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PRELIMINARY 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

SUMMARY

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (staff) review and evaluation of the amendment request to amend Certificate of Compliance (CoC) No. 1032 for the HI-STORM Flood/Wind (FW) Multipurpose Canister (MPC) Storage System (hereafter HI-STORM FW system) submitted by Holtec International (Holtec) by letter dated June 15, 2018 (Holtec, 2018a), and supplemented on September 20, 2018 (Holtec, 2018b), April 1, 2019 (Holtec, 2019a), April 30, 2019 (Holtec, 2019b), June 14, 2019 (Holtec, 2019c), October 4, 2019 (Holtec, 2019d), October 21, 2019 (Holtec, 2019e), and December 18, 2019 (Holtec, 2019f). Holtec proposed the following changes:

1. (a) Add new heat load patterns for the MPC-89 and MPC-37 (long, standard, and short length). (b) Revise the minimum required cooling time for fuel to 1 year for MPC-89 and MPC-37.
2. Add four new fuel types, 10x10I, 11x11A, 7x7C, and 8x8G, to the approved contents listed in CoC Appendix B.
3. Allow an exception to the American Society of Mechanical Engineers (ASME) Code to use certain duplex stainless steels in the HI-STORM FW system.
4. Use FLUENT to revise the calculation for evaluating effective fuel conductivities.
5. Add the use of damaged fuel isolator (DFI) in CoC Appendix A.
6. Add two versions of the standard HI-TRAC VW: Version V has a natural circulation feature, and Version V2 has the option for removable neutron shield.
7. Add the option of using cyclic vacuum drying for all MPCs.
8. (a) Add fuel assemblies containing blended low enriched uranium (BLEU) as approved contents. (b) Add the definition for BLEU fuel assemblies to final safety analysis report (FSAR) Glossary Section and the definition section in the CoC. (c) Add the required shielding evaluation to FSAR Section 5.4.8 for storing BLEU fuel assemblies in HI-STORM FW system.

Holtec also proposed the following clarifications and editorial and minor changes:

E1. Modify the definition of repaired/reconstituted fuel assembly in CoC Appendix A to clarify that when dummy stainless steel rods are present in the loaded spent fuel

assemblies, the dummy/replacement rods will be considered in the site-specific dose calculations.

E2. Add hafnium rods in CoC Appendix B, Table 2.1-1 and clarify that control rod assemblies (CRAs) are not limited to those with hafnium.

E3. Add the definition of DFI in CoC Appendix A and FSAR.

E4. Allow minor deviation from the prescribed loading pattern to CoC Appendix B, Section 2.3 to allow one slightly thermally-discrepant fuel assembly per quadrant to be loaded, as long as the peak cladding temperature is below the limit in SFST-Interim Staff Guidance (ISG)-11, Revision 3.

E5. Correct typographical error in CoC Appendix B, Table 2.1-2 under the 16x16C fuel class to correct the number of fuel rod locations to 235.

E6. Correct typographical error in CoC Appendix A, Table 3-1 to bring into agreement with FSAR Table 4.5.19.

E7. Clarify CoC Appendix B, Section 2.3.1 that vacuum drying system (VDS) is permitted for high burnup fuel with drying time limits as provided in CoC Appendix A, Table 3-1.

The staff did not evaluate the proposed change identified in the June 15, 2018, amendment request to add an alternative vent and drain port cover configuration using dual closures and to remove the leak testing requirement for the alternate vent and drain port cover configuration because Holtec removed this proposed change from Amendment No. 5 in a letter dated September 20, 2018 (Holtec, 2018b). The staff also did not review the changes that were described in the June 15, 2018, amendment request as changes made in accordance with 10 CFR 72.48.

This revised CoC, when codified through rulemaking, will be denoted as Amendment No. 5 to CoC No. 1032.

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 Dry Cask Storage Systems at a General License Facility," July 2010 (NRC, 2010). 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 nor previous amendments to the CoC.

1.0 GENERAL INFORMATION EVALUATION

The purpose of the review is to ensure that the applicant has provided in its documentation for the spent fuel storage system a non-proprietary description, or overview, that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

In FSAR Chapter 1, "General Description," the applicant provided the description of HI-TRAC VW transfer cask; cask content for DFI, MPC-37, and MPC-89; new loading patterns for MPC-37 and MPC-89; and Alloy X. Staff determined that the proposed description in general information is adequate to allow staff's detailed evaluation as documented in other sections of this SER.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

The objective of evaluating the principal design criteria related to structures, systems, and components (SSCs) important to safety is 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 new loading patterns, provide DFI design features, and update Table 2.1.10 for burnup and cooling time fuel qualification requirement. The staff's evaluation of the principal design criteria for new loading patterns is documented in Sections 4, 6, and 8 of this SER and, as described in those sections, the staff determines the three new loading patterns acceptable. The staff discussed the principal design criteria of DFI in Sections 3, 4, 7, and 8 of this SER, and, as described in those sections, determines the proposed DFI is acceptable for use with the proposed content.

3.0 STRUCTURAL EVALUATION

The staff reviewed the proposed changes to verify that the applicant has performed adequate structural evaluation 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 seeks reasonable assurance that the system will maintain confinement, subcriticality, radiation shielding, and retrievability or recovery of the fuel, as applicable, under all credible loads of normal and off-normal conditions, accident conditions, and natural phenomenon events.

The following proposed changes are applicable to the structural evaluation:

- Proposed Change #2: Add four new fuel types, 10x10I, 11x11A, 7x7C and 8x8G to the approved contents in CoC Appendix B.
- Proposed Change #5: Add DFI to CoC Appendix A.
- Proposed Change #6: Add two versions of the HI-TRAC VW, Versions V and V2.

3.1 Addition of Four New Fuel Types, 10x10I, 11x11A, 7x7C, and 8x8G to the Approved Contents In CoC Appendix B

The staff reviewed the structural designs of the fuel assemblies and found that the total weight of the storage system with the new fuel types is bounded by the maximum allowable weight of the storage system previously reviewed and approved by the NRC in the original certificate and subsequent amendments, and the weight of each new fuel assembly is bounded by the fuel assembly limits provided in Table 2.1-1 of the FSAR, Revision 6 (Holtec, 2019g). In addition, there is no change in the center of gravity of the fuel assembly with the new fuel types because there are no changes in length, width, and height limits of the fuel assembly presented in Appendix B, Table 2.1-1. Because Amendment No. 5 presents no changes to the structural designs and center of gravity of the fuel assembly, and the total weight of the storage system with the new fuel types is bounded by the previously approved maximum allowable weight of the storage system; the staff determines that the addition of the four new fuel types as the approved contents in CoC No. 1032, Appendix B is acceptable.

3.2 Addition of DFI to CoC Appendix A

The staff reviewed the structural designs of the DFI and found that the structural design of the DFI were previously reviewed and accepted by the staff for use in HI-STORM 100, CoC

No. 1014, Amendment No. 14 (NRC, 2019a). The staff concluded that the applicant's description of the DFI design and design basis was sufficient to demonstrate that the DFIs constrained fissile material during credible normal operations, off-normal operations, accident conditions, and natural phenomena events. Since (i) there are no design changes in the DFI, (ii) the total weight of the storage system with the DFI is bounded by the maximum allowable weight of the storage system, (iii) there is no change in the center of gravity, and (iv) the DFI inclusion does not result in increase of temperatures, pressures, and weights beyond those used in the previous design basis structural calculations, the staff determines that the addition of the DFI to CoC No. 1032, Appendix A is acceptable.

3.3 Addition of Two Versions of the HI-TRAC VW, Versions V and V2

The HI-TRAC VW is a transfer cask that provides a missile and radiation barrier during transport of the MPC from the fuel pool to the HI-STORM FW system overpack. In CoC No. 1032, Amendment No. 0, the staff reviewed and accepted the structural design bases, acceptance criteria, and design and analysis of the HI-TRAC VW under normal and off-normal operations, accident conditions, and natural phenomena events. In Amendment No. 5, the applicant proposes to use two new versions of the HI-TRAC VW, Versions V and V2. The HI-TRAC VW Versions V and V2 are similar structures to the HI-TRAC VW except that the Version V adds a water jacket to provide neutron shielding with a natural ventilation feature, and the Version V2 removes the water jacket and uses a neutron shield cylinder (NSC) for neutron shielding. The applicant indicated that the Versions V and V2 are designed to ensure that there is no change in the structural capacity of the HI-TRAC VW.

The applicant performed structural analyses for the HI-TRAC VW Versions V and V2 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 VW Versions V and V2 are identical to the ones previously used, and found acceptable by the staff, for the analysis of the HI-TRAC VW.

3.3.1 Stress Analyses for HI-TRAC VW Versions V and V2 under Heavy Lifting Operations

The applicant performed stress analyses of the HI-TRAC VW Versions V and V2 for lifting operations. As the applicant previously evaluated the HI-TRAC VW, it evaluated all structural members of the HI-TRAC VW Versions V and V2 in the load path of the maximum lifted weight. The following stresses were calculated:

- 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 used the same analytical approach, which was previously reviewed and accepted by the staff for the HI-TRAC VW, for the stress analyses of the HI-TRAC VW Versions V and V2. FSAR Tables 3.4.2, 3.4.2A, and 3.4.2B summarize the stress analyses results for the HI-TRAC VW, HI-TRAC VW Versions V and V2, respectively, under the maximum lifted load.

The staff reviewed the stress analyses and found that the results in FSAR Table 3.4.2 for the shear stress in the welds between the top flange and the inner and outer shells, and the primary membrane stress in the inner and outer shells bound the shear stresses of the HI-TRAC VW

Versions V and V2. In addition, as shown in FSAR Tables 3.4.2, 3.4.2A, and 3.4.2B, because the calculated stresses were compared with the allowable stresses and the calculated factors of safety are greater than 1.0, the staff determines that the analyses regarding the HI-TRAC VW Versions V and V2 under heavy lifting operations are acceptable.

3.3.2 Overturning Analyses for HI-TRAC VW Versions V and V2 under Large Missile Impact and Tornado Wind

The applicant performed overturning analyses of the HI-TRAC VW Versions V and V2 under large missile impact and tornado wind. The applicant used the same analytical approach, which was previously reviewed and accepted by the staff for the HI-TRAC VW in the initial application, for the overturning analyses of the HI-TRAC VW Versions V and V2. FSAR Tables 3.4.5, 3.4.5A, and 3.4.5B summarize the overturning analysis results for the HI-TRAC VW, HI-TRAC VW Versions V and V2, respectively, under the large missile impact and tornado wind.

The staff reviewed the overturning analyses and found that the results in FSAR Tables 3.4.5, 3.4.5A, and 3.4.5B show that the HI-TRAC VW Versions V and V2 remain in a vertical upright position (i.e., no overturning) in the aftermath of large missile impact and tornado wind. Because the calculated rotations were compared with the allowable rotations, and the calculated factors of safety were greater than 1.0, the staff determines that the designs and analyses of the HI-TRAC VW Versions V and V2 under large missile impact and tornado wind are acceptable.

3.3.3 Sliding Analyses for HI-TRAC VW Versions V and V2 under Large Missile Impact and Tornado Wind

The applicant performed sliding analyses of the HI-TRAC VW Versions V and V2 under large missile impact and tornado wind using the same analytical approach used for the HI-TRAC VW. FSAR Table 3.4.16 summarizes the sliding displacements of 1.133 ft, 1.193 ft, and 0.958 ft for the HI-TRAC VW, HI-TRAC VW Versions V and V2, respectively, under the large missile impact and tornado wind. Because the calculated sliding displacements of the HI-TRAC VW Versions V and V2 are about the same or less than the sliding displacement of the HI-TRAC VW, the staff determines that the HI-TRAC VW Versions V and V2 are stable and acceptable with respect to sliding under the large missile impact and tornado wind.

3.3.4 Penetration Analyses for HI-TRAC VW Versions V and V2 under Small and Intermediate Missiles

The applicant performed penetration analyses of the HI-TRAC VW Versions V and V2 under the small and intermediate missiles impact to determine the extent to which they will penetrate the HI-TRAC VW Versions V and V2 and cause potential damage to the MPC enclosure vessel. The same analytical approach used for the HI-TRAC VW for the penetration analysis was adopted for the penetration analysis of the HI-TRAC VW Versions V and V2.

The analysis results were documented in FSAR Tables 3.4.6A and 3.4.6B and show that the depth of penetration of the small missile is less than the thinnest section of material on the exterior surface of the HI-TRAC VW Versions V and V2. 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 HI-TRAC VW Versions V and V2 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 HI-

TRAC VW Versions V and V2 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 conclusion of the analytical results is identical to those for impacts from small and intermediate missiles on HI-TRAC VW. Based on the analytical results, the staff determined that the penetration analyses for HI-TRAC VW Versions V and V2 due to impacts from small and intermediate missiles are acceptable.

3.3.5 Stress Analyses for Non-Mechanistic Heat-Up of the HI-TRAC VW Water Jacket

The applicant performed stress analyses of the HI-TRAC VW Versions V and V2 to demonstrate that the stresses in the water jacket and its welds shall be below the limits set by the ASME Boiler and Pressure Vessel (B&PV) Code, Section II, Class 3 for the Level D service condition (ASME, 2007). The same analytical approach used for the HI-TRAC VW for the water jacket stress analysis was adopted for the stress analysis of the HI-TRAC VW Versions V and V2. The accident pressure inside the water jacket is given in Table 2.2.1 of the FSAR, Revision 6 (Holtec, 2019g), which was reviewed and accepted by the staff in the CoC No. 1032, Amendment No. 0.

FSAR Table 3.4.9A summarizes the stress analysis results for the various HI-TRAC VW Version V water jacket components including the connecting welds. For HI-TRAC VW Version V2 neutron shield cylinder, the same approach used to evaluate HI-TRAC VW water jacket is used because neutron shield cylinder serves the same function as that of water jacket and its configuration is similar to the water jacket. FSAR Table 3.4.9B summarizes the stress analysis results for the various HI-TRAC VW Version V2 neutron shield cylinder components including the connecting welds.

The staff reviewed the stress analyses and found that the calculated stresses in FSAR Tables 3.4.9, 3.4.9A, and 3.4.9B are less than the allowable stresses with the large factors of safety in a range of 6.30 to 13.66, and, therefore, the staff determines that the designs and analyses of the water jackets of the HI-TRAC VW Versions V and V2 are acceptable.

3.4 Evaluation Findings

F3.1 The SAR adequately describes structures, SSCs (i.e., HI-TRAC VW Versions V and V2) that are important to safety and provides drawings and text in sufficient detail to allow evaluation of the structural effectiveness.

F3.2 The staff determines the addition of the DFI is acceptable because the design of the DFI, which was previously reviewed and accepted by the NRC in HI-STORM 100 Amendment No. 14, meets the requirements in 10 CFR 72.124(a), 72.124(b), 72.236(b), 72.236(c) and 72.236(l).

F3.3 The applicant has met the requirements of 10 CFR 72.236(b). The SSCs that are important to safety are designed to accommodate all credible loads of normal and off-normal conditions, accident conditions, and natural phenomenon events with an adequate margin of safety and are found to be within limits of applicable codes, standards, and specifications. The staff has reasonable assurance that the addition of the HI-TRAC VW Versions V and V2 in HI-STORM FW system is acceptable.

F3.4 The applicant has met the requirements of 10 CFR 72.236(l) that the design analysis and bases used for evaluation demonstrate that the casks and other systems important

to safety will reasonably maintain confinement of radioactive material under all credible loads of normal and off-normal operational conditions, accident conditions, and natural phenomenon events.

Based on the review of the applicant's description, design criteria, appropriate use of material properties and adequate structural analyses of the relevant SSCs, the staff concludes that the SSCs that are important to safety of the HI-STORM FW system are in compliance with 10 CFR Part 72 regulations.

4.0 THERMAL EVALUATION

The thermal review ensures that the cask components and fuel material temperatures will remain within the allowable values under normal, off-normal, and accident conditions. This review includes confirmation that the fuel clad temperatures for fuel assemblies stored in the HI-STORM FW system will be maintained below specified limits throughout the storage period in order 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.

This review was conducted under the regulations described in 10 CFR 72.236, which identify the specific requirements for the regulatory approval, fabrication, and operation of spent fuel storage cask designs. The unique characteristics of the spent fuel to be stored in the HI-STORM FW system are identified, as required by 10 CFR 72.236(a), so that the design basis and the design criteria that must be provided for the SSCs important to safety can be assessed under the requirements of 10 CFR 72.236(b).

The staff also reviewed the application to determine whether the HI-STORM FW system design fulfills the acceptance criteria listed in Chapters 2, 4, and 12 of NUREG-1536, Revision 1 (NRC, 2010), as well as applicable ISG documents.

The following proposed changes are applicable to the thermal evaluation:

- Proposed Change #1(a): Add new heat load patterns for the MPC-89 and MPC-37 (long, standard, and short length).
- Proposed Change #4: Use ANSYS FLUENT® analysis model to revise the calculation for evaluating effective fuel conductivities.
- Proposed Change #6: Add two versions of the standard HI-TRAC VW transfer cask. Version V adds a natural circulation feature (FSAR Section 1.5), while Version V2 adds the option for a removable neutron shield (FSAR Figure 9.2.7B).
- Proposed Change #7: Add the option for cyclic vacuum drying for all MPCs.

4.1 HI-STORM FW System Thermal Model

The applicant used the ANSYS FLUENT® computer-based analysis program to evaluate the thermal performance of the HI-STORM FW system. ANSYS FLUENT® is a finite volume computational fluid dynamics (CFD) program with capabilities to predict fluid flow and heat transfer phenomena in two and three dimensions. FSAR Section 4.5.2.1 provides a general description of the HI-TRAC VW Versions V and V2 thermal models. These two versions (which only differ in the type of neutron shield employed in the design) have natural ventilation features.

The staff reviewed the applicant's description of the HI-STORM FW system thermal model. Based on the information provided in the application regarding the thermal model, the staff determined that the application is consistent with guidance provided in NUREG-1536, Section 4.4.4, "Analytical Methods, Models, and Calculations." Therefore, the staff concludes that the description of the thermal model is acceptable, as the description is consistent with NUREG-1536, and satisfies the regulatory requirements of 10 CFR 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

4.2 Thermal Evaluation for Normal Conditions of Storage

The applicant used the three-dimensional thermal model, described in FSAR Section 4.4, to determine temperature distributions under long-term normal storage conditions. The applicant performed screening calculations to determine which MPC type and loading pattern would result in the highest fuel cladding temperature.

FSAR Table 4.4.2 shows computed temperatures for the licensing basis minimum fuel height. These temperatures bound all heat loading patterns. Therefore, the previously approved licensing basis models continue to be applicable to the new heat load patterns for either the MPC-37 or MPC-89, and no further evaluation of the new heat load patterns is required.

The applicant calculated the maximum gas pressure in the MPC for a postulated release of fission product gases from fuel rods into the free space of the MPC for the new heat load patterns for both MPC-37 and MPC-89. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the total resulting pressure is determined from the ideal gas law. Based on fission gas release fractions (NUREG-1536 criteria), the fuel 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 the FSAR Table 4.4.5. The maximum computed gas pressures reported in the FSAR are all below the MPC internal design pressures for normal, off-normal, and accident conditions, as specified in the FSAR Table 2.2.1. These pressures correspond to the bounding MPC design.

The staff reviewed the applicant's thermal evaluation of the HI-STORM FW system during normal conditions of storage for the addition of new heat load patterns to the MPC-37 and MPC-89 canister designs. 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 NUREG-1536, Section 4.4.4, "Analytical Methods, Models, and Calculations," and is therefore acceptable.

4.3 Thermal Evaluation for Short-Term Operations

4.3.1 Vacuum Drying

The applicant's methodology for performing cyclic vacuum drying for all MPCs is summarized in FSAR Section 4.5.2.3. The applicant also provided a calculation using the MPC-37 at design maximum heat load, and a summary of those results (maximum temperature, environment, time, etc.) is provided in FSAR Tables 4.5.6, 4.5.7, 4.5.20, and 4.5.21. The results indicate that all temperatures are below the allowable limits defined by the applicant. The limits chosen by the applicant for its analyses are lower than current allowable limits described in Section 4.4.2 of NUREG-1536.

The staff reviewed the applicant's thermal evaluation of the HI-STORM FW system during drying operations. 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 4.4.4 of NUREG-1536 and is therefore acceptable.

4.3.2 Onsite Transfer

The applicant's calculated results, summarized in FSAR Tables 4.5.2 and 4.5.5, show that, from the thermal safety standpoint, the previously approved Version V provides a larger margin of safety than the predecessor HI-TRAC VW versions adopted in this FSAR. The applicant's calculated results for the HI-TRAC VW Version V2 show also that, for the limiting thermal scenario under heat load pattern A, the computed fuel cladding temperature and MPC cavity pressure summarized in FSAR Tables 4.5.23 and 4.5.24 are essentially the same as those in FSAR Tables 4.5.2 and 4.5.5 for the HI-TRAC VW standard Version V which was reviewed and accepted by the staff in the CoC No. 1032, Amendment No. 0.

The staff reviewed the applicant's thermal evaluation of the HI-STORM FW system during on-site transfer. 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 and is therefore acceptable.

4.4 Off-Normal and Accident Events

4.4.1 Off-Normal Events

The proposed changes in Amendment No. 5 do not impact the response of the MPC-37 to off-normal events, and the previous analyses results continue to bound all MPC types, including the new transfer cask versions, therefore, the previous evaluation continues to be acceptable to the staff.

4.4.2 Accident Events

The proposed changes in Amendment No. 5 do not impact the response of the MPC-37 to accident events, and the previous analyses results continue to bound all MPC types, including the new transfer cask versions, therefore, the previous evaluation continues to be acceptable to the staff.

For the HI-TRAC VW Versions V and V2 transfer casks, the applicant states in FSAR Section 4.6 that 100% blockage of inlet passages is not credible; however, based on a defense-in-depth approach, the applicant provided an evaluation assuming that inlet flow passages were 100% blocked. As a result of the considerable thermal inertia of the storage overpack, a significant temperature rise is possible if the inlets are substantially blocked for an extended duration of time. The applicant stated, however, that this accident condition is a short duration event that is identified and corrected through scheduled periodic surveillance.

Based on the results obtained from a steady state evaluation for on-site transfer, the applicant concluded, in FSAR Section 4.6.2.7, that the HI-TRAC VW Version V2 bounds Version V. The applicant provided the analysis and results for this event along with the necessary limiting condition for operation (LCO) and surveillance requirements. The results show that all predicted temperatures and pressures remain below the allowable limit provided the condition of blockage is identified and corrected within the established time, as stated in the application.

The staff reviewed the applicant's thermal evaluation of this accident event. Based on the information provided in the application regarding the thermal analysis CFD model and evaluation, the staff determined that the application is consistent with guidance provided in Section 4.4.4 of NUREG-1536 and is therefore acceptable.

4.5 Confirmatory Analyses

The staff reviewed the applicant's thermal CFD models used in the analyses, checked the code input in the calculation packages submitted, and confirmed that the proper material properties and boundary conditions were used. 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 FW system geometry for the analyzed conditions.

The staff also reviewed the engineering drawings to verify that system geometry and dimensions were adequately represented in the CFD analysis models. The material properties presented in the FSAR were reviewed to verify that they were appropriately referenced and applied. The staff confirmed 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.

4.6 Other Proposed Changes

The staff also review the following proposed changes which are applicable to the thermal evaluation:

<u>Proposed Change #5: Add the use of DFI in CoC No. 1032, Appendix A.</u>

With the use of damaged fuel in DFIs, the peak cladding temperatures (PCTs) of damaged fuel in DFI and fuel debris in DFC are below SFST-ISG-11 (NRC, 2003a) guidance of 400°C for HBF and 570°C for LBF or MBF. Therefore, the staff determines the use of DFI has no negative impact to the thermal performance of the system.

<u>Clarification E4: Allow a minor deviation in CoC Appendix B, Section 2.3.</u>

The applicant proposed to add a minor deviation from the prescribed loading pattern in CoC Appendix B, Section 2.3 to allow one slightly thermally-discrepant fuel assembly per quadrant to be loaded, as long as the PCT for the MPC remains below the limits in SFST-ISG-11, Revision 3. Since the PCT will be below the limits in SFST-ISG-11, there would be negligible impact to the thermal performance of the system. Therefore, the staff determines the proposed minor deviation is acceptable.

<u>Clarification E6: Correct typographical error in CoC Appendix A, Table 3-1.</u>

The applicant provided clarification in CoC Appendix A, Table 3-1 regarding MPC heat load Pattern A or Pattern B for MPC-37 when one or more assemblies is more than 45,000 MWD/MTU. The staff verified that this is not a change, but a clarification for consistency with FSAR Table 4.5.19.

Clarification E7: Add clarification to CoC Appendix B, Section 2.3.1 on VDS.

The applicant proposed to add clarification to CoC Appendix B, Section 2.3.1 to clarify that VDS is permitted for high burnup fuel with decay heat values and drying time limits as provided in CoC Appendix A, Table 3-1. This clarification is consistent with the guidance in SFST-ISG-11, Revision 3. Therefore, the staff determines the clarification is acceptable.

4.7 Evaluation Findings

F4.1 Chapter 2 of the FSAR describes SSCs important to safety to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

F4.2 The HI-STORM FW system, Amendment No. 5, is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask system (MPC, Transfer Cask and Overpack) is designed to provide adequate heat removal capacity without active cooling systems.

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

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 future processing or disposal.

F4.5 The staff finds that the thermal design of the HI-STORM FW system, Amendment No. 5, 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 for safe storage of spent nuclear fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guidance, applicable codes and standards, and accepted engineering practices.

5.0 CONFINEMENT EVALUATION

The confinement review ensures 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 in this amendment application and determined that all eight proposed changes have no negative impact to the confinement evaluation as discussed below. Therefore, the staff determined that the HI-STORM FW system Amendment No. 5 continues to satisfy the confinement acceptance criteria as described in Section 5.4 of NUREG-1536 (NRC, 2010).

5.1 Proposed Change #1—New Heat Load Patterns and Minimum Cooling Time

The applicant proposed to add new heat load patterns for the MPC-89 and MPC-37 (long, standard, and short length), and revise required minimum cooling time for fuel to 1 year for MPC-89 and MPC-37. The applicant presented the peak cladding temperatures (PCTs) in FSAR Table 4.4.2 and the maximum MPC internal pressures in FSAR Table 4.4.5 for new heat load patterns for the MPC-89 and MPC-37.

The staff reviewed FSAR Tables 4.4.2 and 4.4.5 and confirmed that the PCTs and maximum MPC internal pressures are below the corresponding design limits. Under this premise, staff determined that the proposed change #1 has no negative impact to confinement evaluation.

5.2 Proposed Change #2—Four New Fuel Types

The applicant proposed to add four new fuel types, 10x10I, 11x11A, 7x7C, and 8x8G to the approved contents in CoC No. 1032, Appendix B.

The staff reviewed FSAR Chapters 4 and 6 and concludes that the new fuel types do not increase the heat load and continue to be bounded by the previous thermal analyses. Therefore, the staff confirmed that the proposed change #2 has no negative impact to confinement evaluation.

5.3 Proposed Change #3—Exception to ASME Code

The applicant proposed to add an exception to the ASME Code to allow the use of certain duplex stainless steels in the HI-STORM FW system. The staff reviewed FSAR Chapter 4 and Section 3.3.1 and found that the maximum temperature of duplex stainless-steel grade of Alloy X used for confinement boundary does not exceed 600°F under all service modes. At the temperature below 600°F, it prevents the formation of the detrimental intermetallic phases, which can deteriorate the mechanical (toughness) and corrosion properties of the confinement weld. Therefore, the staff determined that the proposed change #3 has no negative impact to confinement evaluation.

5.4 Proposed Change #4—Use FLUENT for Evaluating Effective Fuel Conductivities

The applicant proposed to use FLUENT to revise the calculation for evaluating effective fuel conductivities. The staff recognized that the NRC has accepted FLUENT code for the evaluation of fuel conductivities in the HI-STAR 180D (Docket No. 71-9376). The staff determined that the proposed change #4 has no negative impact to confinement evaluation.

5.5 Proposed Change #5—Use of DFI

The applicant proposed to add DFI to CoC No. 1032, Appendix A. FSAR Section 4.4.1 states that a limited number of fuel assemblies classified as damaged fuel or fuel debris placed in damaged fuel containers (DFCs) or damaged fuel placed in DFIs are permitted to be stored in certain interior locations of MPC-37 and MPC-89 under heat load charts defined in FSAR Figures 1.2.3a thru 1.2.3c, 1.2.4a thru 1.2.4c, 1.2.5a thru 1.2.5c, 1.2.6a thru 1.2.6b, and 1.2.7a thru 1.2.7b.

The staff reviewed Notes 5 and 7 to FSAR Table 4.4.2 and found that PCT of damaged fuel in DFI and fuel debris in DFC are below SFST-ISG-11 (NRC, 2003a) guidance of 400°C for HBF and 570°C for LBF or MBF. The staff determined that the MPC cavity pressures will remain

bounded, and therefore the proposed change #5 has no negative impact to confinement evaluation.

5.6 Proposed Change #6—Add Two Versions of HI-TRAC VW

The applicant proposed to add two versions (V and V2) of the standard HI-TRAC VW. Version V adds a natural circulation feature, and Version V2 adds the option for removable neutron shield.

The staff reviewed FSAR Tables 4.5.2 and 4.5.5 for HI-TRAC VW Version V and Tables 4.5.23 and 4.5.24 for HI-TRAC VW Version V2, and found that the PCTs, maximum cask component temperatures, and maximum MPC pressures for both HI-TRAC VW Versions V and V2 are below the design limits and are therefore acceptable. The staff determined that the proposed change #6 has no negative impact to confinement evaluation.

5.7 Proposed Change #7—Cyclic Vacuum Drying for All MPCs

The applicant stated in FSAR Section 9.2.1 that for MPCs with high burn-up fuel and higher heat load, cyclic vacuum drying may be performed in accordance with FSAR Chapter 4 and SFST-ISG-11 (NRC, 2003a).

Staff accepts this proposed change with respect to confinement evaluation because the use of cyclic vacuum drying would reduce the MPC cavity temperatures and therefore reduce the MPC cavity pressure. Staff determined that the proposed change #7 has no negative impact to confinement evaluation.

5.8 Proposed Change #8—Add Fuel Containing BLEU as Content

The applicant stated in FSAR Table 2.1.2 that any number of fuel rods in an assembly can contain BLEU fuel. If the BLEU rods are present, the site-specific dose and dose rate analyses performed under 10 CFR 72.212 should include considerations for the presence of such rods.

As stated in Sections 6.8 and 7.2, BLEU assemblies are essentially identical to a UO_2 assembly except it includes a higher cobalt impurity which does not affect heat load. Therefore, the staff accepts this proposed change with respect to confinement evaluation because adding BLEU as approved contents does not affect the heat load limit and the fuel and MPC component temperatures are below their design/service limits and would be consistent with the guidance in SFST-ISG-11 (NRC, 2003a). The staff determined that the proposed change #8 has no negative impact to confinement evaluation.

5.9 Evaluation Findings

F5.1 The staff concludes that the proposed changes have no negative impact on the confinement evaluations and that the HI-STORM FW system, Amendment No. 5 continues to meet the confinement requirements of 10 CFR 72.236(l).

6.0 SHIELDING EVALUATION

The staff reviewed the proposed changes with respect to the adequacy of the HI-STORM FW system's shielding design. The shielding review evaluates the ability of the proposed shielding features to provide adequate protection against direct radiation from the dry storage system

contents. The review seeks to ensure that the shielding design is sufficient and reasonably capable of meeting the operational dose requirements of 10 CFR 72.104 and 72.106 in accordance with 10 CFR 72.236(d). In reviewing these changes to the shielding design, the staff followed the guidance in Chapter 6 of NUREG-1536 (NRC, 2010).

The staff evaluated all proposed changes and clarifications with respect to shielding evaluation as discussed below.

6.1 Proposed Change #1–New Heat Load Patterns

The applicant requested to add new heat load patterns to the MPC-89 and MPC-37 with revised minimum required cooling times for fuel from 3 years to 1 year for MPC-89 storage canister and from 3 years to 1 year for the MPC-37 storage canister.

Currently, the HI-STORM FW system is authorized for regionalized loading patterns of the spent fuel in the MPC-37 for PWR fuel or MPC-89 for BWR fuel. For the MPC-37 and MPC-89 canisters, the spent fuel basket cells are divided into 3 regions: inner region, middle region, and peripheral region, as specified in CoC Technical Specifications (TS) Appendix B, Figures 2.1-1 and 2.1-2. TS Appendix B, Tables 2.3-1A through 2.3-1C (MPC-37) and Tables 2.3-2A through 2.3-2B (MPC-89) specify the allowable heat load for each cell within each region.

The HI-STORM FW system can also be loaded in a uniform loading pattern, i.e., the allowable decay heat is the same in each cell throughout the MPC-37 and MPC-89. The heat load data are specified in TS Appendix B, Tables 2.3-3 (MPC-37) and 2.3-4 (MPC-89).

The dry storage system design allows for loading of damaged fuel or fuel debris but the damaged fuel or fuel debris must be loaded in sealed DFCs and the DFCs with damaged fuel or fuel debris must also be loaded in specific fuel cell locations of the MPC as specified in TS Appendix B, Table 2.1-1, Section I.B for the MPC-37 and Section II.B for the MPC-89.

As part of the proposed amendment, the applicant proposes to add new loading patterns that contain damaged fuel in DFCs or DFIs. The shielding evaluation of the addition of the DFI is discussed in Section 6.5 of this SER. The applicant proposes to revise the TS Appendix B to add loading patterns for MPC-37: Figures 2.3-1, 2.3-2, and 2.3-3 for “short” fuel; Figures 2.3-4, 2.3-5, and 2.3-6 for “standard” fuel; and Figures 2.3-7, 2.3-8, and 2.3-9 for “long” fuel. For MPC-89, the new loading patterns are not differentiated by length. These special requirements are specified in TS Appendix B, Figures 2.3-10, 2.3-11, 2.3-12, and 2.3-13. Damaged fuel in DFCs and DFIs are also permissible for both MPC-37 and MPC-89. The loading patterns have multiple regions, up to 5 regions for the MPC-37. The MPC-89 maintains symmetry of each quarter but otherwise each basket cell has a unique allowable decay heat.

For the MPC-37, the allowable heat loads are higher for the longer length fuel to account for the additional decay heat produced by the extra fuel length. The applicant used these bounding heat loads for selecting source terms as documented in Holtec’s proprietary Report, HI-2094431, Revision 21, “HI-STORM FW and HI-TRAC VW Shielding Analysis.” The shielding models used the “standard” length fuel assembly as the representative fuel to calculate the dose rates. This may result in some differences in dose rates when using a longer fuel assembly. The staff finds that using the shorter fuel length in the shielding models is conservative because the radiation sources per fuel segment are larger, i.e., the sources are more concentrated in each segment of the fuel in the shielding analysis models. The staff also found this modeling acceptable for the “short” length fuel which has an even lower allowable decay heat, and

modeling the “standard” fuel with a much higher radiation source per segment would be conservative. The applicant divided the fuel into several segments and distributed the total sources into each segment based on the relative burnup of each segment. Thus, assuming a shorter fuel length for the longer fuel will arbitrarily make the source terms higher in each segment, and therefore make the calculation more conservative.

6.1.1 Shielding Design Description

The applicant did not make any changes to the HI-STORM FW system design that would change the cask shielding capability as previously reviewed and approved by the NRC in the original certificate and subsequent amendments, such as thicknesses of the canister, storage overpack. The applicant included two new transfer casks, the HI-TRAC VW Version V and Version V2 design. The Version V does not include any changes to the shielding that would affect the dose rates. The changes associated with implementing the HI-TRAC VW Version V2 and the impact on shielding are discussed in this SER, Section 6.6.

6.1.2 Radiation Source Definition

To ensure that the bounding radiological source term was used in performing shielding evaluations, the applicant proposed new fuel qualification requirements that are applicable to all MPC-37 and MPC-89 contents, including the proposed new heat loading patterns. The applicant used the same method in Amendment No. 4 of the HI-STORM FW system for analyzing the MPC-32ML 16x16D fuel.

In this method, the applicant specified the required minimum cooling time as a function of burnup using a polynomial correlation in CoC TS Appendix B, Section 2.5 (also in FSAR Section 2.1.6.1). The correlations are defined by sets of correlation coefficients in CoC TS Appendix B, Table 2.5-2 (FSAR Table 2.5-2) for multiple reference decay heats for MPC-37 and MPC-89. The reference decay heats encompass the allowable decay heats for each loading zone as defined by the loading patterns in CoC TS Appendix B, Figures 2.3-1 through 2.3-9 for the MPC-37 and Figures 2.3-10 through 2.3-13 for the MPC-89. The applicant has added additional language in FSAR Section 9.2.3 to clarify the procedure for determining allowable fuel loading, and the staff found the additional language would clarify the procedure.

Cooling times computed by the correlation in TS Appendix B, Section 2.5 for some burnup values in TS are below the minimum allowable cooling times from TS Appendix B, Table 2.1-1, Sections I.A.c, II.A.d.ii, and III.A.c. Since the cooling times from TS Appendix B, Table 2.1-1 are the minimum allowable cooling times, the applicant clarified in Appendix B, Table 2.1-1 and in FSAR Section 9.2.3 that the cooling time must meet both this minimum value and those from TS Appendix B, Section 2.5. Note that Appendix B, Table 2.1-1, Section III.A.c was modified with similar language for consistency although it only applies to MPC-32ML which is not the subject of this amendment.

The 8x8F assembly has a longer minimum cooling time and a lower maximum burnup than the rest of the assemblies allowed in the MPC-89. This is conservative from a radiological source term perspective because longer cooling times and lower burnups produce a lower source term than what was analyzed by the applicant and the staff found that the inclusion of these assemblies in the new loading patterns are acceptable.

6.1.2.1 Spent Fuel Source Term

The applicant showed the burnup, enrichment, and cooling time (BECT) combinations used to determine the source term in FSAR Table 5.0.3 for the MPC-37 and in Tables 5.0.4a and 5.0.4b for MPC-89. The staff verified that the cooling times used by the applicant in the shielding evaluation are conservative (shorter) with respect to the cooling times that are calculated from the correlation in TS Appendix B, Section 2.5. The exception is for some fuel assemblies with lower burnup values, the correlation from TS Appendix B, Section 2.5 gives cooling times shorter than 1 year. Fuel cooled shorter than a year is not allowed for storage per the definition in 10 CFR 72.3 and is further restricted by the minimum allowable cooling time of 1 year for the MPC-37 in TS, and is prohibited from loading. Therefore, the staff found using a cooling time of 1 year for the shielding evaluation in these cases is conservative. The applicant included additional language in the FSAR operating procedures, Section 9.2.3, to clarify this specific restriction.

Similar to the methodology used for the MPC-32ML basket with 16X16D fuel in HI-STORM FW system Amendment No. 4, enrichment is not used as a controlling parameter for fuel specification. Although enrichment is important for determining the source term, the applicant has instead used a conservative value for each analyzed burnup based on data collected from U.S. Department of Energy's Nuclear Fuel Data Survey Forms RW-859 and GC-859. These forms are used to collect information on spent nuclear fuel (SNF) for all commercial reactors. Form RW-859 is for all SNF discharged before 2002. Form GC-859 includes assemblies discharged 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 discussed this method in Holtec's proprietary report HI-2188480, "Lower Bound Fuel Enrichment Based on Industry Data," Revision 0. The applicant determined the enrichment that would bound 99% of the discharged fuel population within 5 GWd/MTU burnup groups (e.g., 0-5 GWd/MTU, 5-10 GWd/MTU, 10-15 GWd/MTU, etc.).

The applicant shows the initial enrichment for 99% of the fuel data for each 5 GWd/MTU burnup group in report HI-2188480. 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, 2001a), "Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks." The enrichment at this high burnup level could have a significant impact on the source term. However, based on report HI-2188480, there are fewer data for assemblies with burnup over 55,000 MWd/MTU.

The applicant used an enrichment value in the shielding evaluations for BWR fuel at 65 and 70 GWd/MTU from FSAR Table 5.2.17 (as well as Tables 5.0.4.a and 5.0.4b); the enrichment level is non-conservative because the enrichment value used is higher than the enrichment value that bounds 99% of the fuel shown in Report HI-2188480. The staff performed independent calculations as discussed in Section 6.1.5.1 of this SER using the enrichment value that bounds 99% of the assemblies at 70 GWd/MTU in Report HI-2188480 and compared it to the enrichment value used by the applicant in the HI-STORM FW system shielding evaluations in FSAR Table 5.2.17. The staff found that the difference in gamma source was as much as 4% higher and the difference in neutron source term was about 20% higher. This shows that the difference in source term due to using the applicant's higher enrichment value is non-conservative and would produce a lower source term.

The maximum dose rate contribution is primarily from gamma radiation as shown in FSAR Tables 5.1.1, 5.1.2a, 5.1.2b, 5.1.4a, 5.1.4b, 5.1.4c, 5.1.5, 5.1.6a, 5.1.6b, 5.1.7, 5.1.8a, and

5.1.8b. This means that the system is better at shielding neutrons than gammas, so a larger neutron radiation increase would have less of an effect on system dose rates. It also means that the more limiting source terms are not at these higher burnup values and are more likely at lower burnup with lower cooling time values that produce higher gamma sources and that the difference in source term due to the higher enrichment would not have a significant impact on dose rates. For example, FSAR Table 5.1.1 shows that at the bottom of the transfer cask that the dose rate contribution from gammas and neutrons is about 30% of the total dose rate. Considering the contributions from fuel and Co-60 (hardware and inserts) gammas, a 4% increase in fuel gamma source term and a 20% increase in fuel neutron source term would result in an overall increase in dose rate of less than 7%. Therefore, the staff found that the increase in source term resulting from using the enrichment value in Report HI-2188480 would have a small impact on the system dose rates and determined the value used by the applicant acceptable. In addition, the maximum burnup for the BWR fuel is limited to 65 GWd/MTU per TS Appendix B, Table 2.1-1, Section II.d.ii, and the source term would be slightly lower at 65 GWd/MTU than at 70 GWd/MTU.

Therefore, the staff found that defining bounding minimum enrichments in this manner to be acceptable for this application, in lieu of having enrichment as part of the fuel specification within the technical specifications.

The applicant analyzed a number of specific BECT points along the burnup-cooling time curve for each correlation corresponding to a decay heat as shown in FSAR Tables 5.0.3, 5.0.4a, and 5.0.4b. In some cases, the applicant combined decay heat zones, i.e., used a BECT from a higher decay heat correlation curve for a cell that was restricted to a lower decay heat. From a shielding perspective, this is a bounding and simplified approach and reduces the number of calculations. The applicant established the bounding source term among the chosen points on the burnup-cooling time correlation by determining, for each region, the BECT combination that produced the highest dose rate for each physical location around the cask that the dose rate was evaluated. The staff found that this process is capable of determining the bounding source term for such a large and varied population of possible fuel loadings.

6.1.2.2 Accident Condition Source Term

The changes to the loading patterns and fuel qualification affect the accident condition source terms as well. The applicant has not proposed any changes from Amendment No. 4 that would otherwise impact the accident condition source term. The changes to the accident condition source terms associated with the implementation of the HI-TRAC VW Version V2 transfer cask are discussed in Section 6.6 of this SER.

The applicant shows the gamma source term from SNF for the accident condition in FSAR Tables 5.2.3 and 5.2.5 for the MPC-37 and MPC-89, respectively. For neutron radiation from SNF, the applicant shows the source terms in FSAR Tables 5.2.12 and 5.2.14 for the MPC-37 and MPC-89, respectively. The staff reviewed the burnup and cooling times for these source terms and found that they are consistent with those produced from the correlation and correlation coefficients outlined in TS Appendix B, Section 2.5 of the HI-STORM FW system as amended.

6.1.2.3 Non-Fuel Hardware Source Term

Non-Fuel hardware is allowed to be stored in the MPC-37 per TS Appendix B Table 2.1-1 Sections I.C. and I.D. One neutron source assembly (NSA) is allowed along with up to 30

burnable poison rod assemblies (BPRAs) per cask. Note 1 to TS Appendix B, Table 2.1-1, Section I states that fuel assemblies containing BPRAs, thimble plug devices (TPDs), 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 axial power shaping rods (APSRs), rod cluster control assemblies (RCCAs), control element assemblies (CEAs), control rod assemblies (CRAs) (including, but not limited to those with hafnium), or NSAs may only be loaded in fuel storage Regions 1 and 2 (two inner regions).

As part of Amendment No. 5, the applicant reduced cooling time and maximum burnup of certain non-fuel hardware. This includes a minimum cooling time of 1 year (reduced from 3 years) for MPC-37, except for NSAs, APSRs, RCCAs, CRAs, and CEAs which was decreased from 5 years to 2 years of cooling time. The burnup is the same as previously authorized, except for TPDs, water displacement guide tube plugs, and orifice rod assemblies, the burnup was reduced from 630 GWd/MTU to 225 GWd/MTU. The applicant modified FSAR Table 2.1.1a to include loading requirements for the MPC-37 and included the minimum cooling time and maximum burnup requirements for the non-fuel hardware.

The applicant re-calculated the Co-60 source terms with the new burnup and cooling time requirements for the non-fuel hardware and shows the new increased Co-60 activity for the non-fuel hardware BPRAs and the TPDs in FSAR Table 5.2.16b. The calculated Co-60 activity for the lower cooling time increased more than the staff expected, given the 2-year difference in cooling time (with Co-60 having a half-life of about 5.27 years) and maximum TPD burnup decreasing (from 630 GWd/MTU to 225 GWd/MTU, FSAR Table 2.1.1a). Modeling a larger Co-60 dose is conservative with respect to predicting dose and dose rates, and therefore, the staff did not investigate why the calculated Co-60 activity for the lower cooling time increased more than the staff expected. The staff found that the proposed minimum cooling time and maximum burnups of non-fuel hardware acceptable.

6.1.2.4 Computer Codes for Radiation Source Definition

The applicant updated the depletion code used to calculate the source term to TRITON and ORIGAMI/ORIGEN modules in the SCALE 6.2.1 system. ORIGEN is considered acceptable to the staff per the guidance in Section 3 of NUREG/CR-6802 (NRC, 2003b), “Recommendations for Shielding Evaluations for Transport and Storage Packages.” ORIGAMI (ORIGEN Assembly Isotopics) is a newer code within SCALE 6.2.1 and was developed after the publication of NUREG/CR-6802. It computes detailed isotopic compositions for LWR assemblies containing UO_2 fuel by using the ORIGEN code with pregenerated ORIGEN libraries for a specified assembly power distribution. TRITON is also a newer code, developed after the publication of NUREG/CR-6802 and represents more detailed 2-D reactor physics models as compared to the methods recommended in NUREG/CR-6802. The staff found the use of these codes acceptable for the HI-STORM FW system.

6.1.2.5 Other Parameters Affecting Source Terms

The staff found specific information about PWR reactor operations that affect the source term within the sample ORIGAMI files submitted by the applicant in FSAR Appendix 5.A (proprietary) as well as the proprietary Attachment 11 to the response to the NRC’s request for additional information (Holtec, 2019d). The staff compared the values used for specific power and the moderator density to the range reported in Appendix B of NUREG/CR-6802 (NRC, 2003b) and found that they are reasonably conservative. Soluble boron concentration is set within the pre-

calculated ORIGAMI cross section libraries and is based on a cycle average as documented in the TRITON input templates with the SCALE 6.2.3 code distribution. Based on Table B.2 of NUREG/CR-6802, boron concentration does not have a large effect on the source term and the staff found the use of pre-calculated values acceptable.

The applicant did not submit any sample input files for the staff to review the BWR input parameters that affect the source terms, and the staff did not find any information on reactor operations assumed for the BWR depletion models in Report HI-2094431, Revision 21. These parameters are not considered to have a strong influence on dose rates as documented in NUREG/CR-6716 (NRC, 2001a). Therefore, the staff has assurance that reasonable parameters were used based in its own calculations (see Section 6.1.5.1 of this SER).

6.1.3 Shielding Model Specification–Configuration of the Shielding and Source

The applicant also proposed changes to the lid that it modeled in the shielding analysis in Amendment Nos. 0 to 4). Currently, there are three lids that can be used in the HI-STORM FW system. They are the standard lid, the XL lid, and a domed lid which is a thicker version of the XL lid. Previously, all dose and dose rate evaluations were performed by the applicant using the standard lid. The applicant stated in Appendix K of Report HI-2094431, Revision 21 that the average dose rate across the top is lower for the XL lid than for the standard lid.

In the proposed changes to FSAR Section 5.0, the applicant, however, states that unless otherwise noted that all dose rates calculations use the “XL lid.” Since the applicant states that the shielding performance of the XL lid is better than that of the standard lid, this would mean that the calculated dose rates at the lid are non-conservative for a design basis loading if the standard lid is used.

If a user uses the standard lid, it will have to load fuel with a lower source term than that of the design basis fuel in order to meet the TS dose rate limits. The staff finds that although using the XL lid is non-conservative within the evaluation, as long as the TS dose rate limit at the lid is met, this would ensure that the HI-STORM FW system remain capable of meeting the site boundary dose limits in 10 CFR 72.104. As discussed in Section 6.1.4.1 of this SER, users are required to perform a site-specific analysis that ensures that TS dose rate limits are established specific to any particular site and the loading configuration which would include the lid being used.

The staff reviewed the calculated maximum dose rates for the HI-STORM FW system under normal conditions at the lid in FSAR Table 5.1.5 (MPC-37) and Tables 5.1.6a and 5.1.6b (MPC-89). The staff found that the maximum dose rate (15 mrem/hr) specified in TS Appendix A, Section 5.3.4 that the applicant established at the lid is appropriate for the XL lid and the new content specification and loading patterns and that this limit is sufficient to ensure that the system is capable of meeting the regulatory dose limits in 10 CFR 72.104.

The staff did not verify the modeled dimensions of the XL lid within the MCNP model against the drawings. However, the staff found the applicant’s modeling of the XL lid acceptable based on the staff’s own modeling of the XL lid as discussed in Section 6.1.5.2 of this SER which was based on drawings and information in the FSAR.

6.1.3.1 Damaged Fuel

The new loading patterns include damaged fuel. With respect to damaged fuel modeling the applicant assumes that the damaged fuel is the same as intact fuel and justifies this by using a comparison documented in Supplement 5.II.4.3 of the HI-STORM 100 FSAR (Holtec, 2019h, response to NRC's request for supplemental information). The staff reviewed the information and found that the study is applicable to the HI-STORM FW system based on the number and location of the damaged fuel assemblies. The results of the study show that dose rates increase with the inclusion of the damaged fuel model (in comparison to modeling it as intact fuel). The largest increase is at the top of the cask, and this is because the way damaged fuel was modeled causing 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 were to experience collapse 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 at the bottom side of the cask. The staff determined that this increase is small enough and it is unlikely that all of the damaged fuel assemblies would fail in the worst scenario, therefore, the staff has reasonable assurance that the system meets regulatory annual dose limits.

Notes 12, 13, and 14 to TS Appendix B, Table 2.1-3 states enrichment limits for damaged fuel for certain BWR assemblies. These enrichment limits are maximum enrichment limits for purposes of criticality safety and are above that used for determining source terms for medium to high burnup assemblies as shown in FSAR Tables 5.0.4a and 5.0.4b. Therefore, the staff found that these maximum enrichment limits for damaged fuel would not impact the ability of damaged fuel with respect to meeting dose limits using the new fuel qualification method.

The applicant proposed the addition of DFIs. The staff's shielding evaluation of this addition is discussed in Section 6.5 of this SER.

6.1.3.2 Material Properties

The material properties remain unchanged from Amendment No. 4 with the exception of the fuel density and composition provided in FSAR Table 5.3.2 for the evaluations involving the XL lid. In Report HI-2094431, Revision 21, the applicant showed that the fuel composition change results in an insignificant change in dose rates, and therefore the staff found it acceptable. Other material properties related to the implementation of the HI-TRAC VW Version V2 transfer cask are discussed in Section 6.6.3.2 of this SER.

The modeling of the XL lid is new in this amendment. The staff reviewed the concrete density and found it is consistent with the minimum density specified in FSAR Table 1.2.5 and determined it appropriate.

6.1.4 Shielding Analyses

In Amendment No. 5, the applicant used the same computer codes, MCNP-5, that the staff previously found acceptable in Amendment Nos. 0 through 4 for the shielding analyses, and the staff found that the MCNP-5 code continues to be acceptable for this amendment application. The staff's determination is based upon NRC prior approvals, and because it is cited as a well-established code commonly used for spent fuel dry storage system shielding evaluations that the staff has found to be acceptable in Section 6.5.4.1 of NUREG-1536 and Section 4.3 of NUREG/CR-6802. The applicant also used the same flux-to-dose-rate conversion factors, from

ANS/ANSI-6.1.1, “Neutron and Gamma-Ray Flux-to-Dose Conversion Factors,” that the staff previously found acceptable in Amendment Nos. 0 through 4. The staff found that flux-to-dose-rate conversion factors continue to be acceptable for this amendment application. The staff’s determination is based upon the NRC prior approvals, and because the flux-to-dose-rate conversation factors for spent fuel dry storage system shielding evaluations was found acceptable in Section 6.5.4.2 of NUREG-1536 and Section 4.3 of NUREG/CR-6802.

6.1.4.1 Dose Rates

The staff reviewed the new and updated dose rate tables in FSAR Sections 5.1 and 5.4. The dose rate limit in TS Appendix A, Section 5.3.4 for the top of the overpack is 15 mrem/hr at the center of the lid. The staff found that this limit is slightly higher compared with the calculated dose rate of 11.2 mrem/hr in FSAR Table 5.1.5. Although the limit in TS, 15 mrem/hr, is slightly higher, the staff does not believe it will impose a significant risk for exceeding regulatory limit in 10 CFR 72.104 because the radiation reaching the controlled area boundary with a minimum distance of 100 meters, per 10 CFR 72.106, would be significantly reduced.

The staff has found that the dose limit in TS Appendix A, Section 5.3.4, for the side of the overpack of 300 mrem/hr at the specified measurement locations described in Appendix A, Section 5.3.8b continues to be acceptable based on the maximum calculated dose rate at mid-height from FSAR Table 5.1.5 (292 mrem/h).

Based on the off-normal conditions in Chapter 12 of the FSAR, the staff found the above dose rates to be representative of both normal and off-normal conditions. The applicant included a dose-to-distance curve in FSAR Figure 5.1.3 for the MPC-37 which has the highest dose rates in the HI-STORM FW system overpack configuration. The staff reviewed the figure and found that the applicant identifies the minimum distance required to meet the annual dose in 10 CFR 72.104 for a single cask and for various array sizes. The staff found that using the number of occupancy hours of 24 hours a day and 365 days a year to be conservative and demonstrates that the design basis system is capable of meeting the normal condition annual dose limit in 10 CFR 72.104.

The staff found that the dose rate limit in TS Appendix A, Section 5.3.4 for the side of the transfer cask of 3,500 mrem/hr at the specified measurement locations described in TS Appendix A, Section 5.3.8c is acceptable based on the maximum calculated dose rate at mid-height from FSAR Table 5.1.2b (5,898.2 mrem/hr). The limit of 3,500 mrem/hr is conservative at this location. The regulations in 72.212(b)(5) requires a user to perform a written evaluation that ensures that the cask when loaded will conform to the conditions in the CoC (including TS). This ensures that the cask will not be loaded with fuels with a source term as high as design basis fuel or that the variable weight transfer cask will include additional shielding to reduce the dose rate to this value.

The applicant calculated the accident condition dose based on the loss of the neutron shield for the transfer cask which was determined to be the limiting consequence with respect to dose for the events discussed in Chapter 12 of the FSAR. The limiting configuration is the MPC-89 with the loading pattern from FSAR Figure 1.2.7 (corresponding to TS Appendix B, Figure 2.3-13). The applicant’s calculation showed that it would take 59 days to reach the regulatory dose limit in 10 CFR 72.106. The staff found that this is sufficient time to return the system to normal operating conditions from the accidents involving the loss of the neutron shield by unloading the cask or repairing the neutron shield and determined that the applicant has demonstrated that it will meet the regulatory dose limit in 10 CFR 72.106 for accident conditions.

6.1.5 Confirmatory Calculations

6.1.5.1 Source term Confirmatory 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 the following tables in FSAR: Tables 5.2.2, 5.2.3, 5.2.4, 5.2.5, 5.2.11, 5.2.12, 5.2.13, and 5.2.14. Although these source terms are representative, the staff found that confirming these data provides additional assurance that the calculation process is consistent with the staff's expectation.

The staff used the Westinghouse 17x17 assembly for the MPC-37 and the GE 10x10 assembly for the MPC-89 because these are stated in FSAR Section 5.2.4 as the design basis assemblies. The staff used the same burnup and cooling time from the above listed tables (which are consistent with FSAR Tables 5.0.3, 5.0.4a, and 5.0.4b) and the enrichments from FSAR Table 5.2.17. The staff used the specific power from the sample ORIGAMI file in FSAR Appendix 5.A for both the PWR and BWR calculations. The sample input file is for a PWR, and the staff found the values sufficiently higher than the typical values of BWR fuel power density and considered it conservative to use the values for the BWR assembly. The staff used the moderator density for the PWR assembly from Report HI-2094431, Revision 21. This values in the report is lower than that from the sample ORIGAMI file, and a lower moderator density is more conservative with respect to maximizing source terms as reported in NUREG/CR-6716 (NRC, 2001a). For the BWR assembly, the staff chose 40% void fraction which is commonly considered a core average for BWRs.

The staff independently confirmed that the gamma and neutron source terms in the tables above are reasonable. Although the applicant did not provide reactor operating parameters it used in the depletion calculations, the staff has reasonable assurance that the reactor operating parameters were reasonably bounding based on the staff's own source term calculations as the reactor operating parameters it used in its calculations resulted in source terms not appreciably different from the applicant's.

As discussed in Section 6.1.2.1 of this SER, the staff performed a calculation using the enrichment from Report HI-2188480 to evaluate the impact of the difference in source terms on calculated dose rates. As discussed above, the staff modeled the BWR fuel assembly, GE 10x10, used specific power from Report HI-2094431, Revision 21, and assumed a void fraction of 0.4. The staff calculated the gamma and neutron source terms at 70 GWd/MTU, using the enrichment from Report HI-2188480, and compared to the applicant's calculated dose rates using enrichment in FSAR Table 5.2.17. The staff found that using the lower enrichment has an insignificant increase on the gamma source (4% or less) term but a more significant increase on the neutron source term (approximately 20%).

6.1.5.2 Shielding Confirmatory Calculations

The staff performed confirmatory calculations using The Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) code Version 4.0 available through the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory (ORNL). UNF-ST&ARDS is a comprehensive integrated data and analysis tool being developed for the US Department of Energy (DOE) Office of Nuclear

Energy (NE) Spent Fuel and Waste Disposition (SFWD) program with support from the NRC. UNF-ST&DARDS simplifies and automates performance of spent fuel analyses.

This code uses ORIGAMI for source term evaluations and Monaco/MAVRIC for the dose calculations. The staff's model was built by ORNL using design basis data from the FSAR and FSAR drawings. One of the notable differences between the staff's calculation method and that of the applicant is that the staff's model represents the fuel rods/pins explicitly versus using a homogenized fuel mixture. The staff's burnup profile is represented by depleting each axial zone individually rather than using an adjustment factor, and the staff's model includes the full loading pattern with all fuel assemblies modeled simultaneously using the design basis BECT combinations.

The staff's model locates the tallies at roughly the same locations as shown in FSAR Figures 5.1.1 and 5.1.2 and used the same BECT as used by the applicant. The staff calculated the dose rates for the following configurations:

- Side surface of the HI-TRAC VW under normal conditions with the MPC-89 using the loading pattern from FSAR Figure 1.2.7 and the BECT provided in Report HI-2094431, Revision 21 for Location 2.
- Bottom side and side (mid-height) surfaces of the HI-STORM FW system XL overpack with the MPC-37 using the loading pattern from FSAR Figure 1.2.5a and the BECT provided in Report HI-2094431, Revision 21 for Locations 1 and 2.
- Top surface of the HI-STORM FW system XL overpack with the MPC-37 and the XL lid using the loading pattern from FSAR Figure 1.2.5a and the BECT provided in Report HI-2094431, Revision 21 for Location 4 at the center of the lid.
- At 1 meter and 100 meters from the side of the HI-TRAC VW under accident conditions with the MPC-89 using the loading pattern from FSAR Figure 1.2.7 and the BECT provided in Report HI-2094431, Revision 21 for Location 2.

The results from the staff's model are in agreement with that of the applicant's calculations. This provides additional assurance that the HI-STORM FW system, with the proposed changes in Amendment No. 5, is capable of meeting the regulatory dose requirements in 10 CFR 72.104 and 72.106.

For all of its dose and dose rate calculations, the applicant used a BECT combination that maximized the dose rate for each location, which means that the maximum dose rate reported for the top of a cask configuration may use a different BECT than that reported for the side or bottom of a cask. The staff did not perform calculations with the same BECT as the applicant for each location; however, the staff calculated the dose rates for all of the reported locations for the BECT combinations identified above. Even with different BECT combinations, the staff's results exhibit the same behavior and are roughly in line with the applicant's results, which provides the staff confidence that the applicant's model is adequately representing the dose rates for the HI-STORM FW system and the HI-TRAC VW.

6.2 Proposed Change #2—Four New Fuel Types

The applicant proposed to add four new BWR fuel types, 10x10I, 11x11A, 7x7C, and 8x8G as authorized contents in CoC No. 1032, Appendix B. FSAR Section 5.1 states: "The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are Westinghouse (W) 17x17 and the General Electric (GE) 10x10, for PWR and BWR fuel types,

respectively.” The fuel parameters for the GE 10x10 are listed in FSAR Table 5.2.1. The staff compared the new fuel assembly design parameters as proposed by the applicant in FSAR Table 2.1.3 and found that the mass of all the proposed new fuel assemblies is higher as compared to the design basis in FSAR Table 5.2.1. The staff found that the highest is for fuel type 7x7C which is about 10% higher based on the allowable fuel pellet diameter, height, and number of rods.

ORNL performed a sensitivity study on the effect of uranium mass on dose rates in NUREG/CR-6716 (NRC, 2001a). 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 sensitivity study where a 10% increase in uranium mass results in a 3% increase in dose rates. Although the cask used in the sensitivity study are not the same as the HI-TRAC VW, HI-TRAC Versions V and V2, or HI-STORM FW system, the staff found that both have dose rates dominated by gamma contributions. For the HI-STORM FW system, this conclusion is confirmed by the dose rates calculated by the applicant for normal conditions in FSAR Section 5.1 for the HI-TRAC VW, HI-TRAC VW Versions V and V2, and the HI-STORM FW system. NUREG/CR-6716 concludes that fuel mass affects doses from neutron radiation more than doses from gamma radiation, and the sensitivity study shows a small change in overall dose because major dose contribution of the system under study was from gamma radiation. Therefore, the staff found that the results of NUREG/CR-6716 are applicable to the HI-STORM FW system and that the new fuel designs, although higher in mass by as much as 10%, would only have a small impact on system dose rates.

The accident condition dose rates are more equal from both gamma and neutron radiations, and therefore the increased mass in the new assemblies would have a larger effect on the accident condition dose rates. NUREG/CR-6716 states in Section 3.4.2.3 that the neutron dose rate increases by about 6% for a 10% increase in fuel mass. This increase is still small compared to the accident dose margin discussed in Section 6.1.4.1 of this SER and the staff found that the inclusion of these fuel types is acceptable as it would be represented by the accident analyses discussed to support Proposed Change #1 in Section 6.1 of this SER. The staff found the inclusion of the new fuel types would not prohibit the system’s ability to meet regulatory dose requirements in 10 CFR 72.104 and 72.106, and nor the occupational dose requirements in 10 CFR 20.1201.

6.3 Proposed Change #3—Exception to the ASME Code

The applicant proposed to add an exception to the ASME Code to allow the use of certain duplex stainless steels in the HI-STORM FW system.

The staff evaluated the effect of this change on the system’s shielding performance. The MPC enclosure vessels are the only shielding components that are specified as Alloy X. The density of the proposed duplex stainless steel is slightly less (approximately 1.5%) than that of the stainless steel modeled within the shielding evaluation. This is non-conservative as lower density materials provide less shielding than higher density materials.

The staff did a sensitivity study in Microshield justifying the difference in shielding capability of the MPC lid with a reduced density of 1.5% as this is the thickest part of the MPC. The staff used the gamma spectra from FSAR Table 5.2.2 and found an insignificant change in dose rates with the reduced density. Therefore, the staff concludes that the inclusion of Alloy X does not negatively impact the HI-STORM FW system capability to meet regulatory dose limits in 10 CFR 72.104 and 106.

6.4 Proposed Change #4—Use FLUENT to Evaluate Effective Fuel Conductivities

The applicant proposed to use FLUENT to revise the calculation for evaluating effective fuel conductivities. This proposed change has no effect on the shielding of the package or the dose estimate analysis.

6.5 Proposed Change #5—Damaged Fuel Isolator

The applicant proposed to add the DFI to CoC Appendix A. Based on the definition of the DFI, this is only for fuel that can be handled by normal means and whose structural integrity is such that geometric arrangement is not expected. Therefore, the fuel condition is the same as is assumed in the intact fuel shielding evaluations and does not affect the shielding or the dose rate calculation. The applicant models damaged fuel as intact, as documented in Section 6.1.3.1 of this SER, and hence, if the fuel experiences reconfiguration, the analysis remains bounding. The DFI itself could potentially reduce dose rates because the shell of the DFI provides additional shielding, and therefore the staff found that the use of DFI will not negatively affect the package's ability to meet regulatory dose limits.

6.6 Proposed Change # 6—HI-TRAC VW Versions V and V2

The applicant proposed to add two versions of the standard HI-TRAC VW: Version V which adds a natural circulation feature and Version V2 which adds the option for a removable neutron shield. The Version V does not include any changes to the shielding that would affect the dose rates, and therefore, this part of the SER will focus on the acceptability of the Version V2 of the HI-TRAC VW.

6.6.1 Radiation Source Definition

The allowable fuel assemblies for the HI-TRAC VW Version V2 are discussed in Section 6.1 of this SER. The staff's evaluation of the source term in SER Section 6.1 is applicable to the HI-TRAC VW Version V and V2.

6.6.2 Shielding Model Specification

The HI-TRAC VW Version V2 transfer cask does not have a water jacket but has a removable NSC made of Holtite-A as the neutron shielding material.

Chapter 12 of the FSAR discusses the potential off-normal conditions and their effect on the HI-STORM FW system. The applicant states in FSAR Section 5.1.1 that none of the off-normal conditions have any impact on the shielding analysis. The staff reviewed the information and confirmed that the off-normal conditions do not have any impact on the shielding analysis of the HI-TRAC VW Version V2.

The accident condition that affects the HI-TRAC VW Version V2 is a fire where the Holtite-A could be lost. As indicated in FSAR Table 3.2.2, the total radial thickness of steel and lead is thicker for the Version V2 than that of the standard HI-TRAC VW, and the applicant states that the Version V2 loss of shielding event is bounded by the HI-TRAC VW accident analysis where the liquid neutron shield is lost. The staff found the applicant's assessment reasonable because under an unlikely fire event, the complete burnout of the polymer-based Holtite-A is not likely as the fire would have to burn 100% of the polymer. On this basis, the staff found that the results

of the applicant's shielding calculation as documented in Chapter 6 of the applicant's FSAR and in Section 6.1 of this SER for HI-TRAC VW bound the dose rate of the HI-TRAC VW Version V2 under accident scenarios. The staff also found that there is reasonable assurance that the HI-TRAC VW Version V2 will enable the HI-STORM FW system as amended to meet the regulatory dose requirement in 10 CFR 72.106.

6.6.3 Shielding Model Specifications—Configuration of the Shielding and Source

The staff reviewed the MCNP model of the HI-TRAC Version V2 as depicted by MCNP VisEd in FSAR Figures 5.3.14 and 5.3.15 and in Report HI-2094431, Revision 21, and found it was modeled in sufficient detail to accurately represent the HI-TRAC VW Version V2 cask. The applicant states in Report HI-2094431, Revision 21, that it modeled the lead thickness using the minimum dimensions consistent with parameters in FSAR Table 3.2.2. The neutron shielding dimensions in FSAR Drawing 11283, Revision 0, are consistent with the minimum dimensions in FSAR Table 3.2.2. In addition, the staff performed independent shielding calculations of the HI-TRAC VW Version V2, using information from FSAR drawings and design information as discussed in Section 6.6.5 of this SER. On these bases, the staff has determined that the nominal dimensions of the neutron shielding component are the same as the minimum dimensions, and that the applicant's model is appropriately representative and capable of calculating reasonably accurate dose rates for the neutron shielding component.

6.6.3.1 Material Properties

The staff reviewed FSAR Table 5.3.2 for the composition of the Holtite-A neutron shield material used in the shielding evaluation for the HI-TRAC VW Version V2. The staff compared the assumed composition in Table 5.3.2 to the information in Appendix 1.B of the HI-STORM 100 FSAR (Holtec, 2019i) and the acceptance criteria specified in HI-STORM FW FSAR Section 10.1.3, "Acceptance Tests and Maintenance Programs." The staff found that the applicant assumed a lower density than specified in Appendix 1.B of the HI-STORM 100 FSAR. The staff found that this is a conservative assumption because a lower density material provides less shielding and a calculation with this assumption would result in a higher dose rate than a higher density material and is therefore acceptable.

The applicant also provided a reference document, HI-2002396, "Holtite-A: Development History and Thermal Performance Data" (Holtec, 2000) to provide more detailed information on the material properties and its qualification testing. The staff verified that the material properties of Holtite-A are consistent with those used to represent this material in the shielding evaluation.

6.6.4 Shielding Analyses

For the shielding modeling for the HI-TRAC VW Versions V and V2, the applicant used the same computer codes and flux-to-dose-rate conversion factors as it used to model the HI-TRAC VW, which is the same one used for modeling the new heat load patterns. The staff's detailed evaluation of the HI-TRAC VW is documented in Section 6.1.4 of this SER.

6.6.4.1 Dose Rates

The staff compared the dose rates of the HI-TRAC VW Version V2 in FSAR Table 5.1.10 to that of the HI-TRAC VW in FSAR Table 5.1.2b. The comparison shows that the total dose rates for the Version V2 are mostly lower than that of the HI-TRAC VW. This result is reasonable because Version V2 has more gamma shielding, and therefore it is expected to have reduced

gamma dose rates. The dose rates from the neutron radiation has increased; however, there is a reduction in dose rates from gamma radiation, thus the overall dose rates have decreased.

The staff reviewed the dose rate limits in the TS Appendix A for the HI-TRAC VW Version V2. The staff found that the TS dose rate limit in TS Appendix A, Section 5.3.4 for the side of the transfer cask of 3,500 mrem/hr at the specified measurement locations described in Section 5.3.8c is acceptable because the maximum calculated total dose rate at this location (mid-height) is 5,898.2 mrem/hr as shown in FSAR Table 5.1.2b. Because this is a lower dose rate than what was calculated in the design basis, the staff determines that the transfer cask is capable of meeting 10 CFR Part 20 occupational dose limits, and this is further discussed in Chapter 11 of this SER.

The applicant's shielding analyses also show that the dose rates at the bottom of the HI-TRAC VW Version V2 are higher than that of the HI-TRAC VW when the shielding pedestals are not present. The dose rate is as high as 7,178 mrem/hr in comparison with dose rate of 1,164 mrem/hr when the pedestals are installed (FSAR Table 5.1.10). This comparison shows that the pedestals or some equivalent level shielding must be installed based on the TS dose rate limit of 3,500 mrem/hr. Detailed discussions of the of the pedestals are in Section 11.6.2, "Radiation Protection", of this SER.

6.6.5 Shielding Confirmatory Calculations

The staff performed confirmatory calculations using the UNF-ST&DARDS code. The staff calculated the dose at the surface for the bottom side of the HI-TRAC VW Version V2 with the MPC-89 using the loading pattern from FSAR Figure 1.2.7 and the BECT provided in Report HI-2094431, Revision 21. The staff also calculated the dose rate at this same location but with the pedestals present using the BECT provided in Report HI-2094431, Revision 21.

The staff's calculated dose rate, without the pedestals, is significantly higher than the applicant's reported value at this location. With the pedestals, the staff's calculation was in better agreement with the applicant presented in the SAR. Not knowing the source of the discrepancy, the staff recognized that the pedestals are very important for ALARA (as low as reasonably achievable) purposes when using the HI-TRAC VW Version V2. As discussed in Section 11.6.2 of this SER, there is no configuration where this location would be exposed, and therefore the staff determined that the differences were not consequential. However, the outside of the pedestals would be an area where occupational workers are exposed. Although the staff's calculation shows higher dose rate, the value is lower than the side surface of the cask at mid-height and lower than the HI-TRAC VW. Therefore, this gave the staff confidence that the HI-TRAC VW Version V2 is bounded by the HI-TRAC VW.

6.7 Proposed Change #7—Option for Cyclic Vacuum Drying

The applicant proposed to add an option for cyclic vacuum drying for all MPCs. This has no effect on the shielding design of the dry cask storage system. Therefore, the staff did not perform further evaluation of the shielding design with respect to this proposed change.

6.8 Proposed Change #8—BLEU Fuel

The applicant proposed to add fuel assemblies that are made from BLEU as approved contents.

The applicant states that BLEU assemblies are essentially identical to a UO_2 assembly except it includes a higher cobalt impurity. The applicant performed an analysis and showed the results in Report HI-2094431, Revision 21. As a result, the minimum cooling time requirement for BLEU fuel has been increased by 1 year in the TS, as shown in Note 1 to CoC Appendix B, Table 2.5-2. The staff found this to be acceptable because there is no difference between normal fuel and BLEU fuel in terms of the fuel hardware and the fuel composition except that some of the BLEU may have included Co-59 impurity from the down-blending process. The staff determined that it is conservative to account for the additional Co-59 impurity because its inclusion within a dose rate evaluation would result in a higher dose rate. Having conservatively accounted for the Co-59 impurity in the calculations, the staff finds it acceptable to assume a higher Co-60 level in the BLEU fuel.

6.9 Clarifications and Editorial and Minor Changes in the CoC/FSAR

6.9.1 E1—Modify Repaired/Reconstituted Fuel Assembly Definition

The applicant modified the definition for repaired/reconstituted fuel assembly in CoC Appendix A, Section 1.1, to clarify that if dummy stainless steel rods are present in the loaded spent fuel assemblies, the dummy/replacement rods will be considered in the site-specific dose calculations.

Irradiated stainless steel rods can have an effect on the system dose and dose rates. FSAR Section 5.4.6 discusses an analysis that was performed to justify the inclusion of the stainless steel rods by increasing the amount of Co-60 to account for the irradiated stainless steel rods. The staff reviewed the analysis and found that the analysis is conservative because the stainless steel rods replace irradiated fuel and typically have less radiation than the fuel rods in the same assembly which typically produce much stronger source terms, e.g., spontaneous fission neutron and (α, n) reaction produced as well as the gammas associated with these neutron emitting reactions. Adding additional Co-60 to the current spent fuel source term is a conservative way to account for the radiation source from these rods. There would be some loss of self-shielding when a UO_2 rod is replaced by a stainless steel rod and this was not evaluated by the applicant; however, this is compensated, to some extent, by the loss of the irradiated fuel, As modified in Amendment No. 5, the irradiated stainless steel rods will be considered in the site-specific dose calculations, and the TS dose rate limits would prohibit an assembly with a significant increase in dose due to the presence of the stainless steel rods. On this basis, the staff found the definition modification acceptable.

6.9.2 E2—Hafnium Rods

Hafnium absorber rods are explicitly added to Appendix B of the CoC as CRA absorber materials. FSAR Section 5.2.3.2 states that hafnium rods are bounded by the AgInCd (silver-indium-cadmium) rods because the activation of the Ag in the AgInCd rods produces a greater source term than the hafnium rods. Because this an accurate description of the reaction rates of neutrons with hafnium and AgInCd, the staff determined that the inclusion of the hafnium rods would not result in an increase in dose rates.

6.9.3 E3—Add Definition of DFI to Appendix A and FSAR.

The applicant added definition of DFI to support the use of the DFI. The use of the DFI is discussed in Section 6.5 of this SER.

6.9.4 E4—Allow Slightly Thermally Discrepant Fuel Assembly per Quadrant

The applicant has included language in CoC Appendix B, Section 2.3 to allow one slightly thermally-discrepant fuel assembly per quadrant to be loaded as long as the peak cladding temperature for the MPC remains below the SFST-ISG-11, Revision 3 (NRC, 2007a) requirements. The staff determined that one out of 9 or 10 for the MPC-37 or one out of every 22 or 23 fuel assemblies for the MPC-89 that has a slightly higher heat load would not affect dose rates significantly. The staff found that this language should prevent a fuel assembly with a significantly higher source term from been loaded and would expect that the discrepant assembly to exceed the allowable decay heat by only a small fraction of the currently allowable decay heat. This would mean that the reference decay heat to determine allowable burnup/cooling time would likely be the same since Holtec generally uses a slightly higher reference decay heat for generating design basis loading patterns. As noted in Section 4.0 of this SER, the staff determined that the additional language would have negligible impact on the cask performance.

6.9.5 E5—Correct Number of Fuel Rod Locations for the 16x16C

In CoC Appendix B, the applicant corrected a typographical error in Table 2.1-2 under the 16x16C fuel class to change the number of fuel rod locations from 236 to 235. The staff found that this has no impact on dose rates as the number of fuel rod locations is not as significant for shielding evaluation as the assembly is represented as a homogenized mass of fuel and hardware within the shielding evaluation. This fuel assembly type is represented by the WE 17x17 in Table 5.2.1 of the SAR and reducing number of fuel rod locations is still bounded by the mass in this table which is the most significant consideration with respect to fuel geometry in determining the source term.

6.10 Evaluation Findings

F6.1 Sections 1, 2, and 6 of the FSAR describe the SSCs important to shielding safety in sufficient detail to allow evaluation of their effectiveness.

F6.2 Sections 1, 2, and 6 of the FSAR provide reasonable assurance that the radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F6.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The HI-STORM FW system shielding features are designed to assist in meeting these requirements.

The staff has reasonable assurance that the design of the shielding system of the HI-STORM FW system Amendment No. 5 is in compliance 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 FW system Amendment No. 5 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 this review is to ensure that the applicant had performed adequate criticality evaluation to demonstrate the system will remain subcritical under all credible normal, off-normal, and accident conditions during handling, packaging, transfer, and storage. The staff's review involved ensuring that the requested changes meet the regulatory requirements of 10 CFR 72.124(a), 72.124(b), 72.236(c), and 72.236(g).

The following proposed changes are applicable to the criticality evaluation:

- Proposed Change #2: Add four new fuel types, 10x10I, 11x11A, 7x7C, and 8x8G, to the approved contents listed in Appendix B.
- Proposed Change #5: Add the use of DFI in CoC Appendix A.
- Proposed Change #6: Add two versions of the standard HI-TRAC VW: Version V has a natural circulation feature, and Version V2 has the option for removable neutron shield.
- Proposed Change #8: Add fuel assemblies containing BLEU as approved contents.

In reviewing these changes on the HI-STORM FW system's criticality safety design, the staff followed the guidance in Chapter 6 of NUREG-1536, Revision 1 (NRC, 2010).

7.1 Criticality Design Criteria and Features

The HI-STORM FW cask system consists of a welded metallic MPC that is contained in a steel and concrete overpack. The previously approved MPCs for the HI-STORM FW system include the MPC-32, MPC-37, MPC-32ML, and MPC-89. The HI-STORM FW system uses the previously approved HI-TRAC VW transfer cask and proposed two new versions of the cask as described above (Proposed Change #6). The transfer cask designs rely on soluble boron in the MPC for PWR fuel as listed in the TS for criticality safety control during both loading and unloading operations, with the minimal soluble boron concentration. The operations for use of the new transfer casks, HI-TRAC VW Versions V and V2, are similar to the previously approved VW transfer cask and no impact on the criticality safety of the HI-STORM FW system was noted by staff.

The applicant's analysis in support of the changes sought in this amendment builds upon the NRC's previously approved application and subsequent approved amendments; and relies on those analyses as a foundation for the proposed changes. The applicant's safety analysis report focuses on the proposed changes with respect to this amendment.

As previously reviewed and approved by the staff in prior amendments to this system, the HI-STORM FW system relies on controlling the contents, the use of favorable MPC geometry, the minimal borated neutron poison (B-10) loading of the poison plates, and soluble boron for criticality safety. The MPC is composed of a steel cylinder and borated Metamic-HT plates to form the fuel basket. The specific areal density of B-10 in the neutron poison plates and the structural integrity of the Metamic-HT has been evaluated and approved by the NRC in the previous amendments of this system.

Damaged fuel may be stored in the HI-STORM FW system using the DFCs that have been previously approved in the initial application. This application seeks to add the ability to use DFIs in the cask using the MPC-37, MPC-89, and MPC-32ML baskets. As defined by the applicant, damaged fuel stored in a DFI must be able to be handled by normal means. If

damaged fuel is not able to be handled by normal means, then the damaged fuel must be stored in a DFC as previously approved by the NRC.

7.2 Fuel Specification

The applicant provided the specifications for the new fuels that will be allowed to be stored in the HI-STORM FW system in the various approved MPCs in the FSAR. The fuel characteristics include: number of fuel rods, maximum fuel pellet outer diameter, minimum fuel clad outer diameter, maximum fuel clad inner diameter, maximum fuel rod pitch, lattice geometry, maximum active fuel length, number of guide and/or instrument tubes, maximum guide and/or instrument tube thickness, and maximum U-235 enrichment.

NRC staff reviewed the new fuel parameters for the 10x10I, 11x11A, 7x7C, and 8x8G fuel types and found that the applicant provided the information necessary for the staff to perform a detailed criticality safety review of this amendment. The cask design does not take any burnup credit for the fuels. The addition of BLEU fuels is treated the same as other normal UO_2 fuels for criticality analysis because BLEU fuels and normal UO_2 fuels are identical from criticality perspective. The exception is that BLEU fuels may have higher cobalt impurity levels, which has a positive impact on the criticality safety because higher Co-60 reduces the reactivity.

7.2.1 Non-fuel Hardware

As described in the TS for the MPC basket designs, non-fuel hardware is an allowable content in the HI-STORM FW system. The applicant analyzed the effects on reactivity for displaced moderator and borated water for the new proposed fuel types and found that the displacement of soluble boron versus moderator tends to have a negative effect on the system reactivity when at the levels indicated in the FSAR. The staff finds this acceptable because it is consistent with the fundamentals of reactor physics.

7.2.2 Fuel Condition

The HI-STORM FW MPCs are designed to store intact fuel assemblies, damaged fuel assemblies and fuel debris in DFCs, and damaged fuel assemblies that are able to be handled by normal means in DFIs. The applicant performed additional criticality safety analyses to support the use of DFIs for the various fuel types in specific configurations as specified in loading patterns. The results are shown in FSAR Tables 6.1.7 and 6.1.8 for each combination of the MPC design and DFC/DFI location pattern, and specify maximum number of DFC/DFIs, the fuel assembly classes, the bounding maximum k_{eff} value, the associated maximum allowable enrichment, and, if applicable, the minimum soluble boron concentration.

NRC staff reviewed the applicant's criticality safety analyses for the new fuel types in the various MPC configurations. The staff finds that the applicant used the same methodology and assumptions used in previously approved HI-STORM FW system amendments and the safety analysis results show that system remains subcritical under normal, off-normal, and accident conditions.

7.3 Model Specification

The applicant provided detailed criticality analyses for the new fuel design classes (i.e., 10x10I, 11x11A, 7x7C, and 8x8G) in the applicable MPCs. For each canister and allowable fuel type, the applicant used a fresh fuel assumption to maximize reactivity and provide a substantial

conservatism for all of the HI-STORM FW system analyses since spent fuel would contain far less fissile material than that of the fresh fuel.

The applicant's models take credit for 90% of the B-10 in the Metamic-HT poison plates in all of the criticality calculations for all of the MPC designs, which has been previously approved by the NRC for the HI-STORM FW cask design in the initial certificate. Computer models evaluate the MPCs in the transfer cask under normal, off-normal, and accident conditions, and includes fabrication tolerances with respect to their most reactive configurations. The soluble boron concentrations used in the models was as specified in the TS for the various PWR fuel canister configurations.

The applicant used the same modeling approach for the new fuel classes as was used in the analyses for the previously approved canister designs. The models define the fuel rods and cladding, guide tubes, neutron absorbers, as well as the MPC shell and overpack. Flooded conditions assumed fresh water present in the fuel rod pellet-to-clad gaps since this was the bounding configuration. For evaluation of the DFIs with damaged fuel assemblies, the MPC basket cell ID is assumed as fuel boundary since this maximizes the area of the optimum moderated fuel.

The staff reviewed the applicant's models (Holtec, 2018a, Attachment No. 10) and the drawings included in FSAR Chapter 1 to ensure that the appropriate dimensions important to criticality safety were used in all calculations.

7.3.1 Model Configuration

Based on the previously approved structural analysis, neither the canister nor the fuel in the canister will experience any changes in geometry during normal and off-normal operating conditions that would affect the criticality safety of the HI-STORM FW cask. Therefore, the applicant used the cask and fuel geometry dimensions as designed for the criticality safety analyses. Staff finds this acceptable since the geometry would not change under these conditions.

Under the accident conditions specified in 10 CFR 72.122, the structural analyses demonstrate that neither the fuel nor the MPC basket would deform to the extent it would affect the geometry of the package for criticality purposes. Staff finds this acceptable since the models used to evaluate the criticality safety of the HI-STORM FW system are consistent with the damaged conditions as demonstrated in the structural analyses.

The applicant modeled the MPCs containing the new fuel types (i.e., 10x10I, 11x11A, 7x7C, and 8x8G) as specified in FSAR Table 2.1.3. The applicant evaluated full and partial flooding for the various MPCs, moderator density, and flooding in the pellet-to-clad gap using the same methodology used in previous applications. The staff finds the applicant's analyses and assumptions conservative for the proposed fuels and cask configurations because the amount of interstitial moderation is maximized for each fuel type. Because of this conservatism, and because the methods used in the evaluation were previously reviewed and approved by the NRC, the staff finds the analyses acceptable.

The applicant also modeled the new fuels, as well as the use of DFIs and DFCs for the various MPC designs and provided the results in FSAR Tables 6.1.2, 6.1.4(a), 6.1.4(b), 6.1.5, 6.1.7(a), 6.1.7(b), and 6.1.8. The applicant also provided the results of their studies on the impact of moderator density and indicated the maximum reactivity of each configuration and the minimum

levels of boron in the water for each PWR MPC configuration. Staff found that the reactivities were reasonable in all instances and finds the applicant's conclusions acceptable and in close agreement with staff confirmatory calculations.

The applicant considered flooding effects in the fuel rod pellet-to-clad gap regions and provided conservative assumptions similar to those used in previously approved analyses. The applicant calculated the k_{eff} for each of the cask configurations with the new fuels and the use of DFIs in the same locations as had been previously approved for DFCs, and provided the results in FSAR Tables 6.4.6, 6.4.7, and 6.4.11. In all instances the resultant neutron multiplication factors were below the upper subcriticality limit, including all bias and bias uncertainty. The staff finds that the calculated k_{eff} values are below the acceptance criteria for criticality safety as described in NUREG-1536, Revision 1, and therefore finds that the HI-STORM FW cask design for the storage of the proposed new fuel types, as well as the use of DFIs for damaged fuel that are able to be handled by normal means, meets the criticality safety requirements of 10 CFR 72.236(c).

7.3.2 Material Properties

In support of the requested changes to the HI-STORM FW system Amendment No. 5, the applicant provided updated material compositions and densities for all additional materials used in the applicant's computer models in Table 6.3.4. Staff verified the material properties are comparable with the SCALE computer code material property library that is an industry standard for standard compositions used in nuclear reactivity calculations.

As previously approved by the NRC, the applicant assumes 90% of the B-10 content for the Metamic-HT fixed neutron absorber used in the MPCs. The acceptance testing to assessment of the material is unchanged and continues to comply with regulatory requirements.

The applicant used the same fuel density of the normal UO_2 fuel for all fuel rods, including those containing gadolinia poison. Gadolinia was not taken credit for in any fuel configurations. This approach yields a modeled fuel that assumes more UO_2 per fuel assembly, this assumption is conservative because it will result in a higher calculated k_{eff} due to the higher fuel density. Therefore, staff find that this approach is conservative and appropriate for this application.

Staff finds that the material properties used in the criticality safety models for the proposed changes to the HI-STORM FW system Amendment No. 5 are consistent with the material compositions typically used for reactivity modeling and meet the guidance in NUREG-1536 for material properties.

7.4 Criticality Safety Analysis

7.4.1 Computer Programs

The applicant used MCNP5 three-dimensional Monte Carlo code and continuous energy cross sections in performing its criticality safety analyses. The MCNP5 code and cross sections are developed and validated by Los Alamos National Laboratory and is listed in NUREG-1536, Revision 1 (NRC, 2010) as an appropriate code for criticality safety analyses for dry cask systems. On these bases, the staff finds the use of this code appropriate for this application.

7.4.2 Effective Neutron Multiplication Factor, k_{eff}

The applicant provided the results of its criticality safety analyses for the new fuel types and DFI/DFC configurations in FSAR Tables 6.1.2, 6.1.4(a), 6.1.4(b), 6.1.5, 6.1.7(a), 6.1.7(b), 6.1.8, 6.4.6, 6.4.7, and 6.4.11. The calculated multiplication factors for each analyzed configuration indicated that the maximum k_{eff} for all evaluated scenarios was below the 0.95 limit with all bias and bias uncertainty included, which provides reasonable assurance that the cask system would remain subcritical under all operating conditions, including normal, off-normal, and postulated accidents. The staff finds the maximum multiplication factor for all configurations meets the criticality safety guidance as specified in NUREG-1536, Revision 1.

7.4.3 Confirmatory Analyses

NRC staff performed confirmatory calculations using the UNF-ST&DARDS. This tool utilizes the SCALE system of Monte Carlo code analysis to run parametric studies on various storage and transportation designs. The staff models for the various new fuel types that were built by ORNL using the design basis data from the FSAR. The UNF-ST&DARDS tool provides an explicit model of each fuel rod throughout the storage cask and allows staff to alter the design basis models to account for rod shifting, boron concentration levels, rod/assembly replacement, moderation levels, and fuel tolerances.

Staff evaluated the bounding fuel assemblies and the use of DFI/DFCs in the HI-STORM FW cask for the various configurations of MPCs using SCALE 6.2 and ENDF/B-VII continuous energy cross section library and found that the results agreed with the analyses performed by the applicant. The staff's analyses confirm that the applicant's criticality safety analyses are appropriate for the systems modeled and provides reasonable assurance of the criticality safety of the HI-STORM FW system with the proposed changes the applicant has requested.

7.4.4 Computer Code Benchmarking

The applicant provided comprehensive benchmarking analyses for the MCNP5 code and cross section set. This benchmarking method and analysis is provided in FSAR Appendix 6.A.

NRC staff reviewed the applicant's analyses and found that the benchmarking that had been previously approved for the HI-STORM FW system was still applicable for the new fuel types proposed in this amendment, and therefore continue to be acceptable in both the applicability and bias determination.

7.5 Evaluation Findings

The staff finds the following with respect to the criticality safety of the HI-STORM FW system design with the proposed changes of Amendment No. 5:

F7.1 The structures, systems, and components important to criticality safety are described in sufficient detail in the FSAR to enable an evaluation of their effectiveness.

F7.2 The spent fuel transfer systems (HI-TRAC VW, Version V, and Version V2) of Amendment No. 5 meet the requirements of 10 CFR 72.236(c), and that the systems remain subcritical under all expected normal and off-normal conditions of operations, as well as design basis accident conditions.

F7.3 The criticality safety design is based on favorable geometry, fixed neutron poisons, and soluble boron poison. The evaluated fixed neutron poison has demonstrated that they will remain effective and there is no credible means for the poisons to significantly degrade for the design basis life of the cask, and therefore, there is no need to provide a positive means to verify their continued efficacy under 10 CFR 72.124(b).

F7.4 The analyses and evaluation of the criticality design and performance have demonstrated that the HI-STORM FW system Amendment No. 5 will be adequate to allow the storage of spent fuel for the term specified in the CoC.

NRC staff concludes, based on its review, that the criticality design features for the HI-STORM FW System Amendment No. 5, is in compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c). The evaluation of the criticality safety design provides reasonable assurance that the system, as amended, will allow for the safe storage of spent nuclear fuel. These findings were reached on the evaluation of the applicant's changes and the staff confirmatory calculations, and considered the regulations of 10 CFR Part 72, applicable regulatory guides, codes and standards, and accepted engineering practices.

8.0 MATERIALS EVALUATION

The staff reviewed the proposed changes to the HI-STORM FW CoC Amendment No. 5 against the appropriate regulations as described in 10 CFR 72.236 to verify that the applicant performed adequate materials evaluation to ensure adequate material performance of components important to safety under normal, off-normal, and accident conditions. The staff's review followed the guidance in Sections 8 of NUREG-1536, Revision 1 (NRC, 2010), as well as associated ISG documents, to reach reasonable assurance of adequate materials performance under normal, off-normal, and accident-level conditions. The staff's review of the application identified a limited number of changes associated with the materials evaluation areas listed in Section 8.2 of NUREG-1536, Revision 1.

The following proposed changes are applicable to the materials evaluation:

- Proposed Change #1: (a) Add new heat load patterns for the MPC-89 and MPC-37 (long, standard, and short length). (b) Revise the minimum required cooling time for fuel to 1 year for MPC-89 and MPC-37.
- Proposed Change #2: Add four new fuel types, 10x10I, 11x11A, 7x7C, and 8x8G, to the approved contents listed in Appendix B.
- Proposed Change #3: Allow an exception to the ASME Code to use certain duplex stainless steels in the HI-STORM FW system.
- Proposed Change #5: Add the use of DFI in CoC Appendix A.
- Proposed Change #6: Add two versions of the standard HI-TRAC VW: Version V has a natural circulation feature, and Version V2 has the option for removable neutron shield.
- Proposed Change #7: Add the option of using cyclic vacuum drying for all MPCs.

The changes proposed in Amendment No. 5, including the new heat load patterns for the MPC-89 and MPC-37, did not result in changes to the maximum temperatures and pressures for the HI-STORM FW system as previously reviewed and approved by the staff in the original certificate and subsequent amendments to this system. The operating environmental conditions are unchanged for the HI-STORM FW system MPCs, transfer casks, and storage overpacks from the previously approved systems.

The applicant provided the updated technical specifications, updated FSAR, and proposed revisions to the CoC to support the proposed Amendment No. 5 changes.

8.1 Cask Design/Materials, Engineering Drawings, and Environmental Conditions

This section addresses the following proposed changes:

- Change #1: Addition of new heat load patterns for the multipurpose canister (MPC)-89 and MPC-37. Revision of minimum required cooling time for fuel to 1 year for MPC-89 and MPC-37. The new heat load patterns include locations for damaged fuel at higher per cell heat load limits.
- Change #5: Add DFI to CoC No. 1032 Appendix A.
- Change #6: Add two versions of the standard HI-TRAC VW transfer cask. Version V adds a natural circulation feature. Version V2 adds the option for removable neutron shield.
- Change #7: Add option for cyclic vacuum drying for all MPCs.

8.1.1 Damaged Fuel Isolators

The applicant's description of the proposed changes includes the addition of the loading of damaged fuel assemblies confined with a DFI that functions as top and bottom end caps in the fuel basket to confine the damaged fuel assemblies within individual basket cells. The top and bottom end caps are provided with screens at the bottom and top to contain fuel debris and allow filling/drainage of water during loading operations. The applicant provided schematic drawings for the endcaps and identified the design features and critical characteristics. The applicant stated that the DFI is constructed from corrosion resistant materials such as stainless steel or a nickel-based alloy.

The applicant stated that the DFI is a set of specially designed barriers at the top and bottom of a storage cell space used to prevent the migration of fissile material in bulk or coarse particulate form from the nuclear fuel stored in its cellular storage cavity. DFIs are not used to handle the fuel assembly and do not provide assistance in the ability to handle the fuel assembly during normal, off normal, or accident conditions. The applicant identified the allowed locations for using DFIs for the MPC-89 and the MPC-37.

The staff reviewed the description of the DFI provided by the applicant, including the schematic drawings, the design features, critical characteristics, and materials of construction. The staff notes that SFST-ISG-1, Revision 2, "Classifying the Condition of Spent Nuclear Fuel for interim Storage and Transportation Based on Function," (NRC, 2007a), considers gross breach of fuel cladding as any cladding breach greater than 1 millimeter. Gross breaches of fuel cladding may permit the release of fuel particulates. The staff determined that the DFI design which specifies that the cap walls shall have perforation with a maximum size of 1 millimeter would keep any gross particulate fissile material inside the basket cell. The staff determines that the design of the DFI is acceptable because the perforation size of 1 millimeter allows the fuel to be dried and is sufficient to retain fuel particulates released from the basket cell with the damaged fuel.

8.1.2 HI-TRAC VW Transfer Cask Version V and Version V2

The applicant described the addition of two variations of the HI-TRAC VW transfer cask design, the Version V and the Version V2. The HI-TRAC VW Version V adds a natural ventilation feature to the transfer cask design. The applicant stated that the natural ventilation pathway integrated into the design slightly increases the inner diameter of the HI-TRAC and adds an air inlet on the HI-TRAC's bottom lid. The applicant stated that the inlet and HI-TRAC annulus are configured to minimize leakage of radiation, and the Version V is designed to ensure there is no change in the structural capacity of the HI-TRAC. The applicant provided licensing drawings that include dimensions, tolerances, and material specifications. The applicant also provided component safety classification information for the HI-TRAC Version V SSCs (FSAR Table 2.0.11). The applicant provided the HI-TRAC Version V licensing drawing which shows the natural ventilation passages installed on the standard HI-TRAC VW configuration.

The applicant described the HI-TRAC Version V2 as a second variation to the HI-TRAC VW transfer cask design which adds a removable NSC. The applicant stated that the Version V2 with the removable NSC is designed to ensure radiation protection to the loading crew and accommodate weight limitations and geometric constraints of the cask pit. The applicant provided licensing drawings that include dimensions, tolerances, and material specifications. The applicant also provided component safety classification information for the HI-TRAC Version V2 SSCs (FSAR Table 2.0.12).

The staff reviewed the revised licensing drawings and safety classification information for the HI-TRAC Version V and Version V2 transfer casks and determined that the revised drawings and safety classification information provided by the applicant are acceptable because they follow the guidance included in NUREG/CR-5502 (NRC, 1998) and NUREG/CR-6407 (NRC, 1996). The staff has determined that the engineering drawings and safety classification tables include an adequate description of the storage system, material specifications contents and details of the storage system design features that provide the important to safety functions of the system.

8.1.3 Storage System Component Temperature Limits

The applicant identified temperature limits for the HI-STORM FW system components in FSAR Table 2.2.3. Specifically, the applicant updated the information contained in Table 2.2.3 to address the addition of the HI-TRAC Version V2 including temperature limits for the NSC Steel and the Holtite-A neutron absorber material. In addition, the applicant added FSAR Table 1.A.6 which identifies temperature limits for the MPC confinement boundary shell constructed from duplex stainless steel.

The applicant provided maximum steady state temperatures for the MPC components and HI-TRAC VW and Version V components in FSAR Table 4.5.2. The applicant showed that the MPC component temperatures and transfer cask component temperatures were below the temperatures limits specified in FSAR Table 2.2.3. The applicant showed that the MPC component temperatures and transfer cask component temperatures were reduced for the HI-TRAC VW with Version V features compared to the temperatures using the standard HI-TRAC VW. The applicant provided maximum steady state temperatures for the fuel cladding using the HI-TRAC VW Version V2 in FSAR Table 4.5.23 and showed that the MPC component temperatures and transfer cask component temperatures were below the temperatures limits specified in FSAR Table 2.2.3.

The applicant provided analyses of maximum temperatures for moderate burnup fuel cladding, high burnup fuel cladding, and MPC and HI-TRAC components under vacuum drying conditions in FSAR Tables 4.5.6 and 4.5.7. In addition, the applicant provided maximum temperatures for moderate burnup fuel in the MPC-37 (loading pattern 37C1 shown in FSAR Figure 1.2.3a) in FSAR Table 4.5.20. and moderate burnup fuel in the MPC-89 (loading pattern 89A1 shown in FSAR Figure 1.2.6a) in FSAR Table 4.5.21. The applicant also showed that the maximum temperatures of the MPC and Transfer cask components during a 100% vent blockage of the HI-TRAC Version V in FSAR Table 4.6.8 and for the HI-TRAC Version V2 in FSAR Table 4.6.10. In all cases, the applicant stated that the fuel cladding, MPC component temperatures, and transfer cask component temperatures were below the temperatures limits specified in FSAR Table 2.2.3.

The staff reviewed the information provided by the applicant in FSAR Table 2.2.3 and compared the allowable temperature limits to the allowable limits for these materials in the ASME B&PV Code Section II Part D (ASME, 2007) or supporting information provided by the applicant for non-code materials such as Holtite-A and Metamic-HT. The staff determined that the temperature limits listed by the applicant in FSAR Table 2.2.3 for the cask carbon steel and low alloy steel in the HI-TRAC transfer cask were acceptable because the allowable temperatures were consistent with the temperature limits for these materials in ASME B&PV Code Section II Part D. The staff compared the temperature limits included in FSAR Table 1.A.6 to the temperature limits in ASME Code Case N-635-1 which provides temperature limits for the use of duplex stainless steel unified numbering system (UNS) S31803 for ASME Section III applications. The staff determined that the allowable temperature for the MPC confinement boundary using UNS S31803 duplex stainless steel provided by the applicant was acceptable because the temperature limit was consistent with the limits listed in ASME Code Case N-635-1 (ASME, 2003), which is approved by the NRC in Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 34, issued in October 2007 (NRC, 2007b).

8.1.4 Radiation Effects

The applicant did not provide an analysis of the potential degradation due to irradiation of the stainless steel MPC, carbon steel components of the HI-STORM FW system, or aluminum components of the MPC. Previous assessment of neutron fluence have been conducted for dry storage systems. For dry storage systems, a neutron flux of 10^4–10^6 neutrons/square centimeter-second (n/cm^2-s) [6.5×10^4–6.5×10^6 neutrons/square inch-second (n/in^2-s)] is typical (NUREG/CR-7116, NRC, 2011a). At these flux levels, the accumulated neutron fluence after 60 years is about 10^{13}–10^{15} neutrons/square centimeter (n/cm^2) [6.5×10^{13}–6.5×10^{15} neutrons/square inch (n/in^2)].

NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report" (NRC, 2019b) includes an assessment of the effects of neutron radiation on stainless steels, carbon steels, and aluminum alloy materials. For stainless steels, Gamble (2006) found that neutron fluence levels greater than 1×10^{20} n/cm^2 [6.5×10^{20} n/in^2] are required to produce measurable degradation of the mechanical properties. Caskey et al. (1990) also indicates that neutron fluence levels of up to 2×10^{21} n/cm^2 [1×10^{22} n/in^2] were not found to enhance SCC susceptibility. For carbon and alloy steels, neutron irradiation has the potential to increase the tensile and yield strength and decrease the toughness of carbon and alloy steels (Nikolaev et al., 2002). Neutron fluence levels greater than 10^{19} n/cm^2 [6.5×10^{19} n/in^2] are required to produce a measurable degradation of the mechanical properties (Nikolaev et al., 2002; Odette and Lucas, 2001). For aluminum alloys, Farrell and King (1973) showed that pure aluminum had increased strength

but decreased ductility after being irradiated to fast fluences in the range of 1 to 3 × 10^{22} n/cm^2 [6.5 to 18 × 10^{22} n/in^2] from a research reactor for 8 years. Alexander (1999) showed that irradiation at 10^{22} n/cm^2 [6.5 × 10^{22} n/in^2] simulating reactor conditions affected the mechanical properties of aluminum alloy 6061-T651.

To verify the conservatism of the previous estimate of accumulated neutron fluence in NUREG/CR-7116 (NRC, 2011a), the NRC staff performed an independent calculation of the maximum potential accumulated neutron fluence on the dry storage system components (NRC, 2019b). The staff considered components most directly exposed to the radiation source (middle of the fuel basket) and assumed fuel is loaded immediately after it is removed from the reactor vessel and stored for 100 years. To further provide a bounding estimate, the staff assumed a cask design that uses 40 Westinghouse 17×17 PWR fuel assembles with an average burnup of 70 Gwd/MTU and 4.0% fuel enrichment. The staff calculated the neutron source term for neutrons with energy at or greater than 1 MeV using the Origen/Arp computer code of the SCALE 6.1 computer code system. At this location, the total accumulated neutron fluence after 100 years of storage was calculated to be 2.63 × 10^{16} n/cm^2 [1.70 × 10^{17} n/in^2]. This worst-case estimate is greater than that calculated using the flux levels reported in NUREG/CR-7116 (NRC, 2011a); however, the staff determined the fluence level is still three orders of magnitude below the levels reported to degrade the fracture resistance of carbon and alloy steels, stainless steels, and aluminum alloys (NRC, 2019b). Therefore, the staff concluded that the changes proposed in the application are acceptable because the neutron fluence is insufficient to result in a degradation of material properties of the storage system components.

8.2 Materials Selection, Material Properties, Applicable Codes and Standards

This section addresses the following proposed changes:

- Change #3: Add an exception to the ASME B&PV Code to allow the use of certain duplex stainless steels in the HI-STORM FW system.
- Change #5: Add DFI to CoC No. 1032 Appendix A.
- Change #6: Add two versions of the standard HI-TRAC VW transfer cask. Version V adds a natural circulation feature. Version V2 adds the option for removable neutron shield.

8.2.1 MPC Duplex Stainless Steel, Material Properties, Weld Design and Inspection

The applicant stated that as an alternative to ASME SA-240 and SA-182 austenitic stainless steels (Types 304, 304LN, 316, and 316LN), the MPC shell material may be constructed using UNS S31803 duplex stainless steel, which has improved corrosion resistance. The UNS S31803 duplex stainless steel is not included in ASME B&PV Code, Section II, Part D, Subpart 1, Tables 2A and 2B, for design stress intensity. However, the applicant stated that UNS S31803 has been accepted for Class 1 components by ASME B&PV Code Case N-635-1 (ASME, 2003). The applicant included the use of duplex stainless steel UNS S31803 as an ASME Code Alternative in HI-STORM FW FSAR Table 2.2.14.

The applicant provided a general description of the duplex stainless steels in FSAR Appendix 1.A. Tables 1.A.1 through 1.A.5, which tabulated the design stress intensity, tensile strength, yield stress values, coefficient of thermal expansion and thermal conductivity for both wrought plate (ASME SA-240) and forged (ASME SA-182) materials, were revised to include the mechanical properties of duplex stainless steel S31803. The applicant also added Table 1.A.6, which includes temperature limits for duplex stainless steel materials, and Table 1.A.7, which

has the moduli of elasticity for the austenitic stainless steel grades and the duplex stainless steel as a function of temperature. Mechanical property values for the austenitic stainless steels were obtained from ASME B&PV Code Section IID (ASME, 2007) and values for the duplex stainless steels were obtained from ASME B&PV Code Case N-635-1 (ASME, 2003).

As described in the revision to FSAR Appendix 1.A, the optional duplex stainless steels and their welds are evaluated for susceptibility to brittle fracture by Charpy V-notch fracture toughness testing in accordance with ASTM A923-14 (ASTM, 2015) which states that for UNS S31803 duplex stainless steels, the minimum Charpy impact energy is 40 ft-lb (54 Joules [J]) for the base metal and the weld heat affected zone, and 25 ft-lb (34 J) for the weld metal.

The applicant has previously provided a revised calculation of the critical flaw size for the MPCs that will be manufactured from duplex stainless steel (Holtec, 2016a, 2016b). The revised critical flaw size calculation accounts for the acceptance criteria for welded duplex stainless steels in ASTM A923-14 and the potential loss of ductility at low temperatures for duplex stainless steels with a microstructure having approximately 50% ferritic phase.

The staff reviewed the composition requirements for the duplex stainless steels and confirmed that UNS S31803 is an accepted ASME B&PV Code alternative for Class 1, 2, and 3 systems. The staff finds that the material selection, materials properties, applicable codes and standards specified by the applicant using ASME B&PV Code Case N-635-1 (ASME, 2003) as an alternate material for the MPC is acceptable because the specifications are sufficient to control the chemical and physical properties of the material. The staff finds that the applicants use of ASME B&PV Code Case N-635-1 (ASME, 2003) which is endorsed by NRC Regulatory Guide 1.84 (NRC, 2007b) is acceptable and consistent with the recommendations in NUREG-1536, Revision 1 (NRC, 2010).

The staff reviewed the information contained in the FSAR sections and the information presented in the FSAR drawings to determine whether the selected materials are acceptable for their structural applications. The staff reviewed the mechanical property data provided by the applicant and verified that the values stated are consistent with the values in the ASME B&PV Code Section IID (ASME, 2007). The staff found the material properties used in the applicant's structural analyses to be acceptable and appropriate for the expected load conditions because the staff independently verified that the properties are based on consensus codes and standards or other technical references commonly used and accepted in the materials industry.

The staff reviewed the available information on fracture toughness testing of duplex stainless steels welds (Sieurin and Sandstrom, 2006; ASTM, 2015) and determined that Charpy V-notch impact testing is adequate to evaluate the effects of welding processes on the fracture toughness of duplex stainless steel-base alloys and welds. The staff reviewed the applicant's analysis of fracture toughness for duplex stainless steels and determined that the approach used by the applicant is adequate to account for the potential decreased fracture toughness of the duplex stainless steel-base alloys and welds observed at temperatures below 0°C (32°F). The staff determined that the applicant's analysis shows that the UNS S31803 duplex stainless steel-base alloy and welds will have sufficient fracture toughness to prevent brittle fracture at low temperatures under accident conditions.

The staff has previously reviewed the applicant's revised critical flaw size calculation (NRC, 2018) and determined that the calculation was acceptable because (1) the minimum fracture toughness of the duplex stainless steel weld exceeds the stress intensity factor corresponding to full circumferential 50% through-thickness crack oriented to maximize the potential for Mode

II (shear) failure; (2) the calculation considered the acceptance criteria for welded duplex stainless steels in ASTM A923-14 (ASTM, 2015); and (3) the calculation used a correlation equation (Roberts and Newton, 1981) that accounted for the potential loss of ductility at low temperatures for duplex stainless steels with a microstructure having approximately 50% ferritic phase.

8.2.2 Damaged Fuel Isolators

The applicant stated that the DFI design and the fabrication and inspection of DFI are in compliance with ASME BPVC, Section III Subsection NF. The DFI is made up of two end caps, along with the four cell walls, and comprise the fuel isolation space. The bottom cap is a prismatic box with a flat baseplate which fits inside the storage cell space with a small clearance (for ease of installation). The sidewalls of the bottom cap have perforations or wire mesh to permit transmigration of gases but not fuel fragments or gross particulates and is equipped with a flexible permeable barrier against the storage cell walls for sequestration of coarse particulate matter. The applicant provided temperature limits for the damaged fuel isolators under normal, off-normal, and accident conditions in FSAR Table 2.2.3.

The staff determined that the design and construction of the DFIs are appropriate for the intended use as an isolator for damaged fuel assemblies that can be handled by normal means. The staff determined that using DFIs instead of a DFC is an acceptable alternative for storing damaged fuel as described in Section 8.6.C of NUREG-1536, Revision 1 (NRC, 2010) because the use of DFIs is limited to fuel that (1) can be handled by normal means and (2) will not undergo geometric rearrangement under normal, off normal, and accident conditions. The staff determined that the materials of construction for the damaged fuel end caps and the FFCs were adequate because (1) the design includes screens at the bottom and top to contain fuel debris and allow filling/drainage and (2) these components are constructed to meet the loading requirements of ASME Code Section III Subsection NF and using ASME code approved materials with adequate temperature limits.

8.2.3 HI-TRAC Version V and Version V2

The applicant provided licensing drawings that included a bill of materials for the HI-TRAC Versions V and V2. The drawings refer to FSAR Tables 1.2.6 and 1.2.7 which identify the ASME Code paragraphs for the design and manufacturing of the transfer casks. In FSAR Table 1.2.6, the applicant identified that the transfer cask Versions V and V2 are designed and constructed in accordance with applicable portions of ASME Section III Subsection NF (ASME, 2007). The applicant also provided the safety classification for the H-TRAC Version V (FSAR Table 2.0.11) and for the HI-TRAC Version V2 (FSAR Table 2.0.12). Mechanical properties including design stress intensity, yield stress ultimate (tensile) stress modulus of elasticity, and mean coefficient of thermal expansion for the materials used in the important to safety components of the HI-TRAC Version V and Version V2 have previously been used in ITS components for the HI-STORM FW system and are included in Chapter 3 of FSAR, Revision 6 (Holtec, 2019g).

The applicant identified bolts for the H-TRAC Version V and Version V2 in the licensing drawings. The bolting materials are the same as the materials used in previously approved HI-TRAC transfer cask versions. The material properties of the bolting material including design stress intensity, yield stress ultimate (tensile) stress modulus of elasticity, and mean coefficient of thermal expansion are included in Chapter 3 of FSAR Revision 6 (Holtec, 2019g).

The NRC has previously approved storage systems designed in accordance with the ASME B&PV Code. Specifically, the NRC has accepted the design of confinement SSCs fabricated in accordance with ASME B&PV Code, Section III, “Rules for Construction of Nuclear Facility Components,” Subsection NB, “Class 1,” criteria (ASME, 2007) and the NRC has accepted the design of transfer casks to ASME B&PV Code Section III, Subsection NC, “Class 2,” criteria. For other safety related structures, the NRC has accepted components designed in accordance with ASME B&PV Code Section III, Subsection NF, “Supports.”

The NRC has previously reviewed and approved the HI-TRAC VW transfer cask that is designed and constructed in accordance with the 2007 ASME B&PV Code Section III Subsection NF (ASME, 2007). The staff reviewed the applicant’s description of the HI-TRAC Version V and Version V2 transfer cask design including the referenced code requirements for the design and construction identified in FSAR, Revision 6, Table 1.2.6 (Holtec, 2019g). The staff determined that the certification of material requirements that refence ASME Section III NF-2130(a) and (b) are appropriate for the HI-TRAC Version V and Version V2 which are designed and constructed in accordance with the 2007 ASME B&PV code Section III Subsection NF.

The staff reviewed the HI-TRAC Versions V and V2 bolting materials and confirmed that the materials specified are in ASME Section II Part D Table 3 which is appropriate for use in components designed and constructed in accordance with ASME Section III Subsection NF per NF-2121 (ASME, 2007). The staff determined that the materials selection and material specifications for bolts in the HI-TRAC Version V and Version V2 are acceptable because the applicable codes and standards specified by the applicant are sufficient to control the chemical and physical properties of the materials and appropriate for the referenced codes and standard for the transfer cask.

The staff reviewed the HI-TRAC Versions V and V2 drawing and the included bill of materials and safety classifications for the transfer cask components. The staff also reviewed previously approved HI-TRAC transfer casks for the HI-STORM FW system and determined that the HI-TRAC Version V and Version V2 transfer casks will use materials, design, and construction methods common to the previously approved designs. The staff determined that the materials selection and material specifications using applicable codes and standards specified by the applicant for the HI-TRAC Versions V and V2 included in the amendment application are acceptable because the specifications are sufficient to control the chemical and physical properties of the materials and consistent with the recommendations in NUREG-1536, Revision 1 (NRC, 2010).

8.2.4 HI-TRAC Version V2 Neutron Shielding Materials

The HI-TRAC Version V2 uses a Holtec proprietary neutron shielding material, Holtite-A. The applicant provided material properties for the Holtite-A neutron shielding material in a revision to FSAR Table 1.2.5. The applicant has previously used the Holtite-A neutron shielding materials in the HI-TRAC 125 and 125D transfer casks in the HI-STORM 100 system (CoC No. 1014). The material properties provided by the applicant are consistent with the information in Section 1.2.1.3.2 of HI-STORM 100 FSAR Revision 18 (Holtec, 2019i). The applicant revised FSAR Table 2.2.3 to include the short-term temperature limit for Holtite-A to be 300°F which is consistent with the design temperature of 150°C [302°F] for Holtite-A in HI-STORM 100 FSAR Revision 18, Table 1.B.1.

The staff have previously reviewed the use of Holtite-A as a neutron shielding material in HI-TRAC transfer casks. The staff confirmed that the information provided by the applicant on

Holtite-A is consistent with the information provided in previous applications for the use of Holtite-A. In HI-STORM FW system Amendment No. 5, there is no change to the material properties or design temperature for the Holtite-A neuron shielding material from that previously reviewed and approved in HI-STORM 100 System (CoC No. 1014).

8.2.5 Other Components and Materials

The MPC-89 and the MPC-37 baskets are constructed from Metamic-HT which is a Holtec proprietary (non-ASME code) material and the basket supports are fabricated from an aluminum alloy. In HI-STORM FW system Amendment No. 5, there is no change to the basket material specifications, materials testing, fabrication methods, non-destructive examination requirements, or acceptance criteria from that reviewed and approved by the staff in the original certificate and subsequent amendments.

Structural steel portions of the HI-STORM FW system overpacks are fabricated and inspected in accordance with the ASME B&PV Code, Section III, Subsection NF requirements (ASME, 2007) as shown in FSAR Table 1.2.6. Concrete portions of the overpack reference ACI 318-05 (ACI, 2005). The concrete density is specified in HI-STORM FW FSAR Table 1.2.5. Testing of the concrete portions of the overpack is performed in accordance with applicable ACI and ASTM standards identified in Section 1.D of the HI-STORM 100 FSAR (Holtec, 2019i). Alternatives to codes and standards are provided in FSAR Section 2.2.4 and Table 2.2.14. In HI-STORM FW system Amendment No. 5, there is no change to the overpack material specifications, materials testing, fabrication methods, non-destructive examination requirements, or acceptance criteria from that reviewed and approved by the staff in the original certificate and subsequent amendments.

8.3 Corrosion

This section addresses the following proposed changes:

- Change #2: Addition of four new fuel types, 10x10I, 11x11A, 7x7C and 8x8G to the approved contents for the MPC-89.
- Change #3: Add an exception to the ASME B&PV Code to allow the use of certain duplex stainless steels in the HI-STORM FW system.
- Change #5: Add DFI to CoC No. 1032 Appendix A.
- Change #6: Add two versions of the standard HI-TRAC VW transfer cask. Version V adds a natural circulation feature. Version V2 adds the option for removable neutron shield.

8.3.1 Addition of New BWR Fuel Types

The applicant proposed to add four new fuel types, 10x10I, 11x11A, 7x7C and 8x8G to the approved contents for the MPC-89. These BWR fuels use zirconium alloy cladding materials. All other hardware and non-cladding component of these fuel assemblies use materials such as zirconium alloys, stainless steels, or nickel-base alloys similar to the fuel assembly types that are approved for storage in the MPC-89. As such, in HI-STORM FW system Amendment No. 5, there is no change to the fuel assembly material specifications from that reviewed and approved by the staff in the original certificate and subsequent amendments.

The staff reviewed the four new fuel types for the MPC-89. The staff determined that the materials of construction for the new fuel types are similar to materials used in fuel assemblies that are currently approved for the MPC-89. The staff notes that NUREG-1536, Revision 1, Section 8.4.8.2, "Canister Contents" states that the staff has previously reviewed a number of non-fuel hardware components and materials for compliance with 10 CFR 72.120(d), meaning, compatibility with a canister interior composed of stainless steel and aluminum components, including zirconium alloy and stainless steel cladding and components. The staff has previously determined that nickel alloys used in some fuel assembly hardware components are also acceptable as they are compatible with canister interior components composed of stainless steel and aluminum alloys.

8.3.2 MPC Duplex Stainless Steel

The applicant stated that the use of duplex stainless steels is an option for the MPC shell (FSAR Section 1.2.1.1, "Multi-Purpose Canisters"). No change to the MPC weld design was included in Amendment No. 5. Construction of MPCs using duplex stainless steel S31803 will be similar to MPCs constructed from the previous austenitic stainless steel (Alloy X) materials. The applicant stated that while duplex stainless steels have increased resistance to stress corrosion cracking (SCC), however, the resistance to SCC is reduced when the microstructure of the steel is altered due to prolonged exposure to elevated temperatures or as a result of improper fabrication methods. The applicant identified fabrication and welding parameter controls including a range of weld heat inputs and a maximum interpass temperature necessary to prevent alteration of the weld and weld heat affected zone microstructures. The applicant also specified standardized evaluation tests with defined acceptance criteria to ensure that degradation of the mechanical properties and corrosion resistance of duplex stainless steels will not occur as a result of MPC construction and closure welding operations.

The staff also reviewed the available information on the chloride induced stress corrosion cracking (CISCC) resistance of duplex stainless steels (Tseng et al., 2003; Cottis and Newman, 1993). The staff determined that UNS S31803 duplex stainless steels are more resistant to CISCC compared to austenitic stainless steels; however, CISCC has been reported in offshore applications when welding practices were used that altered the microstructure of the alloy (Leonard, 2003). The staff reviewed studies conducted by Liou et al. (2002), which showed that increased nitrogen content in Type 2205 stainless steels was beneficial for maintaining a favorable ratio of ferrite to austenite phases that is necessary for CISCC resistance. The staff also found that cooling rates have a significant effect on the resulting microstructure of duplex stainless steels. Cooling rates above 0.41 °F/s are necessary to avoid embrittlement from the formation of sigma phase, but cooling rates above 90 °F/s result in an unfavorable ratio of ferrite to austenite and diminish CISCC resistance (Sieurin and Sandstrom, 2007). The staff reviewed the applicant's fabrication and welding parameter controls for duplex stainless steels and the applicant's specified standardized evaluation tests and acceptance criteria. Based on the review of information in the fabrication of duplex stainless steels, the staff concludes the fabrication and welding parameter controls and implemented evaluation tests specified by the applicant are adequate to ensure that degradation of the mechanical properties and corrosion resistance of duplex stainless steels will not occur as a result of MPC construction and closure welding operations.

8.3.3 Damaged Fuel Isolators

The applicant's description of the proposed changes includes the addition of the loading of damaged fuel assemblies confined with a DFIs that function as top and bottom end caps in the

fuel basket to confine the damaged fuel assemblies within individual basket cell. The top and bottom end caps are provided with screens at the bottom and top to contain fuel debris and allow filling/drainage of water during loading operations. The applicant provided schematic drawings for the damaged fuel endcaps, identified the design features and critical characteristics. The applicant stated that the DFI is constructed from corrosion resistant materials such as stainless steel or a nickel-based alloy.

The staff reviewed the applicant's description of the DFIs. The staff determined that the materials of construction for the DFIs are similar to materials used in currently approved MPCs. The staff notes that NUREG-1536, Revision 1, Section 8.4.8.2, "Canister Contents" states that the staff has previously reviewed a number of non-fuel hardware components and materials for compliance with 10 CFR 72.120(d), meaning, compatibility with a canister interior composed of stainless steel and aluminum components including zirconium alloy and stainless steel cladding and components. The staff has previously determined that nickel alloys used in some fuel assembly hardware components are also acceptable as they are compatible with canister interior components composed of stainless steel and aluminum alloys.

8.3.4 HI-TRAC Version V and Version V2 Materials and Coatings

The applicant provided licensing drawings that included a bill of materials for the HI-TRAC Version V and Version V2. The materials of construction are primarily carbon and low alloy steel along with some stainless steels. The transfer casks also have neutron and gamma radiation shielding materials that are encased in carbon and low alloy steel materials. The carbon and low alloy steel materials are coated to prevent corrosion while exposed to spent fuel pool water during loading operations. The applicant has previously described the coatings used for the transfer casks in Chapter 8 of FSAR Revision 6 (Holtec, 2019g), and in the application for HI-STORM FW system Amendment No. 5, there is no change to the coatings specifications for the transfer casks.

8.4 Cladding Integrity/Fuel

This section addresses the following proposed changes:

- Change #1: Addition of new heat load patterns for the multipurpose canister (MPC)-89 and MPC-37. Revision of minimum required cooling time for fuel to 1 year for MPC-89 and MPC-37. The new heat load patterns include locations for damaged fuel at higher per cell heat load limits.
- Change #2: Addition of four new fuel types, 10x10I, 11x11A, 7x7C and 8x8G to the approved contents for the MPC-89.
- Change #5: Add DFI to CoC No. 1032 Appendix A.
- Change #7: Add option for cyclic vacuum drying for all MPCs.

8.4.1 Cladding Temperature Limits

The applicant provided analyses of maximum temperatures for moderate burnup fuel cladding, high burnup fuel cladding, MPC and HI-TRAC components under vacuum drying conditions in FSAR Tables 4.5.6 and 4.5.7. In addition, the applicant provided maximum temperatures for moderate burnup fuel in the MPC-37 (loading pattern 37C1 shown in FSAR Figure 1.2.3a) in FSAR Table 4.5.20 and moderate burnup fuel in the MPC-89 (loading pattern 89A1 shown in FSAR Figure 1.2.6a) in FSAR Table 4.5.21. In all cases, the applicant stated that the fuel

cladding, MPC component temperatures, and transfer cask component temperatures were below the temperatures limits specified in FSAR Table 2.2.3. The applicant stated that the permissible time for heatup/cooldown cycles is a function of cask specific heat loads. At lower heat loads, the duration of vacuum drying cycles is increased and if the heat load is low enough, then the peak cladding temperature may remain below the SFST-ISG-11, Revision 3 (NRC, 2003a) limit under vacuum conditions indefinitely eliminating the need for cycling. However, with sufficient heat loads, the applicant stated that continuous vacuum drying may not be possible. In this case, the applicant stated that the recommendations of SFST-ISG-11, Revision 3 are to be followed which limits repeated thermal cycling to less than 10 cycles, with cladding temperature variations less than 65°C (117°F) each cycle.

The applicant provided maximum steady state temperatures for the fuel cladding with the HI-TRAC VW and Version V transfer cask in FSAR Table 4.5.2. The applicant showed that the fuel cladding temperatures were below the temperatures limits specified in FSAR Table 2.2.3. The applicant showed that fuel cladding temperatures were reduced for the HI-TRAC VW with Version V features compared to the temperatures using the Standard HI-TRAC VW. The applicant provided maximum steady state temperatures for the fuel cladding using the HI-TRAC VW Version V2 in FSAR Table 4.5.23 and showed that the fuel cladding temperatures were below the temperatures limits specified in FSAR Table 2.2.3.

The staff reviewed the applicant's calculated cladding temperatures to confirm that there is reasonable assurance that creep will not cause gross rupture of the cladding and that hydride reorientation will not degrade the mechanical properties of the cladding. The guidance in NUREG-1536, Revision 1 establishes a maximum fuel cladding temperature limit for normal storage conditions, short-term loading operations, off-normal, and accident conditions. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad). However, for low burnup fuel, a higher short-term temperature limit may be used, if the applicant can show by calculation the best estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed. For all fuel burnups, the cladding should be limited to a maximum temperature of 570°C (1,058°F) for off-normal and accident conditions. The staff reviewed the applicant's thermal analyses and confirmed that the applicant's calculated temperatures are below these maximum temperature limits. With respect to the applicant's cladding temperature for low burnup fuel under vacuum drying operations, the staff reviewed the analysis referenced by the applicant (Lanning and Beyer, 2004) and the similar work reported by Brown et al. (2004). The staff determined that the temperatures for the low burnup fuel during drying that exceed 400°C (752°F) but remain less than 570°C (1,058°F) are acceptable because the estimated cladding hoop stress is equal to or less than 90 MPa (13,053 psi) which is consistent with the guidance in NUREG-1536, Revision 1.

The staff evaluated the applicant's thermal cycling analysis to ensure that the loading operation will not result in conditions that could promote creep or hydride reorientation. The staff confirmed the loading operations addressed the recommendation in NUREG-1536, Revision 1 to avoid hydride reorientation by limiting thermal cycling in loading operations to less than 10 cycles where cladding temperature variations are more than 65°C. For these reasons, the staff found the applicant's loading operations acceptable.

8.4.2 Storage of Damaged Fuel

The applicant clarified that the DFI can be used for the storage of fuel assemblies classified as damaged because of physical defect, such as a breach in the fuel cladding or a structural failure in the grid strap assembly. Damaged fuel stored in DFIs may contain (1) missing or partial fuel rods (empty fuel rod locations that are not filled with dummy fuel rods) and/or (2) fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole. The applicant clarified that the DFI may only be used if the damaged fuel assembly can be handled by normal means and its structural integrity is such that geometric rearrangement of fuel is not expected under normal and off normal conditions. The applicant stated that damaged fuel that does not meet the criteria for using DFI must be stored using a DFC. The DFC design and its allowed contents have been previously approved by the NRC in the original certificate (NRC, 2011b).

The staff reviewed the application and the guidance included in NUREG-1536, Revision 1, Section 8.4.17.2 for fuel classification. The applicant has limited the storage of damaged fuel with the use of basket endcaps to fuel assemblies that can be handled by normal means after normal and off normal events. The applicant's approach, which included the use of endcaps to contain debris for damaged fuel is consistent with the guidance in NUREG-1536, Revision 1. The staff determined that the application was acceptable because the content of the application with respect to fuel classification was consistent with the guidance in NUREG-1536, Revision 1, Sections 8.4.17.2 and 8.6.C.

The staff determined that the applicant's specifications for damaged fuel and the functions of the DFIs meet the regulatory requirements of 10 CFR 72.236(h) and (m) and allow the system users to meet the regulatory requirements of 10 CFR 72.122(h)(1) and (h)(5). The thermal, shielding and criticality evaluations for the storage of damaged fuel is included in Sections 4, 6, and 7 of this SER, respectively.

8.5 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 important to safety in sufficient detail to support a safety finding.

F8.2. The applicant has met the requirements in 10 CFR 72.124(b). Neutron absorbing materials are demonstrated to effectively control criticality without significant degradation over the storage life.

F8.3. 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.

F8.4. The applicant has met the requirements in 10 CFR 72.236(h). The materials of the SNF storage container are compatible with their operating environment such that there are no adverse degradation or significant chemical or other reactions.

F8.5. 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.

9.0 OPERATING PROCEDURES EVALUATION

The objective of this review ensures 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 technical specifications.

The applicant revised FSAR Chapter 9, Operating Procedures, to address the following proposed changes:

- Proposed Change #5: Add the use of DFI in CoC Appendix A.
- Proposed Change #6: Add two versions of the standard HI-TRAC VW. Version V adds a natural circulation feature. Version V2 adds the option for removable neutron shield.
- Proposed Change #7: Add the option for using cyclic vacuum drying for all MPCs.

The applicant discussed specific changes to the MPC fuel loading procedure in FSAR Section 9.2.3 to address loading patterns, decay heat, cooling time, fuel condition and the use of DFCs and DFIs. The applicant revised the MPC unloading procedure in FSAR Section 9.4.4 to address the removal of the DFIs. The applicant also revised FSAR Table 9.2.4, MPC Inspection Checklist, to include the inspection of the DFIs if used. The staff reviewed the applicant’s revised loading and unloading procedure and determined that the procedures are acceptable because they address the required steps to load and unload the MPC using the revised loading patterns and the use of DFIs.

The applicant revised procedures to include the use of the HI-TRAC VW Version V and Version V2 transfer casks. The applicant stated that operationally, the procedures are similar to the originally approved HI-TRAC VW transfer cask. The applicant revised the procedures in FSAR Section 9.2.4, MPC Closure, to address the use of the NSC which uses Holtite-A neutron absorber instead of water for the HI-TRAC Version V2 transfer cask. The applicant included specific steps for the inspection of the inflatable annulus and monitoring of the ventilation channels for the HI-TRAC VW Version V transfer casks in FSAR Section 9.2 for loading operations as well as FSAR Section 9.4 for unloading operations. The staff reviewed the applicant’s revised procedures and determined that the procedure revisions are acceptable because they include the necessary steps and inspections for the loading and unloading operations with the HI-TRAC VW Version V and Version V2 transfer casks.

The applicant stated in FSAR Section 9.2.4, MPC Closure, that drying of high burnup fuel may be performed using vacuum drying with a time limit as specified in FSAR Chapter 4. The applicant stated that the vacuum drying procedure follows the guidance on maximum cladding temperature and thermal cycling provided in SFST-ISG-11, Revision 3 (NRC, 2003a). The staff reviewed the applicant’s revision to the drying procedure and determined that the revisions are acceptable because they are consistent with the guidance on temperature limits for high burnup fuel and thermal cycling during loading and drying operations included in SFST-ISG-11, Revision 3.

10.0 ACCEPTANCE TESTS AND MAINTANANCE PROGRAM EVALUATION

The objective of this review ensures 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 following proposed change:

- Proposed Change #6: Add two versions of the standard HI-TRAC VW. Version V adds a natural circulation feature. Version V2 adds the option for removable neutron shield.

The applicant stated in FSAR Section 1.2.1.4 that Holtite-A neutron shielding material will be used in the HI-TRAC VW Version V2. The applicant stated that each manufactured lot of material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) as indicated in FSAR Table 1.2.5.

The NRC has previously reviewed the testing data provided by the applicant on Holtite-A (Holtec, 2000) and approved the Holtite neutron shielding material for used in the HI-STAR 100 storage system (NRC, 2001b). The staff notes that Holtite-A has also been approved for used as a neutron absorber material in transfer casks used in the HI-STORM 100 system (Holtec, 2019i). There is no change to the material properties or design temperature for the Holtite-A neutron shielding material from what was previously reviewed and approved in the HI-STORM 100 System (CoC No. 1014). Therefore, the staff determined that the previously approved compositional requirement and acceptance tests for the Holtite-A neutron shielding material are acceptable for the HI-TRAC VW Version V2 as proposed in Amendment No. 5.

11.0 RADIATION PROTECTION EVALUATION

11.1 Proposed Change #1—New Heat Load Patterns

The applicant proposed to add new heat load patterns to the MPC-89 and MPC-37 (long, standard, and short length) and revise the minimum required cooling time to 1 year for MPC-89 and MPC-37.

11.1.1 Occupational Exposures

The staff reviewed FSAR Chapter 9, Operating Procedures, and Chapter 5, Shielding Evaluation, and determined that the data is appropriately used in Chapter 11, Radiation Protection.

To support the new heat load patterns and reduced cooling times, the applicant proposed a new fuel qualification strategy as discussed in Section 6.1.2 of this SER. The applicant updated FSAR Table 11.3.2, “Estimated Person-mrem Dose for Loading the HI-STORM FW system,” to account for the proposed source terms. The applicant adjusted the design basis source terms by a factor equivalent to the reduction in the TS dose rate at the side of the transfer cask. For example, the calculated maximum side dose rate for the transfer cask is in the FSAR Table 5.1.2b (5,898.2 mrem/hr). Even though this dose rate is very high, users would not practically be able to load a transfer cask that had this dose rate because they are limited by the TS dose rate in Appendix A Section 5.3.4.c (3,500 mrem/hr). All other dose rates used in FSAR Table 11.3.2 to calculate occupational dose were reduced by this factor (0.593 = 3,500/5,898.2). The staff found this scaling factor to be reasonable as a user would not be allowed to exceed the 3500 mrem/hr dose rate, despite the allowable contents.

The applicant has determined the maximum dose rates by using a different loading pattern that maximizes dose rate at each location, therefore the dose rates for the various locations wouldn’t necessarily scale linearly with the reduction in the dose rate at the side. In other words, a side dose rate decreased by a factor of 0.593 does not necessarily mean the dose rate at other locations would have the same amount of reduction. This would be especially true if the reason

for the decreased dose rates is a result of increasing the lead shielding in the variable weight transfer cask; in this case the dose rate reduction would only be seen at the side of the cask and not necessarily at the top.

On these bases, the staff found this to be a reasonable approach for determining that the HI-TRAC VW with the new loading patterns can meet occupational dose in 10 CFR 20.1201 for the following reasons:

- The applicant has calculated the maximum dose rates in a conservative way by finding a different loading pattern that maximizes dose rates at each location and representing each location (before applying the scaling factor) for the worst possible configuration across all loading patterns and MPCs. In fact, not all highest possible dose rates could exist at the same time as they are from different scenarios.

- If the reduced dose rates are due to fuel loadings with source term below the design basis, the side dose rate is expected to be the highest on the transfer cask and other locations would be reduced by some amount from the design basis as well.

- If the reduced dose rates are due to an increase in the lead shielding for the variable weight transfer cask, most of the dose rate locations in FSAR Table 11.3.2 are shielded somewhat by the lead in the side of the transfer cask. The only location that would not see any reduction in dose rates is directly at the top and bottom of the cask. Step 5, “MPC Closure Ring Installation,” in FSAR Table 11.3.2 is the only step that is directly at the top, and the staff finds that a user would continue meeting the occupational dose in 10 CFR 20.1201 when loading with a design basis dose rate for this step.

11.1.2 ALARA

FSAR Table 11.3.2 discusses several supplemental shielding components that can be used to maintain exposures ALARA. None of these are new to this amendment request, and none are credited in the dose rates reported in FSAR Table 11.3.2. The applicant updated FSAR Section 9.2.4 to clarify the installation of the annulus shield. The staff found that the applicant has adequately described the components needed for maintaining exposures ALARA. Additional components needed for the HI-TRAC Version V2 transfer cask are discussed in Section 11.6.2 of this SER.

11.2 Proposed Change #2—Four New Fuel Types

The applicant proposed to add four new fuel types, 10x10I, 11x11A, 7x7C and 8x8G to the approved contents in CoC No. 1032, Appendix B.As discussed in Section 6.2 of this SER, the additional fuel types do not have a significant increase on dose rates, and therefore does not have any impact on radiation protection.

11.3 Proposed Change #3—Exception to the ASME Code

The applicant proposed to add an exception to the ASME Code to allow the use of certain duplex stainless steels in the HI-STORM FW system. As discussed in Section 6.3 of this SER, this change has an insignificant impact on the dose rates, and therefore does not impact the system’s ability to meet regulatory requirements related radiation protection.

11.4 Proposed Change #4—Use FLUENT to Evaluate Effective Fuel Conductivities

The applicant proposed to use FLUENT to revise the calculation for evaluating effective fuel conductivities. This change does not impact the system's ability to meet regulatory requirements related radiation protection.

11.5 Proposed Change #5—Use of DFI

The applicant proposed to add the DFI to CoC No. 1032 Appendix A. As discussed in Section 6.5 of this SER, this change does not impact the system's ability to meet regulatory requirements related to radiation protection.

11.6 Proposed Change #6—HI-TRAC VW Versions V and V2

The applicant proposed to add two versions of the standard HI-TRAC VW: Version V which adds a natural circulation feature, and Version V2 which adds the option for a removable neutron shield. The Version V does not include any changes to the shielding that would affect the dose rates, and therefore this part of the SER will focus on the acceptability of the Version V2 of the HI-TRAC VW.

11.6.1 Occupational Exposures

The staff documented the evaluation of the dose rates calculated for the HI-TRAC VW Version V2 in Section 6.6 of this SER. It found, as stated in Section 6.6.4.1 of the SER, that the dose rates calculated for the HI-TRAC VW are applicable for the HI-TRAC VW Version V2 and therefore the staff finds that the occupational exposure estimate in FSAR Table 11.3.2 is applicable to the HI-TRAC VW Version V2.

11.6.2 ALARA

The only location where the dose rate for the HI-TRAC VW Version V2 is not bounded by the dose rates for the HI-TRAC VW is Location 1 at the side near the bottom of the cask. This area does have a shield ring pedestal that is used to reduce dose at this location. Based on the comparison of dose rates in FSAR Table 5.1.10 and the staff's own calculations discussed in Section 6.6.5 of this SER, the staff found that the inclusion of the pedestals are critically important to maintain exposures ALARA. The applicant clarified the operating procedures in FSAR Sections 9.2.2 and 9.2.4 to state that the pedestals are required to maintain exposures ALARA and must be attached, unless transferring the HI-TRAC VW Version V2 to the mating device, which would then provide the shielding needed in this location. The staff found that this demonstrates that ALARA principles have been incorporated into operating procedures.

11.7 Proposed Change #7—Option for Cyclic Vacuum Drying

The applicant proposed to add an option for cyclic vacuum drying for all MPCs.

This change does not impact the system's ability to meet regulatory requirements related radiation protection as prescribed in 10 CFR 20.1201, 20.1301, 72.104 and 72.106 because it does not affect the radiation source term or shielding design.

11.8 Proposed Change #8—BLEU Fuel

The applicant proposed to add fuel assemblies containing BLEU as approved contents. As discussed in Section 6.8 of this SER, the applicant has added additional cooling time when loading these assemblies. This means the inclusion of BLEU fuel is bounded by the analysis for non-BLEU fuel, from a radiation protection perspective, and the staff found this change does not impact the system's ability to meet regulatory requirements related radiation protection.

11.9 Clarifications and Editorial and Minor Changes in the CoC/FSAR

As discussed in Section 6.9 of this SER, the staff did not find that any of the clarifications and editorial and minor changes had a significant effect on source term or shielding to cause any effect in dose rates, and therefore do not impact the system's ability to meet regulatory requirements related radiation protection.

11.10 Evaluation Findings

F11.1 The HI-STORM FW system Amendment No. 5 provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106.

F11.2 The design and operating procedures of the HI-STORM FW system Amendment No. 5 provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the objective of maintaining exposures ALARA.

The staff concludes that the design of the radiation protection system of the HI-STORM FW system Amendment No. 5 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 FW system Amendment No. 5 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 performed accident analysis for the proposed changes and has documented the results in Section 4.4 (thermal) and Sections 6.1 and 6.6 (shielding) of this SER.

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, technical specifications, and operating controls and limits (in FSAR Chapter 13) for CoC No. 1032, Amendment No. 5 would be in accordance with the requirements of 10 CFR Part 72. The staff reviewed the proposed changes to the Technical Specifications 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 Technical Specifications:

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

Page Number	Reference	Description	Proposed Change
CoC, Page 2	Description	Add neutron shield cylinder as part of the transfer cask.	6
Appendix A Table of Content	N/A	Updated.	N/A
Appendix A 1.1-1	Definition	Add definition for DFI.	E3
Appendix A 1.1-2	Definition	Add definition for BLEU fuel.	8(b)
Appendix A 1.1-3	Definition	Modify the definition of repaired/reconstituted fuel assembly.	E1
Appendix A 3.1.2-2	Surveillance Requirement (SR) 3.1.2	Revise surveillance requirement for overpacks containing MPC-89s.	6
Appendix A 3.1.4-1 to 3.1.4-2	LCO 3.1.4 and SR 3.1.4	Add LCO 3.1.4 and SR 3.1.4 for the HI-TRAC VW Versions V and V2 heat removal system	6
Appendix A 3.4-1 to 3.4-3	Table 3-1	Add MPC heat load limits for the new loading patterns for MPC-37 and MPC-89. Add Note 5 for guidance in using vacuum drying.	1, 7, E6
Appendix A 3.4-4	Table 3-2	Add decay heat limits for the new loading patterns for MPC-37 and MPC-89.	1
Appendix A 5.0-3 to 5.0-5	Section 5.3	Update 5.3.4 measure dose rate limits, and add a statement on dose rate measurements for transfer cask with a neutron shield.	6
Appendix B Table of Content	N/A	Updated.	N/A
Appendix B 2-5 to 2-6	Table 2.1-1, Section I	Revise information for MPC-37: (1) change minimum required cooling time to 1 years, (2) identify cell locations for DFCs based on heat load patterns, (3) add hafnium fuel, and (4) add Note 2 regarding DFI.	1, 5, E2, E3
Appendix B 2-7 to 2-8	Table 2.1-1, Section II	Revise information for MPC-89: (1) change minimum required cooling time to 1 year, (2) identify cell locations for DFC based on heat load patterns, and (3) add Note 2 to describe DFI.	1, 5, E3

Page Number	Reference	Description	Proposed Change
Appendix B 2-9 to 2-10	Table 2.1-1, Section III	Revise information for MPC-32ML: (1) make conforming change regarding cooling time, and (2) add Note 2 to describe DFI.	1, 5, E3
Appendix B 2-11 to 2-15	Table 2.1-2	(1) Add Note 6 to describe fuel rod replacement by irradiated or unirradiated Steel or Zirconia rods, (2) add Note 7 to include the required shielding evaluation for storing BLEU fuel assemblies, and (3) correct typographical error on the number of fuel rod locations for 16x16C fuel assembly.	8(c), E5
Appendix B 2-16 to 2-20	Table 2.1-3	(1) Add four new fuel types, (2) add Note 15 to clarify the number of fuel rod location does not have water rods, (3) add Note 16 to describe fuel rod replacement by irradiated or unirradiated Steel or Zirconia rods, (4) add Note 17 to include the required shielding evaluation for storing BLEU fuel assemblies.	2, 8(c)
Appendix B 2-21	Section 2.3.1	Add additional description on VDS drying, especially for high burnup fuel drying time limit.	E7
Appendix B 2-21	Section 2.3.1	Add a paragraph to allow minor deviation from prescribed loading pattern, as long as the peaking cladding temperature is below the limit in SFST-ISG-11, Revision 3.	E4
Appendix B 2-24	Table 2.3-6	Add table for PWR fuel length categories.	1, 6
Appendix B 2-25 to 2-34	Figures 2.3-1 to 2.3-9	Add loading patterns for MPC-37.	1
Appendix B 2-35 to 2-38	Figures 2.3-10 to 2.3-13	Add loading patterns for MPC-89.	1
Appendix B 2-41	Table 2.4-2	Include the use of DFI in the burnup credit configurations.	5
Appendix B 2-43	Section 2.5	Update burnup and cooling time fuel qualification requirements.	1
Appendix B 2-44	Table 2.5-2	Add Note 1 regarding BLEU fuel.	8(a)
Appendix B 3-4	Table 3-1	Include an ASME code alternative for MPCs.	3
Appendix B 3-10	Section 3.4.1	Allow the site's yearly average ambient temperature to be used for site-specific analysis.	N/A
FSAR 13-2	Table 13.1.1	Include transfer cask heat removal system.	6
FSAR 13.3	Table 13.1.2	Include transfer cask heat removal system.	6

Page Number	Reference	Description	Proposed Change
FSAR 13.A-2	Bases Table of Content	Updated to include transfer cask heat removal system.	6
FSAR 13.A-25 to 13.A-28	B 3.1.4	Add Transfer Cask Heat Removal System.	6

The staff finds that the proposed changes to the Technical Specifications for the HI-STORM FW system Amendment No. 5 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 FW system Amendment No. 5 will continue to allow safe storage of spent nuclear fuel.

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. 1032, Amendments No. 0 through 4. 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 FW system were considered:

- Proposed Change #1a: Add new heat load patterns for the MPC-89 and MPC-37 (long, standard, and short length).
- Proposed Change #1b: Change the minimum required cooling time for fuel to 1 year for MPC-89 and MPC-37.
- Proposed Change #2: Add four new fuel types, 10x10I, 11x11A, 7x7C, and 8x8G, to the approved contents listed in Appendix B.
- Proposed Change #3: Add an exception to the ASME Code to allow the use of certain duplex stainless steels in the HI-STORM FW system.
- Proposed Change #4: Use FLUENT to revise the calculation for evaluating effective fuel conductivities.
- Proposed Change #5: Add the use of DFI in CoC Appendix A.
- Proposed Change #6: Add two versions of the standard HI-TRAC VW. Version V adds a natural circulation feature. Version V2 adds the option for removable neutron shield.
- Proposed Change #7: Add the option for using cyclic vacuum drying for all MPCs.
- Proposed Change #8a: Add fuel assemblies containing BLEU as approved contents.
- Proposed Change #8b: Add the definition for BLEU fuel assemblies to FSAR Glossary Section and the definition section in the CoC.
- Proposed Change #8c: Add shielding evaluation requirement to FSAR Section 5.4.8 for storing BLEU fuel assemblies in HI-STORM FW system.
- Clarification E1: Modify the definition of repaired/reconstituted fuel assembly in CoC Appendix A to clarify that when dummy stainless steel rods are present in the loaded spent fuel assemblies, the dummy/replacement rods will be considered in the site-specific dose calculations.

- Clarification E2: Add hafnium rods in CoC Appendix B, Table 2.1-1 and clarify that CRAs are not limited to those with hafnium.
- Clarification E3: Add the definition of DFI in CoC Appendix A and FSAR.
- Clarification E4: Add the definition of minor deviation from the prescribed loading pattern to CoC Appendix B, Section 2.3.
- Clarification E5: In CoC Appendix B, correct typographical error in Table 2.1-2 under the 16x16C fuel class to correct the number of fuel rod locations to 235
- Clarification E6: Correct typographical error in CoC Appendix A, Table 3-1 to bring into agreement with FSAR Table 4.5.19.
- Clarification E7: In CoC Appendix B, Section 2.3.1, clarify that VDS is permitted for high burnup fuel with drying time limits as provided in CoC Appendix A, Table 3-1.

Based on the statements and representations provided by the applicant in its amendment application, as supplemented, the staff concludes that the changes described above to the HI-STORM FW MPC Storage System Amendment No. 5 do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. Amendment No. 5 for the HI-STORM FW MPC Storage System should be approved.

Issued with Certificate of Compliance No. 1032, Amendment No. 5
on __________.

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SUBJECT: PRELIMINARY 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

DOCUMENT DATE: February 14, 2020

ADAMS Accession No. ML20014E621 * concurrence via email

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