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FINAL SAFETY EVALUATION REPORT

DOCKET NO. 72-1032
HOLTEC INTERNATIONAL
HI-STORM FLOOD/WIND
MULTI-PURPOSE CANISTER STORAGE SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1032
AMENDMENT NO. 7

SUMMARY

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (staff) review and evaluation of Holtec International's (hereafter Holtec or applicant) request to amend Certificate of Compliance (CoC) No. 1032 for the HI-STORM Flood/Wind (FW) Multi-purpose Canister (MPC) Storage System (hereafter HI-STORM FW system). Holtec submitted its request by letter dated May 6, 2021 (Holtec, 2021a), and supplemented it on October 15, 2021 (Holtec 2021b), July 11, 2022 (Holtec, 2022a), July 13, 2022 (Holtec, 2022b), July 29, 2022 (Holtec, 2022c), September 15, 2022 (Holtec, 2022d), October 3, 2022 (Holtec, 2022e), December 1, 2022 (Holtec, 2022f), January 6, 2023 (Holtec, 2023a), May 8, 2023 (Holtec, 2023b), June 30, 2023 (Holtec, 2023c), July 11, 2023 (Holtec, 2023d), August 15, 2023 (Holtec, 2023e), November 17, 2023 (Holtec, 2023f), February 16, 2024 (Holtec, 2024a), and April 8, 2024 (Holtec, 2024b). On February 17, 2022 (Holtec, 2022g), Holtec requested to separate the RIRP-I-16-01 CoC Reorganization, also known as graded approach (NRC, 2020a), from Amendment No. 7. Therefore, this SER does not include evaluation of the RIRP-I-16-01 CoC Reorganization.

Holtec proposed the following proposed changes (PCs):

PC #1 Add a new unventilated high density (UVH) overpack, HI-STORM FW UVH, which includes high density concrete for shielding. The UVH is to be used with MPC-37 (standard), MPC-89 (standard), and the new MPC-44.

PC #2 Modify vent and drain penetrations to include the option of second port cover plate.

PC #3 Allow automated equipment to perform leak test of the MPC materials and welds in the fabrication shop.

PC #4 Change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only. Remove post hydrostatic test liquid penetrant (PT) and magnetic particle (MT) examination.

PC #5 Include the ability to use computational fluid dynamics (CFD) analysis to evaluate site-specific fire accident scenario.

PC #6 Use updated methodology for tornado missile stability calculations for freestanding HI-STORMs and HI-TRACs. Clarify the weights to be used for varying heights of HI-TRACs.

PC #7 Add a new MPC, MPC-44, with continuous basket shim (CBS) and to hold 44 pressurized-water reactor (PWR) fuel assemblies of certain 14x14 fuel class. It is

to be used with Version E and UVH overpacks. Note that there is no standard version of MPC-44 for FW storage system.

PC #8 Add a new MPC, MPC-37P, with CBS and to hold 37 PWR fuel assemblies of certain 15x15 fuel class. It is to be used with version E overpack. Note that there is no standard version of MPC-37P for FW storage system.

PC #9 Add HI-DRIP ancillary system. HI-DRIP is an optional ancillary system designed to prevent water within the MPC from boiling during loading and unloading operations while loading MPC in the HI-TRAC.

PC #10 Include the ability to use CFD analysis to evaluate site-specific burial-under-debris accident scenario.

PC #11 Include the ability to use water without glycol in the HI-TRAC water jacket during transfer operations below 32°F based on the site-specific MPC total heat loads.

PC #12 Add new 10x10J fuel type to approved contents in the HI-STORM FW system.

PC #13 Update bounding fuel variables for 8x8F and 11x11A boiling water reactor (BWR) fuel types in CoC appendix B.

PC #14 Other editorial changes.

PC #15 Adopt a stress-based structural design criterion as documented as option 2 in the September 8, 2023 conversation record (NRC, 2023h).

PC #16 Establish specific criteria on allowable interference due to differential thermal expansion (DTE).

This revised CoC, when codified through rulemaking, will be denoted as Amendment No. 7 to CoC No. 1032.

This SER documents the staff's review and evaluation of the proposed amendment. The staff notes that the two proposed new MPCs, MPC-44 and MPC-37P, are equipped with CBS. The staff also notes that the applicant incorrectly added MPC-37-CBS and MPC-89-CBS to this CoC using the process described in Title 10 of the *Code of Federal Regulations (10 CFR)* 72.48. Independent from this amendment, the staff's review of the changes made using the 72.48 process is documented in Inspection Report No. 07201014/2022-201, dated September 12, 2023 (ML23145A175), and Notice of Violation EA-23-044, dated January 30, 2024 (ML24016A190). Therefore, although the tip-over accident structural analysis results for these two fuel baskets are cited in this SER, and used in the standard and Version E overpacks, the staff is not approving the MPC-37-CBS and MPC-89-CBS variants in this amendment, as the applicant did not include them in the scope of the amendment request. The staff reviewed the applicant's tip-over analysis results because the applicant referenced these analyses in demonstrating the safety and regulatory compliance of other basket and overpack combinations proposed in this amendment.

The staff followed the guidance in NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities," April 2020 (NRC, 2020b). 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 final safety analysis report (FSAR) or previous amendments to the CoC.

1.0 GENERAL INFORMATION EVALUATION

The staff reviewed the proposed changes in the application chapter 1, “General Description,” to ensure that the applicant has provided a non-proprietary description, or overview, in its documentation for the spent fuel storage system that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

In the application chapter 1, the applicant added the descriptions related to the new MPC-44 and MPC-37P, including their figures and drawings, the descriptions for the optional second port cover, the descriptions and conditions for using ethylene glycol in HI-TRAC water jacket, a new section 1.2.1.7 to describe HI-DRIP auxiliary cooling system, and the new manufacturing facility in Camden, New Jersey.

Table 1-1 lists all the proposed overpack type, MPC type, fuel type, loading pattern, and heat load limit in this amendment application.

Table 1-1
HI-STORM FW Amendment No. 7
Proposed Overpack, MPC, Fuel Type, Loading Pattern, and Heat Load Limit

No.	Overpack Type	Version	MPC Type	Fuel Type	Loading Pattern	Decay Heat Limit (kW)	Appendix B Table	Appendix B Figure
1	Ventilated	Version E only	MPC-37P	HBF[1] LBF[2]	Regionalized Pattern A	44.09	2.3-1A	2.1-1
2	Ventilated	Version E only	MPC-37P	HBF LBF	Regionalized Pattern B	45.0	2.3-1A	2.1-1
3	Ventilated	Version E only	MPC-37P	HBF LBF	Regionalized with DFC[3]/DFI[4]	45.0	2.3-7A	2.1-4 2.3-14 2.3-15
4	Ventilated	Version E only	MPC-37P	HBF LBF	Uniform	33.3 (Threshold)	2.3-7B	2.1-4
5	Ventilated	Version E only	MPC-44	HBF LBF	Uniform	44.0 (without DFC/DFI) 43.4-44.0 (with DFC/DFI)	2.3-8A	2.1-5
6	Ventilated	Version E only	MPC-44	HBF LBF	Uniform	30.0	2.3-8B	2.1-5
7	Unventilated	UVH	MPC-44	HBF LBF	Uniform	28.0 (without DFC/DFI) 27.6-28.0 (with DFC/DFI)	2.3-13	2.1-5
8	Unventilated	UVH	MPC-37 (Standard)	HBF LBF	Uniform	29.0	2.3-9A	2.1-1
					Regionalized	29.0	2.3-9B	
9	Unventilated	UVH	MPC-89 (standard)	HBF LBF	Uniform	29.0	2.3-10A	2.1-2
					Regionalized	29.0	2.3-10B	

Notes:
1. HBF – high burnup fuel.
2. LBF – low burnup fuel.
3. DFC – damaged fuel can.
4. DFI – damaged fuel isolator.

The staff determined that the proposed description in general information is adequate for the staff to conduct its evaluation as documented in the rest of this SER. Therefore, it satisfies the requirements for the general description under 10 CFR Part 72.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

The staff reviewed the proposed changes in the application chapter 2, “Principal Design Criteria,” to ensure the principal design criteria related to structures, systems, and components (SSCs) important to safety (ITS) comply with the relevant general criteria established in the requirements in 10 CFR Part 72.

In the application chapter 2, the applicant added the descriptions related to the new MPC-44 and MPC-37P, the descriptions of the optional second port cover, descriptions for the new 10x10J fuel type and additional details for other fuel types, descriptions for the HI-DRIP supplemental cooling system, discussions related to short-term operations in section 2.2.3 in the event of environmental phenomena, updated temperature limits in table 2.2.3, and changes to the design criteria for fuel baskets in section 2.2.8.

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

3.0 STRUCTURAL EVALUATION

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

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

- PC #1 – Add a new unventilated overpack with high density concrete, HI-STORM FW UVH.
- PC #6 – Add a new methodology for tornado missile stability calculations for freestanding HI-STORMs and HI-TRACs.
- PC #7 – Add the new MPC-44 to be used with Version E and UVH overpacks.
- PC #8 – Add the new MPC-37P to be used with the Version E overpack.
- PC #15 – Adopt a stress-based structural design criterion.
- PC #16 – Establish specific criteria on allowable interference due to DTE.

3.1 PC #1 – HI-STORM FW UVH

3.1.1 Description and Design Criteria of the HI-STORM FW UVH

The applicant proposed adding an alternative version of the HI-STORM FW overpack, the UVH overpack, which is unventilated and constructed with high density concrete. The applicant described the structural design of the HI-STORM FW UVH in chapter 3.I of the application with a general description and design criteria in chapter 1.I and 2.I, respectively.

The applicant listed the principal components associated with the UVH overpack in FSAR table 1.I.1.2, which include the MPC-37, MPC-89, MPC-44, and all three versions of the HI-TRAC VW transfer cask. The applicant depicted the HI-STORM FW UVH in Drawing No. 11897 that is included in section 1.I.5 of the application. The UVH overpack is a metal overpack with a steel shell weldment formed by an inner shell, an outer shell, and a base plate. Steel ribs connect the shell plates and concrete fills the space between the shell plates for shielding. The lid for the UVH overpack also consists of a weldment of steel plates with steel ribs and shielding concrete between the plates. The UVH overpack does not contain inlet or outlet vents, and the bolted lid is installed with a concentric metallic gasket to seal the overpack. The applicant classified the components of the UVH overpack as ITS Category B with the exception of the lid gasket seal being ITS Category C. The staff finds these component categorizations acceptable, as they are consistent with or exceed the guidance for concrete shielded dry spent fuel storage systems presented in NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

The applicant listed the design basis loadings in table 2.I.2.1 of the application. The only proposed changes to the structural design bases that the applicant made for the HI-STORM FW UVH are related to pressures in the overpack annulus. Due to the unventilated design of the UVH overpack, the applicant evaluated new bounding pressure variations as described in section 2.I.2 of the application. The applicant listed the design pressure values in table 2.I.2.3 of the application including maximum and minimum internal pressures for normal, off-normal, and accident conditions and an accident external pressure. The applicant also proposed changes to the maximum concrete compressive strength of the independent spent fuel storage installation (ISFSI) pad for the HI-STORM FW UVH to 6,000 psi as listed in table 2.I.0.1 of the application. The applicant did not propose changes to the ISFSI pad design basis parameters for the maximum thickness and modulus of elasticity of the subgrade listed in FSAR table 2.2.9, and these parameters remain applicable to the UVH pad design.

The applicant defined the acceptance criteria for the UVH overpack shell weldment, the base plate, and the overpack lid as demonstrating that the stress limits from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III, Division 1, Subsection NF are not exceeded under all loadings.

The staff finds the structural design criteria of the HI-STORM FW UVH is consistent with guidance on acceptable design criteria in NUREG-2215. Therefore, the staff finds that the design of the HI-STORM FW UVH meets the requirements of 10 CFR 72.236(b).

3.1.2 Structural Evaluation of the HI-STORM FW UVH

The applicant evaluated the HI-STORM FW UVH using both hand calculations and finite element analyses. The applicant submitted to the staff, and incorporated into the application by reference, three proprietary calculation packages to support the structural analyses: HI-2094418, revision 37, "Structural Calculation Package for HI-STORM FW System," HI-2210313, revision 3, "Analysis of the Non-mechanistic Tip-over Event of the Loaded HI-STORM FW Version UVH Storage Cask," and HI-2094392, revision 13, "Tornado Missile Analysis for HI-STORM FW System." The applicant described the structural evaluations in supplement 3.I of the application.

The applicant's structural evaluations of the HI-STORM FW UVH consisted of analyses of the UVH closure lid, the credible accident conditions for the HI-STORM FW (i.e., flood, explosion, earthquake, tornado, and tip-over), the additional pressure loadings from the new UVH design basis pressures, and the MPC subjected to new temperature contours under normal and off-normal pressure conditions. For the other design basis conditions, the applicant performed analyses to demonstrate that the previous evaluations for the HI-STORM FW were applicable or bounding for the HI-STORM FW UVH.

3.1.2.1 Evaluation of the MPC under Normal and Off-normal Conditions

The applicant analyzed the MPC for normal and off-normal pressure conditions under the temperature contours developed for the UVH as described in section 3.I.3.1 of the application. The applicant described the temperature contours in sections 4.I.4 and 4.I.6.1 of the application for the normal and off-normal conditions, respectively. The analysis used the previously approved normal and off-normal pressures for the HI-STORM FW listed in FSAR table 2.2.1.

The applicant followed the established methodology for analyzing the MPC pressure described in FSAR section 3.4.4.1.5. For this analysis, the applicant used the ANSYS finite element analysis (FEA) model of the MPC previously used for the HI-STORM FW system and described in FSAR section 3.1.3.2. The MPC FEA model is a quarter-symmetry model depicted in FSAR figure 3.4.1 and was created using the MPC dimensions listed in FSAR table 3.1.14. The applicant used the linear elastic material properties of Alloy X provided in section 3.3 of the application and an elastic modulus that varies with the temperature of the UVH stress contours. As described in FSAR section 3.1.3.2, the applicant previously validated the FEA model of the MPC for accurately analyzing the primary and secondary membrane and bending stresses including the stresses at locations of discontinuity. The applicant applied the design internal pressure to all internal surfaces of the MPC storage cavity in the FEA model and applied boundary conditions to the FEA model fixing the center node on the top of the MPC and the two vertical symmetry planes against translation. The applicant depicted the applied loading and boundary conditions in FSAR figure 3.4.31.

The applicant depicted the stress contours from the internal pressure loading on the FEA model of the MPC in figures 3.I.3.1 and 3.I.3.2 of the application. The applicant evaluated these stresses following the methodology described in FSAR section 3.4.4.1.5 with further details for the UVH evaluation in supplement 1 of HI-2094418. For this stress analysis, the applicant compared the maximum primary and secondary stress intensities to the previously established design criteria stress limits for the MPC following ASME B&PV Code Section III, Division 1, Subsection NB; Level A stress limits for the normal condition and Level B for the off-normal condition. The applicant listed the key results of the stress analysis in the application tables 3.I.3.2 and 3.I.3.3 including the primary and secondary stresses in the MPC lid, shell, and baseplate. These tables also included the allowable stress limits from the ASME B&PV code for each component and the safety factors (i.e., a ratio of the allowable value to the value computed by analysis) comparing the FEA stress results to the allowable stress limits. All calculated safety factors were greater than one, indicating that the induced stresses from the UVH internal pressure were less than the allowable stresses for normal and off-normal conditions.

The staff finds the applicant's methodology for FEA and stress analyses performed for an MPC subjected to the normal and off-normal pressures in the HI-STORM FW UVH to be consistent with the previously approved methodology for the HI-STORM FW system, the applicable sections of the ASME B&PV design code, and the guidance on FEA and stress analysis in NUREG-2215. Based on the results of the stress analysis demonstrating safety factors greater than one, the staff finds that an MPC stored in HI-STORM FW UVH has sufficient structural integrity to maintain the confinement boundary under normal and off-normal conditions and meets the requirements in 10 CFR 72.236(d) and (l) for maintaining confinement under normal and off-normal conditions.

3.1.2.2 Lifting and Handling

The applicant described the lifting requirements in FSAR sections 1.2.1.5 and 2.2.3.a and in technical specification (TS) 5.2 in appendix A of the CoC. The applicant made no changes to lifting requirements established for the HI-STORM FW system in Amendment No. 5 and previous amendments to the CoC. These required the loaded cask or transfer cask to be lifted with equipment meeting the design requirements in TS 5.2.c.1 through 5.2.c.3. The applicant concluded that these lifting requirements were sufficient to prevent an uncontrolled lowering of the load such that further analysis of a handling accident for the HI-STORM FW was not required.

In Amendment No. 6 to the HI-STORM FW CoC, the applicant added a lifting height limit of 11 inches based on the drop height analyzed in a drop analysis of the loaded HI-STORM FW cask and HI-TRAC VW transfer cask. This analysis for Amendment No. 6 concluded that when lifting below this limit, the overpack and transfer cask could be lifted with equipment that did not meet the design requirements for lifting equipment in the TS. However, in this amendment, the applicant did not perform a drop analysis for the HI-STORM FW UVH and thus the TS changes added in Amendment No. 6 do not apply to the UVH. The HI-STORM FW UVH must be lifted with equipment that meets the design requirements for lifting equipment in the TS even when lifting below heights of 11 inches.

For the HI-STORM FW UVH, the applicant proposed changes to the lifting requirements in CoC appendix A, TS 5.2.c and table 5-1 to clarify that the use of equipment that did not meet the design requirements in TS was not permitted for unventilated overpacks like the UVH. Such non-conforming lifting equipment is only permitted for transfer casks and ventilated HI-STORM FW casks when lifted below the 11-inch height limit and supported by a drop analysis. Following the previously approved lifting requirements for the HI-STORM FW, a drop analysis would not be required for the HI-STORM FW UVH, since the applicant has proposed requiring unventilated overpacks be lifted only with equipment designed for redundant drop protection. The staff finds that this approach is consistent with the guidance on lifting in NUREG-2215 and NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” July 1980 (NRC, 1980).

The staff reviewed the applicant’s analysis for the normal and off-normal lifting conditions for the HI-STORM FW UVH, which included the UVH design basis pressure loads, as discussed in section 3.1.2.3 of this SER. Based on the staff’s review of the lifting analysis and the requirements for lifting equipment for unventilated HI-STORM FW overpacks, the staff finds the applicant has adequately considered the lifting and handling conditions and demonstrated that the HI-STORM FW UVH has an adequate structural design to meet the requirements of 10 CFR 72.236 under those conditions.

3.1.2.3 Pressure Load

The applicant analyzed the UVH overpack for normal, off-normal, and accident pressure conditions as described in section 3.I.3.2 of the application with further details in supplement 43 of HI-2094418. The analysis considered the following six pressure load cases (four normal, including lifting, and two accident) to envelope all the design basis pressure loadings listed in table 2.I.2.3 of the application:

- A 20-psi internal pressure applied with the loaded MPC weight applied uniformly to the overpack baseplate to envelope the design maximum internal pressure.
- A (-)14.7-psi internal pressure and 10-psi external pressure applied without the MPC weight to envelope the design minimum internal pressure.
- A 25-psi external pressure applied without the MPC weight to consider a design-level external pressure.
- A 25-psi internal pressure applied with the loaded MPC weight applied uniformly to the overpack baseplate to envelope the accident condition internal pressure.
- A 75-psi external pressure applied without the MPC to envelope the accident condition external pressure.
- A lifting load case consisting of a 20-psi internal pressure, the loaded MPC weight applied uniformly to the overpack baseplate, and a constant gravitational acceleration applied in the vertical direction accounting for the UVH overpack self-weight.

To determine the stresses induced by the six pressure loading cases, the applicant created a half-symmetry FEA model of the UVH overpack in the ANSYS FEA code, as depicted in figure 3.I.3.10 of the application. The applicant applied fixed support conditions to the overpack baseplate edge, except under the lifting load case where the cask anchor points were constrained vertically. The staff considers these to be conservative boundary conditions that maximize stress in the UVH overpack. The applicant used material properties for the overpack steel for bounding temperatures of 500°F for normal and off-normal conditions and 600°F for accident conditions. These temperatures conservatively exceed the maximum calculated temperatures for the UVH overpack from section 4.I of the application. The applicant based the dimensions of the UVH FEA model on Drawing No. 11897 of the application.

The applicant performed a stress analysis comparing the primary membrane and membrane plus bending stresses in UVH overpack inner and outer shells, base plate and lid against the design criteria Level A stress limits for normal and off-normal conditions and level D stress limits for accident conditions following ASME B&PV Code Section III, Division 1, Subsection NF. The applicant depicted the stress distribution among the six pressure loading cases from the FEA model in the application figures 3.I.3.11 and 3.I.3.12 from the governing load case. The applicant also summarized the key stresses for the governing load case in the application table 3.I.3.11 along with the ASME allowable stress limit and the safety factor. The stress results in table 3.I.3.11 showed that all safety factors are greater than one.

To check the possibility of buckling of the UVH overpack outer shell, the applicant created a quarter-symmetry FEA model of the UVH overpack in ANSYS. This buckling analysis model consisted of portions of the outer shell, inner shell, radial ribs, and baseplate as depicted in HI-2094418 figure 43.12. The applicant applied the 75-psi accident condition external pressure to external surface of the outer shell. The FEA demonstrated that the applied load resulted in a safety factor of 3.02 before reaching the first mode of buckling. The applicant noted that this was greater than the minimum safety factor of 1.34 against buckling required by ASME B&PV Code, Code Case N-284, "Metal Containment Shell Buckling Design Methods, Class MC, TC, and SC Construction." The staff finds the results of this buckling analysis sufficiently demonstrate that the outer shell of the overpack does not buckle under the accident condition external pressure load.

The staff finds the applicant's methodology for FEA and stress analyses performed for the UVH overpack subjected to the normal, off-normal, and accident condition pressure loads to be consistent with the applicable sections of the ASME B&PV design code and the guidance on FEA and stress analysis in NUREG-2215. Based on the results of the stress analysis demonstrating safety factors greater than one and the buckling analysis demonstrating that buckling does not occur, the staff finds that the HI-STORM FW UVH overpack has sufficient structural integrity to withstand the normal, off-normal, and accident condition pressures and maintain design basis shielding. Therefore, the staff finds the HI-STORM FW UVH overpack has adequate structural integrity to meet the requirements of 10 CFR 72.236(d) under the design basis pressure loadings.

3.1.2.4 HI-STORM FW UVH Cask Lid Lifting

The applicant evaluated the HI-STORM FW UVH cask lid under the design basis lid lifting normal conditions as described in section 3.I.3.3 of the application with further details in supplement 46 of HI-2094418. The applicant followed the established methodology for analyzing the HI-STORM FW overpack lids for lifting described in FSAR section 3.4.3.2.c, case (ii).

For the cask lid lifting conditions, the applicant performed hand calculations analyzing the lifting lugs and lifting lug welds in supplement 46 of HI-2094418 using a strength-of-materials-based approach and following ASME B&PV Code Section III, Division 1, Subsection NF. The applicant applied a 1.15 dynamic load factor consistent with the established HI-STORM FW lid lifting analyses. The applicant also created an ANSYS FEA model of the lid based on Drawing No. 11897 of the application to evaluate the global stresses in the lid components. Figure 3.I.3.7 of the application depicts the FEA model of the UVH cask lid. For the analysis, the applicant constrained the four lift points vertically and applied a downward acceleration of 1.15 g to account for gravity and the dynamic load factor. The applicant showed the stress distribution results from the FEA model in figure 3.I.3.8 of the application. The applicant conservatively took the maximum calculated stresses at any location on the lid, including secondary stress contributions, for the stress analysis.

The applicant's stress analysis compared the hand calculated stresses for the lifting lugs and lifting lug welds to the ASME B&PV Code Section III, Division 1, Subsection NF Level A stress limits. The applicant compared the FEA stresses against the Level A stress limits for primary membrane and primary membrane plus bending following ASME B&PV Code Section III, Division 1, Subsection NF. The applicant presented the maximum stresses, ASME stress limits, and corresponding safety factors in table 3.I.3.10 of the application. The results in table 3.I.3.10 showed that all safety factors were greater than one.

The staff finds the applicant's methodology for the strength of materials hand calculations, FEA, and stress analyses performed for the UVH overpack lid subjected to the lifting loads to be consistent with the applicable sections of the ASME B&PV design code and the guidance on hand calculations, FEA, and stress analysis in NUREG-2215. Based on the results of the stress analysis, the staff finds that the HI-STORM FW UVH overpack lid has sufficient structural integrity to withstand the normal condition cask lid lifting load and to maintain design basis shielding. Therefore, the staff finds the HI-STORM FW UVH overpack lid has adequate structural integrity to meet the requirements of 10 CFR 72.236(d) under the design basis cask lid lifting load.

3.1.2.5 HI-STORM FW UVH Cask Lid Snow Load

The applicant evaluated the HI-STORM FW UVH cask lid under the design basis normal condition snow load as described in section 3.I.3.3 of the application with further details in supplement 46 of HI-2094418. The applicant followed the established methodology for analyzing the HI-STORM FW overpack lids for snow loads described in FSAR section 3.4.4.1.10.

To analyze the snow load, the applicant used the FEA model of the UVH cask lid from the lid lifting analysis described in the previous section of this SER and applied the design basis 10 psi snow load to the top of the lid. The applicant depicted the resulting stress distribution in the lid in figure 3.I.3.9 of the application. The applicant performed a stress analysis following ASME B&PV Code Section III, Division 1, Subsection NF which conservatively compared the maximum calculated stresses at any location on the lid, including secondary stress contributions, to the Level A stress limits for the membrane and membrane plus bending stresses. In the application, the applicant noted that the safety factors for the UVH cask lid subject to the snow load condition were bounded by those for the cask lid lifting conditions listed in table 3.I.3.10 of the application. Supplement 46 of HI-2094418 presented the calculated safety factors of 3.50 for the membrane stress and 5.25 for the membrane plus bending stress.

The staff finds the applicant’s methodology for the FEA and stress analysis performed for the UVH overpack lid subjected to the snow loads to be consistent with the applicable sections of the ASME B&PV design code and the guidance on FEA and stress analysis in NUREG-2215. Based on the results of the stress analysis demonstrating safety factors greater than one, the staff finds that the HI-STORM FW UVH overpack lid has sufficient structural integrity to withstand the normal condition snow load and maintain design basis shielding. Therefore, the staff finds the HI-STORM FW UVH overpack lid has adequate structural integrity to meet the requirements of 10 CFR 72.236(d) under the design basis snow load.

3.1.2.6 Flood Accident Condition

The applicant evaluated the HI-STORM FW UVH cask assembly under the flood accident condition as described in section 3.I.3.4 of the application with further details in supplement 45 of HI-2094418. As described in FSAR section 3.1.2.1.c, the applicant previously calculated the maximum hydrostatic pressure from the design basis flood accident condition as 54.2 psi. This is less than the 75-psi external pressure evaluated for pressure accident condition on the UVH overpack, and the staff finds that the external pressure analysis, discussed in section 3.1.2.3 of this SER, bounds the hydrostatic pressure from the flood accident for the HI-STORM FW UVH.

The applicant followed the established methodology for analyzing the HI-STORM FW cask assemblies for kinematic stability under flood accident conditions described in FSAR section 3.4.4.1.1. For this methodology, the applicant considered the UVH cask dimensions, a conservatively low estimate of the weight of the UVH overpack with an MPC without fuel, and the design basis maximum submerged depth due to flooding of 125 ft taken from FSAR table 2.2.8 and chosen by the applicant to bound storage sites. The applicant’s design criteria required the safety factor to be greater than 1.1 for stability, which is consistent with the guidance in NUREG-2215. The applicant then calculated the maximum permissible flood velocity that would result in a safety factor greater than 1.1 against overturning and sliding. As shown in table 3.I.3.4 of the application, the maximum permissible flood velocity for a site storing the HI-STORM UVH is 37.5 ft/sec, which is greater than the 30.5 ft/sec permissible flood velocity for the ventilated standard HI-STORM FW.

The staff finds the applicant's methodology for the stability analysis performed for the HI-STORM FW UVH subjected to flood accident conditions to be consistent with the previously approved methodology for the HI-STORM FW system and the guidance in NUREG-2215 on flood accident conditions and overturning and sliding analyses. Based on the maximum permissible flood velocity and the results of the stability analyses demonstrating safety factors greater than 1.1, the staff finds that the applicant has demonstrated that the HI-STORM FW UVH is not susceptible to overturning or sliding during the flood accident conditions. Based on the staff's review of the stability analysis and hydrostatic pressure evaluation for flooding, the staff finds the applicant has adequately considered the flood accident and demonstrated that the HI-STORM FW UVH has an adequate structural design to meet the requirements of 10 CFR 72.236 under the flood accident condition.

3.1.2.7 Explosion Accident Condition

The applicant evaluated the stability of the HI-STORM FW UVH cask assembly under the explosion accident condition as described in section 3.I.3.5 of the application with further details in supplement 44 of HI-2094418. The applicant performed hand calculations following the established methodology for analyzing the tipping and sliding of the HI-STORM FW system from an explosion described in FSAR section 3.1.2.1.d.

This methodology for stability during an explosion simplified the analysis by considering a static pressure acting on one side of the overpack instead of a dynamic pressure wave. The staff notes this is a conservative simplification when limited to analyzing tipping and sliding resulting from an explosion. The applicant considered upper bound and lower bound heights of the UVH cask assembly and estimated corresponding minimum weights of the UVH overpack and MPC, conservatively without fuel. The applicant's design criteria required the safety factor to be greater than 1.1 for stability, which is consistent with the guidance in NUREG-2215. The applicant then calculated the allowable pressure from an explosion that would result in a safety factor greater than 1.1 against tipping and sliding. As shown in table 3.I.3.5 of the application, the maximum allowable external pressure on the HI-STORM UVH from an explosion at a site storing the cask is 5.45 psi.

The staff finds the applicant's methodology for the stability analysis performed for the HI-STORM FW UVH subjected to the explosion accident condition to be consistent with the previously approved methodology for the HI-STORM FW system and the guidance in NUREG-2215 on explosions and overturning and sliding analyses. Based on the maximum allowable external pressure and the results of the stability analyses demonstrating safety factors greater than 1.1, the staff finds that the applicant has demonstrated that the HI--STORM FW UVH is not susceptible to tipping or sliding during the explosion accident condition.

Part of the previously approved methodology for evaluating the explosion accident for the HI-STORM FW system, the applicant addressed other structural effects of the explosion pressure in FSAR section 3.1.2.1.d. The applicant referenced the analysis of the static external pressure accident condition, which includes a stress analysis of the overpack and MPC structures, as bounding the structural analysis of the dynamic pressure wave of the explosion accident condition. Given the significantly higher pressures evaluated in the external pressure accident condition and the staff's review of that analysis for the HI-STORM FW UVH discussed in section 3.1.2.3 of this SER, the staff finds this approach acceptable for the HI-STORM FW UVH.

Based on the staff's review of the explosion accident stability analysis and bounding external pressure accident analysis, the staff finds the applicant has adequately considered the explosion accident and demonstrated that the HI-STORM FW UVH has an adequate structural design to meet the requirements of 10 CFR 72.236 under the explosion accident condition.

3.1.2.8 Earthquake Accident Condition

The applicant applied the previously approved design criteria and analysis methodology for the earthquake accident conditions established for the HI-STORM FW system to the HI-STORM FW UVH. The applicant described the design criteria for earthquakes in FSAR section 2.2.3.g and the analysis methodology in FSAR section 3.4.4.1.2. The evaluation of the earthquake accident condition defines two categories of seismic hazard: low intensity and high intensity. The applicant defined a low intensity earthquake as having a zero-period acceleration (ZPA) that would not cause sliding or tipping of the cask in a static equilibrium analysis, and a high intensity earthquake as any greater ZPA that could cause tipping or sliding. The applicant calculated these limiting ZPAs for each HI-STORM FW and established a methodology for a site-specific dynamic seismic analysis required for any high intensity site in FSAR section 3.4.4.1.2.

For the HI-STORM FW UVH, the applicant calculated the limiting ZPA for a low intensity seismic hazard as described in section 3.I.3.6 of the application with further details in supplement 47 of HI-2094418. For the UVH overpack, the applicant performed hand calculations to determine the limiting ZPA values associated with a low intensity seismic event following the established methodology described in FSAR section 2.2.3.g. The applicant considered the UVH cask dimensions, a conservatively low estimate of the weight of the UVH overpack with an MPC loaded with one fuel assembly. The applicant's design criteria required the safety factor to be greater than 1.1 for tipping and sliding, which is consistent with the guidance in NUREG-2215. The applicant then calculated the limiting ZPA for a low intensity seismic hazard that would result in a safety factor greater than 1.1 against tipping and sliding. As shown in table 3.I.3.6 of the application, the combination of limiting ZPA for a low intensity seismic hazard for the HI-STORM UVH is 0.356 g in the horizontal direction and 0.237 g in the vertical direction, where 'g' is the magnitude of gravitational acceleration.

The staff finds the applicant's methodology for calculating the ZPA that would cause sliding or tipping of the HI-STORM FW UVH to be consistent with the previously approved methodology for the HI-STORM FW system and the guidance in NUREG-2215 on earthquake accident conditions and overturning and sliding analyses. Based on the limiting ZPA for a low intensity seismic hazard and the results of the stability analyses demonstrating safety factors greater than 1.1, the staff finds that the applicant has sufficiently demonstrated that the HI-STORM FW UVH is not susceptible to tipping or sliding during the earthquake accident conditions and the HI-STORM FW UVH has an adequate structural design to meet the requirements of 10 CFR 72.236(c), (d), and (l) under the earthquake accident conditions.

3.1.2.9 Tornado Accident Condition

The applicant evaluated the HI-STORM FW UVH cask under the tornado accident condition as described in section 3.I.3.7 of the application with further details in appendices I and J of HI--2094392. The applicant performed hand calculations following the established methodology for analyzing the tornado stability and missile penetration of the HI-STORM FW system described in FSAR section 3.4.4.1.3. The applicant described the established design basis and design criteria for the tornado accident in FSAR sections 2.2.3.e and 3.1.2.1.e. This methodology has the applicant analyze the HI-STORM FW system for kinematic stability and structural integrity of the confinement boundary under bounding combinations of the windspeeds and pressure drop specified in FSAR table 2.2.4 and the tornado-borne missiles in table 2.2.5.

The applicant analyzed the stability of the HI-STORM FW UVH for the tornado accident in appendix I of HI-2094392 to demonstrate the cask assembly does not tip over or slide greater than the minimum spacing between casks established by the layout listed in FSAR table 1.4.1 and the UVH dimensions. The applicant analyzed the minimum loaded weight of the maximum height HI-STORM FW UVH as bounding conditions. The applicant applied the conservative assumptions established for the tornado stability methodology listed in FSAR section 3.4.4.1.3. The applicant then calculated the rotation of the cask (i.e., the angle at which the cask tips) for two load cases: the tornado missile striking along with the tornado wind and the tornado missile striking along with a pressure drop caused by the passing tornado. The applicant listed the resulting cask rotations and safety factors against the allowable rotation that would cause a tip-over in table 3.I.3.7 of the application. The critical safety factor against tipping was 9.69. The applicant separately calculated the displacement of the cask from the tornado load cases. The applicant calculated a total displacement of 0.277 ft and a safety factor of 14.1 compared to the nearly 4 ft of space between HI-STORM FW UVH casks.

The applicant analyzed the structural integrity of the HI-STORM FW UVH for the penetration of the small and intermediate tornado missile in appendix J of HI-2094392 to demonstrate that the containment boundary of the MPC remains intact. The applicant considered missile impacts to the UVH overpack lid, the outer shell of the UVH overpack, and the portion of the outer shell where the optional jacking assembly would be located. The applicant applied the conservative assumptions established for the missile penetration methodology listed in FSAR section 3.4.4.1.3 and performed plastic analyses balancing the kinetic energy of the missile and the work done deforming the cask. The results of the penetration analysis demonstrated that the small missile did not penetrate the outer plate of the UVH overpack body or lid and the intermediate missile penetrated the outer plate and concrete but not the inner plate of the overpack body or lid. The applicant presented the penetration results in table 3.I.3.8 of the application.

The applicant performed a stress analysis of the HI-STORM FW UVH for the intermediate tornado missile impact in appendix J of HI-2094392 to demonstrate that the stresses in the outer shell of the UVH overpack do not exceed the allowable stress limits. The applicant considered a missile impact to the top of the UVH overpack being resisted by the outer and inner overpack shells acting as a cantilever beam, conservatively assuming the bottom of the cask to have fixed support conditions. The applicant calculated the bending stress induced by the missile which resulted in a safety factor of 2.34 compared to the ASME level D allowable membrane stress intensity.

The staff finds the applicant’s methodology for the stability, penetration, and stress analyses performed for the UVH overpack subjected to the tornado accident conditions to be consistent with the applicable guidance in NUREG-2215 and the ASME B&PV design code. Based on the results of the analyses demonstrating safety factors greater than one and the missiles failing to penetrate the inner plate of the UVH overpack, the staff finds that the structural design of the HI-STORM FW UVH is adequate to meet the requirements of 10 CFR 72.236 under the tornado accident conditions.

3.1.2.10 Tip-over Accident Condition

The applicant evaluated the HI-STORM FW UVH cask for the tip-over accident condition as described in section 3.I.3.8 of the application with further details in the report HI-2210313. The applicant followed the established methodology for analyzing the HI-STORM FW system for a tip-over described in FSAR section 3.4.4.1.4a, with minor alterations for the UVH.

The applicant described the established design basis and design criteria for the tip-over accident in FSAR section 2.2.3.b. The applicant analyzed the tip-over by assuming the HI-STORM FW system rests on its bottom edge with its center of gravity (C.G.) directly over this pivot point and the cask begins to rotate under its own weight from an initial velocity of zero. The analysis is intended to demonstrate: (1) the overpack lid does not dislodge from the overpack body; (2) there is no significant loss of shielding for the overpack; (3) the MPC remains within the overpack; (4) the overpack does not suffer ovalization that would prevent the removal of the MPC; (5) the MPC confinement boundary is not breached; (6) the lateral deflection of the fuel basket within the active fuel region is less than the maximum permanent deflection allowed for the fuel basket listed in FSAR table 2.2.11; and (7) the maximum primary (membrane plus bending) stress in the active fuel region of the fuel basket is less than 90% of the true ultimate strength of the basket material at the design temperature. In FSAR section 2.2.4, the applicant states that the portions of the fuel basket outside of the active fuel region are not subject to the deflection limit since they do not affect the reactivity control function of the Metamic-HT fuel basket; this region varies with the fuel assemblies being stored, as provided in FSAR table 3.2.1. These design criteria ensure the storage system maintains confinement, shielding, and retrievability and prevent criticality.

The applicant analyzed the tip-over accident for the HI-STORM FW UVH using the LS-DYNA FEA code. The applicant created three half-symmetry FEA models of the HI-STORM FW UVH for the MPC variants the applicant has proposed storing in the UVH: MPC-37, MPC-44, and MPC-89. In each model, the applicant added a concrete pad target for the cask to strike using the bounding concrete and subgrade properties in FSAR table 2.2.9, with the exception that the applicant used the lower maximum concrete compressive strength of 6,000 psi for the UVH from table 2.I.0.1 of the application. The applicant altered the analysis for the UVH tip-over from the tip-over analysis described in FSAR section 3.4.4.1.4a in two ways: (1) the applicant did not consider the impact between the MPC and the overpack guide tubes to be significant because of the reduced clearance between the MPC and the overpack for the HI-STORM FW UVH; and (2) the applicant designed a new lid closure mechanism for the UVH that relies on a lid bolting reinforcement plate to transfer shear between the lid and the cask body top plate. The new lid closure mechanism removes the shear load on the cask lid bolts but shifts the shear load to the fillet weld between the cask outer shell and the cask body top plate, which requires that the fillet weld be analyzed and evaluated.

The three FEA tip-over models for the UVH, as described in section 3.I.3.8 of the FSAR, differed by MPC height, basket dimensions, and fuel assemblies to account for the proposed MPC variants. The applicant explicitly modeled all structural members of the loaded cask as independent parts with non-linear material properties in LS-DYNA, except the fuel basket, which was modeled in four parts to capture the different bounding temperatures of the basket regions, and the fuel assemblies, which were modeled as elastic rectangular prisms. The applicant also explicitly modeled the critical weld connecting the MPC enclosure vessel and MPC lid. The applicant modeled the target for each tip-over analysis as the same 36-inch-thick concrete pad target above a 4-inch thick mudmat and deep layer of subgrade soil as shown in figure 3.I.3.3 of the application.

The applicant developed non-linear material properties for the tip-over models in appendix B of HI-2210313. The applicant used the ASME minimum material strength data at various normal condition temperatures to determine true-stress-true-strain curves using Hollomon's power law equation. The applicant derived and validated this previously approved methodology through benchmarking in the referenced proprietary report HI-2210251, "Benchmarking of Material Stress-Strain Curves in LS-DYNA." The applicant also considered strain rate effects in the FEA, except for the fuel basket material. The applicant considered this exception conservative as it would increase the deformation of the fuel basket. The staff notes that, within the range of uniform elongation (i.e., before necking occurs), engineering stress strain curves can be readily converted to true-stress-true-strain curves. The staff reviewed the FEA results and determined that gross necking of the materials did not occur. The staff reviewed the applicant's material property models and finds the methodology of deriving true-stress-true-strain curves and the use of these material curves in the LS-DYNA models to be acceptable for evaluating the structural performance of the HI-STORM FW UVH for tip-over.

From the assumed initial conditions of the tip-over (i.e., zero initial velocity and cask positioned such that the C.G. is directly above the bottom edge rotation point), the applicant calculated the angular velocity at the moment of impact for input into the FEA in appendix A of HI-2210313 using the kinematic equations and methodology described in FSAR section 3.4.4.1.4a.

The applicant depicted the maximum plastic strain results from the FEA in the three MPCs, overpacks and bolts in figures 12-20 to 12-28 of revision 3 of report HI-2210313 and listed the maximum plastic strains for the MPC enclosure vessel, overpack, and cask lid bolts for each model in table 3.I.3.9 of the application. The structural components of the models experienced only local plastic deformation. For the MPC vessel, the applicant observed minor plastic strain at the impact location with overpack inner shell. Local plastic strain developed in the inner shell near the impact of the MPC closure lid. And for the overpack outer shell, local plastic strain occurred at the top of the overpack body and in the outer shell of the overpack lid where they impacted the concrete pad. The applicant determined that all the maximum local plastic strains were less than the material failure strains, which it determined from the ASME material property data in a manner similar to that of the true-stress-true-strain curves.

As part of this amendment, the applicant proposed new design criteria for the fuel baskets described in section 2.2.8 of the FSAR: a maximum permanent deflection limit and a maximum primary stress limit. The staff's review of the validity of these proposed design criteria to demonstrate structural safety is discussed in sections 3.5.1 and 3.5.2 of this SER. The proposed changes required the applicant to demonstrate the new criteria were met for the MPC-37, MPC-89, and MPC-44 approved for storage in the UVH overpack.

For determining the maximum permanent deflection limit, the applicant followed the methodology described in the notes of table 3.4.19 of the FSAR. The staff's review of this methodology is discussed further in section 3.5.2 of this SER. The applicant presented the maximum permanent deflection results along with the deflection limit and the safety factor comparing the results to the limit in table 3.I.3.12 of the FSAR for the MPC-37, MPC-89, and MPC-44 baskets in the UVH overpack. These results all show safety factors greater than one, indicating the fuel baskets meet the maximum permanent deflection criterion and maintain appropriate spacing between fuel assemblies after the tip-over event.

For determining the maximum primary stress limit, the applicant followed the methodology employing the "enhanced" FEA model, as described in FSAR section 3.4.4.1.4e. The staff's review of this methodology is discussed further in section 3.5.2 of this SER. As mentioned in FSAR section 3.I.3.8, the MPC-37 fuel basket is chosen as the most limiting fuel basket employed in the UVH overpack, thus its stress results are considered to bound those that would result for the MPC-89 and MPC-44 baskets. The basis for selection of limiting fuel basket and overpack pairings for stress evaluation is explained in FSAR section 3.4.4.1.4e and is discussed in SER section 3.5.2. The applicant presented the resulting stress contour plots for the MPC-37 fuel basket in FSAR figures 3.I.3.14a to d. These contour plots mostly indicate that stresses in the fuel baskets are below the primary stress limit. However, several figures do show small, localized stress areas exceeding the primary stress limit at discontinuities (e.g., notches where basket panels meet). As discussed in section D.3 of appendix D to revision 3 to HI-2210313, the applicant classified these stresses as secondary stresses, which are not subject to the primary stress criterion. The staff reviewed the results and FEAs for these stresses and finds the applicant's classification of these exceedances as secondary stresses to be consistent with the ASME B&PV Code Section III, Division 1, Subsection NG. As discussed in NUREG-2215, the ASME B&PV Code is an acceptable design code for fuel baskets under accident conditions, and therefore the applicant's stress classification is acceptable. The staff notes that increased stresses near structural discontinuities are typically categorized as secondary stresses. Secondary stresses are self-limiting, which means local yielding and minor distortions can occur and alleviate the stress build-up and a single occurrence of an increased secondary stress is not expected to jeopardize structural integrity.

The applicant evaluated the UVH closure lid bolts using hand calculations in appendix C of HI-2210313. During the tip-over, the lid bolts resist the centrifugal action of the lid that occurs after impact. Using the maximum axial deceleration of the lid from the FEA models, the applicant performed a stress analysis of the lid bolts. Compared to the ASME allowable stress limits for the bolts, the applicant calculated a critical safety factor of 1.26 against internal thread shear. In another hand calculation, the applicant evaluated the fillet weld joint between the cask outer shell and the cask top plate that resists the shear load generated by the deceleration of the UVH closure lid in section 9 of HI-2210313. The applicant also calculated a safety factor greater than one for the weld.

The staff finds the applicant's methodology for the FEA and stress analysis performed for the HI-STORM FW UVH cask subjected to the tip-over accident condition to be consistent with the ASME B&PV design code and the guidance on FEA and stress analysis in NUREG-2215. The staff reviewed the results of the tip-over analysis and concludes the following:

- Based on the FEA results showing the maximum permanent deflection and maximum primary stress of the fuel basket are below the allowable limits, the staff finds the applicant has adequately demonstrated that the fuel baskets in the UVH overpack have sufficient structural integrity to meet the criticality safety requirements of 10 CFR 72.236(c) under the tip-over accident conditions.

- Based on the FEA results showing only minor local plastic strain in the overpack, the staff finds the applicant has adequately demonstrated that the shielding capacity of overpack will not be significantly compromised. Based on the safety factors of the stress analyses for the lid bolts and the outer-shell-to-top-plate weld being greater than one, the staff finds the applicant has adequately demonstrated that the closure lid will remain in place and the MPC will remain within the UVH overpack following the tip-over accident. Therefore, the staff finds the HI-STORM FW UVH has sufficient structural integrity to meet the shielding requirements of 10 CFR 72.236(d) under the tip-over accident conditions.

- Based on the FEA results showing only minor local plastic strain in the overpack, the staff finds the applicant has adequately demonstrated that the overpack will not suffer gross deformation that would affect the retrievability of the MPC. Therefore, the staff finds the HI-STORM FW UVH has sufficient structural integrity to meet the retrievability requirements of 10 CFR 72.122(l) under the tip-over accident conditions.

- Based on the FEA results showing only minor local plastic strain in the MPC vessel, the staff finds the applicant has adequately demonstrated that the MPC confinement boundary will not be breached. Therefore, the staff finds the HI-STORM FW UVH has sufficient structural integrity to meet the confinement requirements of 10 CFR 72.236(d) and 72.236(l) under the tip-over accident conditions.

3.2 PC #6 – Tornado Missile Stability Analysis

The applicant proposed updates and clarifications to the methodology for tornado missile stability analysis used to evaluate the loaded HI-STORM FW casks and HI-TRAC VW transfer casks under the tornado accident conditions. The applicant described the methodology for the tornado missile stability analysis in section 3.4.4.1.3 of the application with further details in HI-2094392. The applicant updated the analyzed weights used in the tornado stability analysis for the HI-TRAC VW transfer cask. The applicant reanalyzed the transfer casks with 50% of the fuel loaded. The applicant listed the bounding HI-TRAC VW weight for the tornado stability analysis in table 3.2.8 of the application. This lighter weight was more conservative than the analysis of the fully loaded transfer cask and bounds the use of the HI-TRAC VW with at least half the loaded fuel for the tornado stability analysis. The applicant listed the calculated sliding displacements for each version of the transfer cask from the tornado missile in table 3.4.16 of the application. The staff notes that the distances in table 3.4.16 determine the minimum distance that should be maintained between the HI-TRAC VW and an obstacle that could affect its stability. The staff finds the applicant’s methodology to be consistent with the guidance in NUREG-2215.

The applicant listed the calculated rotations and safety factors in tables 3.4.5, 3.4.5A, and 3.4.5B of the application. All safety factors from the re-analyses were greater than one. Based on the results of the stability analyses demonstrating safety factors greater than one, the staff finds that the applicant has sufficiently demonstrated that the HI-TRAC VW is not susceptible to overturning during the tornado accident condition and has an adequate structural design to meet the requirements of 10 CFR 72.236 under the tornado accident conditions.

The applicant clarified assumptions about the bounding angle of incidence and velocity of the tornado-borne missile in analyzing the potential overturning caused by the tornado accident. Previously, the applicant had described these conditions as the angle that would maximize overturning. The applicant clarified this bounding assumption as the worst case of a horizontal impact at maximum horizontal velocity or the angle that maximizes the moment arm for overturning with a corresponding component of the maximum horizontal velocity. The applicant depicted the angle of incidence that maximizes the moment arm for overturning in FSAR figure 3.4.7. The staff finds the clarification to be consistent with the tornado missile analyses performed for the HI-STORM FW system.

3.3 PC #7 – Add MPC-44

The applicant proposed adding an MPC for storage of PWR fuel, MPC-44, which consists of an MPC-37 enclosure vessel and a basket design with 44 storage cells and CBS. The applicant described the structural design of MPCs in sections 3.1.1(i) and 3.1.3.2 of the application. The applicant updated FSAR tables 1.0.1 and 1.I.1.2 to list the MPCs and transfer casks that are intended to be used with each overpack, which shows that MPC-44 is proposed for storage in the HI-STORM FW Version E and UVH overpacks.

The applicant depicted the MPC-44 fuel basket in Drawing No. 12288 included in the application section 1.5 and the MPC-37 enclosure vessel used for MPC-44 in Drawing No. 6505. The CBS basket design has the shims bolted to extensions of the basket panels. As described in FSAR section 3.1.1(i), the applicant previously established that the CBS basket configuration only experienced significant loads during the tip-over accident condition, which bounds the structural evaluation of all other credible accident conditions.

The applicant evaluated MPC-44 for the tip-over accident condition as described in section 3.4.4.1.4c of the application. The applicant analyzed the tip-over of MPC-44 in both the HI-STORM FW Version E and the HI-STORM FW UVH casks as detailed in appendix D of Holtec proprietary reports HI-2200503, “Analysis of the Non-Mechanistic Tip-over Event of the Loaded HI-STORM FW Version E Storage Cask,” and HI-2210313, respectively. The applicant followed the established methodology for analyzing the HI-STORM FW for a tip-over described in section 3.4.4.1.4a of the application with some modifications to the methodology to incorporate the CBS basket designs. The applicant described the modelling approach for MPC-44 basket designs in section 3.4.4.1.4c of the application. The staff’s review of the tip-over analysis of the HI-STORM FW UVH with the stored MPC-44 is discussed in section 3.1.2.10 of this SER.

The applicant analyzed the tip-over accident for the HI-STORM FW Version E using LS-DYNA. The applicant modified the existing half-symmetry LS-DYNA tip-over model for the MPC-37 in the version E overpack to include the fuel basket and shims of the MPC-44. The continuous basket shims are modeled using the same approach employed for the MPC-89-CBS, which is described in FSAR subsection 3.4.4.1.4b.

The applicant depicted the stresses in the basket shims from the tip-over model results in figure 3.4.46B, which shows the stresses in the shims to largely be below yield with only localized plastic deformation. From these results, the applicant concluded that the structural design criteria of the shims were met in the tip-over accident, which require the shims to remain attached to the basket and maintain their physical integrity.

The applicant compared the maximum permanent deflections of the fuel basket panels from the tip-over model results to the deflection design criterion in section 2.2.8 of the application. The deflection design criterion and the changes made to the deflection design criterion for this amendment are discussed in section 3.5.1 of this SER. The applicant’s determination of the maximum permanent deflections of the fuel basket panels from the FEAs following the methodology described in the notes of table 3.4.19 of the FSAR is discussed further in section 3.5.2 of this SER. The applicant listed the maximum permanent deflections of the fuel basket panels in table 3.4.20 of the application along with the allowable deflection limit from table 2.2.11 and the safety factor of 3.68, resulting from a comparison of the maximum and allowable deflections. Based on the staff’s acceptance of the permanent deflection determination methodology per SER section 3.5.2, and the deflection safety factor being greater than one, the staff finds the fuel basket meets the permanent deflection criterion and maintains appropriate spacing between fuel assemblies after the tip-over event.

The applicant compared the maximum primary stresses in the fuel basket from the tip-over model results to the stress design criterion in section 2.2.8 of the application. For determining the maximum primary stress, the applicant followed the methodology employing the “enhanced” FEA model, as described in FSAR section 3.4.4.1.4e. The staff’s review of this methodology is discussed further in section 3.5.2 of this SER. Based on the staff’s acceptance of the stress determination methodology per SER section 3.5.2 and the following results, the staff finds that the MPC-44 fuel basket meets the maximum primary stress criterion as the applicant described in section 3.4.4.1.4c of the FSAR:

- stress contour plots for the MPC-44 fuel basket in the Version E overpack are bounded by those presented for the MPC-37-CBS in FSAR figures 3.4.49a to h, and
- stress contour plots for the MPC-44 fuel basket in the UVH overpack are bounded by those presented for the MPC-37 fuel basket in FSAR figures 3.I.3.14a to d.

The applicant listed the maximum plastic strains for the MPC enclosure vessel, overpack, and cask lid bolts in table 9.1 of HI-2200503 for the Version E overpack and FSAR table 3.I.3.9 for the UVH overpacks. The analysis results showed that the MPC enclosure vessel, and overpack experienced local plastic deformation. For the overpack lid, negligible local plastic strain occurred in a closure lid bolt, and the maximum cask lid accelerations were bounded by the maximum lid acceleration previously analyzed for the HI-STORM FW Version E in HI-2094418. The applicant determined that all of the maximum local plastic strains were less than the material failure strains, which it determined from ASME material property data.

The staff finds the applicant’s methodology for the FEA and stress analysis performed for MPC-44 subjected to the tip-over accident condition to be consistent with the applicable portions of the previously approved methodology for the HI-STORM FW system tip-over analysis and with the guidance on FEA in NUREG-2215. The staff reviewed the results of the tip-over analysis for Version E and UVH overpacks and concludes the following:

- Based on the FEA results, the staff finds the applicant has adequately demonstrated that the fuel baskets will not deflect more than the maximum permanent deflection limit and the primary stresses in the baskets will remain below the maximum primary stress limit. Based on the predominately elastic FEA results of the shims and the analysis approach employed for the bolts, the staff finds the shims will remain attached to the basket and maintain their physical integrity. Therefore, the staff finds the HI-STORM FW with the stored MPC-44 has adequate structural integrity to meet the criticality safety requirements of 10 CFR 72.236(c) under the tip-over accident conditions.

- Based on the FEA results demonstrating that no gross plastic deformation occurred in the overpack, the staff finds the applicant has adequately demonstrated that the shielding capacity of overpack will not be significantly compromised. Based on the decelerations in the overpack lid being less than the bounding decelerations for the HI-STORM FW overpack lid, the staff finds the applicant has adequately demonstrated that the closure lid will remain in place and MPC-44 will remain within the overpack following the tip-over accident. Therefore, the staff finds the HI-STORM FW with the stored MPC-44 has sufficient structural integrity to meet the shielding requirements of 10 CFR 72.236(d) under the tip-over accident conditions.

- Based on the FEA results demonstrating that no gross plastic deformation occurred in the inner shell of the overpack, the staff finds the applicant has adequately demonstrated that the overpack will not affect the retrievability of the MPC. Therefore, the staff finds the HI-STORM FW with the stored MPC-44 has sufficient structural integrity to meet the retrievability requirements of 10 CFR 72.122(l) under the tip-over accident conditions.

- Based on the FEA results demonstrating only minor local plastic strain in the MPC-44 vessel, the staff finds the applicant has adequately demonstrated that the MPC confinement boundary will not be breached. Therefore, the staff finds the HI-STORM FW system with the stored MPC-44 has sufficient structural integrity to meet the confinement requirements of 10 CFR 72.236(d) and 72.236(l) under the tip-over accident conditions.

3.4 PC #8 – Add MPC-37P

The applicant proposed adding an MPC for storage of PWR fuel, MPC-37P, which consists of an MPC-37 enclosure vessel and a basket design with CBS. The applicant described the structural design of MPCs in sections 3.1.1(i) and 3.1.3.2 of the application. The applicant updated FSAR table 1.0.1 to list the MPCs and transfer casks that are intended to be used with each overpack, which shows that the MPC-37P is proposed for storage only in the HI-STORM FW Version E overpack.

The applicant depicted the MPC-37P fuel basket in Drawing No. 12283 included in the application section 1.5 and the MPC-37 enclosure vessel used for MPC-37P in Drawing No. 6505. The applicant designed the MPC-37P basket to be similar to the MPC-37-CBS basket, except with thicker basket panels and smaller cell width. The CBS basket design has the shims bolted to extensions of the basket panels. As the applicant described in section 3.1.1(i) of the application, the fuel basket experiences the most significant loads during the tip-over accident condition, which bounds the structural evaluation of other credible accident conditions.

The applicant evaluated the MPC-37P for the tip-over accident condition as described in section 3.4.4.1.4d of the application with further discussion in appendix C of HI-2200503. The applicant determined that the structural design of MPC-37P was bounded by the MPC-37-CBS based on the similarity of the MPCs and the following differences: (1) the thicker basket panels would be less susceptible to bending stress and deflections than the MPC-37-CBS; (2) as shown in tables 2.1.1a and 2.1.1c of the application, the weight of each fuel assembly and total weight of the assemblies is less for the MPC-37P than the MPC-37-CBS; (3) the smaller cell width reduces the gap between the fuel assemblies and the basket panels; and (4) the temperature distribution of the MPC-37P is bounded by the MPC-37-CBS as discussed in the thermal evaluation. Based on the review of the basket geometries, loading, and temperatures, the staff finds the applicant's assessment sufficient to conclude that the structural capacity of the MPC-37P fuel basket will be greater than that of the MPC-37-CBS and the structural demands on the MPC-37P will be less than those on the MPC-37-CBS for the tip-over accident, and therefore, the structural design of the MPC-37P is bounded by the MPC-37-CBS.

Though the applicant did not include MPC-37-CBS as part of this amendment, the staff reviewed the applicant's analyses of MPC-37-CBS where those analyses were cited by the applicant as bounding the design and analysis of the MPC-37P. The applicant described the tip-over evaluation of the MPC-37-CBS and MPC-37P in the Version E overpack in section 3.4.4.1.4d of the application with further discussion in appendix C of HI-2200503. The applicant followed the same methodology for tip-over evaluation of the MPC-37-CBS as that of the MPC-44 discussed in the previous section of this SER.

The applicant compared the maximum permanent deflections of the MPC-37-CBS fuel basket from the tip-over model results to the deflection design criterion in section 2.2.8 of the application. The deflection design criterion and the changes made to the deflection design criterion for this amendment are discussed in section 3.5.1 of this SER. The applicant's determination of the maximum permanent deflections of the fuel basket panels from the FEAs following the methodology described in the notes of table 3.4.19 of the FSAR is discussed further in section 3.5.2 of this SER. The applicant listed the maximum permanent deflections of the fuel basket panels in table 3.4.20 of the application along with the allowable deflection limit from table 2.2.11 and the safety factor of 1.25 comparing the maximum and allowable deflections. Based on the staff's acceptance of the permanent deflection determination methodology per SER section 3.5.2, and the deflection safety factor being greater than one, the staff finds the fuel basket meets the permanent deflection criterion and continues maintaining appropriate spacing between fuel assemblies after the tip-over event.

The applicant compared the maximum primary stresses in the fuel basket from the tip-over model results to the stress design criterion in section 2.2.8 of the application. For determining the maximum primary stress, the applicant followed the methodology employing the "enhanced" FEA model, as described in FSAR section 3.4.4.1.4e. The staff's review of this methodology is discussed further in section 3.5.2 of this SER. The applicant presented the resulting stress contour plots depicting any stresses that exceed the stress limit in figure 3.4.49 of the application for the MPC-37-CBS fuel basket in the Version E overpack. These contour plots mostly indicate that stresses in the fuel baskets are below the primary stress limit. However, several figures do show small, localized stresses exceeding the primary stress limit at discontinuities (e.g., notches where basket panels meet). As discussed in section G.3 of revision 9 to HI-2200503, the applicant classified these stresses as secondary stresses, which are not subject to the primary stress criterion. The staff reviewed the results and FEAs for these stresses and finds the applicant's classification of these exceedances as secondary stresses to be consistent with the ASME B&PV Code Section III, Division 1, Subsection NG, which the staff accepts as design criteria for fuel baskets under accident conditions as discussed in NUREG-2215, and is therefore acceptable. The staff notes that secondary stresses are self-limiting and a single occurrence of a secondary stress increase is not expected to jeopardize structural integrity.

For the MPC-37-CBS, the applicant depicted the stresses in the basket shims from the tip-over model results in figure 3.4.46C of the application, which shows the stresses in the shims to largely be below yield with only localized plastic deformation. From these results, the applicant concluded that the shims remain attached to the basket and maintain their physical integrity. The applicant listed the maximum plastic strains for the MPC enclosure vessel, overpack, and cask lid bolts in table 9.1 of HI-2200503. The analysis results showed that the MPC enclosure vessel, and overpack experienced local plastic deformation less than or equal to the material failure strains listed in table 9.1 of HI-2200503, which it determined from the ASME material property data.

Based on the staff's acceptance of the stress and permanent deflection determination methodology per SER section 3.5.2, and the comparative geometry, loading, and temperature evaluation that demonstrates the structural design of the MPC-37P is bounded by the MPC-37-CBS, the staff finds the applicant has adequately demonstrated that the HI-STORM FW Version E system with the stored MPC-37P has an adequate structural design to meet the acceptance criteria for the tip-over accident listed in section 2.2.3.b of the application and the requirements of 10 CFR 72.236 under the tip-over accident conditions.

3.5 PC #15 - Adopt A Stress-Based Structural Design Criterion PC #16 - Establish Specific Criteria on Allowable Interference Due To DTE

Holtec proposed the following changes in Attachment 1 to its November 17, 2023 submittal (Holtec, 2023f), "Summary of Additional Clarifications/Changes Submitted for Review":

- Adopting a Stress-Based Structural Design Criterion
- Establishing an Allowable Interference Due to DTE

These two proposed changes are generic methods of evaluation. The permanent deflection criterion, coupled with the proposed stress-based structural design criterion, applies to all fuel baskets in the FW storage system, including those in the scope of this amendment. The proposed DTE criterion does not apply to any MPCs in the scope of this amendment.

3.5.1 Fuel Basket Design Criteria

As part of this amendment, the applicant proposed changes to the design criterion for the fuel basket in section 2.2.8 of the FSAR. The existing design criterion for the fuel basket consists of a single limit on the deflection of a fuel basket panel that ensures both the structural integrity of the fuel basket and bounds the initial conditions assumed in the criticality analysis (e.g., fuel assembly spacing). Deflections in the basket are caused by lateral loads; the most significant of which is the tip-over accident. The current description of the deflection criterion is for the maximum total deflection at any location along a basket panel to be limited to 0.5% of the width of the basket panel (i.e., inner width of a storage cell) at all times.

For this amendment, the applicant proposed changing the fuel basket design criteria in section 2.2.8 of the FSAR to consist of two requirements to demonstrate safe performance of the basket: (1) limit the maximum permanent deflection of a basket panel within the active fuel region to 0.5% of the panel width as listed in table 2.2.11 of the FSAR and (2) limit the maximum primary (membrane plus bending) stress within the active fuel region to 90% of the true ultimate strength of the basket material at its design temperature. As discussed in section 2.2.8 of the FSAR, the applicant requires both criteria to be met to demonstrate adequate structural integrity of the fuel basket.

Instead of limiting the total deflection (i.e., elastic plus plastic deflection), the proposed change would limit just the portion of the maximum deflection caused by plastic deformation (i.e., permanent deflection) of the basket panel. The methodology for determining this value is discussed further in section 3.5.2 of this SER. The applicant established the permanent deflection limit to ensure the bounding conditions assumed in the criticality analysis were maintained.

During its review, the staff requested additional information and justification from the applicant about the reliance solely on a permanent deflection value to demonstrate structural integrity through several RAIs and clarification calls (NRC, 2023b through 2023h). In summary, the staff noted that the permanent deflection limit reduced the margin of safety for the structure compared to the total deflection limit, and the relationship between the deflection limit and the structural integrity of the basket varied with basket geometries and temperatures. The staff also noted that a deflection limit does not directly demonstrate structural integrity. The staff expressed concern that the conjunction of these factors could jeopardize the structural integrity of the basket as certain basket geometries and temperatures may not maintain structural integrity at the deflection limit.

To address these concerns and more definitively demonstrate structural integrity for all basket designs authorized for the FW system, the applicant proposed establishing a second criterion for the fuel baskets that limits the primary (membrane plus bending) stresses to 90% of the true ultimate strength of the basket material at its design temperature, which would implicitly include the effects of both elastic and plastic deflections. The staff considers this primary stress criterion to be comparable to the level D stress limits prescribed in the ASME B&PV Code Section III, Division 1, Subsection NG, which the staff accepts as design criteria for fuel baskets under accident conditions as discussed in NUREG-2215. ASME subsection NG allows for level D stress limits to include 90% of the ultimate strength in plastic analyses like those in the FEAs that the applicant performs to evaluate the fuel baskets for accident conditions. The staff finds that the proposed primary stress limit provides a similar margin against structural failure of the fuel basket as the ASME level D stress limits. With this limit on the primary basket stresses, it is implicit that the total effective basket stresses are also limited to the true fracture stress.

Based on the similarity between the primary stress criterion and the basket stress criteria described in NRC guidance, the staff finds the proposed design criteria to be acceptable in demonstrating the structural integrity of the fuel basket to meet the criticality safety requirements of 10 CFR 72.236(c) under accident conditions.

3.5.2 Demonstration of the Fuel Basket Design Criteria

The proposed changes to the fuel basket design criteria (described in section 3.5.1 of this SER) required the applicant to demonstrate that the proposed new criteria were met for the combinations of fuel basket designs, MPCs, and overpacks approved for storage in this amendment. The staff's review of analyses demonstrating the new design criteria for baskets, MPCs, and overpacks added in this amendment is described above in their respective sections of this SER. For combinations of basket designs, MPCs, and overpacks that were previously approved for storage in the HI-STORM FW system, the applicant relied on its previously performed FEA to obtain permanent deflection results for the fuel baskets to demonstrate that the deflection design criterion was met. To demonstrate that the primary stress criterion was met for all combinations of new and existing baskets, MPCs, and overpacks, a selected (e.g., limiting) number of cases were evaluated, employing an "enhanced" FEA model, as discussed in FSAR section 3.4.4.1.4e.

The applicant described the evaluation of the fuel baskets for the permanent deflection design criteria in section 3.4.4.1.4a of the FSAR. These evaluations considered the tip-over accident condition, which, the staff notes, is the most significant accident condition for the structure of the fuel basket and for assessing the design criteria. The applicant relied on finite element models (FEMs) previously created in LS-DYNA to analyze the tip-over accident for the basket deflection determination and design of other components. The applicant did not make changes to these models to support this amendment. The applicant merely extracted results related to the new design criterion from the existing models. The applicant described these FEAs in section 3.4.4.1.4a of the FSAR with further details in the following proprietary FEA reports: revision 21 of HI-2094353 for MPC-37 and MPC-89 in the standard overpack; revision 4 of HI-2166998 for MPC-32ML in the standard overpack; and revision 9 of HI-2200503 for MPC-32ML, MPC-37, and MPC-89 in the version E overpack. Since the applicant did not make changes to the associated FEMs and analyses as part of this amendment, the staff did not perform a full review of each entire FEM but instead focused on verifying the FEA results and applicant's use of those results to demonstrate satisfaction of the fuel basket design criteria.

As described in the notes to table 3.4.19 of the FSAR, the applicant determined the maximum permanent deflections of the fuel basket panels from the tip-over model results using the following method: (1) identifying critical times and locations from the stress and strain contour plots; (2) determining the total (i.e., elastic and plastic) deflection of the panel from the displacement of the midspan relative to the displacement of the end points; (3) determining the deflection at yield by repeating the previous steps for a time-step when the maximum stress in the panel is at yield; and (4) subtracting the deflection at yield from the total deflection to determine the plastic deflection at the critical time and location. The applicant presented the maximum permanent deflection results along with the deflection limit from table 2.2.11 of the FSAR and the safety factor comparing the results to the limit in FSAR table 3.4.19 for the standard overpack (for MPC-32ML, MPC-37 and MPC-89) and table 3.4.20 for the Version E overpack (for MPC-32ML, MPC-37, and MPC-89). Table 3.4.21 presents the results for the two tip-over cases for MPC-37 described in section 3.4.4.1.4a. These results all show safety factors greater than 1, indicating that the MPC-32ML, MPC-37, and MPC-89 meet the maximum permanent deflection criterion. Therefore, the staff finds the fuel baskets continue maintaining appropriate spacing between fuel assemblies after the tip-over event. Refer also to sections 3.1.2.10, 3.3 and 3.4 of this SER for a discussion of permanent deflection results for the other fuel basket and overpack pairings approved in this amendment.

The applicant employed an "enhanced" FEA to determine whether the maximum primary stress criterion for the fuel basket was met by using results from a smaller (e.g., limiting) set of basket and overpack combinations determined to provide limiting stress results. The FEA model enhancements and the reasons for the selection of each basket and overpack combination is described by the applicant in FSAR section 3.4.4.1.4e. As stated earlier in this SER, the staff reviewed the applicant's tip-over analysis results for the MPC-37-CBS and MPC-89-CBS fuel baskets because the applicant referenced these analyses in demonstrating the safety and regulatory compliance of other basket and overpack combinations proposed in this amendment. The staff notes that although the applicant did not include MPC-37-CBS and MPC-89-CBS fuel baskets (used in the standard and Version E overpacks) as part of this amendment, the staff reviewed the applicant's tip-over stress analyses of MPC-37-CBS and MPC-89-CBS where those analyses were cited by the applicant as bounding the design and analysis of basket combinations that were included in this amendment. The staff also notes that since the applicant did not make significant changes to the associated FEMs and analyses as part of this amendment, the staff did not perform a review of all changes to each FEM but instead focused on verifying the FEA results and applicant's use of those results to demonstrate satisfaction of the fuel basket design criteria. The staff finds the "enhanced" modeling approach acceptable as: (1) the changes in basket element formulation provide more realistic structural results, (2) the reduced temperature values for basket thermal zones remain bounding, and (3) the FEA erosion limits chosen produce conservative results.

In FSAR section 3.4.4.1.4e, the applicant explains the basis for the selection of this "limiting" analysis subset considered: (1) basket geometry and applied loading, (2) basket temperatures, (3) basket construction method, and (4) overpack combinations. The following basket and overpack combinations were chosen as limiting and were evaluated using the "enhanced" FEM: MPC-32ML, MPC-37, MPC-37-CBS in the Version E overpack, MPC-89-CBS in the standard overpack, and the MPC-37 in the UVH overpack (see section 3.1.2.10 of this SER for further discussion). The staff finds the applicant's basis for selection of basket and overpack combinations to produce limiting stress results to be acceptable. Additionally, the previous analysis results of these combinations, when employing an FEA model with no enhancements, produced minimal stress margins.

The applicant described the methodology for verifying the stress criterion in section 9 of revision 4 to HI-2166998. Following this methodology, the applicant generated stress contour plots for each temperature region of the analyzed fuel baskets. The staff notes these stress contour plots display total effective stress, not only primary stresses. This means that certain small spots of color may or may not indicate exceedance of the primary stress criterion and would require further evaluation and justification by the applicant. The applicant presented the stress contour plots in the following FSAR figures of the application:

- Figures 3.4.48a to d for the MPC-32ML fuel basket in the Version E overpack,
- Figures 3.4.49a to h for the MPC-37-CBS fuel basket in the Version E overpack,
- Figures 3.4.50a to d for the MPC-37 fuel basket in the Version E overpack,
- Figures 3.4.51a to c for the MPC-89-CBS fuel basket in the standard overpack, and
- Figures 3.I.3.14a to d for the MPC-37 fuel basket in the UVH overpack.

For a discussion of the stress contour plots of the MPC-37 in the UVH overpack, refer to section 3.1.2.10 of this SER. The stress contour plots of the most limiting baskets for the Version E and standard overpacks indicate that the majority of stresses in the fuel baskets are below the primary stress limit. However, several figures do show small, localized stress areas exceeding the primary stress limit at discontinuities (e.g., notches where basket panels meet). For the baskets in the Version E overpack, these exceedances are discussed in section G.3 of Appendix G to revision 9 to HI-2200503, where the applicant classified these stresses as secondary stresses, which are not subject to the primary stress criterion. For the MPC-89-CBS basket in the standard overpack, these exceedances are discussed in section N1.3 of Appendix N1 to revision 21 to HI-2094353, where the applicant also classified the stresses at notches and vertical panel intersections as secondary stresses, which are not subject to the primary stress criterion. The applicant explains that the slight stress exceedances at the bottom of the fuel panel in FSAR figure 3.4.51c are also categorized as secondary (or peak) stresses due to their location at a horizontal panel intersection. The staff reviewed the results of the FEAs for these stresses and finds the applicant's classification of these exceedances as secondary stresses to be consistent with the ASME B&PV Code Section III, Division 1, Subsection NG. As discussed in NUREG-2215, ASME B&PV Code is an acceptable design code for fuel baskets under accident conditions, and therefore the applicant's stress classification is acceptable. The staff notes that increased stresses near structural discontinuities are typically categorized as secondary stresses. Secondary stresses are self-limiting, which means local yielding and minor distortions can occur and alleviate the stress build-up and a single occurrence of a secondary stress is not expected to jeopardize structural integrity.

Based on the FEA results of the maximum permanent deflections and stress contour plots, the staff finds that the applicant has adequately demonstrated that the following fuel baskets satisfy the proposed design criteria, which the staff finds demonstrate the structural integrity of the fuel baskets, as discussed in the previous section of this SER, for the following assemblies: (1) the standard overpack with MPC-32ML, MPC-37 and MPC-89, (2) the Version E overpack with MPC-32ML, MPC-37, MPC-37P, MPC-44, and MPC-89, and (3) the UVH overpack with MPC-37, MPC-44, and MPC-89. Therefore, the staff finds that the above-stated fuel baskets have sufficient structural integrity to meet the criticality safety requirements of 10 CFR 72.236(c) under the tip-over accident conditions. Refer also to sections 3.1.2.10, 3.3 and 3.4 of this SER for a discussion of permanent deflection and stress results for the other fuel basket and overpack pairings approved in this amendment.

3.5.3 Differential Thermal Expansion – Fuel Basket-to-Enclosure Vessel

As part of this amendment, the applicant proposed an additional differential thermal expansion (DTE) methodology and acceptance criteria for addressing situations where the size of the radial gap between the fuel basket with shims and MPC enclosure vessel is exceeded by the DTE of the fuel basket. The applicant described the impacts of this new DTE methodology and acceptance criteria on the structural design in section 3.1.1(i) of the FSAR. The staff notes the applicant did not make changes to the method by which DTE is determined.

The FSARs for amendments 0 through 6 previously approved by staff required that the HI-STORM FW system maintain a radial gap between the fuel basket and the inside wall of the enclosure vessel at the maximum design basis heat load. Maintaining a radial gap at this junction eliminates the possibility of interference stresses caused by the fuel basket expanding into the enclosure vessel. All combinations of fuel baskets and MPC enclosure vessels, reviewed as part of this amendment, maintain the radial gap as discussed in section 4.3.4 of this SER. As part of this amendment, the applicant proposed a generic methodology for allowing and addressing a small amount of interference if a fuel basket were to expand slightly beyond the radial gap and contact the MPC enclosure vessel.

The proposed design criterion for this methodology allows the fuel basket and shims to expand 0.030 inches beyond the size of the radial gap between the basket and enclosure vessel due to DTE. The applicant listed this value in table 3.1.15 of the FSAR as the allowable interference. Based on engineering judgement, the applicant considers that no significant distortions can occur because of the small allowable limit on this interference. The staff notes that a smaller gap or a closed gap means heat transfer would be greater in the radial direction, thus the heat removal function would not be negatively affected. The staff also notes that interference stresses can occur in each component if the combined radial gap is closed. However, the applicant concluded that any amount of interference stresses would be small and would not meaningfully contribute to the ASME B&PV Code strength evaluation, i.e., the stresses resulting from the small amount of allowable interference would be negligible compared to the stress from other loads and would not result in exceeding the ASME B&PV Code stress limit established for the component.

The applicant also concluded that fatigue failure due to DTE interference was not credible because temperature fluctuations inside the cask are sufficiently minor. Thus, the applicant proposed to limit the interference to the value in table 3.1.15 of the FSAR and require no further analysis of the interference.

Based on the small amount of interference allowed, the staff considers that limiting the interference to the value in table 3.1.15 of the FSAR is sufficient to ensure the structural capacity of the MPC enclosure vessel and fuel basket is maintained. The staff notes that limiting the interference to this small amount will ensure that any stresses that result from the interference will also be small and likely negligible. Although the applicant has not quantified the stress from this interference, the staff recognizes that the allowable stress limits ensure a margin against structural failure of the components, which the applicant demonstrates through analysis for other loads. In the staff’s engineering judgement, the small increase in stress from the allowable interference, if not insignificant, is not enough to overcome this margin and cause the structural failure of either component. Additionally, the staff notes that if a loading scenario exceeded the stress limits, this would first result in deformation to the edges of the basket structure, outside of the active fuel region, which would alleviate the interference and the resulting stress.

The staff considers that the MPC and fuel basket are not susceptible to fatigue failure from the allowable DTE interference due to the lack of thermal cycles. Therefore, the staff finds the HI-STORM FW system has sufficient structural integrity to meet the confinement requirement of 10 CFR 72.236(l).

3.6 Evaluation Findings

Based on the analyses performed and the supporting information provided by the applicant, the staff concludes that the structural design of the HI-STORM FW system discussed in chapter 3 of this SER complies with 10 CFR Part 72 and provides adequate protection of the public health and safely. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. The following findings are made:

F3.1 The staff reviewed the structural performance of the ITS SSCs designed to maintain subcriticality and concludes that these SSCs have adequate structural integrity to satisfy the criticality safety requirements of 10 CFR 72.124(a).

F3.2 The staff reviewed the structural performance of the ITS SSCs designed to provide and maintain favorable geometry or permanently fixed neutron-absorbing materials and concludes that these SSCs have adequate structural integrity to satisfy the criticality control requirements of 10 CFR 72.124(b).

F3.3 The staff reviewed the design bases and design criteria of the ITS SSCs and concludes that the applicant met the requirements of 10 CFR 72.236(b).

F3.4 The staff reviewed the structural performance of the ITS SSCs designed to maintain the spent nuclear fuel (SNF) in a subcritical condition under normal, off-normal, and accident conditions and concludes that these SSCs have adequate structural integrity to satisfy the subcriticality requirements of 10 CFR 72.236(c).

F3.5 The staff reviewed the structural performance of the ITS SSCs designed to provide radiation shielding and confinement and concludes that these SSCs have adequate structural integrity to satisfy the radiation shielding and confinement requirements of 10 CFR 72.236(d).

F3.6 The staff reviewed the structural performance of the ITS SSCs designed to provide redundant sealing of confinement systems and concludes that these SSCs have adequate structural integrity to satisfy the requirements of 10 CFR 72.236(e).

F3.7 The staff reviewed the structural performance of the ITS SSCs and concludes that these SSCs have adequate structural integrity to store the spent fuel safely for the term proposed in the application and satisfy the requirements of 10 CFR 72.236(g).

F3.8 The staff reviewed the structural evaluations of the storage cask and its ITS SSCs and concludes that these evaluations considered appropriate tests and means acceptable to the NRC to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions and, therefore, meet the requirements of 10 CFR 72.236(l).

4.0 THERMAL EVALUATION

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

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

- PC #1 – Add a new unventilated overpack, HI-STORM FW UVH.
- PC #5 – Include the ability to use CFD analysis to evaluate site-specific fire accident scenario.
- PC #7 – Add the new MPC-44 to be used with Version E and UVH overpacks.
- PC #8 – Add the new MPC-37P to be used with version E overpack.
- PC #9 – Add HI-DRIP ancillary system.
- PC #10 – Include the ability to use CFD analysis to evaluate site-specific burial-under-debris accident scenario.
- PC #11 – Include the ability to use water without glycol in the HI-TRAC water jacket during transfer operations below 32°F based on the site-specific MPC total heat loads.

4.1 PC #1 – Add Unventilated Overpack, HI-STORM FW UVH

The applicant proposed, in the application chapter 4.I, “Thermal Evaluation,” to add the HI-STORM FW UVH with a minimum thermal conductivity of 1.5 Btu/hr-ft-°F, which supports loading of MPC-37, MPC-44, and MPC-89. The applicant stated the HI-STORM FW UVH does not rely on ventilation passages for its means of cooling, therefore, the limiting condition for operation (LCO) 3.1.2 under section 3.1.2, “SFSC Heat Removal System,” of TS appendix A is not applicable for HI-STORM FW UVH.

The staff accepts the addition of UVH to support loading of MPC-37, MPC-44, and MPC-89 in the FW system based on staff’s evaluation of the proposed heat load patterns in UVH under normal, short-term, off-normal, and accident conditions discussed in the following sections. Based on the reduced heat load and the fuel cladding and cask component temperatures calculated under all storage conditions, the staff also accepts that the HI-STORM FW UVH does not rely on ventilation passages for its means of cooling.

4.1.1 Heat Load Patterns

The applicant presented maximum allowable heat loads in table 4.I.1.1 of the application for storage of MPC-37 (29.0 kW), MPC-44 (28.0 kW), and MPC-89 (29.0 kW) in the HI-STORM FW UVH. The staff documents the evaluation of the heat load patterns in the UVH in the following sections.

The applicant proposed to add a minor deviation from the prescribed loading pattern in CoC appendix B, section 2.3.2 to allow one slightly thermally-discrepant fuel assembly per quadrant to be loaded, as long as the peak cladding temperature (PCT) for the MPC remains below the limits in SFST-ISG-11, revision 3 (NRC, 2003a). The staff finds that there would be negligible impact to the thermal performance of the system because the PCT will be below the limits stated in SFST-ISG-11, the cask component temperatures will be below the required limits specified in FSAR table 2.2.3, and the canister internal pressures will be still below the limits specified in FSAR table 2.2.1. Therefore, the staff determines the proposed minor deviation is acceptable for fuel loading decay heat limits for MPC-37, MPC-44, and MPC-89 loaded in UVH.

4.1.2 Thermal Model

The applicant described, in the application section 4.I.4, "Thermal Model and Evaluation of Normal Conditions of Storage," how heat flows and dissipates throughout the system. The applicant also described, in the application section 4.I.0, "Overview," the special design features used to protect the overpack lid concrete from excessive temperature rise.

The staff reviewed the application section 4.I.4 and License Drawing No. 11897, HI-STORM-FW version UVH, and confirmed the number of steel ribs between outer shell and inner steel shell and the conductivity of the shielding concrete are important parameters for the thermal performance of the system. The UVH is also equipped with steel ribs in the overpack lid concrete. The staff reviewed applicant's thermal model and confirmed that thermal model, including ribs and shielding concrete, are consistent with the license drawings. The staff recognizes that the laminar flow in annulus of the UVH results in lower heat transfer when compared to a turbulent flow in annulus of the ventilated overpack.

4.1.3 Effect of Neighboring Casks

The applicant described, in the application section 4.I.4.5, "Impact of Neighboring Casks," the impact of neighboring casks on heat dissipation through the UVH and stated that the impact of the neighboring casks on the temperatures is exacerbated for the UVH when compared to the ventilated overpack. As shown in table 4.I.4.3 of the application, MPC-37 has the highest PCT and is the most bounding MPC for the UVH overpack. The applicant presented the computed temperatures of MPC-37 in table 4.I.4.1 of the application in which the respective design basis limits, as specified in FSAR table 2.2.3, are met.

The staff compared the PCTs in the application table 4.I.4.1 (including the impact of neighboring casks) and table 4.I.4.3 (stand-alone UVH) for the MPC-37 stored in the UVH and finds that the PCT, for the conditions including the impact of neighboring casks, is increased by 7-10°F, but is below the allowable limit of 752 °F stated in NUREG-2215 and ISG-11, with acceptable margin under normal conditions of storage.

4.1.4 Backfill Pressure Limits

The applicant presented the minimum and maximum initial helium backfill pressures for the MPCs stored in the UVH in table 4.I.1.3 of the application and stated that the annulus between the MPC and the UVH is backfilled such that the annulus air design pressure in the application table 4.I.1.3, is satisfied.

The staff reviewed the application section 4.I.1.4, table 4.I.1.3 and Holtec proprietary report No. HI-2094400, revision 34, "Thermal Evaluation of HI-STORM FW," and accepts that (1) the use of 0 psig as an initial annulus fill pressure at a reference temperature of 70°F for the licensing basis calculation is acceptable for normal conditions of storage; and (2) the initial helium backfill pressure limits of the MPCs, stored in the UVH, are established to ensure the operating pressure remains below design pressures, as shown in FSAR table 2.2.1, under storage conditions of the UVH. Therefore, the maximum annulus fill pressure of the air should be lower than the annulus air design pressures specified in table 4.I.1.3 of the application for MPCs stored in the UVH.

4.1.5 Normal Conditions of Storage

The applicant performed the steady state thermal evaluations for MPC-37 (uniform and regionalized patterns), MPC-44 (uniform pattern), and MPC-89 (uniform and regionalized patterns) stored in the UVH and presented the PCTs in table 4.I.4.3 of the application. The applicant presented the PCT and cask component temperatures in table 4.I.4.1 of the application and the MPC cavity and annulus pressures in table 4.I.4.2 of the application for the bounding MPC-37 stored in the UVH.

The staff reviewed the Holtec proprietary calculation package HI-2200191, revision 3, “Thermal Analysis of HI-STORM FW UVH System,” and FSAR tables 4.I.4.1, 4.I.4.2, and 4.I.4.3 for MPC-37, MPC-44, and MPC-89 stored in the UVH and finds that (1) the PCT and cask component temperatures of MPC-37, MPC-44, and MPC-89 are below the allowable temperature limits specified in FSAR table 2.2.3 which was accepted and approved by the NRC in previous amendments; and (2) the MPC cavity and UVH annulus pressures of MPC-37, MPC-44, and MPC-89 are below the allowable pressure limits specified in FSAR table 2.2.1 (for MPC cavity) and table 2.I.2.3 (for annulus), under normal conditions of storage. Therefore, the staff accepts the applicant’s thermal evaluations of MPC-37, MPC-44, and MPC-89 stored in the UVH for normal conditions of storage.

4.1.6 Short-Term Operations

The applicant stated, in the application section 4.I.5, “Thermal Evaluation for Short Term Operations,” that with the HI-TRAC transfer cask remained unchanged for the UVH, and the acceptance criteria of the short-term operations for MPC-37, MPC-44, and MPC-89 will be met because the maximum heat loads qualified for MPC-37, MPC-44, and MPC-89 in the UVH are significant lower than those stored in the ventilated overpack.

The staff reviewed the heat loads qualified for use in MPC-37 (29.0 kW), MPC-44 (28.0 kW), and MPC-89 (29.0 kW) in the UVH and confirmed that the maximum heat load limit for each MPC used in UVH overpack is much lower than the maximum heat load limit allowed for used in ventilated overpack. During short-term operations, using these MPCs with UVH is bounded by using these MPCs with ventilated overpack. Therefore, the acceptance criteria (temperature and pressure) of the short-term operations for MPC-37, MPC-44, and MPC-89 will be met.

4.1.7 Off-Normal Conditions of Storage

The applicant stated in the application section 4.I.6.1, “Off-Normal Conditions,” that the most bounding PCTs for the UVH with MPC-37, MPC-44, and MPC-89 are the corresponding PCTs for the ventilated overpack with MPC-37, MPC-44, and MPC-89 under normal conditions of storage.

The staff reviewed the heat loads qualified for use in MPC-37, MPC-44, and MPC-89 stored in the UVH and the ventilated overpack and confirmed that, based on bounding correlation for normal conditions of storage, the PCTs of MPC-37, MPC-44, and MPC-89 in the UVH will be also bounded by the PCTs of MPC-37, MPC-44, and MPC-89 stored in the ventilated overpack and will be also below the allowable temperature limits specified in FSAR table 2.2.3 for off-normal conditions of storage. The staff also confirmed that the MPC cavity and annulus pressures of MPC-37, MPC-44, and MPC-89 stored in the UVH will be below the design pressures specified in FSAR table 2.2.1 (for MPC cavity) and table 2.I.2.3 (for annulus) based on the bounding conditions of MPC-37, MPC-44, and MPC-89 stored in the ventilated overpack.

4.1.8 Accident Conditions of Storage

4.1.8.1 Fire Accident

In section 4.I.6.2(a) of the application, the applicant performed the thermal evaluation for a postulated fire event of the UVH using the most bounding, MPC-37 (29 kW), and presented the results in table 4.I.6.1 of the application. The applicant presented the details of the methodology and assumptions in HI-2200191 and the details of the steps to model site-specific fire events in table 4.I.6.4 of the application.

The staff reviewed the application section 4.I.6.2, tables 4.I.6.1 and 4.I.6.4 and HI-2200191. The staff finds that (1) the maximum fuel cladding and MPC’s component temperatures are below the allowable limits for the UVH; and (2) the methodology, assumptions, and modeling steps used for thermal evaluation remains unchanged as approved in the previous applications. Therefore, the staff accepts thermal evaluation of the site-specific fire accident for the UVH.

4.1.8.2 Jacket Water Loss Accident

The applicant described the jacket water loss accident in section 4.I.6.2(b) of the application and stated that the maximum allowable heat load of the MPCs qualified for use in the UVH is significantly lower than those qualified for use in the ventilated overpack, and therefore, the component temperatures and MPC cavity pressure under jacket water loss accident condition for the MPCs qualified for use in the UVH will be bounded by those for the ventilated overpack.

The staff reviewed the description of the jacket water loss accident in section 4.I.6.2(b) and accepts that the fuel and cask component temperatures and MPC cavity pressure under jacket water loss accident condition for the MPCs qualified for use in the UVH will be bounded by those for the ventilated overpack reviewed and accepted by the NRC in previous amendments. This is because the lower heat loads are allowed in MPCs stored in the UVH than in the ventilated overpack.

4.1.8.3 Extreme Environment Temperatures

The applicant stated in section 4.I.6.2(c) of the application that the calculation was performed using the same methodology used for the ventilated overpack for the most bounding scenario under the extreme environment temperatures and the maximum MPC component temperatures for the UVH is presented in table 4.I.6.2 of the application.

The staff reviewed section 4.I.6.2(c) and table 4.I.6.2 of the application and accepts the thermal evaluation for the UVH under the extreme environment temperatures because the methodology used for evaluation was accepted for the ventilated overpack and the maximum MPC component temperatures of the UVH are below the allowable limits in table 2.2.3 of the application.

4.1.8.4 Burial Under Debris

The applicant stated in section 4.I.6.2(d) of the application that the burial-under-debris duration calculation was performed following the same methodology used for the ventilated overpack and the maximum allowable time for the UVH is presented in table 4.I.6.3 of the application.

The staff reviewed section 4.I.6.2 and table 4.I.6.3 of the application and confirmed that the methodology used for calculating the burial-under-debris duration is unchanged and was accepted by the NRC in previous amendments and therefore accepts the maximum allowable time per site-specific basis for the UVH to be buried under debris.

4.2 PC #5 – Use of CFD Analysis for Site-specific Fire Accident Scenario

The applicant stated, in the application section 4.6.2.1, “Fire Accident,” that an alternate method using the FLUENT thermal model, described in the section 4.5 of the application, can be adopted to evaluate HI-TRAC site-specific fire accident event. The applicant described the approach of the alternate method, with principal modeling steps and acceptance criteria defined in table 4.6.11 of the application.

The staff acknowledged that Holtec has been using CFD for all MPC thermal analysis of normal, off-normal, and accident conditions. Since the thermal transfer and heat removal mechanism in all MPCs are identical, the CFD analysis is applicable to all MPCs.

The staff confirmed that the use of the CFD analysis using the FLUENT Code for the site-specific fire accident scenario is applicable to all NRC previously approved MPCs in the HI-STORM FW system and the proposed MPC-37P and MPC-44 for the HI-STORM FW system if the assumptions, parameters, methodology, and boundary conditions used in the CFD analysis follow the principal modeling steps and acceptance criteria in table 4.6.11 of the application and match well with the site-specific fire accident conditions.

4.3 PC #7 – Add MPC-44

The applicant stated, in the application, that MPC-44 is qualified for use in both ventilated overpack (44 kW) and UVH (28 kW). MPC-44 is also qualified for the onsite transfer in all versions of HI-TRAC VW. MPC-44 consists of the Metamic panels and CBS. MPC-44 stores a total of 44 PWR fuel assemblies and is designed for loading of only Westinghouse 14x14 fuel class. The DFCs and DFIs can be stored in the outer peripheral locations of the MPC-44, as shown in table 2.1.1d of the application.

The applicant stated, in table 2.3-13 of appendix B, that MPC-44 is permitted to use only uniform heat load pattern presented in the application table 1.2.3e which shows a 5% decay heat penalty per cell for cells permitted to contain damaged fuel or fuel debris.

The staff accepts the addition of the new MPC-44 to be used with Version E and UVH overpacks as discussed more fully below.

4.3.1 Normal Storage Conditions

As presented in HI-2094400, appendix Z, “Thermal Analysis of MPC-44 in HI-STORM- FW System,” the applicant performed steady state thermal evaluations for MPC-44 stored in the HI-STORM FW system under normal storage conditions, with the heat load pattern presented in table Z.3.1 for the design heat load of 44.0 kW, table Z.3.2 for the threshold heat load of 30.0 kW, and table Z.3.3 for the damaged fuel can with the heat load of 43.4 kW. The applicant tabulated the PCT and cask component temperatures in table 4.4.3 of the application. The results show that the PCT is below the temperature limit of 400°C (752°F) as specified in table 2.2.3 of the application and NUREG-2215 and all the MPC and overpack component temperatures are below the allowable temperature limits when MPC-44 is stored in HI-STORM FW system under normal conditions of storage.

The applicant computed the MPC-44 internal pressures following the methodology described in HI-2094400 appendix B.5.3 and reported the calculated pressure in table 4.4.5 of the application. The MPC-44 internal pressures are below the design pressures specified in FSAR table 2.2.1 of the application.

The staff reviewed the application tables 4.4.3 and 4.4.5 of the application and confirmed that the PCT is below the limit of 400°C (752°F) specified in table 2.2.3 of the application and NUREG-2215 and all the MPC and overpack component temperatures are below their respective temperature limits, specified in table 2.2.3 of the application, for MPC-44 stored in HI-STORM- FW system under normal conditions of storage. The staff also confirmed the internal pressures of MPC-44 remain below the design pressure, as specified in FSAR table 2.2.1 of the application, when MPC-44 is stored within HI-STORM FW system under normal conditions of storage.

4.3.2 Normal Storage Condition with DFCs

The applicant stated, in HI-2094400 appendix Z.5.4, that the thermal model of the MPC-44 with the DFCs in the HI-STORM FW system was built with the key attributes specified in table Z.2.2. The modeling methodology is the same as the staff previously accepted methodology explained in HI-2094400 appendix V.2.3. The applicant performed a steady state simulation for MPC-44 in HI-STORM FW system under the heat load pattern presented in table Z.3.3 and figure Z.3.1 with 12 DFCs to store damaged fuel. The calculated temperatures and MPC pressures are presented in HI-2094400 tables Z.5.4 and Z.5.5, respectively.

The staff reviewed HI-2094400 appendix Z.5.4 and tables Z.5.4 and Z.5.5, and confirmed that the PCT is below the limit of 400°C (752°F) specified in table 2.2.3 of the application and NUREG-2215 and the MPC and overpack component temperatures are below their respective temperature limits, specified in table 2.2.3 of the application, for MPC-44 with DFCs in HI-STORM- FW system under normal conditions of storage. The staff also confirmed that the MPC-44 internal pressures remain below the design pressures, as specified in table 2.2.1 of the application, when MPC-44 with DFCs is stored within HI-STORM FW system under normal conditions of storage.

4.3.3 Vacuum Drying

The applicant stated in the application section 4.5.2.3, “Vacuum Drying,” that MPC-44 containing moderate burnup fuel assemblies is vacuum dried without time limits up to design basis heat load and MPC-44 containing high burnup fuel assemblies is vacuum dried without time limits up to threshold heat loads defined in table 4.5.19 of the application. The applicant performed steady state thermal evaluations of two vacuum drying scenarios, design basis heat load of 44.0 kW for moderate burnup fuel (MBF) and threshold heat load of 30.0 kW for HBF, as described in HI-2094400 appendix Z. The applicant presented the heat load patterns and method of drying in table 4.5.19 of the application and the fuel and cask component temperatures for evaluations of MBF and HBF in table 4.5.32 of the application.

The staff reviewed section 4.5.2.3 of the application and HI-2094400 appendix Z and finds that the PCT and temperatures of other MPC internals, provided in table 4.5.32 of the application, are well below their respective limits as specified in FSAR table 2.2.3, demonstrating that the thermal safety is maintained with the MPC-44 basket design under above-mentioned design basis and threshold heat load scenarios.

4.3.4 Thermal Expansion

In HI-2094400 appendix Z.5.3, the applicant stated that following the same method in HI-2094400 section B.5.2, thermal expansion of fuel basket to MPC shell and MPC shell to overpack inner shell are computed both in radial and axial directions in the EXCEL spreadsheets listed in appendix Z.4. The applicant reported the differential thermal expansion calculation results for MPC-44 and MPC-44 with DFCs in tables Z.5.3 and Z.5.6, respectively. The result shows that the differential thermal expansions of the fuel basket to MPC shell and the MPC shell to overpack inner shell are less than the corresponding cold gaps (the design gaps) and therefore the free expansion criteria are satisfied.

The staff reviewed HI-2094400 appendix Z.5.3, tables Z.5.3 and Z.5.6, and confirmed that the differential thermal expansions of gaps, calculated using the NRC accepted methodology, are less than the corresponding cold gaps. Therefore, the free expansion criteria are satisfied for both MPC-44 and MPC-44 with DFCs.

4.3.5 Off-Normal and Accident Conditions

The applicant stated that, in HI-2094400 appendix Z.5.5, since the MPC-44 component temperatures and cavity pressure are bounded by MPC-37 component temperatures and cavity pressure, the conclusions for all the off-normal and accident conditions applicable for MPC-37 can be extended to MPC-44 without carrying out explicit analyses. The applicant presented the canister pressure in table Z.5.2 for MPC-44 stored in the ventilated system under the accident conditions with 100% of the rod ruptured.

The staff reviewed HI-2094400 appendix Z.5.5 and accepts that thermal evaluations of MPC44 are bounded by thermal evaluation of MPC-37 for off-normal and accident conditions of storage because of the existed bounding correlation for normal conditions of storage. Therefore, the PCT and all the MPC and overpack component temperatures and MPC internal pressures are below the respective temperature limits and pressure limits, as specified in tables 2.2.3 and 2.2.1 of the application, for MPC-44 stored within HI-STORM- FW system under off-normal and accident conditions of storage.

4.4 PC #8 – Add MPC-37P

The applicant stated, in the application section 4.4.1, “Overview of the Thermal Model,” that MPC-37P is qualified for use only in ventilated system with the maximum design basis heat load of 45.0 kW and is also qualified for onsite transfer in all versions of HI-TRAC VW. MPC-37P consists of CBS and is designed to store a total of 37 PWR fuel assemblies of only fuel class type 15x15I. The DFCs and DFIs can be stored in the cells denoted by “D/F” in figures 1.2.9a and 1.2.9b of the application.

The applicant presented MPC-37P heat load data in table 1.2.3a of the application for regionalized heat load pattern and table 1.2.3c for uniform heat load pattern. Figures 1.2.9a and 1.2.9b of the application explicitly provide the allowable decay heat and locations for damaged fuel or fuel debris. The applicant provided in Holtec proprietary report HI-2210379, revision 0, “Thermal Evaluation of MPC-37P in HI-STORM FW and HI-TRAC VW,” section 8.4, the technical basis for these zero penalties. It is noted in table 1.2.8b of the application that MPC-37P basket and basket shims shall be treated to achieve the specified emissivity value.

The staff reviewed HI-2210379 section 8.4 and referred to HI-2094400 for evaluation of the MPC-37 containing damaged fuel and fuel debris. The staff determined that due to similarity between MPC-37 and MPC-37P, the results of thermal evaluation for MPC-37 containing damaged fuel and fuel debris can be applied to MPC-37P containing damaged fuel and fuel debris, and therefore accepts that there is no decay heat penalty per cell for cells permitted to contain damaged fuel or fuel debris in MPC-37P, as shown in figures 1.2.9a and 1.2.9b of the application.

The applicant stated, in the application section 4.4.1, “Overview of the Thermal Model,” that MPC-37P basket has higher heat rejection capabilities than MPC-37 (standard) basket due to increased basket thickness; therefore, temperatures and pressures in MPC-37P for heat load patterns A and B (application table 1.2.3a) will be bounded by those for MPC-37. The applicant summarized the maximum component temperatures and MPC internal pressures of MPC-37P in tables 4.4.3 and 4.4.5 of the application, respectively, under normal conditions of storage. The staff finds that the results show: (1) the PCT is below the temperature limit of 400°C (752°F) specified in table 2.2.3 of the application and NUREG-2215 and all the MPC and overpack component temperatures and are below their allowable temperature limits specified in table 2.2.3 of the application; and (2) the MPC internal pressures are below the design limits under normal, off-normal, and accident conditions of storage. Therefore, the staff finds that the NRC accepted methodology is applicable to the evaluation of the MPC-37P and the resulting maximum fuel and cask component temperatures (application table 4.4.3) and MPC internal pressures (application table 4.4.5) are below the allowable limits under normal, off-normal, and accident conditions of storage.

In the September 15, 2022 letter (Holtec, 2022d), the applicant proposed “MPC-37P Burnup, Enrichment, and Cooling Time (BECT) Alternatives” for the specific characteristics of Palisades fuel assemblies. The applicant noted that some of the assemblies from the last core from Palisades exhibit higher assembly heat loads than initially expected; however, the margin in the thermal pattern, together with some reshuffling of fuel between casks, will allow a timely unloading of the pool into dry storage, and hence no modifications to the thermal loading patterns are necessary.

The staff reviewed the information and table 4.4.3 of the application (MPC-37P stored in the ventilated overpack) and accepts addition of the BECT alternatives to MPC-37P from thermal perspective. This is because the maximum heat loads for MPC-37P remain unchanged and therefore, the maximum fuel and cask component temperatures for MPC-37P loaded with BECT alternatives and stored within the ventilated overpack will be below the allowable limits as specified in table 2.2.3 of the application, under all conditions of storage.

The applicant stated, in the application section 4.5.2.3, “Vacuum Drying,” that MPC-37P containing MBF assemblies is vacuum dried without time limits up to design basis heat load defined in the application table 4.5.19 and MPC-37P containing HBF assemblies is vacuum dried with time limits up to design basis heat load and without time limits up to threshold heat load limits defined in tables 4.5.19. The applicant provided the maximum component temperatures in table 4.5.31 of the application for vacuum drying operations of MPC-37P.

The staff reviewed the methodology used for thermal model of MPC-37, which was reviewed and accepted by NRC, and found it is applicable to MPC-37P and confirmed that the PCT and the MPC and overpack component temperatures (application table 4.5.31) are below the allowable limits presented in table 2.2.3 of the application. The staff accepts the addition of the new MPC-37P to be used with version E overpack as discussed above.

4.5 PC #9 – Add HI-DRIP as an Ancillary System

The applicant stated, in the application section 4.5.7, “HI-DRIP,” that the HI-DRIP is designed to prevent the water in the loaded MPC from boiling during the interval after it has been lifted out of the fuel pool and is subject to surface decontamination, lid welding, and related operations which precede the evacuation of water from the MPC. The applicant described the HI-DRIP in HI-2094400 appendix N section N.5.21 and noted that the operation of HIDRIP does not require any pump or electric power; the motive pressure is provided entirely by the plant’s water supply system.

The applicant stated, in section 4.5.7 of the application, that in an event the plant’s water supply system were to fail, the time-to-boil limits shall be calculated by measuring the water temperature inside the MPC as described in the application section 4.5.3.1, “Bounding Time Limit.” The measured MPC water temperature shall be used as the initial temperature to re-evaluate the maximum allowable time duration for fuel to be submerged in water. Alternately, a forced water circulation can be initiated and maintained to remove the decay heat from the MPC cavity as described in the application section 4.5.3.3, “Forced Water Circulation.” At the end of forced water circulation, the measured temperature of water in the MPC shall be used to re-calculate the maximum allowable time duration for fuel to be submerged in water and update the time-to-boil clock.

The applicant described the sizing methodology and demonstration of the HI-DRIP auxiliary cooling system in HI-2094400 appendix N section N.5.22 as: (1) estimating the required thermal capacity of HI-DRIP cooling system, and (2) establishing the design and operating parameters for HI-DRIP to meet the required thermal capacity. The applicant performed a calculation for HI-TRAC VW transfer cask and recommended to initiate the HI-DRIP cooling system no later than 50% of the computed TTB duration to allow sufficient time for enhanced and effective cooling of the cask.

In section 9.2.4 of the application, the applicant outlined the necessary steps for operations, identified the steps for initiation and termination HI-DRIP, added the cautionary note to ensure initiation of the system within 50% of the time-to-boil duration, and noted the system flow rate established on a site-specific basis.

The staff reviewed section 4.5.7 of the application for description on HI-DRIP operations, bounding time-limit determination, and sizing methodology of water supply, and reviewed HI-2094400 appendix N for detailed methodology for sizing and allowable TTB clock. The staff confirmed that (1) the use of the HI-DRIP as an ancillary system is applicable to the proposed

MPCs in HI-TRAC VW (e.g., MPC-37P and MPC-44) and (2) HI-DRIP will be initiated no later than 50% of the allowable TTB duration for enhanced and effective cooling of the cask. -

The applicant stated, in section 10.1.7.1 of the application, that the HI-DRIP supplemental cooling system thermal acceptance testing shall be performed following fabrication and prior to the first implementation of the HI-DRIP supplemental cooling system, and described the testing procedure, in section 10.1.7.1.1 of the application. The applicant noted that the results of the acceptance test will be compared to a thermal analysis using the methodology in the application chapter 4 along with the temperatures measured from the test.

The staff confirmed that the HI-DRIP supplemental cooling system thermal acceptance testing will be considered acceptable if the measured heat rejection capability during the test is greater than the heat capacity estimated using the methodology in chapter 4 of the application that NRC reviewed and accepted. The staff also confirmed that if the acceptance criteria are not met, then the HI-DRIP shall not be accepted until the root cause is found and the corrective actions are completed and re-tested with acceptable results. With these conditions, the staff accepts the use of HI-DRIP as an ancillary cooling system.

4.6 PC #10 – Use of CFD Analysis for Site-specific Debris Burial Accident Scenario

The applicant stated, in the application section 4.6.2.5, “Burial under Debris,” that an explicit site-specific evaluation may also be performed using CFD to compute the allowable burial time utilizing the site-specific conditions and the following methodology in the CFD analysis.

The applicant used to perform hand calculations to evaluate debris burial accidents. The staff confirmed that use of the CFD analysis for site-specific debris burial accident scenario will be more capable of accounting for complicated scenarios involved in the debris burial accident and therefore is applicable to all MPCs previously approved by the NRC and the proposed MPC-37P and MPC-44 for HI-STORM FW CoC.

4.7 PC #11 – Use of Water Without Glycol in Water Jacket

As described in the application section 4.5.4.3, “Normal On-Site Transfer,” the applicant performed a thermal evaluation to include the ability to use water without ethylene glycol in the HI-TRAC water jacket during transfer operations below 32°F and presented the maximum component temperatures under low ambient temperature conditions in table 4.5.30 of the application. The applicant stated, in section 4.5.4.3 of the application, that the users may perform a site-specific evaluation using the site-specific heat loads, equipment models, and the ambient temperature to establish the minimum allowable heat load for any site. The applicant further stated that users may use the methodology, described in section 4.5.4.3 of the application to establish the minimum allowable heat load for any site. The applicant stated, in notes to the application section 9.2.4, “MPC Closure,” and section 9.4.2, “HI-STORM FW Recovery from Storage,” that if the HI-TRAC VW is expected to be operated in an environment below 32°F, and a minimum heat load requirement was not applied to load the MPC, the water jacket shall be filled with an ethylene glycol solution.

The staff reviewed sections 4.5.4.3 and table 4.5.30 of the application and accepts that (1) addition of ethylene glycol in the water jacket is dependent on MPC heat load and is not required if the MPC heat load is high enough to preclude freezing of the water and the threshold MPC heat load should be determined on a site-specific basis using the methodology, described in section 4.5.4.3 of the application, to establish the minimum allowable heat load for any site;

and (2) the use of water without ethylene glycol in water jacket is applicable to all MPCs previously reviewed and approved by the NRC and the proposed MPC-37P and MPC-44 for HI-STORM FW system.

4.8 Evaluation Findings

The findings listed below are based on a review that considered the regulation itself, appropriate standard review plan, applicable codes and standards, and accepted engineering practices.

F4.1 The staff has reasonable assurance that the thermal design, normal storage, and loading operations of the proposed UVH that are ITS are described in sufficient detail in the application to enable an evaluation of the heat removal effectiveness. The SSCs remain within their operating temperature ranges.

F4.2 The staff has reasonable assurance that the MPC-37, MPC-44, and MPC-89 stored within the UVH, continue to be designed with a heat removal capability having verifiability and reliability consistent with its importance to safety.

F4.3 The staff has reasonable assurance that the fuel cladding in the MPC-37, MPC-44, and MPC-89 stored in the UVH, continues to be protected against degradation leading to gross ruptures by maintaining the cladding temperatures below 400°C (752°F) for short-term operations and normal conditions of storage and 570°C (1,058°F) for off-normal and accident conditions of storage, and other cask component temperatures continue to be maintained below the allowable limits for the accidents evaluated.

F4.4 The staff has reasonable assurance that the MPC-37, MPC-44, and MPC-89, stored in the UVH, will sustain the pressures predicted under normal, off-normal, and accident-level conditions. The maximum canister pressures are below the design pressures of 100, 110, and 200 psig for normal, off-normal, and accident conditions of storage, respectively.

F4.5 The staff has reasonable assurance that the use of CFD analysis for site-specific fire accident is acceptable to all MPCs previously approved by the NRC and the proposed MPC-37P and MPC-44 for HI-STORM FW CoC.

F4.6 The staff has reasonable assurance that the use of CFD analysis for site-specific debris burial accident is acceptable to all MPCs previously approved by the NRC and the proposed MPC-37P and MPC-44 for HI-STORM FW CoC.

F4.7 The staff has reasonable assurance that the use of the HI-DRIP as an ancillary system is acceptable to all MPCs previously approved by the NRC and the proposed MPC-37P and MPC-44 for HI-STORM FW CoC.

F4.8 The staff has reasonable assurance that the use of water without ethylene glycol in water jacket is acceptable to all MPCs previously approved by the NRC and the proposed MPC-37P and MPC-44 for HI-STORM FW CoC. However, users need to perform thermal evaluation using the models and methods consistent with those described in FSAR and using the site-specific heat loads and ambient conditions to determine the minimum heat load limit for using water without ethylene glycol in the HI-TRAC water jacket during transfer operations below 32°F.

F4.9 The staff concludes that adding MPC-37P and MPC-44 to HI-STORM FW ventilated system under the corresponding design heat load limits, is compliant with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The MPC-37P and MPC-44 basket and basket shims shall be coated to achieve the required emissivity.

5.0 SHIELDING EVALUATION

The objective of the shielding review includes review of the shielding design description, radiation source definition, shielding model specification and shielding analyses for the proposed HI-STORM FW system addition. This SER documents the basis for the staff's approval for the proposed changes as it relates to the system's ability to maintain the system within applicable dose limits in 10 CFR Part 72. In reviewing these changes to the HI-STORM FW's shielding design, the staff followed the guidance in chapter 6 of NUREG-2215. The staff's evaluation follows the same order as the requested changes in the application.

The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106(b), 10 CFR 72.212, and 10 CFR 72.236(d). Because 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits is provided in SER section 11.

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

- PC #1 – Add a new unventilated overpack, HI-STORM FW UVH.
- PC #7 – Add the new MPC-44 to be used with Version E and UVH overpacks.
- PC #8 – Add the new MPC-37P to be used with version E overpack.
- PC #12 – Add new 10x10J fuel type to approved content.

5.1 PC #1 – Addition of a New UVH Overpack

The applicant requested to add a new variant of the HI-STORM FW Overpack, designated HI-STORM FW UVH for shielding. The applicant stated that the overpack is a simplified version of the HI-STORM FW system wherein the overpack's inlet and outlet air passages have been removed, resulting in a complete cessation of ventilation in the space between the cask cavity and the stored MPC during the system's operation. To compensate for the removal of the inlet and outlet vents, the total allowable heat load of an MPC is reduced, and higher thermal conductivity concrete is employed to increase system heat transfer capabilities.

The applicant stated that due to the UVH overpack configuration, the dose rates are lower at the site boundary than the HI-STORM FW ventilated cask system overpack designs. The UVH overpack is to be used with MPC-37, MPC-89, and the proposed MPC-44. The proposed contents have been previously approved for use in the HI-STORM FW cask system of casks except for the addition of the MPC-44 and 10x10J BWR fuel as approved content in this amendment. Therefore, this review focuses on the UVH overpack, MPC-44, 10x10J BWR fuel in MPC-89, and any modifications to operations arising from the addition of the UVH overpack to the HI-STORM FW cask system.

5.1.1 Shielding Design Description

The shielding design of the UVH overpack utilizes high density concrete and steel. It is a matrix shield made of high-density components, capable of attenuating both the neutron and gamma radiation. Due to this design feature, the dose rates at the site boundary using a UVH overpack are less than, and bounded by, the dose rates from the HI-STORM FW ventilated overpack previously approved by NRC. The applicant performed the shielding analysis for the UVH overpack containing the MPC-37, MPC-89, and the proposed MPC-44 loaded with regionalized loading patterns of the spent PWR and BWR fuel and determined dose rates for the positions shown in figure 5.I.1.1 of the application. The staff found acceptable the inclusion of CBS versions from shielding perspective based on the facts that the shielding analysis do not include MPC basket shim in the shielding model. This is conservative because the modeling removes the shim that would provide a small increase in shielding. The spent fuel basket cells are divided into three regions: inner region, middle region, and peripheral region, as specified in CoC appendix B figures 2.1-1, 2.1-2, and 2.1-4. The applicant's proposed changes in CoC appendix B tables 2.3-9A and 2.3-9B show MPC-37 heat load data for the UVH overpack and the MPC-37 requirements on developing regionalized heat load patterns for the UVH overpack, respectively. The applicant's proposed changes in CoC appendix B tables 2.3-10A and 2.3-10B show MPC-89 heat load data for the UHV overpack and the MPC-89 requirements on developing regionalized heat load patterns for the UVH overpack, respectively. The applicant's proposed changes in CoC appendix B table 2.3-13 shows the MPC-44 heat load data for UHV overpack for only one region.

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 damaged fuel cans (DFCs) or using a damaged fuel isolator (DFI) that can be stored in the outer peripheral locations of the MPC-37 as shown in CoC appendix B figures 2.3-1 through 2.3-9, for MPC-89 as shown in CoC appendix B figure 2.3-10 through 2.3-13, and for MPC-44 as shown in CoC appendix B figure 2.1-5. The DFCs with damaged fuel or fuel debris must also be loaded in specific fuel cell locations of the MPC as specified in CoC appendix B table 2.1-1, section I.B for the MPC-37 and section II.B for the MPC-89.

Based on information provided by the applicant, the description of the geometric arrangement of shielding and illustrations that identify the spatial relationships among sources, shielding, and design dose rate locations, and the conditions provided in the CoC, appendices A and B, the staff finds that the shielding design features of the HI-STORM FW UVH cask system meets the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72.

5.1.1.1 Shielding Design Criteria

The overall radiological protection design requirements are provided in 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). The applicant analyzed the UVH overpack loaded with spent fuel and hardware having the characteristics described in FSAR section 2.1.9. Although there are no numerical limits in the regulations for surface dose rates, the dose rates on the surface of the cask system serve as design criteria (based on source terms of the contents) to ensure there is sufficient shielding to meet radiological limits in accordance with 10 CFR 72.236(d). Section 5.1.1 of the application describes the maximum surface dose rate criteria for the HI-STORM FW cask system overpacks. Due to the high density of the concrete and the total allowable heat load, the maximum source term of an MPC when used with an UVH overpack is constrained by thermal requirements and reduced. Therefore, the applicant calculated dose rates at the exterior of the UVH overpack with less than the proposed design parameters. The most limiting dose rates occur during transfer operations when loaded into the HI-TRAC transfer cask. Staff noted the dose rates at 1 meter from the transfer cask do not exceed 2 rem per hour. As a result, staff finds that the shielding and source term design criteria defined in the FSAR provide reasonable assurance that the UVH overpack can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72. General licensees will be required to protect personnel and minimize dose in accordance with as low as reasonably achievable (ALARA) principles and the regulations of 10 CFR Part 20. A radiation protection program is defined in CoC appendix A TS 5.3 to ensure compliance with these requirements for the UVH overpack. A dose rate limit based on the bounding shielding analysis is in TS 5.3.4 for the top and side of overpack, and side of the transfer cask of the HI-STORM FW system. Limits for the system contents are incorporated into CoC appendix A.

5.1.2 Radiation Source Definition

The applicant used a bounding radiological source term in performing shielding evaluations and simplifying the fuel qualification procedure. A fuel qualification table (FQT) approach was used to specify the content's BECT. The method used to calculate the source terms is the same method that was used in HI-STORM FW Amendments No. 4 and 5 (NRC, 2020d and NRC, 2022c), previously reviewed and approved by the NRC.

The applicant specified the required minimum cooling time as a function of burnup using a polynomial correlation in CoC appendix B section 2.5 (application section 2.1.6.1) to calculate source terms. The correlations are defined by sets of correlation coefficients in appendix B table 2.5-2 (application table 2.1.10) for multiple reference decay heats for MPC-37 and MPC-89 and in table 2.5-3 for MPC-44. The reference decay heats encompass the allowable decay heats for each loading zone as defined by the loading patterns in appendix B figures 2.3-1 through 2.3-9 for the MPC-37, figure 2.1-5 for the MPC-44 (only one region), and figures 2.3-10 through 2.3-13 for the MPC-89.

Burnup and cooling times fuel qualification requirements for fuel assemblies authorized for loading into the proposed MPC-37P and MPC-44 are provided in TS appendix B table 2.5-3. Since the cooling times from TS appendix B table 2.1-1 are the minimum allowable cooling times, it is specified in appendix B table 2.1-1 and in section 9.2.3 of the application that the cooling time must meet both this minimum value and those from TS appendix B section 2.5.

The staff's review of the source term analyses, and the burnup equation method is documented in the SERs for HI-STORM FW Amendments No. 4 and 5 (NRC, 2020d and NRC, 2022c). Limits on the allowable content's specifications, are incorporated into appendix B of the proposed CoC.

5.1.2.1 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. The staff considered the use of ORIGEN acceptable per the guidance in section 3 of NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages" (NRC, 2003b). ORIGAMI (ORIGEN Assembly Isotopic) is a newer code within SCALE 6.2 developed after the publication of NUREG/CR-6802. ORIGAMI computes detailed isotopic compositions for light water reactor (LWR) assemblies containing UO_2 fuel by using the ORIGEN code with pre-generated 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-dimensional reactor physics models as compared to the methods recommended in NUREG/CR-6802. Both codes have been validated against radiochemical assay samples of spent fuel and are widely used for spent fuel isotopic depletion analyses. Therefore, the staff found the use of these codes acceptable for the HI-STORM FW UVH system.

5.1.2.2 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 that would otherwise influence the accident condition source term.

5.1.2.3 Non-Fuel Hardware Source Term

The applicant has not proposed any changes that would otherwise influence the non-fuel hardware source term.

5.1.3 Shielding Model Specification

5.1.3.1 Configuration of the Shielding and Source

The applicant provided upper bound source terms and corresponding upper bound dose rates for the various relevant locations around the cask. The applicant analyzed two loading cases. Case 1 is a representative configuration for CoC dose rate limits. Representative burnup and cooling time combinations are shown in Holtec proprietary report HI-2094431, revision 30, “HI-STORM FW and HI-TRAC VW Shielding Analysis,” table S.4-1 for MPC-37, MPC-44, and MPC-89. Case 2 is a bounding condition where all cells are loaded with assemblies corresponding to the highest source terms. Bounding burnup and cooling time combinations are shown in report HI-2094431 table S.4-10 for MPC-37, MPC-44, and MPC-89. The staff verified that the cooling times used by the applicant within the shielding evaluation are conservative (shorter) with respect to the cooling times that are calculated from the correlation that determines minimum cooling times for allowable fuel loadings in TS appendix B section 2.5. The considered BECT combinations, derived loading curves and established polynomial coefficients are shown in HI-2094431 figures S4-1, S4-2, and S4-3 for MPC-37, MPC-44, and MPC-89, respectively. The staff found the applicant’s results acceptable because they include a significant conservatism based on the shortened cooling time, which decrease to the nearest available cooling time in the source terms library, while, practically, only the fuel assemblies with the cooling time above the loading curve are qualified for loading.

The applicant analyzed several specific BECT points along the burnup/cooling time curve for each correlation corresponding to a decay heat. 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) and this approach assures the worst-case is bounding. The applicant determined the bounding source term among the chosen points on the burnup/cooling time correlation by determining for each region what BECT combination 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 appropriate for determining the bounding source term because it encompasses the entire loading curve which includes a large and varied population of possible fuel loadings. The staff found that defining bounding source terms in this manner to be acceptable for this application.

5.1.3.2 Material Properties

Tables 5.I.3.1 and 5.3.2 in the application provide the composition and densities of the various materials used in the HI-STORM FW UVH system and HI-STORM FW system, respectively. A minimum 3.2 g/cm^3 (200 pcf) concrete density is required for HI-STORM FW UVH for enhanced thermal conductivity as specified in HI-2094431. As shown in table 5.I.3.1 of the application, the applicant used a reduced concrete density of 2.72 g/cm^3 (170 pcf) in its HI-STORM FW UVH dose rates analysis. Since reducing the density of shielding components in the model will conservatively increase calculated external dose, staff finds the applicant's choice of concrete density appropriate. The rest of the material properties are unchanged from those used by the applicant in previously approved HI-STORM FW system amendments.

5.1.4 Shielding Analyses

5.1.4.1 Computer Codes

The applicant used the MCNP5 computer code to calculate doses at the various locations in the HI-STORM FW UVH system. MCNP5 calculates neutron or photon flux, and these values can be converted into doses using dose response functions. MCNP and the calculational approach used for this amendment is presented in section 5.I.1 of the application. An example of an MCNP5 input file, for the HI-STORM FW UVH overpack, is listed in appendix 5.I.A of the application. Section 5.I.1 of the application shows the results for the HI-STORM FW UVH overpack and HI-TRAC VW with representative bounding loading configurations.

The staff found these loading configurations acceptable because these bounding configurations present a highly conservative bounding case where all cells are loaded with assemblies corresponding to the highest allowable source terms. Bounding burnup and cooling time combinations are shown in table 5.I.4.1 of the application.

5.1.4.2 Flux-to-dose-rate Conversion

The applicant has not proposed any changes that would otherwise influence the shielding model specification. The flux-to-dose conversion factors used for this amendment are consistent with those used in previously approved amendments for the HI-STORM FW.

5.1.4.3 Dose Rates

The staff reviewed the new and updated maximum dose rate for the HI-STORM FW UVH overpack. Maximum surface dose rates for MPC-37, MPC-44, and MPC-89 are shown in supplement I section 5.I.1.1 of the application. The applicant has previously established dose rate limits in TS 5.3.4 in appendix A to CoC. For the top of the overpack at the center of the lid, the applicant calculated dose rate of 2.3 mrem/hr for MPC-37, 2.2 mrem/hr for MPC-44, and 1.5 mrem/hr for MPC-89 as shown in the application tables 5.I.1.2, 5.I.1.3, and 5.I.1.4, respectively. The TS dose rate limit specified in TS 5.3.4a in appendix A for the top of the overpack is 15 mrem/hr; therefore, the staff found that calculated dose rates for the MPC 37, MPC-44, and MPC-89 to be well within the TS limit and acceptable. For the side of the overpack, the applicant calculated maximum dose rate of 61.2 mrem/hr for MPC-37, 73.6 mrem/hr for MPC-44, and 42.8 mrem/hr for MPC-89 as shown in application tables 5.I.1.2, 5.I.1.3, and 5.I.1.4, respectively. The TS dose rate limit specified in TS 5.3.4b in appendix A for the side of the overpack is 300 mrem/hr; therefore, the staff found the calculated dose rates of MPC-37, MPC-

44, and MPC-89 to be well within this limit and acceptable. The TS 5.4.3a and 5.4.3b in appendix A continue to be applicable when using UVH overpack.

Based on the off-normal conditions in FSAR chapter 12, the staff found the above dose rates to be representative of both normal and off-normal conditions. The applicant included loading curves in the application tables 2.1.9, 2.1.10, and 2.1.12 for the MPC-37, MPC-44, and MPC-89, respectively. The staff reviewed the loading curves tables 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.

For the side of the transfer cask, the applicant has previously established a dose limit of 3,500 mrem/hr at the specified measurement locations stated in TS 5.3.8c in appendix A to the CoC. As shown in the application table 5.I.4.4, the applicant calculated maximum total dose rates are 2,101.0 mrem/hr for MPC-37, 2,256.4 mrem/hr for MPC-44, and 2,988.5 mrem/hr for MPC-89. The staff found these calculated dose rates to be well within the TS 5.3.4c limit of 3,500 mrem/hr and is acceptable. The TS 5.4.3c in appendix A continues to be applicable when using UVH overpack. The regulations in 10 CFR 72.212(b)(5) requires a user to perform a written evaluation that ensures that the cask, when loaded, will conform to the terms in the CoC (including technical specifications). This ensures that the cask will not be loaded with 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 the applicant determined to be the limiting consequence with respect to dose for the events discussed in FSAR chapter 12. The limiting configuration is the MPC-89B2 with the loading pattern in the application figure 1.2.7b (corresponding to CoC appendix B figure 2.3-13). The applicant calculated that it would take 62 days to reach the regulatory dose limit in 10 CFR 72.106. The staff found that this is sufficient time to recover from the accidents in FSAR chapter 12 involving the loss of the neutron shield and found that this demonstrates that the applicant will meet the regulatory dose limit in 10 CFR 72.106 for accident conditions.

The applicant has established the radiation protection program in appendix A section 5.3 and stated that the licensees shall establish dose rate limits based on the site-specific evaluations required by 10 CFR 72.212(b)(2)(i)(C).

5.1.4.4 Confirmatory Calculations

5.1.4.4.1 Source term Confirmatory Calculations

The staff used the ORIGEN-ARP code from the SCALE 6.1 code package to verify the spent fuel source term for the loading patterns and fuel qualification strategy. The staff verified the spent fuel gamma and neutron source terms using the data provided for burnup and cooling times from table S.4-10 of HI-2094431, and minimum enrichments from table 5.0.3 for PWR and table 5.0.4a for BWR fuel assemblies since MPC-37 and MPC-89 are the bounding cases.

As described in SER section 5.1.4.4.2, the staff independently confirmed the gamma and neutron source terms in the tables above are appropriate or conservative. The staff found that although the reactor operating parameters used by the applicant in the depletion calculations are unknown, there is reasonable assurance that the reactor operating parameters were reasonably bounding based on staff's independent calculations.

As discussed in SER section 5.1.1, when using the UVH overpack, there are high density concrete shielding, lower heat loading limits, and reduced source terms; therefore, the staff found that the dose rates for the HI-STORM FW UVH overpack with the MPC-37, MPC-44, and PMC-89 are bounded by previous amendments of HI-STORM FW (NRC, 2020d and NRC, 2020c). Thus, the new UVH overpack will meet the dose requirements of 10 CFR 72.104 and 72.106.

5.1.4.4.2 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 at Oak Ridge National Laboratory (ORNL). The UNF-ST&ARDS code is a comprehensive integrated data and analysis tool developed for the U.S. Department of Energy Office of Nuclear Energy Spent Fuel and Waste Disposition program with support from the NRC.

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 for the MPC-44 and the UVH overpack from the application. Some 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 also 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 results from the staff's model agree with those of the applicant's providing additional assurance that the HI-STORM FW UVH system including the changes requested with this amendment can meet the regulatory dose requirements in 10 CFR 72.104 and 72.106.

For all dose and dose rate calculations, the applicant used a BECT combination that maximized the dose rate for each location. 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, for the BECT combinations the staff evaluated, the staff's model calculated the dose rates for all the reported locations. Even with different BECT combinations, when compared to the applicant's calculations, the staff's results reasonable approximate those reported by the applicant which provides the staff confidence that the applicant's model is adequately for generating realistic dose rates for the HI-STORM FW overpacks and the HI-TRAC VW.

5.2 PCs #7 and #8 – Addition of new MPCs, MPC-44 and MPC-37P

The applicant is proposing to add two MPCs, MPC-37P and MPC-44, that would affect the dose rates. The staff's review considered the criteria specified in section 6 of NUREG-2215 and the information provided in the HI-STORM FW system FSAR.

5.2.1 Shielding Evaluation for MPC-44

MPC-44 is authorized only for uniform heat load pattern as defined in table 4.I.1.1 of the application and can store up to 44 undamaged ZR clad PWR fuel assemblies of classes specified in table 2.1.1d of the application. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in figure 2.1.1d of the application with the remaining basket cells containing undamaged fuel assemblies, up to a total of 44. The basket cell wall minimum thickness is 0.49 inch. Table 1.2.3.e of the application shows the MPC-44 heat load data. Specific requirements, such as approved locations for DFCs, DFIs, and non-fuel hardware are provided in section 2.1 of the application. Section 2.1.6.2 describes the radiological parameters for spent fuel and non-fuel hardware in MPC-44. MPC-44 is authorized to store 14x14 spent fuel with burnup-cooling time combinations as provided in table 2.1.12 of the application. Table 2.1.12 shows the burnup and cooling time fuel qualification requirements for MPC-44.

The staff's shielding evaluation for the MPC-44 is in section 5.1 of this SER in conjunction with the HI-STORM FW UVH overpack evaluation.

5.2.2 Shielding Evaluation for MPC-37P

The applicant used MCNP5 models of HI-STORM FW version E using the same FQT methodology used for the UVH overpack for shielding evaluation for MPC-37P. The MPC-37P heat load pattern allows various basket cells to contain fuel assemblies with low and high heat loads. The heat loads for the MPC-37P basket cells are listed in table 1.2.3c of the application. In comparing to the previously approved MPC-37, the box inner diameter (ID) of MPC-37P is slightly decreased, and the box wall thickness is increased, so that the basket cell pitch is maintained. Table 2.1.2 of the application provides the acceptable fuel characteristics for the assembly class 15x15I authorized for storage in the MPC-37P. The MPC-37P can store up to 37 undamaged ZR clad PWR fuel assemblies of classes specified in table 2.1.1c of the application. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in figure 2.1.1c of the application with the remaining basket cells containing undamaged fuel assemblies, up to a total of 37.

The dose rates assessment for HI-STORM FW version E and HI-TRAC VW with 5-region loading pattern in MPC-37P are conservative based on the source terms with BECT combinations which result in a higher total heat load for MPC-37P, and the staff found it acceptable.

5.2.2.1 MPC-37P and BECT Alternatives

The applicant has introduced the MPC-37P as a slightly modified version of the MPC-37. The purpose of the added MPC is to offload all assemblies in the Palisades spent fuel pool into dry storage, after shutdown of the Palisades plant, in the most efficient way. The MPC-37P is specifically optimized from a thermal perspective for the Palisades fuel assemblies, which have a slightly smaller cross-section than standard PWR assemblies.

In section 2.1.6.3 of the application, the applicant provides an alternative to the radiological parameters for the MPC-37 and MPC-37P defined in the application sections 2.1.6.1 and 2.1.6.2 and states it is permissible to use the parameters in tables 2.1.13 and 2.1.14 of the application. These tables also define minimum cooling time with corresponding maximum burnup. All assemblies in a single MPC must either meet the limits from FSAR sections 2.1.6.1 and 2.1.6.2, or the limits in tables 2.1.13 and 2.1.14 of the application; a combination is not permitted.

The dose rate calculations consider the fuel locations of regionalized storage patterns which are guided by the considerations of minimizing occupational and site boundary dose to comply with ALARA principles. The application tables 5.1.17, 5.1.18, and 5.1.19 show the highest dose rates from the content specified in tables 5.0.7 and 2.1.13 for MPC-37P. These values bound the dose rates from using the alternative BECT in the application table 2.1.14 for MPC-37.

5.3 PC #12 – Addition of a new 10x10J to Approved Contents

In section 2.1.6.4 of the application, the applicant states that for array/class fuel 10x10J, modified or additional requirements must be met, one must account for the higher uranium mass of this assembly type compared to the design basis assembly. This is to ensure that when any cask is loaded with one or more 10x10J assemblies, dose rates around the cask are still below a cask loaded with design basis fuel. When loading assemblies corresponding to array/class 10x10J according to FSAR table 1.2.4b, the applicant assumed that the burnup of the 10x10J assembly used in the equation in section 2.1.6.1 needs to be increased by 5,000 MWD/MTU. This will result in a longer required cooling time, hence reduced source terms, which offsets the effect of the higher uranium mass. This adjustment is applied to the burnup to determine the cooling time. For example, for an assembly with a burnup of 45,000 MWd/mtU, the cooling time is calculated using a burnup of 50,000 MWd/mtU instead of the 45,000 MWd/mtU, and that calculated higher cooling time limit is then used for fuel with a burnup of 45,000 MWd/mtU. The increase of 5,000 MWD/MTU is specified in a note to table 2.1.10 of the application and is only required for the array/class fuel 10x10J, not for any other assemblies, even if they are located in the same basket and pattern.

The staff finds reasonable assurance that the shielding design has been adequately described and evaluated and will meet the dose requirements of 10 CFR Part 72 because the 10x10J will either have (1) an adjustment of the burnup and cooling time requirement for this fuel or (2) specified discrete cooling time limits as a function of burnup, which ensures that the dose rates from a cask loaded with 10x10J fuel type will produce dose rates consistent with a cask loaded with design basis fuel.

5.4 Evaluation Findings

Based on its review, the staff has reasonable assurance that the design of the shielding system of the HI-STORM FW system, Amendment No. 7 complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. This finding is reached based on 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.

F5.1 The evaluation of the shielding system design provides reasonable assurance that the HI-STORM FW MPC Storage System will allow continued safe storage of spent fuel in accordance with 10 CFR 72.236(d).

F5.2 The staff has reasonable assurance that the new fuel type, 10x10J, is consistent with the applicable standards for shielding analyses and NRC guidance, and that the package design and contents satisfy the radiation protection requirements in 10 CFR 72.104 and 72.106.

6.0 CRITICALITY EVALUATION

The objective of this review is to ensure the proposed changes to CoC No. 1032 and SNF contents to be placed into the HI-STORM FW system, as amended, will remain subcritical under normal, off-normal and accident conditions involving handling, packaging, transfer, and storage.

Staff reviewed information in section 1, 2, 6, and 9 of the application and verified the information is consistent as well as all descriptions, drawings, figures, and tables are sufficiently detailed to support an in-depth evaluation. Staff reviewed the changes proposed to the HI-STORM FW by the applicant and determined the changes discussed below are applicable to criticality safety.

The following proposed changes are applicable to the criticality evaluation:

- PC #7 – Add the new MPC-44 to be used with Version E and UVH overpacks.
- PC #8 – Add the new MPC-37P to be used with version E overpack.
- PC #12 – Add new 10x10J fuel type to approved content.

In reviewing these changes on the HI-STORM FW's contents and criticality safety design, the staff followed the guidance in chapter 6 of NUREG-2215. This SER documents the basis for the staff's approval for the proposed changes with respect to criticality safety and meeting the applicable requirements of 10 CFR 72.124(a), 72.124(b), 72.236(c), and 72.236(g).

6.1 PC #7 – Add MPC-44

The applicant proposed to add a new MPC design called MPC-44. The MPC-44 holds 44 14x14 class of PWR fuel assemblies.

6.1.1 Criticality Design Criteria and Features

Criticality within the proposed MPC-44 is controlled by fixed geometry provided by the basket structure and neutron poison present in the Metamic-HT basket material. The applicant relies on the soluble boron within the refueling water during loading and unloading operations for criticality control within the transfer cask. The applicant placed administrative controls on allowable soluble boron concentrations, which are described in FSAR section 9, Operating Procedures. The applicant proposed no changes to the TS limits on soluble boron concentration surveillance requirements during loading and unloading and staff's previous review in Amendment No. 5 found it acceptable (NRC, 2020c). Staff finds the applicant continues to have sufficient controls to preclude the credible loss of soluble boron. For this reason, staff finds the applicant's reliance on soluble boron acceptable.

6.1.2 Fuel Specification

The new MPC-44 has two fuel assembly types as authorized contents, the 14x14A and 14x14B, which were previously approved for use in the initial application for the HI-STORM FW system (NRC, 2011). The applicant specified these fuel assembly characteristics in FSAR table 2.1.2 and in CoC appendix B table 2.1-2. The applicant followed the same assumptions related to the criticality analysis of the 14x14 assembles as with prior spent fuel assemblies. These include UO_2 density, allowance of annular pellets in top and bottom 12" of active fuel length, and locations of guide tubes and water rod locations. The applicant did not propose any changes to the fuel specifications. Therefore, staff finds the assumptions approved during the initial issuance remain acceptable and applicable for the new MPC-44.

6.1.2.1 Non-Fuel Hardware

The applicant allows for the inclusion of non-fuel hardware as authorized contents in the MPC-44 when contained within a spent fuel assembly. This hardware includes burnable poison rod assemblies (BPRAs), thimble plug devices, wet annular burnable absorbers, water displacement guide tube plugs, orifice rod assemblies, and vibration suppressor inserts with or without instrument tube tie rods. The applicant proposed no restriction on basket locations for any non-fuel hardware, however the applicant limits the total number of BPRAs to 22 in any single MPC-44.

One of the applicant’s criticality assumptions in FSAR section 6.4.7 is that the guide tubes are voided. There are competing effects with this assumption. The presence of moderator increases reactivity in under-moderated systems like modern fuel assemblies, but in this case the moderator also contains soluble boron, a neutron poison that reduces reactivity. The applicant evaluated guide tube voiding in the fuel assemblies in previous amendments of the HI-STORM FW. For the 14x14-type fuel assemblies, the filled guide tubes give a slightly higher system reactivity, k_{eff}, than voided guide tubes (an effect of about 1%). Considering that the applicant has also included the conservative, fresh fuel assumption (i.e., no burnup credit) for the MPC-44, staff finds the potential impact from guide tube voiding in 14x14-type assemblies will be more than accounted for with the fresh fuel assumption and thus acceptable for this MPC-44 analysis.

6.1.2.2 Fuel Condition

The applicant allows for the storage of damaged fuel and fuel debris in MPC-44. Damaged fuel, where the structural integrity is intact to the point which can be handled by normal means and geometric rearrangement of fuel is not expected, may be stored in locations designated for DFCs or DFIs. Up to 12 damaged fuel assemblies in DFCs and/or DFIs may be stored within the MPC-44 in locations described in appendix B to the CoC.

The applicant used the same bare array assumptions that were used to model damaged fuel in the MPC-37 and MPC-89 in previously approved HI-STORM FW evaluations. This means the applicant’s model includes no cladding or structural material of any kind. The applicant performed analysis to verify the model is conservative and maximizes reactivity. Prior staff review in the initial application (Amendment No. 0, NRC, 2011) found this assumption conservative, and staff finds its applicability continues to apply to the MPC-44.

6.1.3 Model Specification

6.1.3.1 Configuration

The applicant provided information on the MPC-44 basket dimensions and the dimensions used in its criticality evaluation in table 6.3.3 of the application. The staff verified these are consistent with the drawings. The applicant did not propose any changes to the HI-TRAC VW transfer cask in FSAR table 6.3.7. Structural analyses in chapter 3 of the application show that accident conditions result in credible deflection of the MPC-44 basket walls. The applicant accounted for this deflection in its criticality model by assuming maximum inward deflection for the entire length of the basket across all basket locations. This is the same modeling assumption that the applicant used for the MPC-37 and MPC-89 in previous amendments which staff found acceptable. The applicant showed in chapter 3 of the application that the MPC-44 basket integrity complies with the previous deflection criteria shown in FSAR table 2.2.11. As a result, staff finds the applicant sufficiently represented the bounding condition of the MPC-44 basket expected during normal, off-normal, and accident conditions in its criticality model.

The applicant selected the 14x14B fuel assembly as the bounding assembly for the MPC-44. The applicant presented the results of criticality calculations in table 6.2.3 of the application which showed the 14x14B fuel assembly to be more reactive than the 14x14A fuel assembly. Since these are the only two assembly types allowed in the MPC-44, staff finds the applicant acceptably demonstrated that the 14x14B is bounding. The bounding assembly characteristics described in FSAR section 6.2.1 remain unchanged and apply to all fuel assembly classes. In addition, the applicant used the same code (MCNP5) to determine maximum reactivity. As a result, staff finds the methodology for the bounding assembly classes for the MPC-37 are acceptable for the 14x14 assemblies in the MPC-44 (NRC, 2011).

The applicant modeled fuel assembly configuration with the assemblies shifted toward the center of the basket. In the initial application (Amendment No. 0 to the FW MPC storage system), staff found this assumption conservative and justified for the MPC-37 and MPC-89 (NRC, 2011). Staff finds this assumption also applies to and is also conservative for the MPC-44, and thus acceptable.

The applicant performed analyses with water at partial density or with the MPC partially flooded. In FSAR section 6.4.2.4, the applicant states it is possible for a DFC to remain partly filled with water while the remainder of the MPC is dry. The applicant performed criticality evaluations with just the DFCs flooded to determine the reactivity in this condition. The applicant's results showed that the most reactive configuration is when all assemblies within the MPC are fully flooded. The applicant evaluated partial flooding in both vertical and horizontal positions for the MPC-44 and presented the results in table 6.4.4 of the application. The applicant found the fully flooded condition gives the highest reactivity and presented the results of its calculations in table 6.4.2 of the application.

The applicant also addressed the reactivity effects of different external moderator density in previously reviewed revisions. With a fully flooded cask, the applicant varied the external moderator for the 10x10A and 17x17B fuel assembly types for the MPC-89 and MPC-37, respectively. The applicant showed the results of its new external moderator evaluations for 14x14B fuel assembly for the MPC-44 in table 6.4.1. Staff reviewed the applicant's results and finds that the difference continues to be small (i.e., within the calculational uncertainty) and staff finds the applicant's continued use of full density external moderator acceptable.

The applicant considered flooding in the pellet-to-clad gap regions with un-borated water and showed the results of these calculations in FSAR tables 6.2.3 and 6.4.3. Following the same assumption, the applicant calculated the reactivity effect of flooding for MPC-44. Previous evaluations have shown this to be conservative since it adds moderator without adding neutron absorbing material, and staff finds this assumption is applicable to MPC-44 and acceptable.

6.2.3.2 Material Properties

The applicant provided the compositions and densities for all materials used in the computational model in FSAR table 6.3.4. Staff reviewed the applicant's compositions and finds that they conform to standard materials used in computational analyses. Small variations in the compositions may have an effect on calculated reactivity (e.g., neutron absorption in iron); however, the compositional differences among accepted standards in the software used by the applicant and the staff's confirmatory analyses are minor. As a result, the staff finds that these variations will not significantly impact the calculated reactivity and uncertainties will be negated by the applicant's conservative modeling assumptions.

The applicant made no changes to the Metamic-HT that comprises the MPC-44 basket. The applicant assumes 90% of the ^{10}B content for the Metamic-HT fixed neutron absorber. This is consistent with NUREG-2215 and prior staff review in Amendment No. 5 (NRC, 2020c) verified the applicant has appropriate acceptance testing and assessment of Metamic-HT to ensure its continued efficacy. Staff finds the applicant’s previous analysis showing no significant degradation of the ^{10}B content over 60 years still applies, and thus staff finds the MPC-44 will meet the requirements of 10 CFR 71.124(b).

6.1.4 Criticality Analyses

6.1.4.1 Computer Programs

The applicant calculated the k_{eff} of its various configurations with MCNP 5-1.51. The applicant used cross-section libraries distributed with the code that are based on ENDF/B-V, ENDF/B-VI, and ENDF/B-VII nuclear data in MCNP Manual LA-UR-03-1987 (LANL, 2003). MCNP is a three-dimensional Monte Carlo code which can model complex geometry, such as that found with the HI-STORM FW system. The code and cross-section libraries have a long history of use in spent fuel storage applications and staff finds their use here acceptable.

6.1.4.2 Multiplication Factor

The applicant presented the results of its criticality calculations for the 14x14B fuel assembly type in the MPC-44 in tables 6.4.1 through 6.4.5 and 6.4.12 of the application. The applicant’s calculated multiplication factors for each analyzed configuration indicate the maximum k_{eff} for the evaluated scenarios remains below the 0.95 limit with all biases and bias uncertainties included. Using the criticality safety guidance specified in NUREG-2215, staff finds this provides reasonable assurance that the HI-STORM FW system will remain subcritical under normal, off-normal, and accident conditions.

6.1.4.3 Independent Staff Calculations

Staff used SCALE 6.2.3 with continuous energy cross-section libraries based on ENDF/V-VII nuclear data. Staff modeled the Metamic-HT MPC-44 basket inside a cylindrical steel shell with the dimensions of the MCP-44 cavity while ignoring all other internal components. Staff’s model also omitted the overpack. Staff simplified its criticality model since the evaluation was comparative in nature to verify the bounding conditions selected by the applicant. Staff modeled a 14x14B type PWR assembly in each basket location and used this base model to determine the change in reactivity. The staff’s model assumed a flooded condition, both internal and external to the canister, with the specified minimum dissolved boron concentration inside the canister. Staff compared the k_{eff} for the 14x14A and 14x14B and confirmed the applicant’s selection of the 14x14B fuel assembly as the bounding contents. Staff evaluated the effect of basket thickness while maintaining constant fuel separation and confirmed that the minimum basket thickness is bounding. Staff evaluated the effect of the basket ID while maintaining constant basket wall thickness and confirmed that a minimum basket ID is bounding. Staff repeated the evaluation with the 14x14B guide tubes voided and confirmed the applicant’s conclusion that assuming flooded guide tubes with borated water results in a higher calculated k_{eff}.

6.1.5 Criticality Code Benchmark Comparisons

The applicant used the same software and cross-section libraries as in the previous amendments. Since the contents remain to be UO_2 LWR assemblies with boron neutron absorbers present, the applicable benchmarks will not change. As a result, staff finds the applicant's prior bias and uncertainty calculations are applicable and acceptable for its analysis of the MPC-44.

6.1.6 Burnup Credit

The applicant did not take credit for any reactivity decrease due to irradiation of the fuel assemblies (i.e., fresh fuel assumption).

6.2 PC #8 – Add MPC-37P

The applicant proposed to add a new MPC design called MPC-37P. This MPC is with CBS and holds 37 PWR fuel assemblies of the 15x15I fuel class using the same burnup and enrichment specifications as the MPC-37. The applicant did not perform any additional criticality safety calculations for the MPC-37P. The applicant evaluated the 15x15I type assembly in previous amendments and determined them to have lower reactivity than the design basis assemblies. As a result, staff finds the applicant's prior analysis remains applicable to the contents of the MPC-37P. The MPC-37P is the same as the MPC-37 but with the box ID slightly decreased, and the box wall thickness increased. The additional wall thickness reduces the amount of moderator. The basket is made of Metamic-HT, which acts as an absorber, so increasing the basket wall thickness increases the amount of absorber (boron) available. Both of these effects will reduce reactivity. The staff compared the dimensions of the MPC-37P basket in drawing No. 12283 revision 0 to that of the MPC-37 in drawing No. 6506 revision 16 and confirmed that the characteristics of MPC-37P would have reduced reactivity as compared to the MPC-37 and found the burnup credit analyses performed for the 15x15I to support inclusion within the MPC-37 in Amendment No. 3 (NRC, 2017) are applicable and acceptable to that of the MPC-37P.

6.3 PC #12 – Add 10x10J Fuel Type

The applicant proposed to add a new 10x10 fuel type to approved contents for the MPC-89, named 10x10J. The fuel parameters for this new assembly type are provided in CoC appendix B table 2.1-3, including limits on enrichment with no Gd_2O_3 credit, cladding outer and inner diameters, pellet diameter, pitch, and active fuel length. The arrangement of full-length, short and long partial length fuel rods, and water rods for this new BWR class is shown in the application appendix 6.B.4.

6.3.1 Criticality Design Criteria and Features

The features of the MPC-89 remain unchanged.

6.3.2 Fuel Specification

The applicant provided the specification for the 10x10J assembly class characteristics in table 2.1-3 of CoC appendix B. The applicant included limits on 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, and maximum ^{235}U enrichment. Staff reviewed the parameters and finds the applicant provided the information necessary for the staff to perform a criticality safety review of the 10x10J assembly class in the MPC-89.

6.3.3 Model Specification

The addition of new contents to the MPC-89 does not change the storage system's response to normal, off-normal, or accident conditions. The applicant included the same previously approved bounding deformation and manufacturing tolerances in its criticality evaluation of the MPC-89 in Amendment No. 5 (NRC, 2020c). Aside from the new 10x10J assembly, the configuration and material properties remain unchanged.

6.3.4 Criticality Analyses

The applicant performed a criticality analysis for the requested new fuel contents in the MPC-89 canister using the criticality model and material properties from Amendment No. 5 (NRC, 2020c). The applicant conservatively ignores the presence of Gd_2O_3 normally incorporated in BWR fuel stored in the HI-STORM FW system, i.e., no gadolinium credit.

Prior analysis of radially and axially varying fuel enrichments showed that modeling the maximum planar average initial enrichment produces a higher k_{eff} than modeling the actual enrichment distribution. The difference with the 10x10J from other BWR assembly types is the location and size of the water rods, and the presence of two different-length partial rods. The applicant found that modeling the partial length rods as designed, rather than omitting them as with other 10x10 assembly classes, yielded the highest calculated reactivity. Due to this, the applicant performed additional analyses to determine a bounding, most reactive configuration. The applicant presented the results of this study in table 6.2.9 of the application and determined the bounding configuration to be one where the long, partial length rods are assumed to run the full length of the assembly and the short, partial length rods are omitted from the model. The applicant also assumed the central water rod, which has a variable diameter over the length of the assembly, maintained its minimum diameter. Staff performed independent calculations and confirmed the applicant's assumptions with its own analyses (SER section 6.3.5); therefore, staff finds the applicant's 10x10J model acceptable.

6.3.5 Independent Staff Calculations

Staff modeled the MPC-89 basket loaded with 10x10J BWR assemblies without channels. Staff used SCALE 6.2.3 with continuous energy cross-section libraries based on ENDF/V-VII nuclear data. Staff modeled the Metamic-HT basket and the MPC-89 canister shell but did not model any other internal components. Staff assumed the canister was flooded with un-borated water and that the exterior was fully reflected by un-borated water. Staff repeated the full-length array assumptions made by the applicant and confirmed that the configuration with the short partial length rods removed and the long partial length rods extended to full length was the most reactive configuration. Staff also confirmed the smaller water rod was more reactive than the larger, central water rod.

6.3.6 Criticality Code Benchmark Comparisons

The applicant used the same software and cross-section libraries as in the previous amendments. Since the contents remain to be UO_2 LWR assemblies with boron neutron absorbers present, the applicable benchmarks will not change. As a result, staff finds the applicant's prior bias and uncertainty calculations are applicable and acceptable for its analysis of the 10x10J BWR assemblies in the MPC-89.

6.3.7 Burnup Credit

No burnup credit is taken for BWR fuel.

6.4 Evaluation Findings

Based on the evaluation of the applicant's design changes and review of the applicable application sections, 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, staff reached the following findings:

F6.1 The applicant has described the systems, structures, and components important to criticality safety in sufficient detail in sections 1, 2, and 6 of the application to enable an evaluation of their effectiveness in accordance with 10 CFR 72.236(b).

F6.2 The applicant has designed the HI-STORM FW system, including its transfer cask for MPC-44 and MPC-37P canister-based systems, to be subcritical under all credible conditions in accordance with 10 CFR 72.124(a) and 72.236(c).

F6.3 The applicant based the criticality design of the MPC-44 in the HI-STORM FW system on fixed neutron poisons and soluble poisons. The applicant's evaluation of the fixed neutron poisons in the storage container has shown that the fixed neutron poisons will remain effective for the storage term of the CoC and there is no credible way for the fixed neutron poisons to significantly degrade during the storage term of the CoC. Therefore, there is no need to provide a positive means to verify their continued efficacy as required in 10 CFR 72.124(b).

F6.4 The design and proposed use of the MPC-44 and MPC-37P in the HI-STORM FW system, including SSCs involved in the handling, packaging, transfer, and storage of the radioactive materials to be stored, acceptably ensure that the materials will remain subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The applicant's analyses in the application and confirmatory analysis by the NRC adequately show that acceptable margins of safety will be maintained in the nuclear criticality parameters commensurate with uncertainties in the data and methods used in calculations, and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment under accident conditions in compliance with 10 CFR 72.124(a) and 72.236(c).

F6.5 The proposed CoC conditions, including the technical specifications, include those items necessary to ensure nuclear criticality safety in the design, fabrication, construction, and operation of the MPC-44 and MPC-37P in the HI-STORM FW system consistent with what is considered necessary to ensure compliance with 10 CFR 72.236(a), 72.236(b), and 72.236(c).

F6.6 The FSAR provides specifications of the spent fuel contents to be stored in the MPC-44 and MPC-37P within the HI-STORM FW system in sufficient detail that adequately defines the allowed contents and allows evaluation of the HI-STORM FW system nuclear criticality safety design for the proposed contents of the MPC-44 and MPC-37P.

The FSAR includes analyses that are adequately bounding for the proposed contents' specifications incompliance with 10 CFR 72.236(a).

F6.7 The applicant has designed the HI-STORM FW system, including its transfer cask for MPC-44 and MPC-37P canister-based systems, for criticality safety purposes, to be compatible with wet and dry loading and unloading facilities and, to the extent practicable, removal of the stored spent fuel from the site and transportation in accordance with 10 CFR 72.236(h) and 72.236(m).

F6.8 The applicant's analysis and evaluation of the criticality design and performance of the MPC-89 in the HI-STORM FW system have demonstrated that it will enable the storage of 10x10J BWR assembly class SNF for the term of the CoC in compliance with 10 CFR 72.236(g).

F6.9 The FSAR provides specifications of the spent fuel contents to be stored in the MPC-89 within the HI-STORM FW system in sufficient detail that adequately defines the allowed contents and allows evaluation of the HI-STORM FW system nuclear criticality safety design for the proposed contents of the MPC-89. The application includes analyses that are adequately bounding for the proposed contents' specifications incompliance with 10 CFR 72.236(a).

The staff concludes, based on the review of the proposed amendment, that the criticality design features for the HI-STORM FW cask, as amended, are in compliance with regulatory requirements in 10 CFR 72.124(a), (b), and 10 CFR 72.236(c) and (g). The applicant's analysis and evaluation of the criticality design and performance provide reasonable assurance that the HI-STORM FW will continue to allow for the safe storage of spent nuclear fuel.

7.0 CONFINEMENT EVALUATION

The objective of this review is to ensure that the confinement for the HI-STORM FW system Amendment No. 7 meets the regulatory requirements for containment performance in 10 CFR 72.236(e) and (l), and complies with the standards in ANSI N14.5, 2014, Radioactive Materials – Leakage Tests on Packages for Shipment (ANSI, 2014), as far as the applicant has committed to implement those standards.

Three proposed changes are pertaining to the confinement review:

- PC #2 – Modify the vent and drain penetrations to include the option of a second port cover plate.
- PC #3 – Allow automated equipment to perform leak test of the MPC materials and welds in the fabrication shop.
- PC #4 – Change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only, along with removing post hydrostatic test liquid PT examination.

7.1 PC #2 – Modify Vent and Drain Penetrations to Include the Option of a Second Port Cover Plate

The staff reviewed the request to modify both the vent and drain penetrations to include the option of a second port cover plate. The applicant's rationale for this proposed change was to remove the need to do field helium leak testing of these cover plates. The applicant provided justification in the proprietary attachment “Proposed Design Change for MPC Lid Port Covers” that would demonstrate improved ALARA and ruggedness of the MPC for increased reliability of confinement integrity. The applicant also provided proposed changes in the HI-STORM FW CoC, MPC drawings, and FSAR chapters 9 and 10 for staff’s review.

Staff issued a request for supplemental information (RSI) and a request for additional information (RAIs) asking how this change will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. Staff also stated in the RSI and RAIs that the weld on the port cover plate cannot have been executed under conditions where the root pass might have been subjected to pressurization from the helium fill in the canister itself. When executing vent and drain connection cover plate welds, one should not assume that the fill and drain closure valves quick-disconnects, or similar, are leak tight without performing helium leak testing. It is assumed that mechanical closure devices (e.g., a valve or quick-disconnect) permit helium leaks.

The applicant agreed with the staff that mechanical closures such as valves and quick connect fittings may not provide definitive closure against the leakage of helium required to make a leak-tight closure weld. To guard against this vulnerability and to prevent leakage of helium, the applicant employs an engineered device known as the remote valve operating assembly that uses a metallic (impermeable) seal ring to establish a high integrity “mechanical seal” whose sole purpose is to prevent loss of helium from the MPC while the welding operations are carried out.

In response to the RAIs, the applicant determined that their response with a confinement evaluation on the redundant port cover plates will take a significant amount of time; therefore, the applicant rescinded the request to remove helium leak testing when using the optional second port cover plate in this amendment application, but the new second port cover plate design will remain in the application as an alternate option and will be helium leak tested in the same manner as the current single port cover design.

The staff reviewed the application and response to the RAIs and confirmed that the new second port cover plate design will be helium leak tested in the same manner as the approved single port cover design. The staff finds this practice acceptable.

7.2 PC #3 – Allow Automated Equipment to Perform Leak Test of the MPC Materials and Welds in the Fabrication Shop

The staff reviewed the proposed change to allow the use of automated equipment to perform leak test of the MPC materials and welds in the fabrication shop. The rationale for this change was that the acceptance criteria for the leakage test will remain unchanged when implementing the automated process. The applicant also stated that by automating the leakage testing process, there will be more reliable and repeatable testing and eliminate possibility for human error. This proposed change is described in FSAR chapter 10.

After reviewing the proposed changes in FSAR chapter 10, along with the explanation provided by the applicant in the summary of proposed changes, staff finds the use of automated equipment to perform leak test of the MPC materials and welds in the fabrication shop acceptable given that the tests provide leak rate values that does not exceed the leaktight rate values stated in ANSI N14.5, 2014.

7.3 PC #4 – Change the Hydrostatic Pressure Test of the MPC Acceptance Criteria to Be Examination for Leakage Only. Remove Post Hydrostatic Test Liquid PT Examination

The staff reviewed the request to change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only, as well as the removal of the post hydrostatic test liquid PT examination. The rationale for this change was that the post hydrostatic pressure test liquid PT examination is not an ASME code requirement and causes incurred dose without corresponding safety benefit. This proposed change is described in the CoC and FSAR chapters 9 and 10.

The staff discussed with the applicant regarding the operating experience with these post-pressure liquid PT examinations. For the liquid PT examinations, the applicant stated that they have been performing hydrostatic test and there has never been any issue with the post-pressure liquid PT examinations. Almost 2,000 systems have been loaded, in which more than 1,000 MPCs are for HI-STORM 100 system and more than 250 MPCs are for HI-STORM FW system after performing hydrostatic pressure tests on these MPCs (NRC, 2023a).

After reviewing this proposed change listed in the CoC, FSAR chapters 9 and 10, and the applicant's operating experience with the post-pressure test liquid PT examinations, staff finds it acceptable to change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only, provided that the leak rate values do not exceed the leak rate values stated in ANSI N14.5, 2014. In addition, staff finds the removal of the post hydrostatic test liquid PT examination acceptable provided that the HI-STORM FW system does not exceed the dose limits in 10 CFR 72.104 and 10 CFR 72.106. Note that testing to other parts of the confinement boundary remain unchanged.

7.4 Evaluation Findings

The HI-STORM FW system Amendment No. 7 storage container confinement system has been evaluated (by appropriate tests or by other means acceptable to the NRC) to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions, in accordance with 10 CFR 72.236 (e) and (l).

The staff concludes that the design of the confinement system of the HI-STORM FW is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the HI-STORM FW will allow for the safe storage of SNF. This finding is reached based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, applicant's analysis and operating experience, and accepted engineering practices.

8.0 MATERIALS EVALUATION

The staff evaluated the materials performance of the MPC-37P and MPC-44 multi-purpose canisters and the unventilated overpack to ensure they meet the requirements of 10 CFR Part 72.

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

- PC #1 – Add a new unventilated overpack, HI-STORM FW UVH.
- PC #7 – Add the new MPC-44 to be used with version E and UVH overpacks.
- PC #8 – Add the new MPC-37P to be used with version E overpack.

The MPCs are comprised of an enclosure vessel and a fuel basket. The staff's review of the MPCs focused solely on the new fuel baskets, as both the MPC-37P and MPC-44 PWR make use of the existing MPC-37 enclosure vessel, as stated in section 3.1.3.1 of the application. The MPC-37P fuel basket is designed to position and support up to 37 PWR fuel assemblies and up to 12 PWR damaged fuel containers. The MPC-44 fuel basket is designed to position and support up to 44 PWR fuel assemblies and up to 12 PWR damaged fuel containers.

The unventilated overpack is a MPC storage system designed to provide shielding and structural protection of the MPC during storage. The principal objective of the new unventilated overpack design is to prevent stress corrosion cracking. This is accomplished by its omission of inlet or outlet vents, limiting the exposure of the overpack internals to the external environment.

8.1 Materials of Construction

As described in the application sections 1.2.1.1, 3.1.1, and the licensing drawings, MPC-37P and MPC-44 fuel baskets are comprised of a rectilinear gridwork of interconnecting Metamic-HT extruded panels. Basket shims fabricated of American Society for Testing and Materials (ASTM) B221 aluminum alloy are bolted to the periphery of the basket to occupy the space between the fuel basket and the inside surface of the enclosure vessel. The staff notes that these bolts are for axial connectivity of the basket shims and are not relied on for structural loading.

As described in the application sections 1.I.2.1.2, 3.1.1 and 3.I.1 and the licensing drawings, the unventilated overpack is comprised of a cylindrical structure consisting of inner and outer shells, a lid, and a baseplate that are all fabricated from SA 516 low carbon steel. Radial ribs connect the inner and outer shells and are also fabricated from SA 516. Plain concrete, meeting the requirements of American Concrete Institute (ACI) 318 (ACI, 2005), is installed in between the inner and outer shells and in the lid for shielding. The closure lid is bolted onto the cask body with studs (SA 540 B23 Class 3, SA 193 B8, or SB 637) and a metallic gasket to limit exposure to the external environment.

Per the above discussion, the staff finds that the applicant's description of the materials of construction to be acceptable. The specific requirements that the materials must meet are fully discussed in sections 8.3. 8.5, 8.6, 8.7, and 8.8 below.

8.2 Drawings

The applicant provided new drawings in the application section 1.5 to incorporate the new MPC-37P and MPC-44 PWR fuel baskets and in section 1.I.5 of the application to incorporate the new unventilated overpack. These drawings include a parts list that provides the material specification of each component and fabrication requirements. The staff notes that the level of detail in the new drawings are consistent with those of the previously approved drawings. The staff reviewed the drawing content with respect to the guidance in NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals" (NRC, 1998), and confirmed that the drawings provide an adequate description of the materials, fabrication, and examination requirements, and, therefore, the staff finds them to be acceptable.

8.3 Codes and Standards

The staff verified that the new MPC-37P and MPC-44 fuel basket structure uses the same proprietary aluminum-based material, Metamic-HT, as the previously approved baskets, whose test program is described in the proprietary Holtec report HI-2084122, “Metamic-HT Qualification Sourcebook,” and which was previously evaluated by the NRC in the SER for HI-STORM FW Amendment No. 0 (NRC, 2011). The fuel basket shims use the same ASTM standards as the previously approved fuel baskets. Similarly, minor changes to other components (e.g., dimensions) did not affect the applicable codes and standards.

The staff verified that the new unventilated overpack uses the same ACI construction codes and ASTM International steel materials as the previously approved overpack. The staff notes that the cited standards are consistent with NRC guidance in NUREG-2215, which states that concrete structures designs may use ACI codes and also states that other safety structures (non-confinement) may be fabricated in accordance with ASME B&PV Code Section III, Subsection NF, “Supports.” Therefore, the staff finds the materials codes and standards to be acceptable.

8.4 Welding

The new unventilated overpack uses the same welding codes and standards as the previously approved designs. The weld design will be in accordance with ASME B&PV code subsection NF, and the welding procedures, processes, and welder qualifications will be in accordance with ASME B&PV Code Section IX. The visual examinations of the welds will be performed in accordance with ASME Code, Section V with acceptance per Section III, Subsection NG, Article NG-5360. The staff notes that the applicant’s use of the cited ASME codes for the design, fabrication, and examination of the welds is consistent with the guidance in NUREG-2215. Therefore, the staff finds the welding criteria to be acceptable.

8.5 Material Properties

For the new MPC-37P and MPC-44 PWR fuel baskets designs, the applicant did not make any changes to the mechanical properties and thermal properties used in the structural analyses and thermal analysis. The staff reviewed the applicant’s thermal analysis to ensure that those material properties remain valid under the service conditions associated with the alternative heat loads for MPC-37P and the MPC-44 heat loads. In section 4 of the application, the applicant evaluated the maximum component temperatures under long-term storage and drying operations. The staff reviewed the applicant’s thermal analysis and verified that the component temperatures remain below each of the material’s allowable limits.

For the new unventilated overpack design, the applicant did not make any changes to the mechanical and thermal properties of the materials used in the structural and thermal analysis. The staff reviewed the new thermal analysis to ensure those materials properties remain valid under the service conditions associated with the new unventilated overpack. In chapter 4.I of the application, the applicant evaluated the maximum temperatures of the fuel cladding, MPC, and cask materials under normal, off-normal, and accident conditions. The staff reviewed the applicant’s analysis and verified that the component temperatures remain below each of the material’s allowable limits. Therefore, the staff finds the mechanical and thermal properties used in the applicant’s structural and thermal analysis to be acceptable.

8.6 Radiation Shielding Materials

The unventilated overpack uses concrete and steel for gamma shielding and concrete for neutron shielding. The staff verified that the unventilated overpack uses the same shielding materials as the previously approved cask. Therefore, the staff finds the applicant's description of shielding materials and geometries to be acceptable.

8.7 Concrete and Reinforcing Steel

As described in section 1.2.1.2 of the application, the new unventilated overpack uses the same concrete as the previously approved casks, complying with the requirements in the HI-STORM 100 FSAR appendix 1.D (type II plain concrete (ASTM C150 or ASTM C595)). As described in table 1.D.1 of HI-STORM 100 FSAR, appendix 1.D, the unventilated overpack follows the temperature requirements of ACI-349 (ACI, 1985). The staff reviewed the thermal analysis in SAR chapter 4.I to ensure theses concrete temperature requirements are met under the service conditions associated with the new unventilated overpack. Therefore, the staff finds the concrete design of the unventilated overpack to be acceptable.

8.8 Bolt Applications

The new unventilated overpack uses four anchor bolts to secure the cask lid to the cask body. The mounting bolts can be fabricated from SA 193 B7 or SB 637 Grade N07718 (same as the previously approved overpack) or a new option of SA 540 B23 class 3, which has been previously evaluated by the staff for the HI-STORM 100 design. Therefore, the staff finds the applicant's bolting materials to be acceptable.

8.9 Corrosion Resistance and Content Reactions

The staff reviewed the amendment changes and verified that they do not introduce any adverse corrosive or other reactions that were not previously considered in the staff's prior review of the HI-STORM FW CoC. The proposed materials of construction and the service environments are bounded by those that were previously evaluated in the CoC. Therefore, the staff finds the applicant's evaluation of corrosion resistance and potential adverse reactions to be acceptable.

8.10 Protective Coatings

In the unventilated overpack design, the applicant used the same coatings that have been previously approved for use in the HI-STORM FW overpack to mitigate atmospheric corrosion of carbon steel. Therefore, the staff finds the coatings to be acceptable.

8.11 Spent Fuel

As described in section 4.4.1.5 of the application, the applicant provided new screening calculations to bound the PCT for the most limiting service conditions for the alternative heat loads for MPC-37P and the MPC-44 heat loads. The staff reviewed the results of these screening calculations to ensure that PCT remain below 400°C, consistent with the guidance in NUREG-2215, section 8.5.15.2.4. Therefore, the staff finds the PCT associated with the new fuel baskets to be acceptable.

8.12 Editorial Change

The applicant proposed to change the CoC description to clearly indicate that only the portions of the components that come into contact with the pool water need to be made of stainless steel or aluminum. This was previously reviewed and approved in HI-STORM FW Amendment No. 8 (NRC, 2022). As stated in the SER for Amendment No. 8, this proposed change is consistent with the storage system design previously reviewed and approved by the staff, and the staff considered it an editorial change to the CoC. The staff determines that the same assessment applies to Amendment No. 7.

8.13 Evaluation Findings

The staff concludes that the design of HI-STORM FW system adequately considers material properties, environmental degradation and other reactions, and material quality controls such that the design is in compliance with 10 CFR Part 72. This finding is reached based on a review that considered the regulation, itself, the NRC's standard review plan, applicable codes and standards, and accepted engineering practices.

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

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

F8.3 The applicant has met the requirements in 10 CFR 72.236(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.

9.0 OPERATING PROCEDURES EVALUATION

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

In Amendment No. 7, the applicant revised SAR chapter 9, Operating Procedures, and added supplemental chapter 9.I, Operating Procedures, to address the following proposed changes:

- PC #1 – Add a new unventilated overpack, HI-STORM FW UVH.
- PC #2 – Modify vent and drain penetrations to include the option of second port cover plate.
- PC #4 – Change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only. Remove post hydrostatic test Liquid PT and MT examination.
- PC #6 – Use updated methodology for tornado missile stability calculations for freestanding HI-STORMs and HI-TRACs. Clarify the weights to be used for varying heights of HI-TRACs.
- PC #7 – Add the new MPC-44 to be used with Version E and UVH overpacks.
- PC #8 – Add the new MPC-37P to be used with version E overpack.

- PC #9 – Add HI-DRIP ancillary system.
- PC #11 – Include the ability to use water without glycol in the HI-TRAC water jacket during transfer operations below 32°F based on the site-specific MPC total heat loads.
- PC #14 – Other editorial changes.

In the application section 9.2.1, “Overview of Loading Operations,” on page 9-8, the applicant removed the requirement to perform an additional liquid penetrant examination on the MPC lid to shell weld after pressure testing to verify structural integrity. Also in this section, on page 9-9, the applicant added an option available to all MPCs to add a second cover plate on the drain and vent ports. Weld inspection requirements are provided for the second outer port cover welds.

In the application section 9.2.4, “MPC Closure,” on page 9-15, the applicant added more guidance on when an ethylene glycol solution must be added to the water jacket of the HI-TRAC transfer cask. In the same section, on page 9-17, the applicant revised the step to perform a liquid penetrant examination on the final pass of the MPC lid to shell weld after pressure testing. The step now will only require checking for leakage during the MPC pressure test. In the event of leakage, lid to shell weld defects are to be repaired and then nondestructive examination (NDE) performed. Further in this section, on page 9-20, the applicant provides new guidance for when the redundant MPC port cover plates are installed, on how to perform the welds, and perform NDE of the welds. If a redundant port cover plate is used, the leak test is only required on the inner port cover plate. If a redundant port cover plate is not used, a leakage test must be performed for single cover plates with no change to the existing requirement.

In the application section 9.4.2, “HI-STORM FW Recovery from Storage,” on page 9-50, the applicant added more guidance when an ethylene glycol solution must be added to the water jacket of the HI-TRAC transfer cask.

Throughout chapter 9 of the application, the applicant added operational steps on the use of the HI-DRIP cooling system and added a line item in table 9.2.1, “HI-STORM FW SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESRIPTION,” to document the HI-DRIP is not ITS.

In the application section 9.I.1, “Technical and Safety Basis for Loading and Unloading Procedures,” the applicant states: “The technical and Safety Basis for loading and unloading the HI-STORM FW identified in Section 9.1 of Chapter 9 are applicable to the HI-STORM FW UVH.”

In the application section 9.I.2.6, “Placement of HI-STORM FW UVH into Storage,” the applicant provided steps for installing a gasket under the UVH closure lid, installing the lid fastening hardware, loosening the lid fastening hardware nuts by 0.5 inches, and replacing the air in the MPC/HI-STORM FW UVH annulus with a non-oxidizing gas.

In the application table 9.I.2.2, “HI-STORM FW SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS,” the applicant added a new line item for pressure gages to ensure correct pressure during HI-STORM FW UVH backfill operations.

In the application table 9.I.2.3, “HI-STORM FW UV SYSTEM OVERPACK INSPECTION CHECKLIST,” the applicant added additional checklist items for the HI-STORM FW UVH overpack lid and HI-STORM FW UVH main body when using the UVH overpack.

In the application section 9.I.3, “ISFSI Operations,” the applicant provided general guidance on the UVH when in storage and the need for minor maintenance or removal of a MPC from the UVH.

In the application section 9.I.4, “Procedure for Unloading the HI-STORM FW UVH Fuel in the Spent Fuel Pool,” the applicant provided procedure steps specific to the UVH when recovering it from storage and unloading the HI-STORM FW UVH fuel in the spent fuel pool.

See chapters 3 through 8 for safety evaluations of the specific proposed changes.

9.1 Evaluation Findings

The staff concludes that the generic procedures and guidance for the operation of the HI-STORM FW UVH and the new canisters, MPC-44 and MPC-37P, are in compliance with 10 CFR Part 72 and sufficient information was provided for a safety evaluation and review. The evaluation of the operating procedure descriptions in the application provides reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F9.1 The HI-STORM FW system with these proposed changes is still compatible with wet loading and unloading, and therefore, is in compliance with 10 CFR 72.236(h). General procedure descriptions for these operations are summarized in the application chapter 9 and chapter 9.I. Detailed procedures will need to be developed and evaluated on a site-specific basis.

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

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

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

F9.5 The application with the Amendment No. 7 changes still includes acceptable descriptions and discussions of the HI-STORM FW operations, operating characteristics, and safety considerations, in compliance with 10 CFR 72.234(f).

10.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM EVALUATION

The objective of this review is to ensure that the applicant’s application includes the appropriate acceptance tests and maintenance programs for the system. A clear, specific listing of these commitments will help avoid ambiguities concerning design, fabrication, and operational testing requirements when the staff conducts subsequent inspections. The acceptance tests demonstrate that the cask has been fabricated in accordance with the design criteria and that the initial operation of the cask complies with regulatory requirements. The maintenance program describes actions that the licensee needs to implement during the storage period to ensure that the cask performs its intended functions.

In Amendment No. 7 the applicant revised SAR chapter 10, “Acceptance Criteria and Maintenance Program,” and added supplement chapter 10.I, “Acceptance Criteria and Maintenance Program,” to address the following proposed changes:

- PC #1 – Add a new unventilated overpack, HI-STORM FW UVH.
- PC #2 – Modify vent and drain penetrations to include the option of second port cover plate.
- PC #3 – Allow automated equipment to perform leak test of the MPC materials and welds in the fabrication shop.
- PC #4 – Change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only. Remove post hydrostatic test Liquid PT and MT examination.
- PC #9 – Add HI-DRIP ancillary system.
- PC #14 – Other editorial changes.

In the application section 10.1.2.2.2, “MPC Confinement Boundary,” on pages 10-9 and 10-10, the applicant discusses the pneumatic or hydrostatic pressure testing of the MPC lid to shell field weld. The applicant has changed the requirement to re-examine the MPC lid to shell weld by liquid penetrant examination to just visually re-examine the weld for leakage after the pressure hold period. During the visual re-examination, any evidence of leakage, cracking or deformation shall be cause for rejection, repair and retesting.

In the application section 10.1.4, “Leakage Testing,” on page 10-12, the applicant added the option for shop leakage tests of MPC base metals and enclosure welds to be performed using automatic leak test equipment to minimize the need for operator actions and interpretations. Also in this section, the applicant added the option on all MPCs for the addition of a second port cover plate on the drain and vent ports. The outer port cover plate is to be welded using a minimum of three weld passes that bridge the weld joint. It also now states that helium leak testing is only required on inner port cover when the redundant port cover design is used.

In the application table 10.1.1, “MPC INSPECTION AND ACCEPTANCE CRITERIA,” the applicant has added in the Maintenance and Operations block for leak test that if the redundant port cover design is used on the vent and drain ports, helium leak testing is only required on inner port cover.

In the application table 10.1.4, “HI-STORM FW NDE REQUIREMENTS,” the applicant added information for weld NDE requirements when the redundant port cover plate option is used.

The applicant added new SAR section 10.1.7.1.1, “HI-DRIP Supplemental Cooling Thermal Acceptance Testing,” to perform testing of the HI-DRIP system and evaluate its heat removal capability.

In the application chapter 10.I, “Acceptance Criteria and Maintenance Program,” in section 10.I.0, “Introduction,” the applicant states: “The acceptance and maintenance program associated with the use of the HI-STORM FW UVH system, described in supplement 10.I, are like the maintenance and acceptance program for the standard HI-STORM FW system. The following sections describe the requirements that are, in any respect, unique to the HI-STORM

FW UVH system and thus supplement the information presented in chapter 10…. The guidance provided in this supplement shall be used along with the procedures provided in chapter 10 to develop the site-specific maintenance program for the HI-STORM FW UVH.” The staff notes that there are little changes to the testing Acceptance Criteria and Maintenance Program when the HI-STORM FW UVH is used.

In the application section 10.I.2.2, “Leakage Tests,” and table 10.I.2.1, “HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE,” the applicant states “The unventilated Storage system lid gasket requires the additional maintenance step of replacement anytime the joint is completely disassembled. A new gasket shall be used upon reassembly.”

See chapters 3 through 8 for safety evaluations of the specific proposed changes.

10.1 Evaluation Findings

The staff concludes that the acceptance tests and maintenance program for the HI-STORM FW ventilated and UVH overpacks are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F10.1 Chapters 10 and 10.I of the the application adequately describe the applicant’s proposed program for preoperational testing, initial operations, and maintenance of the HI-STORM FW ventilated and UVH Systems.

F10.2 SSCs ITS will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform.

F10.3 The applicant will examine and/or test the HI-STORM FW to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness.

11.0 RADIATION PROTECTION EVALUATION

The objective of the radiation protection evaluation is to determine that the applicant has proposed a functional radiation protection program that will effectively manage, monitor, and control radiation exposures and doses to facility workers and members of the public from a dry storage facility in compliance with NRC regulations and acceptance criteria.

The applicant requested some additions that includes a new variant of the HI-STORM FW Overpack UVH, a new MPC-44 and MPC-37P. For the Overpack UVH, components in the design have been principally focused on avoiding exposure to radiation that does not have a direct benefit to the workers based on ALARA during the short-term operations as well as during long-term storage. For the UVH overpack, the absence of vent openings eliminates localized areas of increased dose rates to workers during loading and handling operations. The absence of vent openings also reduces dose rates from stationary casks on an ISFSI and minimizes dose rates at the site boundary. The need for periodic inspection of the vent openings and associated LCOs in the CoC are no longer required thus eliminating this source of radiation dose to the site staff.

Some of the key design features engineered in the system components to minimize occupational dose and site boundary dose are summarized in table 11.I.2.1 of the application. The key design features engineered in the HI-TRAC VW components to minimize occupational dose and site boundary dose are summarized in FSAR table 11.2.1.

11.1 Occupational Exposures

The staff reviewed the application 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 UVH overpack, the applicant used the shielding analysis provided in supplement 5.I of the application. The applicant presented the exposures estimate in the application table 11.I.3.2, "Estimated Person-mrem Dose for Loading the HI-STORM FW UVH System," to account for the proposed source terms.

The applicant has determined the maximum dose rates by using a single 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. This would especially be true if the reason for the dose rates being decreased is a result of increasing the lead shielding in the variable weight transfer cask, in which case the dose rate reduction would only be seen at the side of the cask and not necessarily the top.

The staff found this to be a reasonable approach for determining that the HI-STORM UVH overpack with the authorized contents can meet occupational dose in 10 CFR 20.1201 because the absence of vent openings eliminates localized areas of increased dose rates from the source of radiation emitted from the canister to workers during loading and handling operations. The absence of vent openings also reduces dose rates from stationary casks on an ISFSI and minimizes dose rates to nearby buildings, to the nearby environment, and to the site boundary. The need for periodic inspection of the vent openings and associated LCOs in the CoC is not required.

11.2 ALARA

The applicant provided several supplemental shielding components, with associated estimated dose rates, in table 11.I.3.2 of the application, that can be used to maintain exposures ALARA for loading UVH overpack. These supplemental shielding components are consistent with those described in FSAR table 11.3.2 for loading the ventilated system. The dose rates presented in table 11.I.3.2 are within the regulatory limits. Therefore, the staff found that the applicant has adequately described the components needed for maintaining exposures ALARA.

11.3 Estimated Controlled Area Boundary Dose Assessment

The applicant presented the dose rates at various distances from sample ISFSI arrays of HI-STORM FW UVH for the bounding burnup and cooling time in table 5.I.4.3 of the application, which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in supplement 5.I of the application.

The staff evaluated the minimum distance from the ISFSI to the controlled area boundary from table 5.I.4.3. The applicant's calculations assumed one hundred percent (100%) occupancy

(8,760 hours), and all the calculated dose rates are within the regulatory limits in 10 CFR 72.104. Therefore, the staff found that the applicant has demonstrated that the HI-STORM FW UVH system would be in compliance with 10 CFR 72.104.

11.4 Controlled Area Boundary Dose for Off-Normal Conditions

The staff evaluated the application supplement 12.I, “Off-Normal and Accident Events” postulated for off-normal condition which is elevated off-normal environmental temperature. The staff found that this off-normal condition does not result in the degradation of the HI-STORM FW UVH system shielding effectiveness. The staff concluded that the dose at the controlled area boundary from direct radiation for the elevated off-normal environmental temperature is going to be equal to that of normal conditions. Therefore, there will be no effect to occupational or public exposures as a result of this off-normal condition.

11.5 Controlled Area Boundary Dose for Accident Conditions

The UVH overpack, made of steel and devoid of any vents, emulates a metal cask in respect of critical functions under accident conditions such as the design basis fire. However, because of its larger footprint and greater mass, it is a far superior in respect of shielding capacity and seismic stability in comparison to any peer metal cask.

The design basis accidents analyzed in this amendment have a negligible effect on the HI-STORM FW UVH overpack, but a larger effect on the HI-TRAC, and results of the dose rate is presented in table 5.I.1.9 of the application. The dose rate at the 100-meter controlled area boundary during the accident condition in a period of 30 days is 2.4 rem, which is less than the regulatory limit of 5 rem in 10 CFR 72.106. The results also showed that 62 days are needed to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in supplement 12.I of the application. The staff found these results acceptable because the dose requirement of 10 CFR 72.106 is satisfied.

11.6 Evaluation Findings

The staff concludes that the design of the radiation protection system of the HI-STORM FW UVH, the addition of MPC-44 and MPC-37P, and the addition of 10X10J fuel are 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. 7 will allow safe storage of proposed new 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.

F11.1 The HI-STORM FW system Amendment No. 7 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. 7 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.

12.0 ACCIDENT ANALYSES EVALUATION

The staff's evaluations of the applicant's accident analysis for the proposed changes are documented in sections 3 through 11 of this SER. Therefore, this section need not provide further evaluation.

13.0 TECHNICAL SPECIFICATIONS AND OPERATING CONTROL AND LIMITS EVALUATION

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

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

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

Page Number	Reference	Description	Proposed Change
CoC, pages 1-2	Description	(1) Update the maximum number of pressurized water reactor fuel assemblies to be stored in MPC. (2) Add MPC-37P and MPC-44 to the list. (3) Add description for UVH overpack.	1, 7, 8
CoC, page 1	Description	Update the description of the HI-STORM FW system in the CoC to clearly indicate that only the portions of the components that come into contact with the pool water need to be made of stainless steel or aluminum.	N/A. This change was previously approved in Amendment No. 8 (NRC, 2022)
CoC, page 3	Condition 8	Add a statement that Condition 8 does not apply to UVH.	1
Appendix A, page 1.1-3	Definition	Update the definition for "Overpack" to account for unventilated overpack	1
Appendix A, page 1.1-3	Definition	Add the definition for "Redundant Port Cover Design"	2
Appendix A, pages 1.1-4 & 1.1-5	Definition	Add the definitions for "Unventilated Overpack" and "Ventilated Overpack"	1
Appendix A, page 3.1.2-1	Limiting condition for operation (LOC) 3.1.2, Note	Add a statement that LCO 3.1.2 only applies to ventilated overpacks.	1

Page Number	Reference	Description	Proposed Change
Appendix A, page 3.1.2-3	SR 3.1.2	Add surveillance requirement for MPC-37P and MPC-44.	7, 8
Appendix A, page 3.3.1-1	LCO 3.3.1	Add boron concentration condition for MPC-37P and MPC-44.	7, 8
Appendix A, pages 3.4-1 & 3.4-2	Table 3-1	Add information for: (1) MPC-37 with unventilated overpack. (2) MPC-37P with ventilated overpack. (3) MPC-44 with ventilated and unventilated overpacks. (4) MPC-89 with unventilated overpack.	1, 7, 8
Appendix A, page 3.4-3	Table 3-2	Add information for: (1) MPC-37 with unventilated overpack. (2) MPC-89 with unventilated overpack. (3) MPC-37P with ventilated overpack. (4) MPC-44 with ventilated and unventilated overpacks.	1, 7, 8
Appendix A, page 5.0-2	Section 5.2, subsection c.2	Editorial change to clarify applicable stress limits are for the lifting attachments.	N/A
Appendix A, page 5.0-2	Section 5.2, subsection c.4	Identify lifting heights conditions for transfer cask, ventilated overpack, and unventilated overpack.	1
Appendix A, page 5.0-3	Table 5-1	Specify the lifting requirements is for ventilated overpack.	1
Appendix A, page 5.0-4	Section 5.3.3, subsection d	Add a statement to specify that section 5.3.3.d only applies to ventilated overpack.	1
Appendix A, page 5.0-5	Section 5.3.8, subsection d	Add a statement to specify that section 5.3.8.d only applies to ventilated overpack.	1
Appendix B	Table of Content	Updated	N/A
Appendix B, page 2-1	Section 2.1.2	Add fuel loading information for MPC-37P and MPC-44.	7, 8
Appendix B, page 2-5	Figure 2.1-4	Add MPC-37P cell identification.	8
Appendix B, page 2-6	Figure 2.1-5	Add MPC-44 cell identification.	7
Appendix B, pages 2-8 & 2-12	Table 2.1-1, Sections I.D & III.D	Correct typo for BPRAs.	N/A
Appendix B, pages 2-13 through 2-16	Table 2.1-1, Sections IV & V	Add fuel assembly limit information for MPC-37P and MPC-44.	7, 8

Page Number	Reference	Description	Proposed Change
Appendix B, pages 2-21 through 2-25	Table 2.1-3	(1) Update bounding fuel variables for 8x8F fuel. (2) Update bounding fuel variables for 10x10I fuel. (3) Add new fuel type 10x10J. (4) Update bounding fuel variables for 11x11A fuel. (5) Add notes 18 through 21.	12, 13
Appendix B, pages 2-26 & 2-27	Section 2.3	(1) Update decay heat limits for ventilated overpack, including for the new MPC-37P and MPC-44. (2) Add subsection 2.3.2 for decay heat limits for unventilated overpack, including MPC-37, MPC-89, and MPC-44. (3) Add subsections 2.3.3 and 2.3.4 for variable fuel height information for MPC-37, MPC-44, and MPC-89.	1, 7, 8
Appendix B, page 2-28	Table 2.3-1A	Include MPC-37P to the table.	8
Appendix B, page 2-31	Tables 2.3-7A & 2.3-7B	Add tables for MPC-37P heat load data for ventilated overpack.	8
Appendix B, page 2-32	Tables 2.3-8A & 2.3-8B	Add tables for MPC-44 heat load data for ventilated overpack.	7
Appendix B, page 2-33	Tables 2.3-9A & 2.3-9B	Add tables for MPC-37 heat load data and requirements for developing regionalized heat load patterns for unventilated overpack.	1
Appendix B, page 2-34	Tables 2.3-10A & 2.3-10B	Add tables for MPC-89 heat load data and requirements for developing regionalized heat load patterns for unventilated overpack.	1
Appendix B, page 2-35	Table 2.3-11	Add a table on section heat load calculations for MPC-37.	1
Appendix B, page 2-36	Table 2.3-12	Add a table on section heat load calculations for MPC-89.	1
Appendix B, page 2-36	Table 2.3-13	Add a table for MPC-44 heat load data for unventilated overpack.	1, 7
Appendix B, pages 2-50 & 2-51	Figures 2.3-14 & 2.3-15	Add two loading patterns for MPC-37P.	8
Appendix B, page 2-52	Section 2.4	Include MPC-37P in the burnup credit calculation.	8

Page Number	Reference	Description	Proposed Change
Appendix B, page 2-56	Section 2.5	(1) Update burnup equation for MPC-32M in section 2.5.1. (2) Add section 2.5.2 on burnup equation for MPC-37P and MPC-44. (3) Add section 2.5.3 on alternate burnup and cooling time limits for MPC-37P and MPC-37. (4) Add section 2.5.4 on burnup and cooling time limits for 10x10J assembly class.	7, 8, 12, 14
Appendix B, page 2-57	Table 2.5-2	Add note 2 on information for loading 10x10J assembly class in MPC-89.	12
Appendix B, page 2-58	Table 2.5-3	Add a table on the fuel qualification requirements for MPC-37P & MPC-44.	7, 8
Appendix B, page 2-59	Table 2.5-4	Add a table on the alternative fuel qualification requirements for MPC-37P.	8
Appendix B, page 2-60	Table 2.5-5	Add a table on the alternative fuel qualification requirements for MPC-37.	8
Appendix B, page 2-61	Table 2.5-6	Add a table on fuel qualification requirements for 10x10J fuel assembly class.	12
Appendix B, page 3-2	Sections 3.2.5 & 3.2.6	Add criticality control information for MPC-37P and MPC-44.	7, 8
Appendix B, page 3-9	Section 3.4, subsection 1	Add yearly ambient temperature information for unventilated overpack.	1
Appendix B, page 3-10	Section 3.4, subsection 3c	Editorial change to delete “HI-STORM FW.”	N/A

The staff finds that the other proposed changes to the TS for the HI-STORM FW system Amendment No. 7 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. 7 will continue to allow safe storage of SNF.

14.0 QUALITY ASSURANCE EVALUATION

The applicant did not propose any changes that affect the staff’s quality assurance evaluation provided in the previous SERs for CoC No. 1032, Amendment Nos. 0 through 6. 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:

PC #1 Add a new UVH overpack, HI-STORM FW UVH, which includes high density concrete for shielding. The UVH is to be used with MPC-37 (MPC-37 standard), MPC-89 (MPC-89 standard), and the new MPC-44 (with CBS).
PC #2 Modify vent and drain penetrations to include the option of second port cover plate.
PC #3 Allow automated equipment to perform leak test of the MPC materials and welds in the fabrication shop.
PC #4 Change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only. Remove post hydrostatic test liquid PT and MT examination.
PC #5 Include the ability to use CFD analysis to evaluate site-specific fire accident scenario.
PC #6 Use updated methodology for tornado missile stability calculations for freestanding HI-STORMs and HI-TRACs. Clarify the weights to be used for varying heights of HI-TRACs.
PC #7 Add a new MPC, MPC-44, with CBS and to hold 44 PWR fuel assemblies of certain 14x14 fuel class. It is to be used with Version E and UVH overpacks. Note that there is no standard version of MPC-44 for FW storage system.
PC #8 Add a new MPC, MPC-37P, with CBS and to hold 37 PWR fuel assemblies of certain 15x15 fuel class. It is to be used with version E overpack. Note that there is no standard version of MPC-37P for FW storage system.
PC #9 Add HI-DRIP ancillary system. HI-DRIP is an optional ancillary system designed to prevent water within the MPC from boiling during loading and unloading operations while loading MPC in the HI-TRAC.
PC #10 Include the ability to use CFD analysis to evaluate site-specific burial-under-debris accident scenario.
PC #11 Include the ability to use water without glycol in the HI-TRAC water jacket during transfer operations below 32°F based on the site-specific MPC total heat loads.
PC #12 Add new 10x10J fuel type to approved content in the HI-STORM FW system.
PC #13 Update bounding fuel variables for 8x8F and 11x11A BWR fuel types in CoC appendix B.
PC #14 Other editorial changes.
PC #15 Adopt a stress-based structural design criterion as documented as option 2 in the September 8, 2023 conversation record (NRC, 2023h).
PC #16 Establish specific criteria on allowable interference due to differential thermal expansion (DTE).

Based on the statements and representations provided by the applicant in its amendment application, as supplemented, the changes described above to the HI-STORM FW MPC Storage System in Amendment No. 7 do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. Amendment No. 7 for the HI-STORM FW MPC Storage System should be approved.

Issued with Certificate of Compliance No. 1032, Amendment No. 7
on August 13, 2024.

References

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Holtec, 2021b. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 Responses to Requests for Supplemental Information." October 15, 2021. This package contains 19 attachments, and Attachments 1, 5, 7 through 10, and 18 are *Proprietary Information and Not Publicly Available*. ML21288A521.

Holtec, 2022a. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 Responses to Request for Additional Information Part 1." July 11, 2022. This package contains 9 attachments, and Attachments 1, 5, 7, and 8 are *Proprietary Information and Not Publicly Available*. ML22192A215.

Holtec, 2022b. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 Responses to Requests for Additional Information Part 1 – Additional Supporting Documents." July 13, 2022. This package contains 3 attachments, and Attachments 1 and 2 are *Proprietary Information and Not Publicly Available*. ML22194A953.

Holtec, 2022c. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 Responses to Requests for Additional Information Part 2." July 29, 2022. This package contains 7 attachments, and Attachments 3, 5, and 6 are *Proprietary Information and Not Publicly Available*. ML22210A145.

Holtec, 2022d. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 RAI Responses Part 1 Clarification Call Action Items." September 15, 2022. This package contains 9 attachments, and Attachments 3, 6, and 8 are *Proprietary Information and Not Publicly Available*. ML22258A250.

Holtec, 2022e. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 Responses to Requests for Additional Information Part 3." October 3, 2022. This package contains 3 attachments, and Attachment 1 is *Proprietary Information and Not Publicly Available*. ML22276A281.

Holtec, 2022f. Letter from Holtec International to NRC, "HI-STORM FW Amendment 7 RAI 5-2 Response Clarification." December 1, 2022. ML22336A132

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Holtec, 2023b. Letter from Holtec International to NRC, “HI-STORM FW Amendment 7 Responses to Requests for Additional Information Part 5.” May 8, 2023. This package contains 10 attachments, and Attachments 1, 3, and 5 through 9 are Proprietary Information and Not Publicly Available. ML23128A302.

Holtec, 2023c. Letter from Holtec International to NRC, “HI-STORM FW Amendment 7 RAI Responses Part 5 Clarification Call Action Items.” June 30, 2023. This package contains eight attachments, and Attachments 2, 4, 6, and 7 are *Proprietary Information and Not Publicly Available*. ML23181A192.

Holtec, 2023d. Letter from Holtec International to NRC, “HI-STORM FW Amendment 7 RAI Responses Part 5 Clarification Corrected Attachments 4 and 5.” July 11, 2023. This package contains three attachments, and the attachment labeled as Attachment 4 is *Proprietary Information and Not Publicly Available*. ML23192A031.

Holtec, 2023e. Letter from Holtec International to NRC, “HI-STORM FW Amendment 7 RAI 3-10 Response Clarification Call Action Items.” August 15, 2023. This package contains three attachments, and Attachment 2 is *Proprietary Information and Not Publicly Available*. ML23227A248.

Holtec, 2023f. Letter from Holtec International to NRC, “HI-STORM FW Amendment 7 RAI Response Clarifications (Part 3).” November 17, 2023. This package contains eight attachments, and Attachments 2 and 4 through 7 are *Proprietary Information and Not Publicly Available*. ML23321A245.

Holtec, 2024a. Letter from Holtec International to NRC, “HI-STORM FW Amendment 7 RAI Response Clarifications (Part 4).” February 16, 2024. This package contains five attachments, and Attachments 1, 2, and 4 are Proprietary Information and Not Publicly Available. ML24047A323.

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