIs, and undamaged fuel in remaining locations
- Configuration 4: Damaged fuel or fuel debris in DFCs in up to eight peripheral basket locations, and undamaged fuel in remaining locations
- Configuration 5: Damaged fuel or fuel debris in DFCs in up to 12 peripheral basket locations, and undamaged fuel in remaining locations
- Configuration 6: Damaged fuel or fuel debris in DFCs in up to 16 peripheral basket locations, and undamaged fuel in remaining locations
- Configuration 7: Damaged fuel or fuel debris in DFCs in up to 16 peripheral basket locations, four empty basket cells, and undamaged fuel in remaining locations
- Configuration 8: Damaged fuel in up to four peripheral basket locations with DFIs, and undamaged fuel in remaining locations.

Minimum soluble boron concentrations for each configuration, for fuel enriched up to 4.0 and 5.0 wt. % ^{235}U, are shown in FSAR Tables 2.II.1.9 through 2.II.1.11.

7.1.2 Model Specification

7.1.2.1 Configuration

The addition of the MPC-32M does not change the storage system's response to normal, off-normal, or accident conditions. The structural analysis, evaluated in Section 3 of this SER, shows the potential for small deformations of the Metamic^{TM}-HT basket under accident conditions. These deformations are included in the evaluation of material and fabrication tolerances and are conservatively assumed to exist under all conditions for the purposes of the criticality analysis.

The applicant modeled the MPC-32M using the most reactive configuration of PWR fuel assemblies determined from previous analyses of the MPC-32 (NRC, 2002). The staff finds this acceptable, as the fuel materials and geometry to be loaded into the MPC-32M are identical to those allowed in the MPC-32. For the basket geometry, the applicant performed sensitivity studies, summarized in FSAR Section 6.II.3.1, to determine the most reactive combination of material and fabrication tolerances and fuel position. The most reactive configuration consisted of the minimum basket cell inner dimension; minimum basket cell wall thickness, including potential deformation of the basket walls from accident conditions; and eccentric positioning of fuel assemblies towards the center of the basket. The minimum basket cell inner dimension and minimum basket cell wall thickness are identified in FSAR Table 6.II.3.2. The applicant also conservatively modeled the Metamic^{TM}-HT basket walls with a boron-10 (^{10}B) content of 90% of the required minimum specified in Drawing No. 11425 Revision 0, consistent with the recommendations in Section 7.4 of NUREG-2215. For the fuel assemblies, the model used the most reactive combination of tolerances for the pellet diameter, outer clad diameter, and clad thickness, determined by the applicant in Section 6.II.2.1 of the SAR, as well as the maximum active fuel length. The applicant modeled the fuel material as UO_2 at a stack density of 97.5% theoretical density, without allowance for dishing or chamfer, which conservatively increases the modeled fuel material in the assembly. Additionally, the applicant conservatively assumed that the pellet-to-clad gap is filled with full density moderator.

For damaged fuel in DFCs or basket locations with DFIs, the applicant modeled the fuel as clad rods with varying pitch, with a maximum pitch defined by the inner dimension of the basket cell walls. For fuel debris in a DFC, the applicant continued to model the fuel as unclad rods with varying pitch, with a maximum pitch defined by the inner dimension of the DFC. For all damaged fuel and fuel debris configurations, the damaged or failed fuel is modeled at 5.0 wt. % ^{235}U. This damaged fuel/fuel debris modeling approach is consistent with that previously evaluated and approved by the NRC for DFCs in the MPC-24, MPC-32, MPC-68, and MPC-68M canisters. The staff determined that this approach is appropriate for the MPC-32M, since the fuel and canister materials and geometry for the MPC-32M are largely the same as for those previously approved canisters.

The configurations of the MPC-32M canister analyzed in the criticality analysis are consistent with previously approved analyses of the HI-STORM 100 system. The staff concluded that the applicant has determined the most reactive configuration of the canister basket for each fuel type, for the reasons discussed in the preceding paragraphs.

7.1.2.2 Material Properties

The MPC-32M fuel materials are similar to those previously evaluated and approved for the standard MPC-32. The major difference between the MPC-32 and the MPC-32M is the

Metamic^{TM}-HT basket in the MPC-32M canister, which is identical in material composition to that for the MPC-68M canister. The applicant describes the fuel and basket materials of the MPC-32M in FSAR Section 6.II.3. These descriptions include the composition of the major components of the MPC, including UO_2 fuel, steel and aluminum structural components, and the Metamic^{TM}-HT neutron absorber. The applicant assumes that the fuel pellets have the maximum theoretical density of UO_2, which is conservative since actual UO_2 pellets are less dense. The criticality analyses conservatively assume no more than 90% of the Metamic^{TM}-HT minimum ^{10}B content specified in Drawing No. 11425, Revision 0, consistent with the recommendations in Section 7.4 of NUREG-2215.

The material properties assumed in the MPC-32M criticality analyses are consistent with those assumed in previously approved analyses of the HI-STORM system. The staff concluded that the applicant has determined appropriate material properties for the canister basket and each fuel type, as discussed in the preceding paragraph.

7.1.3 Criticality Analysis

For the MPC-32M criticality analysis, the applicant modeled all the PWR fuel assembly classes previously evaluated for the standard MPC-32 canister, except the 14x14E assembly class. The applicant modeled the fuel and canister basket according to the most reactive combination of fuel and basket parameters determined previously for similar canisters. The applicant performed analyses to determine the most reactive system configuration, similar to those performed for the standard MPC-32. These analyses included eccentric fuel assembly positioning, varying internal and external moderator density, and partial and preferential flooding, as discussed in FSAR Section 6.II.4. The applicant also performed analyses to determine the most reactive configuration of damaged fuel and fuel debris, as described in FSAR Section 6.II.4.4, and summarized in FSAR Section 7.2.2.1. Additionally, the applicant performed analyses to show that non-fuel hardware or neutron source assemblies present in the guide tubes of PWR assemblies are bounded by guide tubes filled with borated water, as discussed in FSAR Sections 6.II.4.6 and 6.II.4.7, respectively. Using the most reactive configuration determined from these analyses, the applicant determined the maximum system k_{eff} for each fuel assembly class with the minimum required soluble boron during loading and unloading. The bounding maximum k_{eff} for each assembly class with undamaged fuel is shown in FSAR Table 6.II.1.1. For configurations with damaged or failed fuel, the maximum calculated k_{eff} values for each assembly class are shown in FSAR Table 6.II.1.2. All of the maximum calculated k_{eff} values reported in Tables 6.II.1.1 and 6.II.1.2 are below 0.95, including all biases and uncertainties.

The staff reviewed the applicant's analyses described above to determine the most reactive configuration of the MPC-32. The staff determined the results of these analyses are appropriate and that the applicant has identified the most reactive fuel and canister configuration.

7.1.3.1 Computer Programs

The applicant used the MCNP5 Version 1.51 three-dimensional Monte Carlo neutron transport code and the ENDF/B-VII continuous-energy cross section library for all k_{eff} calculations for this amendment. The MCNP code is a standard in the nuclear industry for performing Monte Carlo criticality safety and radiation shielding calculations.

The staff performed confirmatory calculations, as discussed below, using the CSAS6 sequence of the SCALE 6.2.3 code system, with the KENO VI three-dimensional Monte Carlo neutron transport program and the continuous-energy ENDF/B-VII.1 cross section library.

7.1.3.2 Multiplication Factor

The applicant demonstrated that k_{eff} values for the MPC-32M canister storage configurations are all below 0.95, including all biases and uncertainties determined for the canister in the benchmarking analysis for the MCNP5 Version 1.51 code and continuous-energy ENDF/B-VII cross section library used in the criticality analysis, as discussed in Section 7.1.3 above. Therefore, the HI-STORM 100 system with the MPC-32M canister will remain subcritical under normal, off-normal, and accident conditions, demonstrating the design of the system is subcritical as required by 10 CFR 72.124(a) and 10 CFR 72.236(c).

The staff performed confirmatory criticality evaluations of the HI-STORM 100 system with the fuel configurations described in Section 7.1.1 of this SER in the MPC-32M canister. Using assumptions similar to the applicant's for fuel and canister materials and geometry, the staff calculated k_{eff} values for select configurations. The staff obtained results within the margin of error of those calculated by the applicant. Thus, the staff confirmed that the storage system is subcritical per the requirements of 10 CFR Part 72.124(a) and 10 CFR 72.236(c).

7.1.3.3 Benchmark Comparisons

The applicant previously provided a benchmarking analysis of MCNP5 Version 1.51 and the continuous-energy ENDF/B-VII cross section library for use in evaluating the MPC-68M canister. This benchmarking analysis is detailed in Appendix C to Holtec proprietary Report HI-2033039 Revision 6, and summarized in FSAR Section 6.A.1.1. The benchmarking analysis determined that the calculated bias did not exhibit any significant trends, and that the MPC-68M was within the range of applicability of the bias and bias uncertainty for all parameters considered. The design of the MPC-32M canister involves similar fuel types and basket, poison, and fuel configurations as the MPC-68M. Therefore, the applicant determined that the previously approved benchmarking analysis performed for the MPC-68M and the code and cross section data used is still applicable in this amendment. There are no significant deviations from the type or concentrations of fuel, moderator, and absorber material used in previously approved evaluations, and all of the parameters of interest to the criticality calculation remain within the area of applicability of the previous benchmarking analysis. Therefore, the staff finds that the previously approved bias and bias uncertainty for the MPC-68M canister modeled with MCNP5 Version 1.51 and the continuous-energy ENDF/B-VII cross section library are appropriate for criticality calculations of the MPC-32M canister.

7.1.4 Conclusion

Based on the above review, the staff finds that the HI-STORM 100 system with the MPC-32M canister will remain subcritical under normal, off-normal, and accident conditions, demonstrating the design of the system is subcritical as required by 10 CFR 72.124(a) and 10 CFR 72.236(c).

7.2 Proposed Change #3 – Include MPC-32 Version 1 and MPC-68 Version 1

For the MPC-32 Version 1 and MPC-68 Version 1, the fuel assembly classes to be stored are identical, in terms of assembly materials and geometry, initial enrichments, and soluble boron requirements (where applicable), to those for the MPC-32 and MPC-68, respectively. The canister and basket materials for the MPC-32 Version 1 and MPC-68 Version 1 are also identical to those for the MPC-32 and MPC-68, respectively. This includes the minimum ^{10}B areal density of the neutron absorber panels. The applicant states in FSAR Section 6.II.1.1 that the MPC-32 and MPC-68 basket designs bound, or are statistically equivalent to, the MPC-32 Version 1 and MPC-68 Version 1 basket designs, respectively, since the larger basket cell dimensions of the Version 1 canisters increase spacing between fuel assemblies, or between fuel assemblies and DFCs in configurations that allow damaged or failed fuel, which decreases system k_{eff}. Additionally, the applicant determined that the decrease in basket wall thickness has a negligible effect on k_{eff}. For this reason, the applicant did not perform any additional criticality analyses for the MPC-32 Version 1 and MPC-68 Version 1 canisters. Pitch expansion for damaged or failed fuel in a DFC is confined by the inner dimensions of the DFC, which has the same dimensions in the MPC-32 Version 1 and MPC-68 Version 1 canisters as for the MPC-32 and MPC-68, respectively. Basket cells with DFIs are not an authorized configuration for the MPC-32, MPC-68, MPC-32 Version 1, and MPC-68 Version 1. The staff concludes the applicant's conclusion that the larger dimension of the basket cells will serve to move fuel assemblies or DFCs farther apart in the basket, which will decrease system k_{eff}, is appropriate, and that the applicant has demonstrated that the decreased basket wall thickness has no statistically significant effect on system k_{eff}. Therefore, the staff finds that no further criticality analysis for the MPC-32 Version 1 and MPC-68 Version 1 canisters is necessary and that the system meets 10 CFR 72.124(a) and 10 CFR 72.236(c).

7.3 Proposed Change #5 – Add three additional BWR fuel types

7.3.1 Fuel Specification

For the MPC-68M canister, the applicant proposed to add three new BWR fuel assembly classes as allowable contents: 10x10I, 10x10J, and 11x11A. The fuel parameters for these new assembly classes are given in FSAR Table 2.III.3, including enrichment limits with and without limited Gd_2O_3 credit. The arrangement of full-length fuel rods, partial-length fuel rods, and water rods for these new BWR fuel assembly classes are shown in FSAR Figure 6.III.4.1.

7.3.2 Model Specification

7.3.2.1 Configuration

The addition of new fuel types to the MPC-68M does not change the storage system's response to normal, off-normal, or accident conditions. The previously approved structural evaluation of the MPC-68M shows the potential for small deformations of the Metamic^{TM}-HT basket under accident conditions. These deformations are included in the evaluation of material and fabrication tolerances and are conservatively assumed to exist under all conditions for the purposes of the criticality analysis.

The applicant modeled the MPC-68M using the most reactive configuration of the canister basket and fuel assemblies determined from previous analyses. The staff finds this acceptable, as the fuel materials and geometry to be loaded into the MPC-68M are identical to those allowed in the MPC-68. For the basket geometry, the configuration consisted of the most reactive minimum basket cell inner dimension and minimum basket wall thickness, including potential deformation of the basket walls from accident conditions, and eccentric positioning of fuel assemblies. The applicant also conservatively modeled the Metamic^{TM}-HT basket walls with 90% of the required minimum ^{10}B content specified in Drawing No. 7195 Revision 16, consistent with the recommendations in Section 7.4 of NUREG-2215. For the fuel assemblies, the model used the most reactive combination of tolerances for the pellet diameter, outer clad diameter, and clad thickness, as well as the maximum active fuel length. The applicant modeled the fuel material as UO_2 at a stack density of 97.5% theoretical density, without allowance for dishing or chamfer, which conservatively increases the fuel material in the modeled assembly. Additionally, the applicant conservatively assumed that the pellet-to-clad gap is filled with full density moderator.

In Amendment No. 15, for damaged fuel in DFCs or basket locations with DFIs, the applicant modeled the fuel as clad rods with varying pitch, with a maximum pitch defined by the inner dimension of the basket cell walls. For fuel debris in a DFC, the applicant modeled the fuel as unclad rods with varying pitch, with a maximum pitch defined by the inner dimension of the DFC. For all damaged fuel and fuel debris configurations, the damaged or failed fuel is modeled at 5.0 wt. % ^{235}U. This damaged fuel/fuel debris modeling approach is consistent with that previously evaluated and approved by the NRC for the MPC-68M canister. The staff finds that this approach is appropriate for the MPC-68M since the fuel and canister materials and geometry for the MPC-68M are largely the same as for the previously approved MPC-68 canister.

The configurations of the MPC-68M canister analyzed in the criticality analysis for Amendment No. 15 are consistent with previously approved analyses of the HI-STORM 100 system. The staff finds that the applicant has determined the most reactive configuration of each canister basket and fuel type, for the reasons discussed in the preceding paragraphs.

7.3.2.2 Material Properties

The MPC-68M canister and basket materials have not changed from the previously approved design and are described in FSAR Section 6.III.3. These descriptions include the composition of the major components of the MPC, including UO_2 fuel, steel and aluminum structural components, and neutron absorber panels. The criticality analyses conservatively assume no more than 90% of the Metamic^{TM}-HT minimum ^{10}B content specified in Drawing No. 7195 Revision 16, consistent with the recommendations in Section 7.4 of NUREG-2215. The fuel materials for the MPC-68M canister have also not changed from the previously approved design. Because the materials have not changed, the staff finds that the applicant has determined conservative material properties for the canister basket and each fuel type.

7.3.3 Criticality Analysis

The applicant performed a criticality analysis for the requested new fuel contents in the MPC-68M canister using the criticality model and material properties described above. The resulting maximum initial enrichments and associated k_{eff} values are reported in FSAR Table 6.III.1.1 for the new 10x10I, 10x10J, and 11x11A fuel assembly classes.

Since BWR fuel typically is designed with radially and axially varying fuel rod enrichments, the applicant performed a study to demonstrate that modeling the fuel at the maximum planar average initial enrichment produces a higher system k_{eff} value than modeling actual distributed enrichments. Similar analyses were performed for the previously approved fuel assembly classes allowed for storage in the MPC-68M canister. For each new fuel assembly class, the applicant performed a series of analyses comparing k_{eff} values of fuel with various configurations of distributed enrichments to those for fuel modeled with a uniform average enrichment. The results of these analyses, shown in FSAR Table 6.III.2.4, demonstrate that in all cases, k_{eff} values for the average enrichment model were higher or statistically the same as for the distributed enrichment model. The applicant therefore conservatively modeled the new fuel assembly classes with a uniform enrichment equal to the maximum planar average for all subsequent models of the MPC-68M canister. The staff reviewed this study and finds that the applicant has demonstrated that modeling the fuel with a uniform maximum planar average enrichment is conservative. These results are consistent with staff's experience with reviews and confirmatory analyses performed for other BWR spent fuel storage systems and are consistent with the guidance in Section 7.5.2.1 of NUREG-2215 regarding the use of lattice-averaged enrichments in BWR fuel assemblies.

The 10x10I, 10x10J, and 11x11A fuel assembly classes have partial-length rod configurations as shown in FSAR Figure 6.III.4.1. Similar to previous analyses for other BWR fuel assembly classes, the applicant performed a series of studies to determine the most reactive configuration of partial-length rods, comparing k_{eff} values for three different configurations: (1) partial-length rods modeled at lengths similar to the actual fuel design, (2) partial-length rods extended to the length of full-length rods, and (3) partial-length rods removed from the fuel lattice. The results of these analyses, shown in FSAR Table 6.III.4.13, demonstrate that, for the 10x10I and 11x11A fuel assembly classes, removing the partial-length rods from the fuel lattice results in the highest k_{eff} value. For these fuel assembly classes, all subsequent criticality analyses conservatively model the fuel with partial-length rods removed from the fuel lattice. For the 10x10J fuel assembly class, however, the configuration with partial-length rods at lengths similar to the actual fuel design was the most reactive. The applicant performed a series of additional calculations with the 10x10J fuel assembly class, independently varying the length of the long and short partial-length rods in the fuel lattice. These additional calculations demonstrate that all other assumed partial-length rod lengths result in k_{eff} values that are lower or statistically the same as for the initial assumed partial-length rod lengths. Therefore, for all subsequent criticality calculations with the 10x10J fuel assembly class in the MPC-68M canister, the initial assumed short and long partial-length rod lengths are conservatively used. Based on the above results, the staff finds that the applicant has identified the most reactive configuration of partial-length rods for the 10x10I, 10x10J, and 11x11A fuel assembly classes.

7.3.3.1 Computer Programs

The applicant used the MCNP5 Version 1.51 three-dimensional Monte Carlo neutron transport code and the ENDF/B-VII continuous-energy cross section library for all k_{eff} calculations for this amendment. The MCNP code is a standard in the nuclear industry for performing Monte Carlo criticality safety and radiation shielding calculations.

The staff performed confirmatory calculations, as discussed below, using the CSAS6 sequence of the SCALE 6.2.3 code system, with the KENO VI three-dimensional Monte Carlo neutron transport program and the continuous-energy ENDF/B-VII.1 cross section library.

7.3.3.2 Multiplication Factor

The applicant demonstrated that k_{eff} values for the new fuel contents for the MPC-68M canister are all below 0.95, including all biases and uncertainties calculated for the canister in the benchmarking analysis for the MCNP5 Version 1.51 code and continuous-energy ENDF/B-VII cross section library used in the criticality analysis, as discussed in Section 7.3.3 above. Therefore, the applicant stated that the HI-STORM 100 system with the MPC-68M canister will remain subcritical under normal, off-normal, and accident conditions, demonstrating the design of the system is subcritical as required by 10 CFR 72.124(a) and 10 CFR 72.236(c).

The staff performed confirmatory criticality evaluations of the HI-STORM 100 system with new fuel contents in the MPC-68M canister. Using assumptions similar to the applicant's for fuel and canister materials and geometry, the staff calculated k_{eff} values for selected configurations. The staff obtained results within the margin of error of those calculated by the applicant. Thus, the staff confirmed that the storage system is subcritical per the requirements of 10 CFR 72.124(a) and 10 CFR 72.236(c).

7.3.3.3 Benchmark Comparisons

As the applicant's requested revisions to the MPC-68M canister involve similar types of fuel and basket poisons (with the exception of Gd_2O_3 credited in some calculations), and fuel configurations to those previously used to perform benchmark comparisons, the applicant determined that the previously approved benchmarking analysis performed for the code and cross section data used in this amendment is still applicable. There are no significant deviations from the type or concentrations of fuel, moderator, and absorber material used in previously approved evaluations, and all of the parameters of interest to the criticality calculation remain within the area of applicability of the previous benchmarking analysis summarized in FSAR Section 6.A.1.1. Therefore, the staff finds that, for storage configurations that do not include Gd_2O_3 credit, the previously approved bias and bias uncertainty for the MPC-68M canister modeled with MCNP5 Version 1.51 and the continuous-energy ENDF/B-VII cross section library are appropriate. This SER evaluates configurations involving Gd_2O_3 credit in Section 7.4.

7.3.4 Conclusion

Based on the review described above, the staff finds that the HI-STORM 100 system with the MPC-68M canister with the 10x10I, 10x10J, and 11x11A fuel assembly classes will remain subcritical under normal, off-normal, and accident conditions, demonstrating the design of the system is subcritical as required by 10 CFR 72.124(a) and 10 CFR 72.236(c).

7.4 Proposed Change #8 – Allow for partial Gd_2O_3 credit for BWR fuel assemblies

7.4.1 Fuel Specification

The applicant proposed to allow limited Gd_2O_3 credit for all 10x10 and 11x11 BWR fuel assembly classes, with a maximum planar average initial enrichment of 5.0 wt. % ^{235}U, in the MPC-68M canister. For fuel loaded with enrichment values up to 5.0 wt. % ^{235}U with limited Gd_2O_3 credit, the minimum Gd_2O_3 rod requirements, in terms of number of required rods, rod placement, and minimum Gd_2O_3 concentration, are given in FSAR Table 2.III.5.

7.4.2 Model Specification

7.4.2.1 Configuration

The addition of Gd_2O_3 credit for some contents of the MPC-68M does not change the storage system's response to normal, off-normal, or accident conditions. The previously approved structural evaluation of the MPC-68M shows the potential for small deformations of the Metamic^{TM}-HT basket under accident conditions. These deformations are included in the evaluation of material and fabrication tolerances and are conservatively assumed to exist under all conditions for the purposes of the criticality analysis.

For undamaged 10x10 and 11x11 fuel assembly classes with limited Gd_2O_3 credit, the applicant modeled the fuel assemblies according to the requirements in FSAR Table 2.III.5. This included crediting no more than 3.0 wt. % Gd_2O_3 in rods that must have Gd_2O_3, no Gd_2O_3 credited in the peripheral row of the fuel assembly, and no Gd_2O_3 credited in damaged or failed fuel assemblies. The models for 10x10C and 10x10G assembly classes credit two Gd_2O_3 rods; the models for all other 10x10 and 11x11 assembly classes credit one Gd_2O_3 rod. The applicant models BWR fuel assemblies with Gd_2O_3 rods as fresh, unburned fuel, since the applicant shows in FSAR Section 6.III.4.8 that for the number of rods credited, fresh fuel is most reactive (i.e., there is no "peak reactivity" at a higher burnup). The applicant evaluated several different configurations of credited Gd_2O_3 rods in order to determine the most reactive (i.e., the configuration where the Gd_2O_3 rods are least effective in reducing k_{eff}).

The configurations of the MPC-68M canister analyzed in the criticality analysis for 10x10 and 11x11 fuel assembly classes with Gd_2O_3 credit are conservative since the number and Gd_2O_3 concentration of rods credited is low relative to typical BWR fuel designs, and the applicant modeled the Gd_2O_3 rods in the least effective assembly location. The staff reviewed these evaluations and finds that the applicant has determined the most reactive configuration of fuel assemblies with Gd_2O_3 credit within the MPC-68M canister.

7.4.2.2 Material Properties

The MPC-68M canister and basket materials have not changed from the previously approved design and are described in FSAR Section 6.III.3. These descriptions include the composition of the major components of the MPC, including UO_2 fuel, steel and aluminum structural components, and neutron absorber panels. The criticality analyses conservatively assume no more than 90% of the Metamic^{TM}-HT minimum ^{10}B content specified in Drawing No. 7195 Revision 16, consistent with the recommendations in Section 7.4 of NUREG-2215. The fuel materials for the MPC-68M canister have also not changed, with the exception of fuel for which limited Gd_2O_3 credit is applied, as discussed in FSAR Section 6.III.4.8. Gd_2O_3 is mixed with UO_2 fuel material at a maximum credited concentration of 3.0 wt. % Gd_2O_3. Although Gd_2O_3 will typically be present in more rods within an actual BWR fuel assembly, the maximum number of rods credited in the criticality analysis is two (for assembly classes 10x10C and 10x10G) or one (for all other 10x10 and 11x11 assembly classes). The staff finds that the material properties assumed in the applicant's analysis of BWR fuel with Gd_2O_3 rods are consistent with the required Gd_2O_3 concentration. All other material properties are consistent with those assumed in previously approved analyses of the MPC-68M canister, and therefore, are acceptable.

7.4.3 Criticality Analysis

Modern BWR fuel assemblies are typically designed with significant amounts of burnable absorber integral to the UO_2 fuel matrix, in the form of Gd_2O_3. The applicant conservatively neglects this absorber for most BWR fuel analyses and models the fuel as fresh UO_2 composition with no Gd_2O_3. The maximum initial planar average enrichment for BWR fuel assemblies without any credit for Gd_2O_3 is 4.8 wt. % ^{235}U. In order to store 10x10 and 11x11 fuel assembly classes with a maximum planar average enrichment up to 5.0 wt. % ^{235}U, the applicant provided analyses crediting a limited amount of Gd_2O_3 typically present in BWR fuel assemblies.

BWR Gd_2O_3 loadings are typically high, in terms of the number of Gd_2O_3 rods and the concentration of Gd_2O_3 within each rod, such that fuel assemblies do not typically have their highest reactivity when fresh. The Gd_2O_3 in each rod burns out over the first cycle of operation and produces a peak in fuel assembly reactivity typically near the end of the first cycle. However, the applicant provided analyses which demonstrate that, for a low number of rods at a low Gd_2O_3 concentration, the fresh fuel composition with Gd_2O_3 is the most reactive (i.e., the fuel does not have a higher reactivity peak at higher burnup). The applicant's analyses for limited Gd_2O_3 credit, as discussed in FSAR Section 6.III.4.8, evaluate one or two rods in each 10x10 and 11x11 fuel assembly class, with a Gd_2O_3 concentration of 3.0 wt. %. Since the location of the Gd_2O_3 rod within the fuel lattice can greatly affect the rate at which the Gd_2O_3 depletes, the applicant evaluated a range of Gd_2O_3 rod locations within each fuel assembly class. The Gd_2O_3 rod locations evaluated for each fuel assembly class are shown in FSAR Figure 6.III.4.2.

To demonstrate that the fresh unburned fuel composition with limited Gd_2O_3 credit is more reactive than the composition of the same fuel at any burnup, the applicant performed multiple two-dimensional infinite lattice burnup calculations using the CASMO5 lattice physics code. These calculations modeled the initial composition of fuel with Gd_2O_3 rods in several configurations and determined the k_{eff} values of the lattice with increasing burnup. The results of these calculations, shown in FSAR Figures 6.III.4.3 through 6.III.4.10, demonstrate that fuel with fresh unburned composition is most reactive in the 10x10C and 10x10G fuel assembly classes with two Gd_2O_3 rods, and most reactive for all other 10x10 and 11x11 fuel assembly classes with one Gd_2O_3 rod.

The applicant then performed a series of calculations with the Gd_2O_3 rods credited as described above to determine the k_{eff} values for each fuel assembly class in the MPC-68M. These calculations model the Gd_2O_3 rods in several lattice locations in order to determine where they are least effective in reducing k_{eff}. Maximum system k_{eff} is then determined using this conservative Gd_2O_3 rod location. The results of these calculations are shown in FSAR Table 6.III.4.15.

Although the applicant modeled the fuel with fresh compositions for the analysis to determine maximum system k_{eff}, the applicant provided a misload analysis, consistent with the guidance in NUREG-2215 for burnup credit analyses. This analysis, described in FSAR Section 6.III.4.8.5, conservatively assumes that all fuel assemblies with Gd_2O_3 rods are replaced with fuel assemblies with no Gd_2O_3 rods. The results of this analysis, given in FSAR Table 6.III.4.15 for configurations with undamaged fuel only and in FSAR Table 6.III.4.18 for configurations with damaged fuel, show that such a misload results in a k_{eff} below 0.98, including biases and uncertainties. This result is consistent with the guidance of NUREG-2215 for misload analyses of burnup credit cask analyses.

The applicant also provided a comparison of the k_{eff} of the MPC-68M with limited Gd_2O_3 credit to the k_{eff} of the cask system containing BWR fuel with more typical Gd_2O_3 loadings. FSAR Section 6.III.4.8.4.3 discusses the configurations considered in the analysis, which include fuel assemblies with 6, 15, or 18 Gd_2O_3 rods in various arrangements. The applicant performed this analysis using the CASMO5 two-dimensional neutron transport code. The results of these analyses, shown in FSAR Table 6.III.4.17, show that the more realistic BWR Gd_2O_3 rod loadings have a k_{eff} of at least 0.05 lower than the configuration modeled for the MPC-68M safety analyses.

The staff evaluated the applicant's use of limited Gd_2O_3 credit for 10x10 and 11x11 assembly classes stored in the MPC-68M canister. The staff finds that the applicant has demonstrated that the number of Gd_2O_3 rods credited, as well as the concentration of Gd_2O_3 credited within each rod, will result in a maximum system k_{eff} with fresh fuel compositions. The staff also finds that the applicant has identified the most conservative arrangement of Gd_2O_3 rods within each BWR assembly lattice. The staff therefore finds the applicant's use of limited Gd_2O_3 credit acceptable because it is conservative.

7.4.3.1 Computer Programs

The applicant used the MCNP5 Version 1.51 three-dimensional Monte Carlo neutron transport code and the ENDF/B-VII continuous-energy cross section library for all k_{eff} calculations for this amendment. The MCNP code is a standard in the nuclear industry for performing Monte Carlo criticality safety and radiation shielding calculations.

The applicant also used the CASMO5 Version 2.00.00 two-dimensional deterministic neutron transport code for select scoping studies for BWR fuel analyses in the MPC-68M canister, including those studies involving burned fuel compositions with Gd_2O_3 rods. This code is not explicitly benchmarked for this specific safety analyses. The staff, however, finds this use acceptable because this code is only used to compare system reactivities and to qualitatively verify that there is no evidence of peak reactivity with fuel irradiation where the reactivity would be higher than that for the fresh unburned fuel condition when limited Gd_2O_3 credit is considered. Any code bias and bias uncertainty determined in a benchmarking analysis would affect each code result in the same way, such that the direction of reactivity changes determined in this analysis would not be affected. Also, the staff notes that CASMO5 is widely used for light water reactor fuel lattice physics and depletion calculations in the nuclear industry and is extensively benchmarked for this purpose.

The staff performed confirmatory calculations, as discussed below, using the CSAS6 sequence of the SCALE 6.2.3 code system, with the KENO VI three-dimensional Monte Carlo neutron transport program and the continuous-energy ENDF/B-VII.1 cross section library.

7.4.3.2 Multiplication Factor

With the above-discussed models, the applicant demonstrated that k_{eff} values for the additional storage configurations for the MPC-68M canister are all below 0.95, including all biases and uncertainties calculated for each canister in the benchmarking analysis for the MCNP5 Version 1.51 code and continuous-energy ENDF/B-VII cross section library used in the criticality analysis. Therefore, the HI-STORM 100 system with the MPC-68M canister will remain subcritical under normal, off-normal, and accident conditions, demonstrating the design of the system is subcritical as required by 10 CFR 72.124(a) and 10 CFR 72.236(c).

The staff performed confirmatory criticality evaluations of the HI-STORM 100 system with additional fuel configurations in the MPC-68M canister. Using assumptions similar to the applicant's for fuel and canister materials and geometry, the staff calculated k_{eff} values for selected configurations. The staff obtained results within the margin of error of those calculated by the applicant. Thus, the staff confirmed that the storage system is subcritical per the requirements of 10 CFR 72.124(a) and 10 CFR 72.236(c).

7.4.3.3 Benchmark Comparisons

The previous benchmarking analysis for the HI-STORM 100 system with MCNP5 Version 1.51 and the continuous-energy ENDF/B-VII cross section library is summarized in FSAR Section 6.A.1.1 and discussed in detail in Appendix C to Holtec proprietary Report HI-2033039 Revision 6. The bias and bias uncertainty from this benchmarking analysis were determined based on the results from 56 critical configurations chosen to match the neutronic characteristics of the canisters and fuel of the HI-STORM 100 system. To capture the potential effects on bias and bias uncertainty of including Gd_2O_3 for configurations of the MPC-68M canister which credited Gd_2O_3, the applicant evaluated 15 additional critical configurations with Gd_2O_3, described in Section C.5 of Report HI-2033039. As described in FSAR Section 6.A.1.1, the applicant compared the bias and bias uncertainty calculated in the previously approved benchmarking analysis to that calculated with the additional 15 Gd_2O_3 critical configurations. Since the total bias and bias uncertainty calculated with the additional 15 configurations is lower than that previously calculated for MCNP5 Version 1.51 and the continuous-energy ENDF/B-VII cross section library, the applicant conservatively applied the previously calculated bias and bias uncertainty to all calculations involving Gd_2O_3 credit. The staff reviewed these evaluations and finds that this approach to benchmarking Gd_2O_3 credit calculations in the MPC-68M canister with MCNP5 Version 1.51 and the continuous-energy ENDF/B-VII cross section library is acceptable because it is conservative.

7.4.4 Conclusion

Based on the review described above, the staff finds that the HI-STORM 100 system with the MPC-68M canister with partial Gd_2O_3 credit for 10x10 and 11x11 fuel assembly classes will remain subcritical under normal, off-normal, and accident conditions, demonstrating the design of the system is subcritical as required by 10 CFR 72.124(a) and 10 CFR 72.236(c).

7.5 Proposed Change #11 – Update Drawing No. 7195 for the MPC-68M

The applicant proposed to update Drawing No. 7195 for the MPC-68M to remove dimensions that are not used in the safety analysis. One of the dimensions that the applicant proposed to remove is the tolerance value for the basket cell wall panel thickness. Since the cell wall panels are made of Metamic^{TM}-HT with a minimum ^{10}B specified in Note 5 of the drawing for reactivity control, this dimension is important for criticality safety. This dimension is, in fact, used in the safety analysis when determining the most reactive dimension of the basket cell wall panel thickness, which the applicant determined to be the minimum thickness, as discussed in FSAR Section 6.III.3. Although the basket cell wall dimension removed from Drawing No. 7195 is used in the safety analysis, the staff finds the removal of this dimension to be acceptable because the minimum dimension of the basket cell wall thickness is specified in FSAR Table 6.III.3.1.

7.6 Evaluation Findings

F7.1 SSCs important to criticality safety are described in sufficient detail in Chapters 2 and 6 of the FSAR to enable an evaluation of their effectiveness.

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

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

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

The staff concludes that the criticality design features for the HI-STORM 100 spent fuel storage system design, as amended, 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 spent fuel storage system design, as amended, 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.

8.0 MATERIALS EVALUATION

The objective of materials review is to ensure that the applicant adequately described and evaluated materials performance, including whether the new SSCs and contents introduce any changes in the service environment (e.g., temperature and radiation) that could affect materials performance of all ITS SSCs.

The staff evaluated the material changes associated with Amendment No. 15 of the HI--STORM 100 system. The staff notes that the new SSCs introduced in the amendment are variations of SSCs that have been previously approved for HI-STORM 100 system. The materials, fabrication, and examination requirements remain essentially unchanged.

8.1 Summary of Proposed Changes Affecting Materials or Material Service Conditions

8.1.1 Proposed Change #1 – Add HI-TRAC MS

FSAR Sections 1.II.2.3 and 2.II.0.3 and Drawing No. 11381 discussed the new transfer cask, HI-TRAC MS, including the components, design, materials, and specifications for fabrication. The applicant stated that the HI-TRAC MS design includes an option for variable weight (thickness) of the lead and water jacket to accommodate variations in site-specific MPCs and adds cooling passages to the bottom lid to allow for storage of canisters containing higher heat loads. The applicant stated that a gap maintained by spacers between the bottom plate and bottom flange will allow for natural air circulation and enhances cooling between the inner transfer cask shell and the canister outer shell. The staff notes that lifting blocks are added to all versions of the HI-STORM 100 HI-TRACs, including the HI-TRAC MS.

8.1.2 Proposed Change #2 – Include MPC-32M

FSAR Section 1.II.2.2 and Drawing No. 11425 discussed the new canister, MPC-32M, including the components, design, materials, and specifications for fabrication. The applicant stated that the MPC-32M is the PWR counterpart to the previously approved BWR MPC-68M and allows for storage of spent fuel assemblies with higher heat loads and reduced cooling times. In addition, the MPC-32M fuel basket is fabricated entirely from Metamic^{TM}-HT. The applicant stated that the use of neutron absorber Metamic^{TM}-HT increases the subcriticality function of the canister to allow loading of more reactive fuel assemblies.

8.1.3 Proposed Change #3 – Include MPC-32 Version 1 and MPC-68 Version 1

FSAR Section 1.II.2.2 and Drawings No. 11572, 11574, and 11578 discussed the new Version 1 of MPC-32 and MPC-68, including the components, design, materials, and specifications for fabrication. The applicant stated that Version 1 of MPC-32 and MPC-68 are modified by slightly enlarging basket cell openings to improve the ability to store DFCs and by slightly increasing the baseplate thickness to allow for increased helium backfill pressure and load bearing capacity. The allowable contents for the Version 1 canisters remain the same as for the previously approved MPC-32 and MPC-68.

8.1.4 Proposed Change #4 – Add HI-STORM 100S Version E

FSAR Sections 1.II.2.1 and 2.II and Drawing No. 11371 discussed the new overpack, HI-STORM 100S Version E, including the components, design, materials, and specifications for fabrication. The applicant stated that the Version E is a variant of the previously approved Version B with an increased number of inlet vents, elevated inlet vents, increased outlet flow area, added bottom supports to elevate the MPC inside the enclosure vessel, improved top lid design to limit shear load on the closure bolts, and increased density of the shielding concrete in the overpack body and top lid. The applicant stated that the Version E allows for storage of higher heat loads by increasing the number of inlet air vents to provide increased cooling and by angling the inlet vents to a higher axial location, allowing continued heat rejection during a flood event. In addition, bottom supports that elevate the MPC inside of the overpack allow air flow underneath the MPC providing additional cooling and placing the MPC in an area of increased shielding. Further, the top lid is designed with an increased outlet flow area, which improves the thermal efficiency of the system, and the closure bolts are designed to reduce the shear load during tipover. The applicant stated that these Version E design changes will result in a lower peak fuel cladding temperature than Version B.

8.1.5 Proposed Change #5 – Add Three BWR fuel types: 10x10I, 10x10J, and 11x11A

FSAR Section 2.III.1 identified new spent fuel assemblies to be stored. The applicant stated that three additional BWR fuel types, 10x10I, 10x10J, and 11x11A, will be added to the approved contents for previously approved MPC-68M only. The applicant stated that these fuel types are currently in use by operating reactors and will require storage. FSAR Section 2.1.9.1 discusses decay heat, burnup, and cooling time limits for zirconium (ZR)-clad fuel, including these three fuel types.

8.1.6 Proposed Change #7 – Add HI-DRIP and DIJ Ancillary Systems

FSAR Sections 1.2.1.6 and 1.2.1.7, and Figures 1.2.19 through 1.2.21 discussed the optional HI-DRIP and DIJ Auxiliary Cooling Systems. The applicant stated that the HI-DRIP auxiliary cooling system is designed to prevent the loaded MPC water from boiling prior to the evacuation of water and backfill of helium. The staff notes that the HI-DRIP consists of a tubular ring equipped with spray nozzles that girdles the transfer cask above its main cylindrical body such that the entire circumference of the transfer cask shell can be wetted by gravity. In addition, the ring is connected to the plant's water supply with a gate valve to regulate the flow of water. The applicant stated that, as long as the external surface of the transfer cask remains wet, the water in the canister will remain below boiling. The applicant stated that, for a high-heat-emitting MPC, a custom-sized DIJ that surrounds all or a portion of the transfer cask water jacket serves as a passive means to augment heat removal and can be used for short-term operations in the vertical, horizontal, or inclined position at a site.

8.2 Materials Review

8.2.1 Applicable Codes and Standards

FSAR Sections 14.II.3 and 14.II.5 discussed applicable codes and standards and welding material and specifications, respectively. The staff notes that the ITS structural and non-structural components are made primarily from ASME SA-516, Grade 70 carbon steel and ASME SA-240 Type 304 stainless steel, Metamic^{TM}-HT, ASTM B221 aluminum alloy, ASTM B29 lead, and concrete designed to ACI 349. In addition, the bolts are ASME SA-193, Grade B7, alloy steel. The typical material specifications used in the fabrication of the lift blocks are ASME SA-965 Grade FXM-11 or SA-336 Grade FXM-11 stainless steel and ASME SB-637 (UNS N07718) nickel alloy for the lift block attachment bolts.

The staff reviewed the applicable codes, standards and specifications described in the applicant’s FSAR and verified that the Amendment No. 15 HI-STORM 100 system SSCs use the same construction codes, materials, welding requirements, and examination methods as the previously approved SSC versions. The staff confirmed that the construction codes, materials, welding requirements, and examination methods remain valid under the service conditions associated with the Amendment No. 15 HI-STORM 100 system ITS SSCs. Therefore, the staff finds the material codes and standards to be acceptable, as required by 10 CFR 72.236(b).

8.2.2 Drawings

The staff reviewed the new license assembly drawings for Amendment No. 15 HI-STORM 100 system SSCs and confirmed that the drawings provided updates to incorporate the proposed changes (i.e., MPC, basket assemblies, overpack, and transfer cask). In addition, the drawings contain a bill of materials, including appropriate consensus code information, consistent with those of previously approved drawings.

The staff reviewed the drawing content with respect to the guidance in NUREG/CR-5502, “Engineering Drawings for 10 CFR Part 71 Package Approvals,” (NRC, 1998) and confirmed that the drawings provide an adequate description of the materials, fabrication, and examination requirements; therefore, the staff finds the drawings to be acceptable, as required by 10 CFR 72.236.

8.2.3 Material Properties

FSAR Sections 2.II, 3.II, and 14.II discussed Amendment No. 15 HI-STORM 100 System SSCs’ principal design criteria, structural safety, and materials, respectively. FSAR Section 3.1 provided allowable stresses, stress intensities, and fracture toughness requirements for the Amendment No. 15 HI-STORM 100 System SSC materials for different service conditions. The staff verified that the mechanical and physical property data for the major structural materials were largely obtained from ASME Code, Section II, Part D; however, some of the values were obtained from other acceptable technical references. In addition, the applicant did not make changes to the material mechanical properties used in the structural analysis.

The staff reviewed the service temperatures described in the applicant's thermal analysis and confirmed that material mechanical properties remain valid under the service conditions associated with the Amendment No. 15 HI-STORM 100 System ITS SSCs. Therefore, the staff finds the mechanical properties used in the applicant's structural analysis to be acceptable, as required by 10 CFR 72.236(b).

8.2.4 Thermal Properties

FSAR Section 14.II.4.2 provided the thermal properties of materials used in the thermal analyses of the HI-STORM 100 system SSCs (i.e., Alloy X, Metamic^{TM}-HT, aluminum shims, helium gas, carbon steel, concrete, lead, fuel (UO_2) cladding, and demineralized water). The applicant stated that these properties include density, thermal conductivity, heat capacity, viscosity, and surface emissivity/absorptivity. In addition, the applicant also provided variations of these properties with temperature. The applicant stated that thermal properties are obtained from standard handbooks, established textbooks and, when variations of thermal properties are observed, the most conservative values are established as input for the design of the HI-STORM 100 System SSCs.

The staff evaluated the applicant's listed thermal properties of the materials credited in the thermal analysis and finds that the thermal properties are consistent with those in the technical literature. FSAR Sections 4.II and 14.II discuss the maximum temperatures of Amendment No. 15 HI-STORM 100 system SSCs under normal, off-normal, and accident conditions, respectively. The staff reviewed the applicant's analysis and finds that the component temperatures remain below those associated with the previously approved HI-STORM 100 system, as well as below each of the material's allowable service temperatures; therefore, the thermal properties are acceptable and meet 10 CFR 72.236(b).

8.2.5 Radiation Effects

The staff notes that the newly added fuel assemblies are not entirely bounded by the previously approved fuel contents. This is because the fuel mass is marginally higher, which means the radiation to which the fuel assemblies are exposed would be marginally higher. The staff notes that the carbon/stainless steel materials selected for the HI-STORM 100 System SSCs have a long, proven history of use in the nuclear industry and are not affected by the radiation levels produced by the SNF. In addition, significant neutron radiation damage of steels does not occur for neutron fluences below 10^{19} neutrons/centimeter squared (n/cm^2) and significant gamma radiation damage to metals only occurs for doses of equal to or greater than 10^{18} rads. The staff notes that these values are much greater than the neutron fluence exposure expected by the HI-STORM 100 System SSCs and the gamma dose produced by SNF in the MPC (approximately 10^{13} n/cm^2 for neutrons and 10^{10} rads for gamma, respectively).

Therefore, the staff finds that the new fuel contents will not result in detrimental radiation effects on Amendment No. 15 HI-STORM 100 system SSC structural steels, based on the above discussion.

8.2.6 Corrosion and Content Reactions

FSAR Section 14.II.12 discussed chemical and galvanic reactions. The applicant stated that materials used in the HI-STORM 100 System do not participate in any chemical or galvanic reactions when exposed to the various environments during normal operating conditions, off-normal, and accident events.

The staff reviewed Amendment No. 15 HI-STORM 100 System SSCs and verified that they do not introduce any adverse corrosive or other reactions that were not previously considered in the staff's prior review of the HI-STORM 100 System. In addition, the materials of construction and the service environments are unchanged. Therefore, the staff finds the applicant's evaluation of corrosion resistance and potential adverse reactions to be acceptable.

8.2.7 Cladding Integrity

The staff notes that the zirconium cladding material for the newly added BWR fuel assembly contents is identical to the cladding that was previously approved for the HI-STORM 100 System.

The staff verified that the cladding temperatures for each fuel type proposed for storage within Amendment No. 15 HI-STORM 100 System are below the allowable temperature limits that would preclude cladding damage possibly leading to gross rupture. Therefore, the staff finds that the fuel cladding for the new fuel contents is acceptable, based on the above discussion.

8.3 Evaluation Findings

F8.1 The applicant has met the requirements in 10 CFR 72.236(b). The applicant described the materials design criteria for SSCs ITS in sufficient detail to support a safety finding.

F8.2 The applicant has met the requirements in 10 CFR 72.236(g). The properties of the materials in the storage system design have been demonstrated to support the safe storage of SNF.

F8.3 The applicant has met the requirements in 10 CFR 72.236(a) and 10 CFR 72.236(m). SNF specifications have been provided and adequate consideration has been given to compatibility with retrieval of stored fuel for ultimate disposal.

The staff reviewed the applicable regulations, appropriate regulatory guidance, applicable codes and standards, and accepted engineering practices. Based on its review, the staff concludes that the HI-STORM 100 Amendment No. 15 design adequately considers material properties, environmental degradation, and other reactions, fuel clad integrity, content retrievability, and material quality controls such that the design is in compliance with 10 CFR Part 72. Therefore, there is reasonable assurance that the HI-STORM 100 Amendment No. 15 will allow safe storage of SNF for a licensed (certified) life of 20 years.

9.0 OPERATING PROCEDURES EVALUATION

The objective of this review is to ensure that the applicant's FSAR presents acceptable operating sequences, guidance, and generic procedures for the key operations. The review also ensures that the FSAR incorporates and is compatible with the applicable operating control limits in the TS.

In Amendment No. 15, the applicant revised FSAR Chapter 8, Operating Procedures, and added FSAR Section 8.II, Operating Procedures, to address the following proposed changes:

- Proposed Change #1 – Add a new version of a transfer cask, HI-TRAC MS
- Proposed Change #2 – Add a new MPC called MPC-32M
- Proposed Change #3 – Add Version 1 of MPC-32 and MPC-68
- Proposed Change #4 – Add HI-STORM 100S Version E overpack
- Proposed Change #7 – Add HI-DRIP and DIJ ancillary system as additional cooling
- Proposed Change #12 – Include DIJ as optional alternate cooling method for short-term operation of a loaded HI-TRAC

The applicant made minor changes to FSAR Chapter 8 which includes the use of optional lift blocks in place of lifts using the trunnions for lifting the various HI-TRAC designs. However, the new HI-TRAC MS design, added in Amendment No. 15, is only lifted using the lift blocks since it does not have trunnions, thus requiring the generic change to state trunnions or lift blocks are used for lifting HI-TRACs throughout FSAR Chapter 8.

FSAR Chapter 8 includes the installation of the optional HI-DRIP system, which is not important to safety, on the various HI-TRAC designs for additional cooling and refers to Section 8.II for HI-TRAC MS pool lid bolt torque values. The applicant added new guidance on when to remove the HI-DRIP system, if used, from the various HI-TRAC designs and when to depressurize the HI-TRAC MS bottom lid seal.

The applicant did not describe the use of the not important to safety DIJ as an alternate HI-TRAC cooling system in the FSAR Chapter 8 and Section 8.II, but did in FSAR Chapter 1, General Description. The use of the DIJ is not an operations issue since it is not important to safety. See Section 4 of this SER for the safety evaluation of the HI-DRIP and DIJ ancillary systems.

The proposed Section 8.II provides clarifications and limitations to the procedures for loading, unloading, and recovery presented in FSAR Chapter 8 as applicable to the HI-STORM 100S Version E, HI-TRAC MS, various MPC versions, and the new canister, MPC-32M. Section 8.II stated that the existing procedure steps outlined in FSAR Chapter 8 for loading, unloading, and recovery remain applicable without any modifications for all the versions of the Alloy X MPC canisters. The operations specific to MPC-68M are contained in the previously reviewed and NRC-approved Sections 1.III and 8.III. Section 8.II addresses the new Metamic^{TM}-HT canister, MPC-32M, by referring to Section 1.II.2.2 which provides an overview of operations. Section 8.II Tables 8.II.0.1 and 8.II.0.2 provide the handling weights for each of the new HI-STORM 100S Version E System components and new HI-TRAC MS major components and the loads to be lifted during the various phases of operation of the HI-STORM 100S Version E System. However, the handling weight values provided are only estimates as the actual size of the MPCs, HI-TRAC MS, and HI-STORM 100S Version E overpack may vary at different nuclear power plant sites. Actual component weights are dependent upon as-built dimensions. Users of these components are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.II.0.3 provides the HI-STORM 100 System bolt torque and sequencing requirements. Figure 8.II.0-1 illustrates the lift yoke engagement with HI-TRAC MS lift blocks. Section 8.II.1 provides operational guidance when loading the MPC-32M with moderate or high burnup spent fuel and the considerations that must be taken with respect to the thermal design basis heat load. Supplemental cooling measures described in Section 2.II shall be employed if indicated by the specific fuel loading thermal analysis and the Technical Specifications.

The staff has reviewed the associated updates in FSAR Chapter 8 for the proposed changes and determined the revised operating procedures are acceptable for using the HI-STORM storage system and satisfy the requirements in 10 CFR 72.236.

9.1 Evaluation Findings

F9.1 The HI-STORM 100S Version E is compatible with wet loading and unloading in compliance with 10 CFR 72.236(h). General procedure descriptions for these operations are summarized in FSAR Chapter 8 and Section 8.II. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F9.2 The welded lids of the MPCs allow ready retrieval of the spent fuel for further processing or disposal as required.

F9.3 The smooth surface of the various MPCs and HI-TRAC MS are designed to facilitate decontamination. Only routine decontamination will be necessary after the HI-TRAC is removed from the spent fuel pool.

F9.4 The content of the general operating procedures described in the FSAR for the applicable proposed changes are adequate to protect health and minimize damage to life and property. Detailed procedures will need to be developed and approved on a site-specific basis.

The staff concludes that the generic procedures and guidance for the operation of the HI-STORM 100S Version E, HI-TRAC MS, various MPC versions, and the new canister, MPC-32M, are in compliance with 10 CFR Part 72 and sufficient information was provided for a safety evaluation and review. The evaluation of the operating procedure descriptions provided in the FSAR provided reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

10.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM EVALUATION

The objective of this review is to ensure that the applicant's FSAR includes the appropriate acceptance tests and maintenance programs for the system. The applicant revised FSAR Chapter 10, Acceptance Criteria and Maintenance Program, to address the allowance of different helium leak tests for Amendment Nos. 2 through 7 (proposed change #10), and the staff's evaluation is documented in Section 5.3 of this SER. The applicant also added a section on the thermal acceptance testing for HI-DRIP and DIJ supplemental cooling, and the staff's evaluation is documented in Section 4 of this SER. None of the other changes require an acceptance tests and maintenance program evaluation.

11.0 RADIATION PROTECTION EVALUATION

The objective of the radiation protection evaluation is to determine that the proposed changes comply with the applicable regulatory requirements for radiation protection and ensure that the proposed changes include reasonable consideration of, and facilitate licensees' compliance with, the requirements that licensees who use the storage system must meet. In reviewing these changes to the HI-STORM 100 system's shielding capability, the staff followed the guidance in Chapter 10b of NUREG-2215 (NRC, 2020a). This SER documents the basis for the staff's approval for the proposed changes with respect to radiation protection and meeting the applicable dose limits in 10 CFR Part 72. The applicant's requested changes and the staff's evaluation follows.

11.1 Proposed Change #1 – Addition of HI-TRAC MS

The applicant proposed a new transfer cask called the "HI-TRAC MS" which includes an option for a variable weight lead shield and water jacket and also adds cooling passages to the bottom lid. The HI-TRAC MS is for use with all previously approved MPCs as well as the MPC-32M, MPC-32 Version 1, and MPC-68 Version 1 that are part of this amendment request (See Proposed Changes #2 and #3 in Sections 11.2 and 11.3 of this SER).

11.1.1 Radiation Protection Design Features

The radiation protection features associated with the HI-TRAC MS are a combination of operational controls and physical features. The physical protection features of the HI-TRAC MS are the variable width lead gamma shield and the variable width water jacket. Other parts of the HI-TRAC MS will also provide shielding, however these two features are considered the main components that provide radiation protection.

The operational controls include a technical specification dose rate limit on the side of the HI-TRAC MS of 4 Rem/hr in CoC Appendix C, Section, 5.3.4.c.

11.1.2 Occupational Exposures

The staff verified that the applicant provided an estimate of doses to workers as a result of dry storage system operation. This includes loading of the HI-STORM 100S Version E using the HI-TRAC MS transfer cask. The estimated operational exposure for the operation is in FSAR Table 10.II.3.1. As discussed in FSAR Section 10.II.3, the applicant used the occupational dose estimates in FSAR Table 10.3.1b which are from loading of the 100-ton HI-TRAC using the MPC-24. The applicant adjusted those estimates to be appropriate for the HI-STORM 100 Version E, the HI-TRAC MS, and the MPC-32M with the bounding content. The dose rates for the HI-TRAC MS correspond to a transfer cask with dimensions adjusted to be consistent with the TS dose rate limit as discussed in Section 6.1.4.3.1 of this SER. The staff reviewed the applicant's method to adjust the dose rates in Appendix E of report HI-2188253 Revision 5 and found it acceptable.

Since there will be at least three operators, as indicated in FSAR Table 10.II.3, the staff finds that the estimated occupational exposure is below the dose limit in 10 CFR 20.1201(a), which provides the staff assurance that the design of the HI-TRAC MS and the operational restrictions meet ALARA objectives for direct radiation levels and meet 10 CFR 72.104(c) which, in turn, meets 10 CFR 72.236(d) in part. This is because the regulations in 10 CFR 72.236(d) require that the radiation shielding features meet the requirements in 10 CFR 72.104.

11.1.3 Exposures at or Beyond the Controlled Area Boundary

The applicant provided an estimate of exposures at or beyond the controlled area boundary due to operations associated with the HI-TRAC MS in FSAR Table 10.II.4.2. Based on the off-normal events considered in FSAR Section 11.II.1, the staff did not find that any of these events cause a reduction in shielding capability of the HI-TRAC MS. Some events, such as #6 "Off-Normal Handling of Heavy Loads" for the HI-TRAC MS may cause the HI-TRAC MS to be operated outdoors for additional lengths of time, which increases the exposure at or beyond the controlled area boundary. The applicant did not state the amount of time it assumed for the HI-TRAC MS operations under normal or off-normal conditions, however the dose rates at or beyond the controlled area boundary in FSAR Table 10.II.4.2 are low enough that the staff found that, with any reasonable amount of time, the HI-TRAC MS will be well within the limits of 10 CFR 72.104(a) for exposures at or beyond the controlled area boundary.

As discussed in Section 6.1.4.3.2 of this SER, the applicant calculated the accident condition dose for the HI-TRAC MS and showed the result in FSAR Table 5.II.1.4. The staff determined that this meets the accident condition dose limit in 10 CFR 72.106(b).

As stated in the above paragraphs, the staff found that the HI-TRAC MS meets the regulatory dose requirements beyond the controlled area boundary in 10 CFR 72.104(a) and 10 CFR 72.106(b) and therefore meets the requirement for the shielding being sufficient to meet these requirements. Therefore, the staff finds that it meets the regulation, in part, in 10 CFR 72.236(d).

11.1.4 As Low as Is Reasonably Achievable (ALARA)

The applicant stated that it considered ALARA for the HI-TRAC MS in the idea that it has variable shielding. This variable shielding means that the applicant can optimize the shielding to meet the needs of an individual site. The applicant has provided additional temporary shielding requirements for the HI-TRAC MS in FSAR Table 10.II.1.1. The staff reviewed the operations procedures in Section 8 of the FSAR and notes that the applicant points out parts of the loading procedures where higher dose rates are expected, and auxiliary shielding may need to be used. The staff finds that this demonstrates that ALARA was appropriately considered in the design and procedures for the HI-TRAC MS and meets the requirement in 10 CFR 72.236(d) which requires that the design meets 10 CFR 72.104(b) in that operational restrictions must be established to meet ALARA objectives.

11.1.5 Conclusion

Based on the above considerations, the staff finds that, with the addition of the HI-TRAC MS, the HI-STORM 100 dry storage system design continues to meet 10 CFR Part 72 regulations pertaining to radiation protection in 10 CFR 72.236(d) and that it is capable of being operated so that 10 CFR Part 20 will be met.

11.2 Proposed Change #2 – Inclusion of MPC-32M

The applicant requested the inclusion of the MPC-32M, a new basket for storing PWR SNF, in the HI-STORM 100 storage system. This basket is made entirely of Metamic^{TM}-HT. The applicant requested the MPC-32M be included in the approval of allowable canisters in the HI-STORM 100S Version E Overpack; all transfer casks previously approved for the HI-STORM 100, and the HI-TRAC MS which is also a part of this amendment. Exposures associated with the MPC-32M within the HI-TRAC MS and HI-STORM 100S Version E are discussed in Sections 11.1 and 11.4 of this SER, respectively. The applicant did not provide an analysis for the MPC-32M within transfer casks other than the HI-TRAC MS. The staff found the analysis for the MPC-32M within the HI-TRAC MS applicable for the other transfer casks based on the discussion in Section 6.2.4.2 of this SER.

Based on the above considerations, the staff finds that, with the addition of the MPC-32M, the HI-STORM 100 dry storage system design continues to meet 10 CFR Part 72 regulations pertaining to radiation protection in 10 CFR 72.236(d) and that it is capable of being operated so that 10 CFR Part 20 will be met.

11.3 Proposed Change # 3 – Inclusion of MPC-32 Version 1 and MPC-68 Version 1

The applicant requested the inclusion of the MPC-32 Version 1 and MPC-68 Version 1 in the HI-STORM 100 storage system. These are slightly modified versions of the previously approved MPC-32 (NRC, 2002) and MPC-68 (NRC, 2000) with larger cell openings to improve the ability to store DFCs. The applicant is seeking approval of the Version 1 baskets with the HI-STORM 100 overpack and the HI-STORM 100S Version E overpack, which is part of this amendment request. The allowable contents remain the same as for the previously approved MPC-32 and MPC-68. Based on the discussion in Section 6.3 of this SER, the staff finds that the addition of these canisters would not have a significant effect on the HI-STORM 100 System's ability to meet regulatory radiation protection requirements in 10 CFR 72.236(d) and that it is capable of being operated so that 10 CFR Part 20 will be met.

11.4 Proposed Change # 4 – Addition of HI-STORM 100S Version E Overpack

The applicant requested the inclusion of the HI-STORM 100S Version E overpack in the HI-STORM 100 storage system. The HI-STORM 100S Version E is a variant of the HI-STORM 100S Version B and is for use with all MPCs approved in the HI-STORM 100 system prior to this amendment's approval. The applicant also requested that the new MPC-32M, MPC-32 Version 1, and MPC-68 Version 1, introduced in this amendment, be used in the HI-STORM 100S Version E overpack.

11.4.1 Radiation Protection Features

The primary radiation protection feature of the HI-STORM 100S Version E is the roughly 30 inches of concrete surrounding the enclosure cavity on the side and the roughly 22 inches of concrete within the lid. The basket and enclosure vessel will also provide some shielding.

11.4.2 Exposures at or Beyond the Controlled Area Boundary

The applicant provided an estimate of exposures at or beyond the controlled area boundary under normal and accident conditions in FSAR Tables 5.II.4.5, 5.II.4.6, and 10.II.4.1. These tables show that the HI-STORM 100S Version E will meet the regulatory limits in 10 CFR 72.104(a) at 300 meters for a single cask and a 2x5 array at 500 meters as discussed in Section 6.4.4.2.1 of this SER. FSAR Section 11.II.2.3.3, as discussed in Section 6.4.4.2.2 of this SER, shows that the HI-STORM 100S Version E meets the regulatory limit in 10 CFR 72.106(b) for accident conditions. The HI-STORM 100S Version E is considered leak tight so the applicant did not consider any contribution to dose from effluents. The staff found that this demonstrates that the system meets the requirement that design has radiation shielding and confinement features sufficient to meet the requirements in 10 CFR 72.236(d).

11.4.3 Conclusion

Based on the above considerations, the staff found that with the addition of the HI-STORM 100S Version E overpack, the HI-STORM 100 System design continues to meet 10 CFR Part 72 regulations pertaining to radiation protection in 10 CFR 72.236(d).

11.5 Proposed Change #5 – Three Additional BWR Fuel Types: 10x10I, 10x10J, and 11x11A

The applicant requested three additional BWR fuel types: 10x10I, 10x10J, and 11x11A be found acceptable for use with the MPC-68M basket. As discussed in Section 6.5 of this SER, the staff did not find that this change affected the systems' ability to meet regulatory dose limits. Therefore, this section need not evaluate this proposed change.

11.6 Proposed Change #6 – Lower Ambient Temperature

The applicant proposed to lower the ambient temperature to 70°F for HI-STORM 100S Version E. This does not affect the cask system's shielding or ability to meet regulatory dose limits. Therefore, this section need not evaluate this proposed change.

11.7 Proposed Changes #7 & #12 – Addition of HI-DRIP and DIJ Ancillary System

The applicant proposed adding HI-DRIP and a DIJ ancillary system that would prevent the water with the MPC from boiling during loading/unloading operations while the loaded MPC is in the HI-TRAC transfer cask to the HI-STORM 100 storage system. Although water in the MPC is important for shielding the contents during loading operations, the staff's assessment on the HI-DRIP and DIJ ancillary system's ability to keep the water from boiling is not discussed here as the system itself does not serve a shielding function.

11.8 Proposed Change #8 – Partial Gadolinium Credit for BWR Fuel Assemblies

The applicant requested partial gadolinium credit for some BWR fuel assemblies within the MPC-68M. This change is requested to demonstrate margin for criticality and does not affect the shielding design of the package and does not affect the package's ability to meet regulatory dose limits. Therefore, this section need not evaluate this proposed change.

11.9 Proposed Change #9

This proposed change was removed.

11.10 Proposed Change #10 – Different Helium Leak Test Requirements for Earlier Amendments

The applicant requested allowance for canisters loaded under earlier amendments to have different helium leak test requirements. This does not affect the cask system's shielding or ability to meet regulatory dose limits. Therefore, this section need not evaluate this proposed change.

11.11 Proposed Change #11 – Remove Dimensions from MPC-68M Drawing

The applicant proposed updating Drawing No. 7195 for the MPC-68M to remove dimensions on the drawing that are not used in the safety analysis.

As discussed in Section 6.11 of this SER, the staff found that this change does not have a significant effect on dose and dose rates and therefore would not significantly impact the system's ability to meet regulatory dose limits. Therefore, this section need not evaluate this proposed change.

11.12 Evaluation Findings

F11.1 The HI-STORM 100 System Amendment No. 15 provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106, in accordance with 10 CFR 72.236(d).

F11.2 The design and operating procedures of the HI-STORM 100 System Amendment No. 15 provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the ALARA objective with respect to exposures, consistent with 10 CFR 20.1101(b).

The staff concludes that the design of the radiation protection system of the HI-STORM 100 System Amendment No. 15 is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM 100 System Amendment No. 15 will allow safe storage of proposed new SNF contents. The staff reached this finding primarily on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted health physics practices, and the statements and representations in the application.

12.0 ACCIDENT ANALYSES EVALUATION

The staff's evaluations of the applicant's accident analysis for the proposed changes are documented in Sections 3 through 8 of this SER. Therefore, this section need not provide further evaluation.

13.0 TECHNICAL SPECIFICATIONS AND OPERATING CONTROL AND LIMITS EVALUATION

The staff reviewed the proposed amendment to determine that applicable changes made to the conditions in the CoC and TS would be in accordance with the requirements of 10 CFR Part 72. The staff reviewed the proposed changes to the TS to confirm the changes were properly evaluated and supported in the applicant's revised safety analysis report.

Table 13-1 lists the applicant's proposed changes to the CoC and TS:

Table 13-1 – Conforming Changes to the Technical Specifications and Operating Control and Limits

Page Number	Reference	Description	Proposed Change
CoC, Page 1	Condition	Clarify the applicability of Appendices A, B, A-100U, and B-100U, as well as the new Appendices C and D.	1, 2, 3, 4
CoC, Page 2	Condition 1b.	Add MPC-32M, MPC-32 Version 1, and MPC-68 Version to the list of MPCs.	2, 3

Page Number	Reference	Description	Proposed Change
CoC, Page 2	Condition 1b.	Above ground systems description: Simplify the overpack description and add HI-STORM 100S Version E.	4
CoC, Page 2	Condition 3	Revise Condition 3 to move operational details to Appendix A, Section 5.8 and Appendix C, Section 5.4.	10
CoC, Page 3	Conditions 5 through 8	Add references to Appendices C or D, as applicable.	1, 4
CoC, Page 3	Condition 9b.	Add MPC-32M.	2
CoC, Page 5	Condition 12	Correct the numbering for this condition from 13 to 12. Add references to Appendices C and D.	1, 4
CoC, Page 5	Attachments	Add references to Appendices C and D.	1, 4
Appendix A	Table of Content	Updated.	N/A
Appendix A, Page 1.1-5	Definition	Add HI-TRAC MS to the definition of transfer cask.	1
Appendix A, Page 3.1.1-1	LCO 3.1.1	Add Note 2 to address casks initially loaded under Amendment Nos. 2 through 7.	10
Appendix A, Page 5.0-8	Section 5.8	Add Section 5.8 to incorporate operational details from CoC Condition 3 and address the helium leak test for casks initially loaded under Amendment Nos. 2 through 7.	10
Appendix B	Table of Content	Updated.	N/A
Appendix B Pages 2-45 to 2-47	Table 2.1-3	Add three BWR fuel types (10x10I, 10x10J, and 11x11A) to the approved content for MP-68M, add partial gadolinium credit for BWR 10x10 and 11x11 assembly classes in MPC-68M, and add associated notes.	5, 8
Appendix B Page 2-49	Table 2.1-9	Add the table on the restrictions for partial gadolinium credit in MPC-68M.	8
Appendix B Page 2-50	Section 2.4	Update section title.	N/A
Appendix B Page 3-16	Section 3.4, subsection 8	Add additional information on ambient temperature limits.	N/A
Appendix B Page 3-20	Section 3.6	Remove “mandatory” for clarification.	N/A
Appendix B Page 3-25	Section 3.9	Clarify the environmental temperature requirements.	N/A

Page Number	Reference	Description	Proposed Change
Appendix C		Add Appendix C to provide TS for the HI-STORM 100S Version E cask.	4
Appendix D		Add Appendix D to provide approved contents and design features for the HI-STORM 100S Version E cask and HI-TRAC MS.	1, 4

The staff finds that the proposed changes to the TS for the HI-STORM 100 system Amendment No. 15 conform to the changes requested in the amendment application and do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. The proposed changes provide reasonable assurance that the HI-STORM 100 system Amendment No. 15 will continue to allow safe storage of SNF.

14.0 QUALITY ASSURANCE EVALUATION

The applicant did not propose any changes that affect the staff's quality assurance evaluation provided in the previous SERs for CoC No. 1014, Amendment Nos. 0 through 14. Therefore, the staff determined that a new evaluation was not required.

15.0 CONCLUSIONS

The staff has performed a comprehensive review of the amendment application, during which the following requested changes to the HI-STORM 100 System were considered:

1. Add a new version of a transfer cask, HI-TRAC MS (maximum shielded), which includes an option for variable weight of the lead and water jacket and cooling passages to the bottom lid. HI-TRAC MS is to be used with all MPCs approved for use in Amendment Nos. 0 through 14 to the HI-STORM 100 System and the newly proposed MPC-32M, MPC-32 Version 1, and MPC-68 Version 1.
2. Include MPC-32M for storage in the HI-STORM 100 System.
3. Include MPC-32 Version 1 and MPC-68 Version 1 for storage in the HI-STORM 100 System.
4. Add the new overpack, HI-STORM 100S Version E (hereafter Version E). The requested permissible MPCs include all MPCs approved for use in Amendment Nos. 0 through 14 to the HI-STORM 100 System and the newly proposed MPC-32M, MPC-32 Version 1, and MPC-68 Version 1 are to be stored in Version E.
5. Add three additional boiling water reactor (BWR) fuel types to the approved content for MPC-68M: 10x10I, 10x10J, and 11x11A.
6. Lower the allowed ambient temperature from 80°F to 70°F for Version E.
7. Add HI-DRIP and dry ice jacket (DIJ) ancillary system as additional cooling when the MPC is loaded into the HI-TRAC transfer cask.
8. Allow for partial gadolinium credit for BWR fuel assemblies types 10x10 and 11x11 assembly classes in MPC-68M.
10. Include allowance for canisters currently loaded under earlier amendments which had different helium leak test requirements.
11. Update Drawing No. 7195 for the MPC-68M by removing dimensions which are not used in the safety analysis.
12. Include DIJ as an optional alternate cooling method for short-term operation of the loaded HI-TRAC.

Based on the statements and representations provided by the applicant in its amendment application, as supplemented, the changes described above to the HI-STORM 100 MPC Storage System in Amendment No. 15 do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. As noted earlier, the staff did not review proposed change #9 in this amendment. Amendment No. 15 for the HI-STORM 100 MPC Storage System should be approved.

Issued with Certificate of Compliance No. 1014, Amendment No. 15
on May 11, 2021.

References

EN, 2010. Energy Northwest letter to NRC, “Columbia Generating Station, Docket No. 50-397, Independent Spent Fuel Storage Installation, Docket No. 72-35, Actions Regarding NRC Enforcement Discretion Letter No. EA-09-190,” October 27, 2010. ADAMS Accession No. ML103070299.

Entergy, 2010. Entergy Operations, Inc. letter to NRC, “Follow-up Action to NRC Enforcement Discretion Letter EA-09-190, Holtec Elimination of Multi-Purpose Canister (MPC) Shop Helium Leak Rate Test,” November 2, 2010. ADAMS Accession No. ML103090653.

Exelon, 2010. Exelon Generation Company, LLC letter to NRC, “Response to NBC's Request for Disposition of Non-Helium Leak Tested Holtec Multi-Purpose Canisters (MPCs) as Discussed in EA-09-190, "Exercise of Enforcement Discretion - Holtec International,” February 4, 2010. ADAMS Accession No. ML100350810.

Holtec, 2009a. Amendment No. 7 to Certificate of Compliance No. 1014 for the Holtec International HI-STORM 100 Cask System. December 28, 2009. ADAMS Accession No. ML093620057.

Holtec, 2009b. Certificate of Compliance No. 1014, Appendix A, “Technical Specifications for the HI-STORM 100 Cask System,” Amendment No. 7. December 28, 2009. ADAMS Accession No. ML093620062.

Holtec, 2009c. Holtec International letter, “Response to EA-09-190.” September 2, 2009. ADAMS Accession No. ML092470363.

Holtec 2012a. Amendment No. 8 to Certificate of Compliance No. 1014 for the Holtec International HI-STORM 100 Cask System. May 10, 2012. ADAMS Accession No. ML12132A038.

Holtec, 2012b. Certificate of Compliance No. 1014, Appendix A, “Technical Specifications for the HI-STORM 100 Cask System,” Amendment No. 8. May 10, 2012. ADAMS Accession No. ML12132A060.

Holtec, 2017. “Final Safety Analysis Report on the HI-STORM FW MPC Storage System,” Revision 5. June 20, 2017. ADAMS Accession No. ML17179A444.

Holtec, 2019a. Letter from Holtec International to NRC, “HI-STORM 100 Multipurpose Canister Storage System Amendment 15 Request.” March 20, 2019. This package contains 27 attachments, and Attachments 7, 9, and 11 through 26 are *Proprietary Information and Not Publicly Available*. Agencywide Document Access and Management System (ADAMS) Accession No. ML19092A192.

Holtec, 2019b. Letter from Holtec International to NRC, “HI-STORM 100 Amendment 15 Response to Request for Supplemental Information.” September 16, 2019. This package contains 14 attachments, and Attachments 7, and 9 through 13 are *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML19277G818.

Holtec, 2019c. “Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System,” Revision 18. May 16, 2019. ADAMS Accession No. ML19150A405.

Holtec, 2020a. Letter from Holtec International to NRC, “Submittal of Responses to NRC’s 1st Round RAIs for HI-STORM 100 Amendment Number 15.” April 28, 2020. This package contains 13 attachments, and Attachments 1, 8, and 9 are *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML20128J292.

Holtec, 2020b. Letter from Holtec International to NRC, “Submittal of Responses to NRC’s RAIs 3-1 and 3-6 for HI-STORM 100 Amendment Number 15.” May 15, 2020. This package contains 4 attachments, and Attachment 3 is *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML20136A475.

Holtec, 2020c. Letter from Holtec International to NRC, “Submittal of Supplemental Information to RAI 8-1 response for HI-STORM 100 Amendment Number 15.” June 12, 2020. This package contains 3 attachments, and Attachment 1 is *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML20164A294.

Holtec, 2020d. Letter from Holtec International to NRC, “Submittal of Response to RAI 4-9 for HI-STORM 100 Amendment Number 15.” June 22, 2020. This package contains 6 attachments, and Attachments 1, 4, and 5 are *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML20174A397.

Holtec, 2020e. Letter from Holtec International to NRC, “Submittal of Supplemental Information to response to RAIs 6-1, 6-2, 6-7, 6-8 and 11-1 for HI-STORM 100 Amendment Number 15.” July 30, 2020. This package contains 8 attachments, and Attachments 1, 4, 6, and 7 are *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML20213C679.

Holtec, 2020f. Letter from Holtec International to NRC, “Submittal of Additional Responses to Amendment 15 RAIs.” August 14, 2020. This package contains 7 attachments, and Attachments 3, 4, 5, and 6 are *Proprietary Information and Not Publicly Available*. ADAMS Accession No. ML20229A001.

Holtec, 2020g. Letter from Holtec International to NRC, “Submittal of Supplemental Information on HI-STORM 100 Amendment 15.” September 1, 2020. ADAMS Accession No. ML20245E463.

Holtec, 2020h. Letter from Holtec International to NRC, “Submittal of Supplemental Information on HI-STORM 100 Amendment 15.” This package contains 4 attachments, and Attachments 1 and 3 are *Proprietary Information and Not Publicly Available*. September 25, 2020. ADAMS Accession No. ML20269A425.

ISG-18, “The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage,” Revision 1. October 3, 2008. ADAMS Accession No. ML082750469.

McConn, 2011. McConn, R., C.J. Gesh, R.T. Pagh, R.A. Rucker, and R. Williams III, “Compendium of Material Composition Data for Radiation Transport Modeling,” PNNL-15870, Revision 1, March 4, 2011.

NRC, 1980. “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” NUREG-0612, U.S. NRC, July 1980. ADAMS Accession No. ML070250180.

NRC, 1989. “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask,” Regulatory Guide 3.61, U.S. NRC, February 1989. ADAMS Accession No. ML003739511.

NRC, 1998. “Engineering Drawings for 10 CFR Part 71 Package Approvals,” NUREG/CR-5502, U.S. NRC, May 1998. ADAMS Accession No. ML20248L098.

NRC, 2000. “Safety Evaluation Report for Holtec International HI-STORM 100 Cask System,” U.S. NRC, May 4, 2000. ADAMS Accession No. ML003711865.

NRC, 2001. “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks,” NUREG/CR-6716, U.S. NRC, March 2001. ADAMS Accession No. ML010820352.

NRC, 2002. “Holtec International HI-STORM 100 Cask System Amendment No. 1, Safety Evaluation Report,” U.S. NRC, July 18, 2002. ADAMS Accession No. ML022000249.

NRC, 2007. “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” NUREG-1864, U.S. NRC, March 2007. ADAMS Accession No. ML071340012.

NRC, 2009a. NRC conversation record, “Follow-up Actions Regarding NRC Enforcement Discretion Letter No. EA-09-190.” December 1, 2009. ADAMS Accession No. ML093510008.

NRC, 2009b. NRC EA-09-190, “Exercise of Enforcement Discretion – Holtec International, Enforcement Discretion Letter.” August 5, 2009. ADAMS Accession No. ML092180140.

NRC, 2010a. “Standard Review Plan for Spent Fuel Dry Cask Storage Systems at a General License Facility,” NUREG-1536, Revision 1, U.S. NRC, July 2010. ADAMS Accession No. ML101040620.

NRC, 2010b. NRC letter, “Response to Holtec International (Holtec) Reply to EA-09-190.” January 12, 2010. ADAMS Accession No. ML100140070.

NRC, 2010c. NRC letter to Southern Nuclear Operating Company, Inc., “Evaluation of Non-Helium Leak Rate Test of Holtec Multi-Purpose Canisters (MPCs),” April 15, 2010. ADAMS Accession No. ML101060436.

NRC, 2010d. NRC letter to Exelon Generation Company, LLC, ““Evaluation of Non-Helium Leak Rate Test of Holtec Multi-Purpose Canisters (MPCs),” June 7, 2010. ADAMS Accession No. ML101600314.

NRC, 2010e. NRC letter to Tennessee Valley Authority, “Evaluation of Non-Helium Leak Rate Test of Holtec Multi-Purpose Canisters (MPCs),” September 1, 2010. ADAMS Accession No. ML102450015.

NRC, 2011a. NRC letter to PSEG Nuclear LLC, “Evaluation of Non-Helium Leak Rate Test of Holtec Multi-Purpose Canisters (MPCs),” July 7, 2011. ADAMS Accession No. ML111880156.

NRC, 2011b. NRC letter to Energy Northwest, “Evaluation of Non-Helium Leak Rate Test of Holtec Multi-Purpose Canisters (MPCs),” July 7, 2011. ADAMS Accession No. ML111880244.

NRC, 2011c. NRC letter to Entergy Operations, Inc., “Evaluation of Non-Helium Leak Rate Test of Holtec Multi-Purpose Canisters (MPCs),” January 26, 2011. ADAMS Accession No. ML110270139.

NRC, 2011d. Safety Evaluation Report, Docket No. 72-1014, Holtec International HI-STORM 100 Cask System, Certificate of Compliance No. 1014, Amendment No. 8. August 21, 2011. ADAMS Accession No. ML12213A203.

NRC, 2018. Memorandum from P. Saverot to Y. Diaz-Sanabria, “Summary of November 13, 2018, Meeting with Holtec International,” November 27, 2018. ADAMS Accession No. ML18331A184.

NRC, 2019a. Final Safety Evaluation Report, Docket No. 72-1014, Holtec International HI-STORM 100 Multipurpose Canister Storage System, Certificate of Compliance No. 1014, Amendment No. 14. November 29, 2019. ADAMS Accession No. ML19295C576

NRC, 2019b. Memorandum from M. Dapas to NMSS Staff, “Key Principles for Nuclear Material Safety and Safeguards Reviews,” January 15, 2019. ADAMS Accession No. ML19015A290.

NRC, 2020a. “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities,” NUREG-2215, Final Report, U.S. NRC, April 2020. ADAMS Accession No. ML20121A190.

NRC, 2020b. Final Safety Evaluation Report, Docket No. 72-1032, Holtec International HI-STORM Flood/Wind Multipurpose Canister Storage System, Certificate of Compliance No. 1032, Amendment No. 5. June 25, 2020. ADAMS Accession No. ML20163A706.

PSEG, 2010. PSEG Nuclear LLC letter to NRC, “Response to NRC's Request for Disposition of Non-Helium Leak Tested Holtec Multi-Purpose Canisters,” May 25, 2010. ADAMS Accession No. ML101550039.

SNC, 2010. Southern Nuclear Operating Company, Inc. letter to NRC, “Edwin I. Hatch Nuclear Plant, Joseph M. Farley Nuclear Plant, Follow-up Actions – NRC Enforcement Discretion Letter EA-09-190, Holtec Elimination of MPC Shop Helium Leak Rate Test.” January 25, 2010. ADAMS Accession No. ML100250991.

TVA, 2010. Tennessee Valley Authority letter to NRC, “Tennessee Valley Authority - Response to Request for Information Regarding NRC Enforcement Discretion Letter No. EA-09-190,” May 12, 2010. ADAMS Accession No. ML101340622.