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NUCLEAR REGULATORY COMMISSION
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PRELIMINARY SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HOLTEC INTERNATIONAL
HI-STORM 100 CASK SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 16

SUMMARY

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (staff) review and evaluation of Holtec International's (hereafter Holtec or applicant) request to amend Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Cask System. Holtec submitted its request by letter dated March 9, 2021 (Holtec, 2021a), and supplemented it on August 11, 2021 (Holtec, 2021b), August 31, 2022 (Holtec, 2022a), September 9, 2022 (Holtec, 2022b), October 3, 2022 (Holtec, 2022c), January 4, 2023 (Holtec, 2023a), January 5, 2023 (Holtec, 2023b), January 13, 2023 (Holtec, 2023c), March 17, 2023 (Holtec, 2023d), and September 20, 2023 (Holtec, 2023e). On February 17, 2022 (Holtec, 2022d), Holtec requested to separate the RIRP-I-16-01 CoC reorganization, also known as graded approach (NRC, 2020a), from Amendment No. 16. Therefore, this SER does not include evaluation of the RIRP-I-16-01 CoC reorganization.

Holtec proposed the following proposed changes (PC):

PC #1 Add a new unventilated high-density (UVH) overpack, HI-STORM 100 UVH, which includes high-density concrete for shielding. The UVH overpack is to be used with the multi-purpose canister (MPC)-32M and MPC-68M.

PC #2 Include the ability to use the computational fluid dynamics (CFD) analysis to evaluate site-specific fire accident scenario.

PC #3 Modify the vent and drain penetrations to include the option of a second port cover plate.

PC #4 Include the ability to use CFD analysis to evaluate site-specific burial under debris accident scenario.

PC #5 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 #6 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 #7 Replace the fuel qualification tables (FQT) in final safety analysis report (FSAR) chapter 2 and the CoC, including the equation for calculation of the maximum allowable burnup as a function of the cooling time and cooling time-dependent coefficients, with simpler sets of burnup and cooling time limits. Reduce the minimum cooling time for pressurized-water reactor (PWR) fuel from 2 years to 1 year.

PC #8 Revise FSAR appendix 1.D to enhance certain requirements, to add certain revised shielding assumptions following a significant thermal event, and to add critical characteristics for concrete employed in the HI-STORM 100 UVH system. Appendix 1.D is renamed to “Specification for Plain Concrete in the HI-STORM Family of Overpacks.”

This revised CoC, when codified through rulemaking, will be denoted as Amendment No. 16 to CoC No. 1014.

This SER documents the staff’s review and evaluation of the proposed 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 FSAR or previous amendments to the CoC.

1.0 GENERAL INFORMATION EVALUATION

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

1.1 Dry Storage System (DSS) Description and Operational Features

The HI-STORM 100 Cask System is a vertical DSS. The system consists of: (i) interchangeable MPCs that contain spent nuclear fuel (SNF), (ii) the overpack (HI-STORM) that contains the MPC during storage, and (iii) the transfer cask (HI-TRAC) that contains the MPC during loading, unloading, and transfer operations. The applicant proposed that a new variant of the HI-STORM 100 overpack, called HI-STORM 100 Version UVH be added to the HI-STORM 100. The UVH overpack is a simplified version of the HI-STORM 100 overpack without inlet and outlet air passages, resulting in no ventilation in the space between the overpack cavity and the stored MPC.

The applicant provided the description of the UVH overpack in the proposed FSAR Supplement IV of the application. The UVH overpack is a dual buttressed steel shell structure with the inter-shell space filled with plain concrete. The key distinguishing feature of the UVH overpack is that it has no inlet or outlet vents. Thus, there is no ventilation flow of air around the MPC. Rather, the UVH overpack is designed to remove the fuel’s decay heat from the external surface of the MPC without the benefit of ventilation flow. Removal of heat from the external surface of the

MPC to the external surface of the overpack is facilitated by a combination of conduction and radiation modes of heat transmission.

The design and specifications for the MPC, transfer cask, and independent spent fuel storage installation (ISFSI) are not affected by the proposed changes in the Amendment No. 16. Since the structural performance of these structures will not be affected, the structural technical evaluation focused primarily on the UVH overpack design and its effects on the structural performance and safety of the DSS.

1.2 Engineering Drawings

The amendment application includes updated engineering drawings for the MPC and overpack that describe the structural modifications made to the DSS for the MPCs and UVH overpack. The updated drawings include:

- Drawing 3923 Revision 44 – MPC Enclosure Vessel
- Drawing 11572 Revision 1 – MPC Enclosure Vessel Version 1
- Drawing 12233 Revision 0 – HI-STORM 100 Version UVH

The optional secondary cover plates on the MPC enclosure vessel are shown on Drawing 3923 Revision 44 and for the MPC Version 1 enclosure vessel on Drawing 11572 Revision 1. The 3/8-inch thick Alloy X redundant port cover plate is attached with a full penetration single bevel weld with a minimum of three passes and visual/penetrant testing of the root, middle, and final pass. The general arrangement, overall size/dimensions, materials, and component safety categories are shown in the drawings.

The UVH overpack is shown on Drawing 12233 Revision 0. The general arrangement, overall size/dimensions, materials, and component safety categories are shown in the drawing. The overpack is similar in construction to the HI-STORM 100S version but does not have inlet or outlet air vents. The overpack design includes the higher strength concrete and eight additional full depth steel ribs welded between the inner and outer shells. The 2-inch body base thickness, ¾-inch outer shell thickness, 31.8-inch body concrete thickness, and 14-inch lid concrete thickness are comparable to the designs of the other overpack versions. It was noted that the inner shell is only ¾-inch thick as compared to 1-1/4-inch thick for the 100S and 1-inch thick for the 100S Version B overpacks, but it is welded to and supported by the additional support ribs. The inner diameter of the inner shell is also permitted to be machined until acceptable radial clearance of 5/32 inch with the MPC is obtained. The material specification for the lid stud was also changed to the higher strength SA-540 B23/Class 3 from SA 193 B7 as used on the 100S and 100S Version B overpacks. The overpack includes a drain assembly to vent the overpack cavity for any internal pressure buildup prior to any UVH overpack lifting activities.

The MPC and overpack dimensions depend on the length of fuel stored, including control components, to provide room for differential thermal expansion and manufacturing tolerances. Minimum dimensions are provided in the proposed FSAR table 3.IV.2.1 of the application. The MPC has an outer diameter of 68-1/2 inches and height of approximately 181.25 inches for a W17x17 PWR fuel.

The overpack cavity has an inner diameter based on a 5/32-inch radial clearance with the MPC outer diameter and maximum height based on a 2-inch clearance with the MPC height. The overpack exterior has an outer diameter of 133 inches and height of approximately 206 inches for a W17x17 PWR fuel. The maximum gross weight of the fully loaded overpack with 200 lb/ft^3 high-density concrete is 410 kip and the maximum weight of the loaded MPC is 90 kip.

The staff reviewed the drawings of the UVH overpack on the general assembly drawings for completeness and accuracy, and finds that the geometry, dimensions, material, components, notes, fabrication details are adequately incorporated throughout the application and are acceptable.

1.3 Findings

The staff reviewed the general description and discussion of the UVH overpack presented in the application with special attention to the design and operating characteristics, unusual design features, and principal considerations important to safety, and finds that the information provided is adequate for the staff to review and evaluate the application.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

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

The applicant states that the principal design criteria for the UVH storage system are unchanged from the design criteria of the HI-STORM 100 Cask System in all respects except for those relating to its functions associated with environment control.

2.1 Classification of Structures, Systems, and Components

The HI-STORM 100 FSAR provides a list of the SSCs that are important to safety in table 2.2.6 (Holtec, 2021c). The UVH overpack design is highly similar to the HI-STORM 100 construction and essentially contains the same SSCs providing the same safety functions (except for the absence of integral overpack ventilation). Three additional structural SSCs that are important to safety (ITS) and unique to the UVH overpack are identified. First, the UVH overpack contains eight additional full depth ribs between the inner and outer shells that are assigned quality category B. The ribs provide enhanced decay heat removal through the overpack thickness. Second, the UVH overpack contains a higher density concrete that is assigned quality category B. The concrete has a higher thermal conductivity to provide enhanced decay heat removal through the overpack thickness. Third, the MPC contains an optional redundant cover for the drain/vent ports and is assigned quality category A.

All SSCs for the HI-TRAC transfer cask are unaffected by the addition of the UVH overpack.

2.2 Design Bases for Structures, Systems, and Components Important to Safety

The HI-STORM 100 Cask System stores PWR and boiling-water reactor (BWR) SNF with various MPCs. The HI-STORM 100 UVH system utilizes a subset of the same fuel and MPCs that have been previously approved. It is allowed to store BWR fuel in the MPC-68M and PWR fuel in the MPC-32M as shown in the proposed FSAR table 2.IV.1.1 of the application.

The applicable codes and standards for the structural SSCs of the HI-STORM MPC and overpack designs are detailed in table 2.2.6 of the FSAR (Holtec, 2021c). Table 2.2.7 summarizes the applicability of the American Society of Mechanical Engineers (ASME) BPVC

Code (ASME, 2010) for the HI-STORM system. The MPC components comprising the confinement boundary are designed, fabricated, and inspected per ASME BPVC Code, Section III, Subsection NB. The MPC components comprising the fuel basket are designed, fabricated, and inspected per Section III, Subsection NG of the code. The overpack components providing structural support and damage fuel containers/isolators are designed, fabricated, and inspected per Section III, Subsection NF of the code. The ASME Code alternatives are provided in table 2.2.15 of the FSAR. The overpack concrete is designed per American Concrete Institute (ACI) 349/ACI 318.1-89(92) and fabricated/inspected per ACI 349 as specified in appendix 1.D. These codes and standards are also applicable to the version UVH storage system.

Principal design criteria unique to the UVH overpack are discussed in the proposed FSAR section 2.IV.2 of the application. The applicant considers load case combinations for normal, off-normal, and accident conditions for the MPC, overpack, and transfer cask as provided in table 2.2.14 of the FSAR (Holtec, 2021c). Temperature, internal pressure, external pressure, dead weight, handling, and snow loads are considered for normal conditions. Pressure in the overpack cavity is evaluated due to the use of a gasket between the overpack body and lid. Off-normal conditions consider temperature, internal pressure, external pressure, and handling loading. Handling, earthquake, fire, missile, wind, flood, explosion, internal pressure, and external pressure loads are considered for accident conditions. The loads and load combinations evaluated in the HI-STORM 100 FSAR for normal, off-normal, and accident conditions are unchanged for evaluation of the UVH overpack. Mechanical loadings, applicability, and evaluation for the UVH overpack is described in the proposed FSAR table 2.IV.2.1 of the application.

2.3 Evaluation Findings

The staff reviewed the principal design criteria of the HI-STORM 100 Version UVH presented in the application and finds that: (i) the application adequately defines the bounding conditions under which the HI-STORM 100 UVH system is expected to operate in accordance with the requirements of 10 CFR 72.236(b), (ii) the application relating to the design bases and criteria for structures categorized as ITS for the HI-STORM 100 version UVH meet the requirements given in 10 CFR 72.236(b), and (iii) the principal design criteria for the HI-STORM 100 Version UVH are acceptable with regard to meeting the regulatory requirements in 10 CFR Part 72.

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 information provided by the applicant and evaluated the following proposed change that are applicable to the structural review:

PC #1 Add a new UVH overpack which includes high-density concrete for shielding. The UVH overpack is to be used for the MPC-32M and MPC-68M.

The staff evaluates the structural design of the HI-STORM 100 UVH storage system. Structural design features and design criteria are reviewed together with evaluations of the structural analyses performed by the applicant to demonstrate the structural performance of the system under normal, off-normal, and accident conditions.

3.1 Description of the Structures, Systems, and Components

The HI-STORM 100 Version UVH design is almost identical to the HI-STORM 100 design and essentially contains the similar SSCs to provide the same safety functions (except for the absence of integral overpack ventilation). The only three additional SSCs related to the structural design (high-density concrete, overpack ribs, optional redundant port cover lids) are added as described in the proposed FSAR Supplement IV of the application.

The SSCs for the HI-STORM 100 Cask System are listed in table 2.2.6 and briefly summarized in section 3.1 of the FSAR (Holtec, 2021c). The proposed FSAR Supplement IV of the application describes the version UVH overpack (section 1.IV.2), safety evaluations that are unaffected by the new design (table 1.IV.0.1), the components ITS (table 1.IV.1.1), principal design criteria (section 2.IV.0), and the mechanical loadings (section 2.IV.2.1 and table 2.IV.2.1).

The applicant states that there is no change to the MPC in the version UVH from the MPCs (MPC-32M and MPC-68M) which were previously reviewed and approved by the staff to be used in the HI-STORM 100 Cask System, thereby, the descriptions of the confinement vessel boundary, penetrations, welds, and redundant sealing system for the MPCs remain valid. The only minor change to the previous MPCs is an addition of the optional redundant cover plates to the MPCs as described in PC #3. However, the addition of the optional redundant cover plate will not negatively affect or reduce the structural integrity of the MPC because (i) the maximum design weight is unchanged, and (ii) the first plate still maintains its intended functionalities.

3.2 Structural Design Criteria

The applicant stated that the structural design criteria used for the evaluations of the version UVH are identical to the design criteria used for the evaluations of the HI-STORM 100 overpack which were previously reviewed and accepted by the staff. The criteria define, in general, the applicable codes and standards, individual loads as related to environmental conditions and natural phenomenon events, load combinations, and stress allowable values for normal, off-normal, and accident conditions. The staff reviewed the structural design criteria provided in the proposed FSAR chapter 2.IV, “Principal Design Criteria,” of the application and chapters 2 and 3 of the FSAR (Holtec, 2021c) and determines that the structural design criteria are consistent with those of NUREG-2215 (NRC, 2020b) and are acceptable.

3.3 Loads

The applicant stated that the structural design loads used for the evaluations of the version UVH are similar to the design loads used for the evaluations of the HI-STORM 100 overpack which were previously reviewed and accepted by the staff in the FSAR (Holtec, 2021c).

Table 2.2.13 in Subsection 2.2.6, “Loads,” of the FSAR lists the individual loads, which were defined from the design criteria. The proposed FSAR table 2.IV.2.1 of the application provides the mechanical loading cases for the version UVH storage overpack and its contents (MPC-32M and MPC-68M). In addition to the environmental conditions and natural phenomenon events, the loads considered include the dead weight, live load, thermal effects, internal and external pressures, handling load, and loads associate with non-mechanistic tip-over accidents.

3.4 Analytical Approach

The structural performance of the HI-STORM 100 Version UVH under normal, off-normal, and accident conditions was evaluated in Holtec's proprietary report HI-2210241, Revision 0, "Structural Calculation Package for HI-STORM 100 Version UVH System," with structural analyses that supplement the analyses to the FSAR (Holtec, 2021c). The applicant used the commercially available finite element (FE) computer codes (ANSYS and LS-DYNA) for analyses and designs. The use of the FE codes was previously accepted by the staff through the Quality Assurance Program with the NRC under Docket No. 71-0784.

3.5 Weights and Centers of Gravity

The bounding weights of the loaded MPC with the maximum fuel assembly weights and the loaded HI-STORM 100 Version UVH overpack are provided in the proposed FSAR table 3.IV.2.2 of the application. Also, the locations of the center of gravity (CG) are provided in table 3.IV.2.3. Those weights and CGs are used in the structural analyses.

3.6 Positive Closure

A description of the positive closure of the HI-STORM 100 Version UVH system is provided in the proposed FSAR section 3.IV.4.1 of the application. There are no quick-connect/disconnect ports in the confinement boundary of the version UVH system. The only access to the MPC is through the storage overpack lid. The lid is fastened to the storage overpack with large bolts. Inadvertent opening of the storage overpack is not feasible because opening a storage overpack requires mobilization of special tools and heavy load lifting equipment. These design features preclude inadvertent opening, thereby ensuring positive closure of the HI-STORM 100 Version UVH overpack system.

3.7 Lifting Device Analysis

The applicant performed the lift analysis for the HI-STORM 100 Version UVH with the bounding weight of the loaded HI-STORM 100 Version UVH set down in the proposed FSAR table 3.IV.2.2 of the application. The stress analysis of the overpack lid under normal handling conditions was performed using the ANSYS FE computer code. The FE model of the overpack closure lid is shown in the proposed FSAR figure 3.IV.4.1 of the application. The resulting stress distribution in the steel structure of the overpack lid under the applied handling load is shown in the proposed FSAR figure 3.IV.4.2 of the application. The calculated stresses and the corresponding safety factors are summarized in the proposed FSAR table 3.IV.4.1 of the application. When the calculated maximum primary membrane stress and the primary membrane plus bending stress in the lid are compared against the corresponding stress limits per Subsection NF of the ASME Code for Level A condition, the calculated factors of safety were 4.97 and 7.46, respectively, indicating that the computed factors of safety have a large safety margin over the allowable value of 1.0.

Based on a review of the above evaluations, the staff concludes that the version UVH is structurally adequate in meeting the stress-based design factor of safety criteria of NUREG-0612 and ANSI N14.6 (ANSI, 1993).

3.8 Normal and Off-Normal Conditions

The applicant performed structural analysis for the version UVH with the pressures and temperatures listed in tables 1.2.2, 2.2.1 and 2.2.3 of the FSAR and the proposed FSAR table 2.IV.2.3 of the application for the normal and off-normal conditions. The applicant used the ANSYS FE model of the MPC described in section 3.4.3.2 of the FSAR, which was previously reviewed and accepted by the staff. The applicant added the components (i.e., inner and outer shells, ribs, baseplate and the lid) to the previous FE model to make a FE model for the version UVH and performed the structural analysis to demonstrate its structural integrity of the version UVH.

The results of the analysis are presented in the proposed FSAR table 3.IV.4.2 of the application. The calculated stresses of the components are compared with the required ASME Code, NB Level A stress limits. Table 3.IV.4.2 shows that all calculated safety factors are greater than 1.0 indicating that the load-carrying capacity of the structural components is greater than the maximum load combinations applied, and therefore the staff concludes that the version UVH will perform its intended functions under the normal and off-normal load conditions.

3.9 Accident Conditions

3.9.1 Internal Pressure and Temperature

The applicant performed structural analysis for the version UVH under internal pressure and temperature of an accident condition. The accident internal pressure loading from the proposed FSAR table 2.IV.2.3 of the application along with the temperatures obtained from the thermal analysis in the proposed FSAR chapter 4.IV of the application were applied to the MPC lid, shell and baseplate in the ANSYS FE model.

The proposed FSAR figure 3.IV.4.4 of the application shows various loadings applied in the ANSYS FE model. The stresses of the lid, shell and baseplate were calculated, and they were compared with the ASME Code, NB Level D stress limits. The results for this loading condition are presented in the proposed FSAR table 3.IV.4.3 and they are showing that all calculated safety factors are greater than 1.0 indicating that the load-carrying capacity of the structural components is greater than the maximum load combinations applied, and therefore the staff concludes that the version UVH will perform its intended functions under the maximum internal pressure and temperature of an accident condition.

3.9.2 Tornado Wind and Missile Impact

The staff reviewed the results of the applicant’s stability evaluation of the version UVH overpack with respect to overturning and sliding under tornado wind and missile impact of an accident condition. The applied loading is described in section 2.2.3.5 and the loading data is provided in tables 2.2.4 and 2.2.5 of the FSAR (Holtec, 2021c).

<u>Overturning Analysis</u>

The overturning analysis of the version UVH overpack under tornado wind load and large missile impact is performed by solving a single degree of freedom dynamic equation of motion. The applicant made the following assumptions in the analysis: (i) the overpack is assumed to be a rigid solid cylinder, (ii) the angle of incidence of the missile is assumed to be such that its overturning effect on the cask is maximized, (iii) the cask is assumed to pivot about a point at the bottom of the base plate opposite the location of missile impact and the application of wind force to conservatively maximize the propensity for overturning, and (iv) inelastic impact is assumed with the missile velocity reduced to zero after impact representing the maximum amount of angular momentum to the overpack by the missile impact.

The results of the analysis are presented in the proposed FSAR table 3.IV.4.4 of the application. When the calculated angular acceleration in table 3.IV.4.4 was compared with the allowable angular acceleration, a factor of safety of more than 7 was obtained indicating that the overpack remains in a vertical upright position (i.e., no overturning) in the aftermath of a large missile impact. The staff evaluated the applicant’s dynamic overturning analysis and finds it acceptable because (i) the applicant has made conservative assumptions for the analysis, and (ii) there will be no overturning based on the result of calculated angular acceleration with a large safety margin.

Sliding Analysis

The sliding analysis of the UVH overpack under tornado wind load and large missile impact is performed by solving the Newton's equation of motion. The applicant made the following principal assumptions in the analysis: (i) the weight of the overpack used in the analysis is assumed to be the lowest bound; (ii) the overpack is assumed to absorb the energy of impact purely by sliding, so that none of the impact energy is dissipated by the noise from the impact; (iii) the missile impact and high wind are assumed to act synergistically to maximize the movement of the cask; (iv) the overpack is assumed to be freestanding on a concrete surface; and (v) the interface friction coefficient is assumed to be 0.53 as previously accepted by the staff.

The results of the analysis are presented in the proposed FSAR table 3.IV.4.4 of the application. The table shows that the calculated maximum sliding displacement of the overpack is less than 1 foot. When the sliding displacement was compared with the allowable sliding displacement, a safety factor of greater than 5 was obtained. The staff evaluated the applicant’s sliding analysis and finds it acceptable because (i) the applicant has made conservative assumptions for the analysis, (ii) small sliding displacement compared with the large dimension of the version UVH, and (iii) a large safety margin with the calculated factor of safety.

3.9.3 External Pressure Resulting from an Explosion

The applicant performed structural analysis for the UVH overpack under external pressure load due to an explosion per the proposed FSAR table 2.IV.2.1 of the application. A methodology of the ASME Code Case N-284, which was previously accepted by the staff, was used for this analysis. The buckling capacity of the shell structure of the overpack was evaluated using conservative weights and an external pressure of 60 psi. The buckling capacity of the overpack was calculated with a classical strength of materials calculation using the MathCAD program. In the analysis, the applicant assumed that the structural support from the concrete shield rings and the effect of support by radial ribs were conservatively ignored.

The results of the buckling analysis for the version UVH overpack are summarized in the proposed FSAR table 3.IV.4.5 of the application. The results show that the factors of safety are greater than the required 1.0 for axial plus hoop compression and axial compression plus shear for both elastic and inelastic buckling based on the ASME Code Case N-284. The staff reviewed the applicant’s buckling analysis and finds it acceptable because the shell structural component of the version UVH has a load-carrying capacity to resist the buckling load of the external pressure due to an explosion.

3.9.4 Snow Load

The applicant performed structural analysis for the UVH overpack lid under snow load condition using the ANSYS FE computer code. The FE model used for the analysis is essentially the same as shown in the proposed FSAR figure 3.IV.4.1 of the application. The snow pressure of 100 lb/ft^2 was used as provided in table 2.2.8 of the FSAR (Holtec, 2021c). The resulting stress distribution in the steel structure of the overpack lid under the applied snow load is shown in the proposed FSAR figure 3.IV.4.26 of the application. The maximum stresses and the corresponding safety factors are summarized in table 3.IV.4.8. The table shows that both the maximum primary membrane stress and the maximum primary membrane stress plus bending stress have a factor of safety of greater than 1.1. The staff reviewed the applicant's analysis under the snow load and finds it acceptable because the UVH overpack lid under snow load condition has a load-carrying capacity that is larger than the required limit of Subsection NF (class 3 structures) of the ASME Code for Level A condition.

3.9.5 Earthquake

The applicant did not perform structural analysis for the UVH overpack under the design base earthquake load. The applicant stated that no new analysis for the design basis earthquake is warranted because the outer diameter and height of the CG of version UVH overpack are essentially identical to the HI-STORM 100 Cask System as identified in the proposed FSAR table 2.IV.2.1 of the application. The applicant previously performed structural analysis for the HI-STORM 100 Cask System under the design base earthquake load in chapter 3 of the FSAR, which was reviewed and accepted by the staff. The staff concludes that the seismic analysis performed in section 3.4.7 of the FSAR for the HI-STORM 100 Cask system is applicable to the version UVH.

3.9.6 Non-Mechanistic Tip-over

The applicant performed structural analysis of the HI-STORM 100 Version UVH during non-mechanistic tip-over using the LS-DYNA FE computer code as documented in the proposed FSAR section 3.IV.4.3.4 of the application and Holtec's proprietary report HI-2210290, Revision 2, "Analysis of The Non-Mechanistic Tip-over Event of The Loaded HI-STORM 100 Version UVH Storage Cask." The FE model consisted of a half-symmetric 3-D representation of the overpack and loaded MPC impacting the ISFSI pad with an initial rotational velocity of 1.489 rad/s. The model evaluated the MPC-32M and the MPC-68M geometries with two different basket orientations. The applicant stated the ISFSI geometry is consistent with prior licensing models as discussed in section 3.IV.4.3.4 of the FSAR but uses bounding ISFSI pad perimeters including a 5 ksi compressive strength based on its 28-day concrete strength as stated in NUREG-2215. Minimum material properties for the overpack materials, bounding fuel assembly lengths, and heaviest fuel assemblies were considered to induce maximum stresses in the structural members.

The plastic strain results and failure limits for the tip-over analysis are summarized in the proposed FSAR table 3.IV.4.6 for the MPC-32M and table 3.IV.4.7 for the MPC-68M. The overpack lid bolts had plastic strains lower than their failure strain, the overpack lid remained attached to the overpack, and localized damage of the overpack shielding concrete at the top of the cask was minimal to not have a significant loss of shielding. The MPC remained in the overpack, the MPC did not impact the overpack lid, there was no gross plastic deformation of the overpack inner shell that prevents removal of the MPC, and plastic strains for the MPC vessel were much less than the failure strain indicating that the steel confinement was not breached. There was no gross plastic deformation of the fuel basket panels. The maximum local plastic strain of the basket was less than the allowable percentage for material failure, and the calculated non-dimensional basket deformation metric (established for the basket's subcriticality safety function) was much less than the 0.005 acceptance value for both MPC-32M and MPC-68M. The staff evaluated the applicant's non-mechanistic tip-over analysis and corresponding FE models and finds them acceptable because (i) the FE models used acceptable modeling techniques, (ii) plastic strains and deformations were less than their allowable limits, and (iii) the predicted damage to the version UVH overpack and MPC will not affect performance of its confinement, shielding, and subcriticality safety functions or retrievability.

3.10 Evaluation Findings

F 3.1 On the basis of the review of the statements and representations in the application, the staff finds that the application adequately describes the HI-STORM 100 UVH storage system to enable evaluations of its structural performance and effectiveness.

F3.2 The staff finds that the application has met the requirements of 10 CFR 72.236(b). The HI-STORM 100 UVH storage system is designed to accommodate the combined loads of normal and off-normal conditions, accident conditions, and natural phenomena events with an adequate margin of safety. Stresses at various locations of the overpack for various design loads are determined by analysis. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and are within the limits given in applicable codes, standards, and specifications.

4.0 Thermal Evaluation

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

The 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 UVH overpack which includes high-density concrete for shielding. The UVH overpack is to be used for the MPC-32M and MPC-68M.
PC #2 Include the ability to use CFD analysis to evaluate site-specific fire accident scenario.
PC #4 Include the ability to use CFD analysis to evaluate site-specific burial under debris accident scenario.
PC #5 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: MPC-32M and MPC-68M stored in HI-STORM 100 UVH

The applicant proposed a new overpack, HI-STORM 100 version unventilated overpack with a high-density concrete, to store MPC-32M with the maximum heat load of 25 kW (see heat load patterns in FSAR tables 2.IV.1.2 and 2.IV.1.3) and MPC-68M with the maximum heat load of 25 kW (see heat load patterns in FSAR tables 2.IV.1.4 and 2.IV.1.5) in this amendment. The UVH overpack is a simplified version of the HI-STORM 100 Cask System without inlet and outlet air passages, resulting in a complete cessation of ventilation in the space between the cask cavity and the MPC during the system's operation. The applicant described the design of UVH overpack in FSAR section 4.IV.1 and presented it in licensing drawings No. 12233.

The staff reviewed the above information and finds that (1) the description for the UVH overpack to store MPC-32M and MPC-68M with the maximum heat load of 25 kW is appropriate for thermal evaluation and (2) the thermal evaluations of the UVH overpack are consistent with the dimensions provided in the proprietary report HI-2210277, revision 0, "HI-STORM 100 UVH Critical Dimension Report."

The staff finds acceptable the addition of UVH overpack to support loading of MPC-32M and MPC-68M in the HI-STORM 100 Cask System based on staff's evaluation of the proposed heat load patterns in UVH overpack 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 concludes that the HI-STORM FW UVH system does not rely on ventilation passages for its means of cooling.

4.1.1 Heat Load Patterns

As described in the proprietary report HI-2210138, revision 2, "Thermal Evaluations of HI-STORM 100 Version UVH," the applicant evaluated MPC-32M uniform heat load pattern and regionalized heat load patterns 1 thru 5 for screening of bounding heat load pattern. The applicant evaluated MPC-68M uniform heat load pattern and regionalized heat load patterns 1 thru 10 for screening of bounding heat load pattern. Based on the results, the applicant stated that the uniform heat load pattern results in higher fuel cladding and component temperatures and is the most bounding pattern for both MPC-32M and MPC-68M.

The staff reviewed the above information and finds that (1) MPC-32M results in higher peak cladding temperature (PCT) and component temperatures than those for MPC-68M stored in the UVH overpack and (2) the uniform heat load pattern of MPC-32M is the bounding pattern in PCT and component temperatures, and therefore is the bounding pattern for canister pressure.

The applicant proposed to add a minor deviation from the prescribed MPC-68M loading pattern in CoC appendix B, section 2.4.5 and a minor deviation from the prescribed MPC-32M loading pattern in CoC appendix D, section 2.4.5 to allow one slightly thermally-discrepant fuel assembly per quadrant to be loaded, as long as the PCT for the MPC remains below the limits in Spent Fuel Storage and Transportation (SFST)-Interim Staff Guidance (ISG)-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," revision 3 (NRC, 2003). Since the PCT will be below the limits in SFST-ISG-11, there would be negligible impact to the thermal performance of the system. Therefore, the staff determines the proposed minor deviation is acceptable for fuel loading decay heat limits for MPC-32M and MPC-68M loaded in UVH overpack.

4.1.2 Backfill Pressure Limits

The applicant listed the minimum and maximum initial helium backfill pressures for MPCs stored in UVH overpack in FSAR table 4.IV.1.3. The annular gap between MPC and UVH overpack is air-backfilled to atmospheric conditions such that the operating pressure under normal long-term storage conditions is within the limits set forth in FSAR section 2.IV, “Principal Design Criteria.” The limits of initial backfill pressure and annulus gas fill pressure for MPC-32M and MPC-68M at ambient 70°F, stored in UVH overpack, are provided in FSAR table 4.IV.1.3.

The staff reviewed the thermal calculations provided in report HI-2210138 and finds that the proposed initial helium backfill pressure ranges and annulus air fill pressure limits, shown in FSAR table 4.IV.1.3, for MPC-32M and MPC-68M stored in UVH overpack are acceptable because the MPC helium backfill pressure limits are calculated using the NRC accepted methodology and are lower than, and bounded by, those limits for MPCs stored in ventilated overpack with heat loads greater than 25.0 kW under normal long-term storage conditions.

4.1.3 Use of Nitrogen or Argon as Backfill Gas in Annulus

The applicant stated in safety analysis report (SAR) section 4.IV.1.5, “Use of Nitrogen or Argon as HI-STORM 100 UVH Annulus Backfill Medium,” that in addition to air, nitrogen or argon is also used as the backfill medium in annulus (between MPC and overpack). Argon has much lower thermal conductivity than air, and the thermal conductivity of nitrogen is very close to that of air. The applicant performed thermal evaluations using argon in the annulus, with the bounding loading pattern of MPC-32M in a cask array, to demonstrate that all components of the storage system still meet their respective temperature and pressure limits under all conditions of storage. The applicant presented the cask component temperatures and the MPC pressures in FSAR tables 4.IV.4.3 and 4.IV.4.4, respectively.

The staff reviewed the thermophysical properties of argon in FSAR tables 4.2.1 through 4.2.6 and confirmed the thermophysical properties are acceptable for thermal evaluations. The staff also reviewed FSAR tables 4.IV.4.3 and 4.IV.4.4 for MPC-32M with argon in the annulus and confirmed (1) MPC-32M component temperatures are below the service limits under long-term storage listed in FSAR table 2.2.3, (2) the MPC-32M pressures are below their design limits under all conditions of storage in FSAR table 1.II.2.3, and (3) thermal evaluations for using argon in the annulus bound thermal evaluations for using nitrogen and air in the annulus for both MPC-32M and MPC-68M stored in the UVH overpack.

4.1.4 Normal Conditions of Storage

The applicant described the thermal model of the MPCs (MPC-32M and MPC-68M) and the UVH overpack in FSAR section 4.IV.4.1, with the laminar flow simulated in the space between the MPC and the UVH overpack and no wind around the UVH overpack. The applicant used nominal array parameters provided in FSAR table 4.IV.4.2 to evaluate the impact of neighboring casks modeled as hollow cylinders with uniform heat flux applied on the outer surfaces of the cylinder (see FSAR section 4.IV.4.1.1 and figure 4.IV.4-1).

The applicant stated, in FSAR section 4.IV.4.4, that storage of UVH casks in an array with the bounding MPC-32M is adopted as the governing long-term storage scenario as it yields the highest fuel cladding and component temperatures. The applicant presented the maximum fuel and component temperatures and MPC’s pressures in FSAR tables 4.IV.4.3 and 4.IV.4.4, respectively. The applicant compared the PCT in table 4.IV.4.3 (MPC-32M in cask array) to the

PCT in table 4.IV.4.1 (MPC-32M in isolated cask) and concluded that impact of neighboring casks on the cask heat-removal performance is not significant.

The staff reviewed and confirmed that the maximum fuel and component temperatures and MPC's pressures are below the allowable limits in FSAR tables 2.2.3, 2.2.1 (MPC-68M), and 1.II.2.3 (MPC-32M). Therefore, the staff concludes that the impact of neighboring casks on heat-removal performance is not significant.

The applicant stated, in FSAR section 4.IV.4.4.2, that the minimum off-normal ambient temperature condition for UVH overpack and MPC is specified to be -40°F with no insolation and zero decay heat load applied to the stored fuel assemblies. Under this cold condition, every component of the system (MPC-32M fuel basket, MPC-68M fuel basket, and the UVH overpack) would be at -40°F at steady state.

The staff reviewed FSAR section 4.IV.4.4.2 for UVH overpack and MPCs under extreme cold conditions and FSAR section 2.2.2.2 for the MPC and overpack components designed for extreme cold conditions of -40°F. The staff confirmed that the structural steel materials used for MPC-32M and MPC-68M stored in the UVH overpack are sustainable under extreme cold conditions of -40°F.

Clearances to Thermal Interferences

The applicant stated, in FSAR section 4.IV.4.5, that to minimize thermal stresses in load bearing members, the UVH overpack is engineered with internal gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. The applicant evaluated the differential thermal expansions of fuel basket-to-MPC radial gap, fuel basket-to-MPC axial gap, MPC-to-overpack radial gap, and MPC-to-overpack axial gap. FSAR table 4.IV.4.5 shows the initial minimum gaps and their corresponding values during long-term storage conditions under MPC-32M storage, and the applicant concluded there is no risk of restraint to free end expansion in the storage system.

The staff reviewed FSAR section 4.IV.4.5 and table 4.IV.4.5 and finds that the differential thermal expansions of gaps, calculated using the NRC accepted methodology, are less than the cold gaps. Therefore, the free thermal expansion design criteria are satisfied for both MPC-32M and MPC-68M stored in the UVH overpack.

Storage of Damaged Fuel Assemblies and Fuel Debris

The applicant stated, in FSAR section 4.IV.4.6, that storage of damaged fuel (in damaged fuel isolators (DFIs)/damaged fuel containers (DFCs)) and fuel debris (in DFCs only) in MPC-32M and MPC-68M is permitted for the UVH overpack subject to the location and heat load penalty specified in FSAR tables 2.IV.1.9. The applicant noted that when DFCs/DFIs are placed in locations specified in FSAR section 2.IV, it has a small impact on the computed temperatures and pressures.

The staff reviewed FSAR sections 4.II.4, 4.III.4, and 4.IV.4.6 and confirmed the applicant's statement that storage of DFCs/DFIs have a small impact on the cask components temperatures and MPC pressures when placed in locations same as that specified in FSAR section 2.IV. Therefore, the staff finds acceptable that storage of DFIs/DFCs in HI-STORM 100 UVH system is bounded by the evaluations presented in FSAR sections 4.II.4 and 4.III.4 for MPC-32M and MPC-68M, respectively, stored in the UVH overpack.

4.1.5 Short-Term Operations

The applicant stated, in FSAR section 4.5, that the HI-TRAC transfer cask remains unchanged for MPC-32M and MPC-68M in the short-term operations. The applicant noted that maximum heat loads qualified for MPC-32M and MPC-68M in UVH overpack are significantly lower than those qualified for use in the standard HI-STORM 100 overpack, and no separate evaluation is needed for MPCs qualified for UVH overpack and the thermal evaluations are bounding for vacuum drying operations, normal onsite transfer operations, post-loading wet transfer operations, and MPC cooldown and reflood for unloading operations.

The staff reviewed FSAR section 4.5 and confirmed that the previously approved thermal evaluations for MPC-32M in Amendment No. 15 (NRC, 2021) are bounding for vacuum drying operations, normal onsite transfer operations, post-loading wet transfer operations, and MPC cooldown and reflood for unloading operations due to lower decay heat load of 25 kW allowed for MPC-32M and MPC-68M stored in the UVH overpack. Therefore, the staff concludes that no separate evaluation is needed for MPC-32M and MPC-68M in short-term operations.

4.1.6 Off-Normal Conditions

The applicant stated, in FSAR section 4.IV.6.1, that the most bounding fuel temperatures for UVH overpack with MPC-32M and MPC-68M under off-normal conditions of storage are bounded by the corresponding MPC-32M and MPC-68M in HI-STORM 100 evaluations under normal conditions of storage. The maximum temperatures of some of the components, such as MPC baseplate, are higher for UVH overpack when compared to those for standard HI-STORM 100 overpack under off-normal condition. The staff finds acceptable that the difference in temperatures is well within the respective margin-to-limits, and concludes no separate evaluations are required for UVH overpack.

The applicant performed MPC's pressure evaluation for a 10% rods rupture event assuming 100% of the rods fill gases and 30% of fission gases release for UVH overpack, which is consistent with NUREG-2215. The applicant tabulated the MPC-32M pressures under long-term conditions in FSAR table 4.IV.4.4 for a cask in array configuration. The staff notes that the MPC-32M pressure under off-normal condition is below the off-normal design pressure set forth in FSAR table 1.II.2.3.

Based on the evaluations for normal long-term storage of MPC-32M and MPC-68M in the UVH overpack, the staff has reasonable assurance that (1) the bounding correlations of temperatures can be extended to the off-normal conditions and (2) the maximum cavity pressure (see FSAR table 4.IV.4.4) remains below the design pressure specified in FSAR table 1.II.2.3 for off-normal conditions.

4.1.7 Accident Conditions

<u>Fire Accident</u>

The applicant stated, in FSAR section 4.IV.6.2(a), that the thermal analysis for a fire accident was performed for the most bounding loading scenario (MPC-32M) with the bounding heat load pattern (25 kW and uniform loading) and with the effect of neighboring casks included. The applicant gives the maximum fuel cladding and component temperatures and the cavity and annulus pressures in FSAR table 4.IV.6.1.

The staff reviewed the above information and confirmed that the fuel cladding and system component temperatures and the pressures in MPC-32M cavity are below the accident limits specified in FSAR tables 2.2.3 and 1.II.2.3, respectively. The staff also confirmed, based on the bounding correlation, that the fuel cladding/component temperatures and the pressures in MPC-68M cavity are also below the accident limits specified in FSAR tables 2.2.3 and 2.2.1.

Jacket Water Loss Accident

The applicant stated, in FSAR section 4.IV.6.2(b), that the maximum allowable heat load of the MPCs qualified for use in UVH overpack is significantly lower than those qualified for use in the standard HI-STORM 100 version. Therefore, the component temperatures and MPC cavity pressure under jacket water loss accident condition for the MPCs qualified for use in the UVH overpack will be bounded by those presented in FSAR section 4.6.2.2 for jacket water loss.

The staff reviewed FSAR section 4.IV.6.2(b) and finds acceptable that as the MPC-32M fuel temperatures in the HI-TRAC are bounded by evaluation of the fuel temperatures in previously certified transfer cask (FSAR table 4.5.6) in Amendment No. 15 (NRC, 2021), the jacket water loss temperatures are still bounded by the HI-TRAC jacket water loss evaluation in main FSAR section 4.6.2.2. The staff also confirmed that the bounding correlation is applicable to MPC-68M stored in the UVH overpack.

Burial Under Debris

The applicant stated, in FSAR section 4.IV.6.2(d), that the licensing basis model from SAR section 4.II.4 can be used to compute the time limit under any postulated site-specific burial accident. The licensing basis 3D model shall be modified to include the site-specific burial conditions.

The staff finds acceptable the statements made in FSAR section 4.IV.6.2(d), and the thermal model described in FSAR section 4.II.4 for determination of the time limit because the licensing basis thermal model mentioned above was accepted by the NRC. The staff determined that the fuel and component temperatures and the canister pressures of MPC-32M and MPC-68M, stored in the UVH overpack, will be below the allowable temperature and pressure limits within the calculated time limits for burial under debris.

Extreme Hot Environment Temperatures

The applicant stated, in FSAR section 4.IV.6.2(c), that starting from the baseline condition evaluated in section 4.IV.4.4., the temperatures of the HI-STORM 100 Cask System are conservatively assumed to rise by the difference between the extreme and normal ambient temperatures (55°F). The applicant reported the MPC component extreme ambient temperatures and MPC pressure computed in this manner for the UVH overpack in FSAR table 4.IV.6.2.

The staff reviewed FSAR section 4.IV.6.2 and table 4.IV.6.2 and confirmed that the fuel cladding/component temperatures and the pressures in MPC-32M cavity are below the accident limits in FSAR tables 2.2.3 and 1.II.2.3, respectively. The staff also confirmed, based on the bounding correlation under normal conditions of storage that the fuel cladding/component temperatures and the pressures in MPC-68M cavity are also below the accident limits specified in FSAR tables 2.2.3 and 2.2.1.

<u>100% Rods Rupture Accident</u>

The applicant stated, in FSAR section 4.IV.6.2(g), that in accordance with NUREG-2215 (NRC, 2020b), a 100% rods rupture accident is evaluated for UVH overpack assuming 100% of the rods fill gases and 30% of fission gases release. The applicant computed and tabulated the MPC-32M pressures under this postulated accident in FSAR table 4.IV.4.4 for a UVH cask under array configuration. The pressures are below the accident design pressure set forth in supplement 2.IV.

The staff reviewed FSAR section 4.IV.6.2 and table 4.IV.4.4 and confirmed that the pressures in MPC-32M cavity and UVH annulus are below the accident limits in FSAR table 1.II.2.3. The staff also confirmed, based on the bounding correlation under normal conditions of storage, that the pressures in MPC-68M cavity and UVH annulus are also below the accident limits.

4.2 PC #2: Use of CFD Analysis for Site-Specific Fire Accident

The applicant stated, in FSAR section 4.IV.6.2(a), that the UVH overpack contains steel inner shell ribs that transfer heat flux due to fire more efficiently into the MPC, therefore the fire evaluations for UVH overpack are not necessarily bounded by those for standard HI-STORM 100. The applicant performed thermal evaluation using CFD on the most bounding MPC-32M with the bounding heat load pattern and the fire conditions of fuel volume of 50 gallons, fuel consumption rate of 0.15 inches/minute, and fire duration of 3.62 minutes, consistent with those adopted in the main FSAR. The applicant presented the maximum temperature and pressure rise of the MPC internal due to a hypothetical fire event in FSAR table 4.IV.6.1 for the bounding condition of UVH cask stored in an array.

The staff reviewed FSAR table 4.IV.6.1, which used CFD to compute maximum temperature and pressure rise of the MPC internal due to a hypothetical fire event. The staff confirmed that the use of the CFD analysis for the site-specific fire accident scenario can be applicable to all MPCs, including MPC-32M and MPC-68M stored in the UVH overpack, for the HI-STORM 100 Cask System. The CFD analysis would be applicable if the assumptions, parameters, methodology, and boundary conditions used in the CFD analysis follow the principal modeling steps and acceptance criteria in FSAR section 4.IV.6.2(a) and are consistent with the site-specific fire accident conditions.

4.3 PC #4: Use of CFD Analysis For Site-Specific Burial

The applicant stated, in FSAR section 4.6.2.5, "Burial under Debris," that the licensing basis model can be used to compute the time limit under any postulated site-specific burial accident. The user should modify the licensing basis 3D model to include the site-specific burial conditions to compute the burial time for the respective MPC heat load such that all cask component temperature and canister pressure limits are satisfied.

The applicant stated, in FSAR section 4.IV.6.2(d), that a bounding thermal analysis of the most bounding MPC-32M was performed for the postulated site-specific burial conditions with assumptions shown in FSAR table 4.IV.6.3. The applicant presented the calculated burial time and the pressures of MPC-32M cavity and UVH annulus in FSAR table 4.IV.6.3.

The staff reviewed FSAR section 4.6.2.5 and table 4.IV.6.3 and finds the use of the CFD analysis for the site-specific burial scenario for all MPCs authorized for use in HI-STORM 100

Cask System acceptable, including MPC-32M and MPC-68M stored in the UVH overpack. This is because the methodology used in the CFD model is consistent with the methodology in the calculation method that was accepted by the NRC.

4.4 PC #5: Use of Water Without Glycol In Water Jacket

Per FSAR chapter 8, ethylene glycol is added to the water jacket if the normal onsite transfer operations are performed under ambient temperatures below 32°F. In FSAR section 4.5.7, the applicant proposed that this requirement does not apply if the MPC heat load is above a minimum value at which freezing of the water in the water jacket is of no concern. The applicant performed thermal analysis with uniform heat load of 10 kW at 0°F ambient temperature, as described in appendix N.5.19 of the Report HI-2043317, revision 47. The applicant calculated a minimum water temperature at a heat load of 10 kW. The result shows that if the heat load is above a certain limit and the temperature of water in the water jacket remains above freezing point of water, the system of MPC-32 and MPC-68 (including MPC-32M and MPC-68M) in HI-TRAC do not need to be filled with ethylene glycol under allowable operating short-term conditions.

The applicant stated, in FSAR sections 8.1.5, “MPC Closure,” and 8.3.2, “HI-STORM Recovery from Storage,” that if the transfer cask is expected to be operated in an environment below 32°F, and a minimum heat load requirement was not applied to loading the MPC, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, users need to perform thermal evaluation, using the site-specific heat loads and ambient conditions and the model/methodology consistent with that presented in FSAR section 4.5.1, 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.

The staff reviewed FSAR sections 4.5.7, 8.1.5 and 8.3.2 and finds the addition of ethylene glycol in the water jacket is not required if the MPC heat load is high enough to preclude freezing of the water and this threshold MPC heat load is determined on a site-specific basis through methodology described in FSAR section 4.5.7. Based on the parameters used by the applicant for the thermal analysis, the staff also determined that the handling operations for loaded HI-TRAC transfer cask should be limited to a working area with ambient temperature greater than or equal to 0°F and MPC decay heat greater than 10 kW.

4.5 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 overpack 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-32M and MPC-68M, stored within the UVH overpack, 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-32M and MPC-68M, stored in the UVH overpack, 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-32M and MPC-68M, stored in UVH overpack, can sustain the pressures predicted under normal, off-normal, and accident-level conditions. The maximum MPC-32M canister pressures are below the design pressures of 110, 120, and 225 psig for normal, off-normal, and accident conditions of storage, respectively. The maximum MPC-68M 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 and site-specific burial accident is acceptable.

F4.6 The staff has reasonable assurance that the use of water without glycol in water jacket is acceptable. 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.

F4.7 The staff concludes that the thermal design of UVH overpack, containing MPC-32M and MPC-68M 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 evaluations of the thermal design provide reasonable assurance that the UVH overpack will allow safe storage of spent fuel during the license period.

5.0 SHIELDING EVALUATION

The objective of this shielding review is to evaluate whether the shielding features of the HI-STORM 100 Cask System design, as amended, will continue to provide reasonable assurance of adequate protection to workers and the public from radiation from the proposed contents. The staff reviewed the applicant's safety analyses for the requested changes to the CoC following the guidance provided in NUREG-2215.

The staff identified the following changes to have potential impact on the shielding performance of the HI-STORM 100 Cask System:

- PC #1 - Add a new overpack, called the HI-STORM 100 UVH.
- PC #3 - Modify vent and drain penetrations.
- PC #5 - Remove glycol from the HI-TRAC water jacket at temperatures below 32°F.
- PC #7 - Replace FQTs with sets of burnup and cooling time limits and reduce minimum cooling time from 2 years to 1 year.
- PC #8 – Add revised shielding assumption related to overpack concrete specifications.

5.1 PC #1: Addition of UVH Overpack

The applicant proposed the addition of a new variant of the HI-STORM 100 overpack, called the HI-STORM 100 UVH, limited for use with the MPC-32M and MPC-68M canisters. This new overpack is unvented and does not have inlet and outlet air passages. To compensate for the removal of convective cooling provided by the vents, the applicant has proposed a reduction in the allowable MPC heat load and an increase in the thermal conductivity with the UVH overpack. To increase the thermal conductivity of the UVH overpack, the applicant proposed using full depth rib plates between the overpack inner and outer shells. In addition, the applicant proposed the use of higher density concrete with higher thermal conductivity and enhanced shielding performance. The applicant has not proposed any changes to the design of MPC 32M and MPC-68M canisters. Since those canisters have been evaluated with the HI-STORM 100 Cask System by NRC staff in prior amendments, staff's review focuses on the changes in the spent nuclear fuel and associated hardware specifications for those canisters stored with the HI-STORM UVH System that have not been previously evaluated.

The HI-TRAC MS is to be used to transfer the canisters stored with the HI-STORM UVH. The applicant has not made any changes to the design of this transfer cask but has re-evaluated dose rates with the proposed spent nuclear fuel characteristics and associated hardware specifications for the HI-STORM UVH.

5.1.1 Radiation Source Definition

The allowable fuel for the HI-STORM UVH includes all the damaged and undamaged spent fuel assemblies along with fuel debris and non-fuel hardware.

The damaged fuel assemblies are allowed in only peripheral locations of the MPC-32M and MPC-68M shown in table 2.IV.1.9 and figures 2.IV.1.2 and 2.IV.1.3 of the application.

5.1.1.1 Spent Nuclear Fuel

For spent nuclear fuel, maximum burnup and minimum cooling time limits were proposed for the HI-STORM UVH in terms of coefficients used for a polynomial correlation that defines minimum cooling time based on maximum allowable burnup values. Although the correlation coefficients are new with this amendment and specific to the HI-STORM UVH and its associated transfer cask (HI-TRAC MS), the strategy for using this type of correlation to define these values was used and accepted by the NRC for the HI-STORM 100 Amendment No. 15. This correlation and the associated coefficients are defined in section 2.IV.1.2 of the application and are translated to loading limits within the technical specifications in table 2.1-4 of the proposed appendix D to the CoC for the MPC-32M and table 2.4-9 of the proposed appendix B to the CoC for the MPC-68M. There are separate correlations for fuel loaded in interior basket locations versus outer basket locations as specified in figure 2.IV.1.1 of the application for the MPC-32M and figure 2.IV.1.2 for the MPC-68M.

To justify that spent nuclear fuel meeting the limits specified by the correlations would meet regulatory requirements related to limiting dose, Holtec provided demonstration calculations using source terms derived from these correlations. The applicant documented the burnup and cooling times evaluated in the proprietary report HI-2201113, revision 1, "HI-STORM 100 Version UVH Shielding Analysis," for burnup from 5,000 to 70,000 MWd/MTU. The applicant evaluated the maximum dose rates for a set of the bounding burnup, enrichment, and cooling time. The results show the dose rates are lower than HI-STORM 100S Version B and Version E due to higher density of concrete and not having the vent ducts in version UVH. The staff finds that the cooling times in these calculations are less than the minimum cooling times required by the loading curves and thus using a lower cooling time is a conservative assumption.

5.1.1.2 Fuel and Non-fuel Hardware

For the source term from non-fuel hardware components, which are only allowed in PWR spent nuclear fuel assemblies, the applicant referenced section 5.II.2.3 of the FSAR which was reviewed and approved during the review of HI-STORM 100 Amendment No. 15 (NRC, 2021) and is not changed and remains valid for the MPC-32M analyses within the HI-STORM 100 UVH.

The method for generating source term from fuel hardware (e.g., top nozzle, end fittings, etc.) was not discussed in the FSAR, but was discussed in the calculation file, HI-2201113, revision 1, for the HI-STORM 100 UVH. The method is consistent with previously approved methods used in the HI-STORM 100 calculations and staff therefore finds it acceptable for this amendment.

5.1.2 Shielding Model Specification

The applicant developed computational models of the HI-STORM 100 UVH and the HI-TRAC MS containing the MPC-32M and MPC-68M. These are documented in report HI-2201113, revision 1.

The applicant did not model damaged fuel within the HI-STORM 100 UVH. Damaged fuel can result in localized increases in dose rate should a fuel assembly fail, such that the locations of higher source term of a fuel assembly could be relocated to parts of a cask where there is less shielding. Damaged fuel was authorized within the MPC-32M in the HI-TRAC MS as part of Amendment No. 15 (NRC, 2021). The opening of a vent has less shielding and would result in higher dose rates. Because the HI-STORM 100 UVH has higher density concrete and no vents, the staff finds that damaged fuel is also acceptable for the HI-STORM 100 UVH and HI-TRAC MS.

5.1.3 Shielding Analyses

The shielding analyses for the HI-STORM 100 UVH and HI-TRAC MS with the MPC-32M and MPC-68M canisters are summarized in chapter 5 of Supplement IV to the application.

The applicant performed a shielding evaluation for HI-STORM 100 UVH with the MPC-32M and MPC-68M with uniform and regionalized loadings described in the proposed FSAR section 2.IV.1 of the application.

5.1.3.1 HI-TRAC MS Under Normal and Accident Conditions

The HI-TRAC MS is the transfer cask that will be used to transfer loaded canisters from the spent fuel pool to the HI-STORM 100 UVH overpack. The HI-TRAC MS contains a variable thickness lead shield. The most conservative modeling of this component would include the limiting spent nuclear fuel specifications allowed from the correlations in section 2.IV.1.2 of the application with the minimum allowable thickness of the lead shield. The applicant has chosen to implement technical specification dose rate limits for the HI-TRAC MS as part of the strategy to limit occupational exposure. For example, using the minimum cooling time derived from correlations in section 2.IV.1.2 of the application for a given maximum burnup to perform a dose rate evaluation with the minimum thickness for the HI-TRAC MS would exceed technical specification dose rate limits. The allowable loading needed to meet technical specification limits will be a combination of contents with increased cooling times beyond the correlation limits and increased lead thickness that would allow a HI-TRAC MS to meet its technical specification dose rate limits. This approach was used in Amendment No. 15 of the HI-STORM 100 and was reviewed and approved by the NRC.

For the normal condition dose rate calculation results presented in section 5.IV.1 of the application, the HI-TRAC MS was modeled with maximum burnup and minimum cooling time limits consistent with the correlations in section 2.IV.1.2 of the application and an amount of lead needed to meet the technical specification dose rate limits. It is expected that less lead would need higher cooling times and the dose from HI-TRAC MS versions with more lead would be bounded by the FSAR (NRC, 2021c) calculations. The applicant performed additional calculations, such as a scenario with minimum lead shielding and additional cooling time than specified by the correlations in section 2.IV.1.2 of the application, with results documented in section 5.IV.4 of the application with other source term and lead thickness. The staff finds that results show that the HI-TRAC MS with minimum lead thickness and the additional cooling time meets the technical specification dose rate limit.

5.1.3.2 Computer Codes

The applicant evaluated the source term using the TRITON and ORIGAMI modules of SCALE 6.2.1, as described in section 5.IV of the application. This is the same code and methodology used to develop the spent nuclear fuel source term as described in supplement 5.II of the FSAR which was reviewed and approved by staff for the HI-STORM 100 Amendment No. 15 and the code remains acceptable for analyzing spent nuclear fuel source terms for the HI-STORM 100 UVH and HI-TRAC MS in this amendment.

The applicant analyzed the dose rates for the HI-STORM 100 UVH and HI-TRAC MS using MCNP5-1.51 code. MCNP5-1.51 was used to calculate the dose rates for the HI-STORM 100 in section 5.II and 5.III and have been previously reviewed and approved by the NRC and the staff finds that this version is acceptable for the HI-STORM 100 UVH and HI-TRAC MS in this amendment.

5.1.3.3 Dose Rates

The results of the bounding dose rates for the HI-STORM 100 UVH and the HI-TRAC MS for normal and accident conditions are summarized in tables 5.IV.1.1 through 5.IV.1.4 of the application. The applicant has demonstrated in section 5.IV.4 of the application that MPC-32M has bounding dose rates over MPC-68M. The results for the HI-STORM 100 UVH in table 5.IV.1.2 of the application shows that for various example array sizes at specified distances, the HI-STORM 100 UVH meets the regulatory dose rate limit in 10 CFR 72.104 at the controlled area boundary. The results in table 5.IV.1-4 in the application shows that under accident conditions, the HI-TRAC MS meets the dose limit in 10 CFR 72.106. The loss of neutron shielding for the transfer cask was determined to be the limiting accident for the HI-STORM 100 Cask System.

For accident conditions, the applicant used the minimum shielding thickness, bounding source term, and complete loss of water in the water jacket. This represents the most conversative condition and the staff finds it acceptable.

Staff finds that Holtec's analyses show that HI-STORM 100 UVH storage cask, the HI-TRAC MS transfer cask with the MPC-32M and MPC-68M canisters, can meet the 10 CFR 72.104 limits and support as low as reasonably achievable (ALARA) practices. Based on the bounding conditions used in the accident analyses, the staff finds that the requirements from 10 CFR 72.106 will be met at 100 meters from the ISFSI.

5.1.4 Evaluation

The staff verified the applicant's code input in the calculation packages and confirmed that the proper material properties and boundary conditions were used. The staff also checked the engineering drawings to verify that proper geometry dimensions were translated to the analysis model. The staff reviewed the material properties stated in the FSAR to verify that they were appropriately referenced.

The staff reviewed the results of the design basis dose rate calculations for normal conditions for the UVH and HI-TRAC MS in tables 5.IV.4.1, 4.2. and 4.3. of the application. Based on the comparisons in application tables 5.IV.4.2 and 5.IV.4.3, MPC-32M would produce higher dose rates than the MPC-68M; therefore, the staff finds using the MPC-32M to calculate the dose rates shown in table 5.IV.4.1 is acceptable.

For the HI-TRAC MS, the shielding thickness was adjusted resulting in a dose rate that meets the dose rate limits in the technical specification (TS) section 5.7.4 of appendix A and section 5.3.4 of appendix C. Therefore, the staff finds the dose rates reported in the proposed FSAR table 5.IV.4.1 of the application to be appropriately representative of the design under normal conditions.

Consideration for all these factors provides the staff with reasonable assurance that the applicant has demonstrated that the HI-STORM UVH and HI-TRAC MS meet regulatory requirements with the MPC-32M or MPC-68M canister. Therefore, the staff finds that the addition of the HI-STORM 100 UVH and changes to the associated spent nuclear fuel and hardware specifications within the HI-TRAC MS meet the requirement of 10 CFR 72.236(d).

Since the transfer cask is used for short-term loading operations, the applicant also uses HI-TRAC MS dose rate calculations to determine if occupational dose limits in 10 CFR 20.1201 can be met. The staff's assessment of this evaluation will be discussed in the section 10 of this SER.

5.2 PC #3: Modify Vent and Drain Penetrations

The applicant proposed the addition of an optional second cover plate for the vent and drain penetrations. Based on a review of Drawing 11572 and a review of the information on the proposed design change for MPC lid port covers, the alternate configuration would have more shielding and, if anything, likely reduce dose rates. The staff finds that the proposed changes to the vent and drain penetrations would have no effect on the system's ability to meet regulations in terms of shielding and limiting dose.

5.3 PC #5: Remove Glycol from The HI-TRAC Water Jacket At Temperatures Below 32°F

The applicant proposed to remove the requirement to use glycol in the HI-TRAC water jacket at temperatures below 32°F. The purpose of the glycol is to prevent the water in the jacket from freezing, potentially rupturing the jacket and resulting in the loss of shielding provided by the water jacket should the temperature rise above 32°F. The staff reviewed the applicant's model and verified within the provided MCNP input files that the HI-TRAC water jacket is modeled as lower density water. This is conservative compared to nominal density water because it would provide less shielding. Therefore, switching from glycol to water is currently bounded by the shielding evaluation, and the staff finds that this proposed change would have no effect on the system's ability to meet regulations in terms of shielding and limiting dose.

5.4 PC #7: Replace FQTs With Sets of Burnup and Cooling Time Limits and Reduce Minimum Cooling Time from 2 Years to 1 Year

The applicant proposed to remove the current fuel qualification from FSAR chapter 2 and the CoC TS for the MPC-24/24E/24F, MPC-32/32F, and MPC-68/68FF/68M when stored in the ventilated overpack including the 100U. Previously, these limits were in terms of tables of correlation coefficients for a 7-coefficient polynomial correlation. The correlation calculated maximum allowable burnup as a function of enrichment and maximum allowable decay heat with correlation coefficients specified in TS loading tables. The applicant proposed to replace these tables of correlation coefficients with a table of allowable maximum burnup and minimum cooling time combinations. Enrichment has been removed from the correlation and treated in a bounding way within the dose rate calculations based on actual discharge data, which is consistent with an approach taken for the approval of the MPC-32M in HI-STORM 100 Amendment No. 15 (NRC, 2021) and therefore staff finds it acceptable for this amendment. Decay heat is no longer a part of the correlation and is now independent from the burnup/cooling time combinations. The applicant also proposed to reduce the minimum cooling time for PWR fuel from 2 to 1 year.

5.4.1 Radiation Source Definition

5.4.1.1 Spent Nuclear Fuel

In the application, table 2.1.28 of the FSAR specifies the maximum burnup and minimum cooling time limits for PWR fuel for the ventilated overpacks. Table 2.1.29 of the FSAR specifies these limits for BWR fuel for ventilated overpacks. These limits are reflected in the proposed section 2.4.6 of the technical specifications in appendix B to the CoC.

For the HI-STORM 100U and reduction of cooling times, the applicant states in FSAR section 2.I.0 of the application that the limits in proposed FSAR table 2.1.28 and 2.1.29 are applicable to the HI-STORM 100U except that the cooling time cannot be less than 3 years consistent with chapter 5.I. These limits are reflected in the proposed table 2.4-3 of the technical specifications for the HI-STORM 100U in appendix B-100U.

For the HI-STORM 100U and the new burnup and cooling time limits, the applicant did not perform new dose rate calculations. The staff reviewed the burnup and cooling time assumed for the HI-STORM 100U dose rate calculations in chapter 5.I. Table 5.I.1 of the FSAR shows that the applicant used burnup and cooling time combinations consistent with the MPC-32 burnup and cooing time limits for the HI-STORM 100U in the proposed table 2.1.29 of the FSAR. The staff finds this acceptable and representative of the MPC-24 and MPC-68 baskets as well as the dose rate discussed in section 5.4.3 of this SER.

5.4.1.2 Fuel and Non-Fuel Hardware

The applicant includes Co-60 source term for fuel and non-fuel hardware (for PWR fuel). The applicant did not change the method of generating these source terms from previously approved amendments, and the method remains acceptable for this amendment.

5.4.2 Shielding Model Specification

There are no changes to the design of the canisters or storage or transfer cask overpacks from previously approved amendments. The applicant modeled the HI-STORM 100S Version B to represent all of the ventilated above ground HI-STORM 100 overpacks authorized with the MPC-24/24E/24F, MPC-32/32F, and MPC-68/68FF/68M canisters. The applicant stated that this overpack produces the highest off-site dose rates and the staff finds that these would adequately represent the highest dose rate for the other HI-STORM 100 overpacks.

The HI-TRAC family of transfer casks will be used to transfer the loaded canisters from the spent fuel pool to the HI-STORM 100 overpacks. There are several different variants of the HI-TRAC that are allowed for use with the MPC-24/24E/24F, MPC-32/32F, and MPC-68/68FF/68M canisters. The 100-ton HI-TRAC has the least shielding and the highest dose rates, and the applicant modeled it for the proposed change. The staff finds the use of 100-ton HI-TRAC to represent other transfer casks acceptable because it provides a bounding analysis.

The applicant summarized the shielding analyses for the HI-STORM 100S Version B to support the proposed burnup and cooling time combinations in the proposed section 5.4.11 of the FSAR with additional calculation details in appendix NN of the proprietary report, HI-2012702 Revision 19, “HI-STORM 100 System Additional Shielding Calculations.” These evaluations include calculated dose rates for three canister sizes, MPC-24, MPC-32, and the MPC-68 (including MPC-68M) at the surface, 1 meter, and distances beyond the controlled area boundary for a single cask and example array sizes. The variants of the MPC-24 and MPC-32 are similar enough that the differences do not have an appreciable effect on dose rates; therefore, the staff finds the selection of the MPC-24 and the MPC-32 to be acceptable to represent other variants. For the MPC-68, the MPC-68M is a Metamic (aluminum based) basket which may provide less shielding than the stainless steel basket, so the applicant calculated dose rates using both MPC-68 and the MPC-68M baskets. Neither basket proved to be bounding over the other, in that some locations had higher dose in one basket, while other locations were higher in the other basket. The highest dose rates reported in the FSAR for the MPC-68 are from the highest dose rate result from all the MPC-68 and MPC-68M calculations. The staff finds this conservative and therefore acceptable.

5.4.2.1 Accident Conditions

For accident conditions, as described in FSAR table 5.1.10, the applicant assumed 75,000 MWd/MTU and 5-years cooling. This is more bounding than any of the burnup/cooling time combinations proposed in this amendment because the 75,000 MWd/MTU is as high as or higher than any other allowed burnup, and the 5-year cooling time is equal to or shorter than any other cooling time allowed for this burnup. Considering the accident of HI-TRAC with the loss of neutron shield has the highest dose rate, the assumption of the highest burnup at 75,000 MWd/MTU is conservative because neutron dose increases proportional to the burnup raised to the fourth power. Therefore, the staff finds the previously approved accident analysis applicable to the new burnup/cooling time combinations.

5.4.2.3 Computer Codes

The applicant evaluated the source term for the changes to the allowable burnup and cooling time combinations using the TRITON and ORGAMI sequences in SCALE 6.2.1, as described in the proposed FSAR section 5.2. This is the same code and methodology used to develop the spent nuclear fuel source term described in FSAR supplement 5.II and the proposed chapter 5.IV, discussed earlier in this SER. The code was reviewed and approved by staff and documented in the SER for the HI-STORM 100 Amendment No. 15 (NRC, 2021) and remains acceptable for analyzing spent nuclear fuel source terms for the ventilated HI-STORM 100 overpacks.

The applicant analyzed the dose rates for the HI-STORM 100S Version B overpack and the 100-ton HI-TRAC with the new burnup and cooling time combinations using MCNP5-1.51 code. This is an updated version of the MCNP-4A code used with previously approved spent nuclear fuel specifications (e.g., burnup and cooling time). The applicant performed a comparison to show that the results using MCNP5-1.51 are consistent with using MCNP-4A. MCNP5-1.51 is the same code and version used for the HI-STORM 100 UVH overpack discussed earlier in this SER, and the staff finds it acceptable for the dose rate calculations for the HI-STORM 100S Version B and HI-TRAC with the MPC-24/24E/24F, MPC-32/32F, and MPC-68/68FF/68M canisters with the new burnup and cooling time combinations.

5.4.3 Dose Rates

The results of the bounding dose rates for the HI-STORM 100S Version B with the bounding source terms derived from the proposed burnup and cooling time tables are shown in FSAR table 5.4.22 at the cask surface and 1 meter from the overpack. In the application, the applicant provided dose rates to a real individual beyond the controlled area boundary in table 5.4.23 of the FSAR for the three canister sizes (MPC-24, MPC-32, and MPC-68) and for a single cask and several example array configurations. These calculations show the distance at which the dose is reduced below 25 mrem/year, the limit in 10 CFR 72.104(a) for the whole body dose. Thus, the staff concludes that the shielding is sufficient to reduce the dose rate to these levels for the example array sizes and distances and meets the regulation in 10 CFR 72.236(d).

The dose rates reported in the proposed FSAR table 5.4.22 are higher than the current TS dose rate limit of 300 mrem/hr for the side of the overpack in section 5.7.4 of appendix A of the CoC. This means that the end user will have to adjust the actual loading to reduce the design basis source terms to maintain the dose rates to be below the TS limits. The staff finds the reported dose rates in the proposed FSAR are conservative and acceptable. The dose rates reported in the proposed FSAR table 5.4.22 are relatively equivalent to the current TS limit of 30 mrem/hr for the top of the overpack. The staff finds that the TS limits remain appropriate or conservative to represent the ventilated above ground HI-STORM 100 overpacks with the proposed burnup and cooling time combinations.

FSAR table 5.I.2 shows that the dose at 100 meters from the HI-STORM 100U with the MPC-32 is 8.5 mrem/year. The FSAR did not include dose rate information for the HI-STORM 100U with the MPC-24 or MPC-68 baskets. The dose rate for the HI-STORM 100U is lower than the HI-STORM 100S Version B at 16.51 mrem/year at 400 meters (proposed table 5.4.23 of the FSAR). The dose rates at the top of the cask are similar to those for the above ground HI-STORM 100S Version B for all three basket sizes (proposed FSAR table 5.4.22) which are bounded by the MPC-32. Since the dose rate is lower than the limit of 25 mrem/year in 10 CFR 72.104(a) at a much closer distance, the staff has reasonable assurance that the dose rates with the new burnup/cooling times for the MPC-24 and MPC-68 baskets will also meet regulatory requirements in 10 CFR 72.104(a).

Dose rates for the 100-ton HI-TRAC transfer cask are shown in the proposed table 5.4.21 of the FSAR. These dose rates are used to inform occupational exposure and ALARA practices for onsite operations and are discussed in section 10 of this SER.

5.4.4 Staff's Confirmatory Calculations

The staff independently evaluated several of the applicant's allowable burnup and cooling time combinations from FSAR tables 2.1.28 and 2.1.29 of the application using enrichment from table NN-3 of the proprietary report HI-2012702, Revision 19, using the STANDARDS Spent Nuclear Fuel Data and Analysis Tool (formerly known as UNF-ST&DARDS) code. The STANDARDS code employs the SCALE/MAVRIC code to calculate dose rates and SCALE/ORIGAMI code to calculate the neutron and gamma source term from fuel and activated hardware (Radulescu, 2017).

The staff's calculations assume the same design basis fuel assembly as the applicant as specified in table 5.2.1 of the FSAR: B&W 15x15 for PWR fuel within the MPC-24 and MPC-32 and GE 7x7 for BWR fuel within the MPC-68.

Non-fuel hardware is allowed to be stored in PWR assemblies per TS in CoC appendix B tables 2.1-1. Similar to the applicant's calculations, the staff modeled Co-60 resulting from the non-fuel hardware as a combination of the burnable poison rod assembly and thimble plug device as described in the FSAR section 5.2.4.1 and assumed the same Co-60 levels for each region of the fuel assembly (e.g., top nozzle, active fuel, etc.) from table 5.2.31 of the FSAR. The staff's calculations also assumed the guide tubes are filled with a homogenized mixture consistent with the B&W BPR design (DOE, 1992).

The staff modeled the HI-STORM 100S-E rather than the 100S-B modeled by the applicant because the modeling for 100S-E was readily available to the staff. The 100S-B is the same as the 100S-E, except 100S-E has a higher concrete density and different vent design (NRC, 2021). The staff reduced the concrete density to match that of the HI-STORM 100S-B to get a better comparison to the applicant's calculations.

The staff performed dose rate calculations for the MPC-24, MPC-32, and MPC-68M. The staff used MAVRIC model to calculate MPC-32 by modeling MPC-32M and changed the basket material to stainless steel.

The dose rate reported in the application FSAR table 5.4.22 is the highest dose rate at each location from every location for every evaluated burnup/cooling time combination. That is, the applicant calculated the dose rate at all of the specified locations around the HI-STORM 100S Version B using each burnup/cooling time combination, and the highest dose rate from the top may be from a calculation using a different burnup/cooling time combination than the highest dose rate from the side. The staff performed the calculations using the burnup/cooling time combinations that produce the highest dose rates at each location as identified in the proprietary report HI-2012702, Revision 19, appendix NN.

The dose rates for different vent design between the HI-STORM 100S-B and 100S-E could not be compared directly. However, the staff compared the surface and 1-meter calculated dose rates. Results of the dose rate calculations on the side and the top of the overpack compared reasonably well with the applicant's calculations. This provides the staff additional assurance that the proposed source terms result in dose rates consistent with what the applicant has reported and that HI-STORM 100 with these source terms meet regulatory dose rate requirements.

Based on the above evaluation and staff's independent calculations, the staff finds the proposed burnup and cooling time combinations acceptable. The minimum cooling time is now governed by these burnup and cooling time combinations (tables 2.4-3 and 2.4-4 in both appendices B and B-100U). According to the tables, the minimum cooling time for the above ground system could be as low as 1 year for both PWR and BWR fuels. Therefore, it is acceptable to reduce the minimum cooling time for PWR fuel from 2 to 1 year so long as the maximum burnup time allows.

5.5 PC #8: Add Revised Shielding Assumption Related to Overpack Concrete Specifications

The applicant proposed changes to FSAR appendix 1.D which provides specifications for the concrete for all HI-STORM 100 overpack versions. The concrete provides the primary neutron and gamma shielding for the storage overpacks. The appendix provides requirements to ensure the concrete continues to maintain its shielding function including design and testing requirements and adherence to standards. To maintain the shielding function, the density of the concrete must remain at or above the specified levels. The staff verified that the density of concrete listed in the proposed table 1.D.1 of the FSAR is at or above the density used in the shielding evaluations or in FSAR section 5.6 (references for chapter 5) to ensure that the acceptance criteria are consistent with the shielding evaluations. Therefore, the staff finds the proposed changes to FSAR appendix 1.D acceptable.

5.6 Evaluation Findings

Based upon its review, the staff has reasonable assurance that the design of the shielding system of the HI-STORM 100 Cask System, Amendment No. 16 complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system design provides reasonable assurance that the HI-STORM 100 Cask System, Amendment No. 16 will allow safe storage of spent fuel in accordance with 10 CFR 72.236(d). 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 FSAR provides specifications of the spent fuel contents to be stored in the HI-STORM 100 Cask System in sufficient detail to adequately define the allowed contents and allow evaluation of the DSS shielding design for the proposed contents. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed contents' specifications. Thus, the applicant has satisfied the requirements in 10 CFR 72.236(a).

F5.2 The FSAR describes the SSCs ITS that are relied on for shielding in sufficient detail to allow evaluation of their effectiveness for the proposed term of storage. Thus, the applicant satisfies the requirements in 10 CFR 72.236(b) and 10 CFR 72.236(g).

F5.3 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of the HI-STORM 100 Cask System's compliance with 10 CFR 72.236(d). This evaluation is described in the radiation protection review in SER chapter 11. Therefore, the applicant satisfies the requirements in 72.236(d).

F5.4 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of consideration of ALARA in the HI-STORM 100 Cask System's design and evaluation of occupational doses. This evaluation is described in the radiation protection review in SER chapter 10.

6.0 CRITICALITY EVALUATION

In reviewing these changes on the HI-STORM 100 Cask System's criticality capability, the staff followed the guidance in chapter 7 of NUREG-2215 (NRC, 2020b). The staff reviewed PC #1 which may be applicable to criticality review. The applicant proposed to add a new variant of the HI-STORM 100 overpack, called HI-STORM 100 UVH, where the "UVH" stands for "unventilated" with "high-density" concrete. The applicant asserted that there is no change to the criticality evaluation because the MPCs are unchanged from the MPCs previously approved MPC-68M in FSAR supplement III and MPC-32M in FSAR supplement II and no new or changed fuel is introduced. Components external to the canister (such as the overpack) have very little effect on k-eff and the criticality evaluation is bounded by the flooded condition inside the transfer cask in the spent fuel pool. Therefore, the staff finds it acceptable that the applicant did not update the criticality evaluation for the UVH overpack and that the previous SER for CoC No. 1014, Amendments No. 0 through 15 documenting the staff's evaluation are applicable to this amendment.

7.0 CONFINEMENT EVALUATION

The objective of this review is to ensure that the confinement for the HI-STORM 100 Amendment No. 16 meets the regulatory requirements for containment performance found in 10 CFR Part 72 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.

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

- PC #3: Modify the vent and drain penetrations to include the option of a second port cover plate.

- PC #6: 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 #3: 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 change was to remove the need to do field helium leak testing of these cover plates. The applicant provided justification in the 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.

Staff issued request for supplemental information (RSI) and requests 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 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 a response with a confinement evaluation on the redundant port cover plates will take a significant amount of time; therefore, the applicant rescinded the request and revised its application 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 #6: 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 proposed change to change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only and to remove the post hydrostatic test liquid PT examination. The applicant's rationale for this change was that the post hydrostatic pressure test PT examination is not an ASME Code requirement and causes incurred dose without corresponding safety benefit. This proposed change is described in FSAR chapter 9.

In response to staff questions, the applicant shared information about the operating experience with 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 Cask 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 described in the HI-STORM 100 CoC and FSAR chapters 8 and 9, 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 does not exceed the leak rate values stated in ANSI N14.5, 2014 (ANSI, 2014). In addition, staff finds the removal of the post hydrostatic test liquid PT examination acceptable provided that the HI-

STORM 100 Cask System does not exceed the dose limits mentioned in 10 CFR 72.104 and 10 CFR 72.106. Note that testing to other parts of the confinement boundary remain unchanged.

7.3 Findings

The HI-STORM 100 Cask System Amendment No. 16 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(l).

The staff concludes that the design of the confinement system of the HI-STORM 100 Cask System complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the HI-STORM 100 Cask System will allow 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 OPERATING PROCEDURES EVALUATION

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

In Amendment No. 16, the applicant revised FSAR chapter 8, Operating Procedures, and added supplemental chapter 8.IV, Operating Procedures, to address the following proposed changes:

- PC #1 Add a new unventilated UVH overpack which includes high-density concrete for shielding.
- PC #2 Modify vent and drain penetrations to include the option of a 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 examination.
- 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 8.1.1, Overview of Loading Operations, on proposed revision 21 page 8-8, the requirement to perform an additional liquid penetrant examination on the MPC lid to shell weld after pressure testing to verify structural integrity has been removed. On proposed revision 21 page 8-9, Holtec 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 8.1.5, MPC Closure, on proposed revision 21A page 8-20, Holtec added more guidance for when an ethylene glycol solution must be added to the water jacket of the HI-TRAC transfer cask.

In the same section, on proposed revision 21 page 8-24, Holtec has 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 non-destructive examination (NDE) performed. Further in this section, on proposed revision 21E page 8-29, Holtec provides new guidance for when the redundant MPC port cover plates are installed, how to perform the welds, and perform NDE of the welds. If redundant port cover plates are used, a leakage test must be performed for the first cover plate, with no change to the existing requirement. If a single port cover plate is used, a leakage test must be performed for the single cover plate, with also no change to the existing requirement.

In the application section 8.3.2, HI-STORM Recovery from Storage, on proposed revision 21 page 8-101 and revision 21A page 102, Holtec added more guidance when an ethylene glycol solution must be added to the water jacket of the HI-TRAC transfer cask.

In the application section 8.IV.0, Technical and Safety Basis for Loading and Unloading Procedures, the applicant states: “The Technical and Safety Basis for loading and unloading the HI-STORM 100 identified in section 8.0 of chapter 8 are applicable to the HI-STORM 100 UVH.”

In the application section 8.IV.1.7, Placement of HI-STORM 100 UVH overpack into Storage, steps are provided for installing a gasket under the UVH overpack 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 100 UVH annulus with a non-oxidizing gas.

In the application table 8.IV.1.6, HI-STORM 100 system Ancillary Equipment Operational Description, line items for the HI-STORM UVH annulus evacuation system and nitrogen (or another non-oxidizing gas) backfill system have been added as not ITS.

In the application table 8.IV.1.7, HI-STORM 100 system Instrumentation Summary for Loading and Unloading Operations, a new line item has been added for pressure gauges to ensure correct pressure during HI-STORM 100 UVH overpack backfill operations.

In the application table 8.IV.1.8, HI-STORM 100 UVH system Overpack Inspection Checklist, additional checklist items have been added for the UVH overpack lid and UVH overpack main body when using the UVH overpack.

In the application section 8.IV.2, ISFSI Operations, Holtec provides general guidance on the UVH overpack when in storage and the need for minor maintenance or removal of a MPC from the UVH overpack.

In the application section 8.IV.3, Procedure for Unloading the HI-STORM 100 UVH Fuel in the spent fuel pool, Holtec provides procedure steps specific to the UVH overpack when recovering it from storage and unloading the HI-STORM 100 UVH fuel in the spent fuel pool.

See section 7, Confinement Evaluation, of this SER for a safety evaluation review of MPC vent and drain penetrations to include the option of a second port cover plate. Also see SER chapter 7 for a review of the change to the MPC hydrostatic pressure test acceptance criteria to be examination for leakage only and remove the post hydrostatic test liquid PT examination.

See section 4, Thermal Evaluation, of this SER for a safety evaluation review of the requirements for when an ethylene glycol solution must be used in the HI-TRAC water jacket.

8.1 Evaluation Findings

The staff concludes that the generic procedures and guidance for the operation of the HI-STORM 100 and HI-STORM 100 UVH systems comply with 10 CFR Part 72 and sufficient information was provided for a safety evaluation and review. The evaluation of the operating procedure descriptions provided in the FSAR 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.

F8.1 The HI-STORM 100 Cask System, with these changes, is still compatible with wet loading and unloading in compliance with 10 CFR 72.236(h). General procedure descriptions for these operations are summarized in FSAR chapters 8 and 8.IV. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 The welded lids of the MPCs allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The content of the general operating procedures described in the FSAR for the applicable proposed changes are adequate to protect health and minimize damage to life and property. Detailed procedures will need to be developed and approved on a site-specific basis.

F8.4 The FSAR with the Amendment No. 16 changes still includes acceptable descriptions and discussions of the existing HI-STORM 100 and new HI-STORM 100 UVH operations, operating characteristics, and safety considerations, to demonstrate compliance with 10 CFR 72.234(f).

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM EVALUATION

The objective of this review is to ensure that the applicant's FSAR includes the appropriate acceptance criteria/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 NRC staff conducts subsequent inspections. The acceptance criteria/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. 16, the applicant revised FSAR chapter 9, Acceptance Criteria and Maintenance Program, and added supplemental chapter 9.IV, Acceptance Criteria and Maintenance Program, to address the following proposed changes:

PC #1 Add a new unventilated overpack, HI-STORM 100 UVH, which includes high-density concrete for shielding.
PC #2 Modify vent and drain penetrations to include the option of a second port cover plate.
PC #6 Change the hydrostatic pressure test of the MPC acceptance criteria to be examination for leakage only. Remove post hydrostatic test liquid PT examination.
PC #14 Other editorial changes.

In the application section 9.1.2.2.2, MPC Confinement Boundary, on proposed revision 21 page 9-8, Holtec discusses the pneumatic or hydrostatic pressure testing of the MPC lid to shell field weld. Holtec 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 9.1.3, Leakage Testing, on proposed revision 21E page 9-10, Holtec added the option on all MPCs for the addition of a second 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.

In the application table 9.1.1, MPC Inspection And Test Acceptance Criteria, on proposed revision 21E page 9-18, for the function of leak tests, Holtec added in the maintenance and operations block for leak tests, new text stating that if the redundant port cover design is used on the vent and drain ports, helium leak testing is only required on the inner port cover.

In the application table 9.1.4, HI-STORM 100 NDE Requirements, on proposed revison 21A page 9-27, Holtec added direction for weld NDE requirements when the redundant port cover plate option is used.

In the application chapter 9.IV, Acceptance Criteria and Maintenance Program, in section 9.IV.0, Introduction, the applicant states in part: “The addition of the Unventilated overpack through supplement IV does not involve the introduction of any new structural or shielding materials, MPCs, or transfer casks to the storage system. Therefore, no change to the areas which cover most of chapter 9 is necessary. Any additional tests, inspections, and maintenance activities are identified in the following sections. The following sections describe the acceptance and maintenance activities that are unique to the HI-STORM 100 UVH system and thus supplement the information presented in chapter 9. …The guidance provided in this supplement shall be used along with the acceptance and maintenance activities provided in chapter 9 to develop the site-specific maintenance procedures for the HI-STORM 100 UVH.”

In the application section 9.IV.2.2, Leakage Tests, and table 9.IV.2.1, HI-STORM System Maintenance Program Schedule, the following additional maintenance step has been added when the UVH overpack is used: “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 section 7, Confinement Evaluation, of this SER for a safety evaluation review of MPC vent and drain penetrations to include the option of a second port cover plate and the change to the MPC hydrostatic pressure test acceptance criteria to be examination for leakage only and remove the post hydrostatic test liquid PT examination.

9.1 Evaluation Findings

The staff concludes that the acceptance tests and maintenance program for the HI-STORM 100 standard and 100 UVH systems comply 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 based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F9.1 Chapters 9 and 9.IV of the FSAR adequately describe the applicant's proposed program for preoperational testing, initial operations, and maintenance of the HI-STORM 100 standard and 100 UVH systems.

F9.2 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.

F9.3 The applicant will examine and/or test the HI-STORM 100 standard and 100 UVH systems to ensure that they do not exhibit any defects that could significantly reduce their confinement effectiveness.

10.0 RADIATION PROTECTION EVALUATION

The objective of the radiation protection evaluation is to determine that the proposed changes comply with the applicable regulatory requirements for radiation protection and ensure that the proposed changes include reasonable consideration of, and facilitate licensees' compliance with, the requirements that licensees who use the storage system must meet. In reviewing these changes to the HI-STORM 100 Cask System's shielding capability, the staff followed the guidance in Chapter 10b of NUREG-2215 (NRC, 2020a).

The two changes that would affect the radiation protection evaluation are:

- PC #1 - Add a new overpack, called the HI-STORM 100 UVH.
- PC #7 - Replace FQTs with sets of burnup and cooling time limits and reduce minimum cooling time from 2 years to 1 year.

10.1 PC #1: Addition of UVH Overpack

As discussed in section 5.1 of this SER, the dose rates from the UVH is lower than those from the ventilated overpack. Therefore, dose rates from the previously approved HI-STORM 100 overpacks bound this system.

HI-TRAC MS is the transfer cask to be used with the HI-STORM 100 UVH, and it contains a variable thickness lead shield. As discussed in section 5.1.3.1 of this SER, the most conservative modeling of this component would include the limiting spent nuclear fuel specifications allowed from the correlations in FSAR section 2.IV.1.2 of the application with the minimum allowable thickness of the lead shield. The applicant has chosen to implement TS dose rate limits for the HI-TRAC MS as part of the strategy to limit occupational exposure. To meet the TS dose rate limits, the allowable loading would be a combination of contents with increased cooling times beyond the correlation limits and/or increased lead thickness for HI-TRAC MS. This approach was used in HI-STORM 100 Cask System Amendment No. 15, and it was reviewed and approved by the NRC.

Because the TS limit for the HI-TRAC MS transfer cask has not changed from Amendment No. 15, the staff finds that the radiation protection evaluation in supplement 10.II is also applicable to the UVH Overpack for the spent fuel specifications in FSAR section 2.IV.1.2 of the application.

10.2 PC #7: Replace FQTs With Sets of Burnup and Cooling Time Limits and Reduce Minimum Cooling Time from 2 Years to 1 Year

The new burnup and cooling time limits result in higher dose rates for the above ground ventilated overpacks. The applicant calculated the dose to a real individual beyond the controlled area with these new burnup and cooling time limits for the three canister sizes (MPC-24, MPC-32, and MPC-68) and documented the results in FSAR table 5.4.23. These calculations assume full occupancy for a real individual, which is conservative. The staff finds that these calculations meet the regulation in 10 CFR 72.236(d) by demonstrating that the shielding is sufficient to reduce the dose rate to the levels in 10 CFR 72.104(a) at the distances prescribed in table 5.4.23 for a single cask and several example array sizes.

Although the storage overpack surface dose rates have increased, the applicant did not propose changes to increase the TS dose rate limits in appendix A to the CoC. The staff considers this conservative because the users would have to load less than design basis source terms to meet the TS dose rate limits.

As discussed in section 5.4.1.1. of this SER, for the HI-STORM 100U, all previously analyzed dose rates are applicable for the proposed burnup and cooling time tables.

The applicant did not update the estimated occupational exposure in FSAR chapter 10 with the calculated dose rates (FSAR table 5.4.21) from the HI-TRAC transfer cask using the new burnup and cooling time limits. Instead, the applicant proposed a TS dose rate limit at the side of the transfer cask of 4 rem/hr (appendix A, section 5.7.4.c). This value is less than the calculated dose rate at this location but is consistent with the dose rate used in the radiation protection operations in chapter 10 of the FSAR. The end user will have to adjust the actual amount of fuel loading to reduce the design basis source terms to maintain the dose rates from the transfer casks to be below the TS limits. Therefore, the staff finds the current radiation protection chapter occupational dose estimates applicable to the new burnup/cooling time combinations and the reduction of cooling time to 1 year for PWR fuel. This is similar to the approach taken in Amendment No. 15 when the HI-TRAC MS was implemented and staff finds it acceptable for the proposed changes in this amendment.

10.3 Evaluation Findings

The staff finds, with reasonable assurance, that the radiation protection design of the HI-STORM 100, including the changes associated with this amendment, complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection design provides reasonable assurance that the HI-STORM 100 Cask System will continue to allow safe storage of SNF. The staff reached this finding based on a review that considered applicable NRC regulations and regulatory guides, codes and standards, accepted health physics practices, statements and representations contained in the SAR, and the staff’s independent confirmatory analyses.

F10.1 The HI-STORM 100 Cask System provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106, in accordance with 10 CFR 72.236(d).

F10.2 The design and operating procedures of the HI-STORM 100 Cask System provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the ALARA objective with respect to exposures, consistent with 10 CFR 20.1101(b).

F10.3 The HI-STORM 100 Cask System is adequately designed to facilitate decontamination in accordance with 10 CFR 72.236(i) and includes, to the extent practical and appropriate, adequate features, operating procedures, and controls that are designed to assist a general licensee to meet the radiological protection criteria in 10 CFR 72.126(a) and 10 CFR 72.126(d).

11.0 ACCIDENT ANALYSES EVALUATION

The staff's evaluations of the applicant's accident analysis for the proposed changes are documented in sections 3 through 7 and 14 of this SER. Therefore, this section does not need to provide further evaluation.

12.0 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 SAR.

Table 12-1 lists the applicant's proposed changes to the CoC and TS:

Table 12-1 – Conforming Changes to the Technical Specifications and Operating Control and Limits

Page Number	Reference	Description	Proposed Change
CoC, page 1	Condition 1b	Update the description of the system in the CoC to clarify that only the portions of MPC 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. 17 (NRC, 2023b).
CoC, page 2	Condition 1b	Add the description on UVH overpack.	1
CoC, page 3	Condition 9a	Add a statement that Condition 9a does not apply to UVH overpack.	1
Appendix A	N/A	Update title, table of content, page numbers, headers, footers, and format.	N/A
Appendix A, page 1.1-4	Definition	1) Revise the definition of overpack for the addition of UVH overpack. 2) Add the definition of redundant port cover design.	1, 3

Page Number	Reference	Description	Proposed Change
Appendix A, page 1.1-6	Definition	Add the definitions of unventilated and ventilated overpacks.	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
Appendix A, page 3.4-1	Table 3-1a	Update the table number and specify this table is for ventilated overpack.	1
Appendix A, page 3.4-3	Table 3-1b	Add table 3-1b for unventilated overpack.	1
Appendix A, page 3.4-4	Table 3-2a	Update the table number and specify this table is for ventilated overpack.	1
Appendix A, page 3.4-6	Table 3-2b	Add table 3-2b for unventilated overpack.	1
Appendix A, page 5.0-5	Section 5.7.2	Correct the reference from 10 CFR 72.212(b)(2)(i)(C) to 10 CFR 72.212(b)(5)(iii).	N/A
Appendix A, page 5.0-5	Section 5.7.3(c)	Add a statement that TS 5.7.3(c) only applies to ventilated overpacks.	1
Appendix A, page 5.0-5	Section 5.7.4(c)	Add dose rate limit on the side of the transfer cask.	1
Appendix A, page 5.0-7	Section 5.7.8(e)	Add a statement that TS 5.7.8(e)only applies to ventilated overpack.	1
Appendix B	N/A	Update title, table of content, page numbers, headers, footers, and format.	N/A
Appendix B, page 2-2	Section 2.1.3	Add a statement to specify applicability for ventilated and unventilated overpacks.	1
Appendix B, page 2-50	Section 2.4.1	Specify section 2.4.1 is for ventilated overpack.	1, 7
Appendix B, page 2-51	Section 2.4.2	Specify section 2.4.2 is for ventilated overpack.	1, 7
Appendix B, page 2-53	Section 2.4.3	Revise burnup calculation equation for ventilated overpack in section 2.4.3.	1, 7
Appendix B, page 2-53	Section 2.4.5	Add section 2.4.5 for fuel loading decay heat limits for unventilated overpack.	1, 7
Appendix B, page 2-53	Section 2.4.6	Add section 2.4.6 for burnup calculation equation for unventilated overpack.	1, 7
Appendix B, page 2-54	Table 2.4-3	Revise PWR fuel assembly burnup and cooling time limits for ventilated overpack.	1, 7

Page Number	Reference	Description	Proposed Change
Appendix B, page 2-55	Table 2.4-4	Revise BWR fuel assembly burnup and cooling time limits for ventilated overpack.	1, 7
Appendix B, pages 2-56 through 2-58	Table 2.4-6a, 2.4-6b, 2.4-7, 2.4-8, and 2.4-9	Add five tables to include thermal and shielding analysis information (heat load data, requirements for developing loading patterns, storage location, and fuel qualification requirements) for MPC-68M in unventilated overpack.	1, 7
Appendix B, page 3-8	Table 3-1, page 5	Revise ASME Code alternatives for radiographic or ultrasonic examination requirements for MPC closure ring, vent, and drain cover plate welds, specifically for the optional second port cover plate.	3
Appendix B, page 3-13	Section 3.4, subsection 1	Revise the maximum average yearly temperature for ventilated and unventilated overpacks.	1
Appendix A-100U	N/A	Update title, page numbers, headers, footers, and format.	N/A
Appendix A-100U, page 5.0-5	Section 5.7.2	Correct the reference from 10 CFR 72.212(b)(2)(i)(C) to 10 CFR 72.212(b)(5)(iii)	N/A
Appendix B-100U	N/A	Update title, table of content, page numbers, headers, footers, and format.	N/A
Appendix B-100U, page 2-28	Section 2.4.3	Revise burnup calculation equation in section 2.4.3.	7
Appendix B-100U, page 2-29	Table 2.4-3	Revise PWR fuel assembly burnup and cooling time limits.	7
Appendix B-100U, page 2-29	Table 2.4-4	Revise BWR fuel assembly burnup and cooling time limits.	7
Appendix C	N/A	Update title, table of content, page numbers, headers, footers, and format.	N/A
Appendix C, page 1.1-4	Definition	1) Revise the definition of overpack for the addition of UVH overpack. 2) Add the definition of redundant port cover design.	1, 3
Appendix C, page 1.1-6	Definition	Add the definitions of unventilated and ventilated overpacks.	1
Appendix C, page 3.1.2-1	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 C, page 3.4-1	Table 3-1a	Update the table number and specify this table is for ventilated overpack.	1
Appendix C, page 3.4-3	Table 3-1b	Add table 3-1b for unventilated overpack.	1
Appendix C, page 3.4-4	Table 3-2a	Update the table number and specify this table is for ventilated overpack.	1
Appendix C, page 3.4-5	Table 3-2b	Add table 3-2b for unventilated overpack.	1
Appendix C, page 5.3-1	Section 5.3.2	Correct the reference from 10 CFR 72.212(b)(2)(i)(C) to 10 CFR 72.212(b)(5)(iii)	N/A
Appendix C, page 5.3-1	Section 5.3.3(c)	Add a statement that TS 5.3.3(c) only applies to ventilated overpacks.	1
Appendix C, page 5.3-3	Section 5.3.8(d)	Add a statement that TS 5.3.8(d) only applies to ventilated overpack.	1
Appendix D	N/A	Update title, table of content, page numbers, headers, footers, and format.	N/A
Appendix D, page 2-1	Section 2.1a	Delete “the HI-STORM 100S Version E” from the sentence.	N/A. MPC-32M is only allowed in HI-STORM Version E, and appendix D is for HI-STORM Version E. This is to remove the redundant statement.
Appendix D, page 2-11	Table 2.1-4	Revise fuel qualification for MPC-32M.	7
Appendix D, page 2-12	Figure 2.1-1	Add a note to designate shaded cells as region 1 (inner).	7
Appendix D, page 2-13	Sections 2.4, 2.4.1, and 2.4.2	Revise the title for section 2.4. Specify that sections 2.4.1 and 2.4.2 apply to ventilated overpack.	1
Appendix D, page 2-14	Section 2.4.5	Add section 2.4.5 to describe decay heat limit for MPC-32M for unventilated overpack.	1, 7

Page Number	Reference	Description	Proposed Change
Appendix D, pages 2-18 and 2-19	Table 2.4-5a, 2.3-5b, 2.4-6, and 2.4-7	Add thermal analysis information (heat load data, requirements for developing loading patterns, heat load calculation equations, and storage location) for MPC-32M in unventilated overpack.	1, 7

The staff finds that the proposed changes to the TS for the HI-STORM 100 Cask System Amendment No. 16 conform to the changes requested in the amendment application and do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. The proposed changes provide reasonable assurance that the HI-STORM 100 Cask System Amendment No. 16 will continue to allow safe storage of SNF.

13.0 QUALITY ASSURANCE EVALUATION

The applicant did not propose any changes that affect the staff's quality assurance evaluation provided in the previous SERs for CoC No. 1014, Amendment Nos. 0 through 15. Therefore, the staff determined that a new evaluation was not required.

14.0 MATERIALS EVALUATION

The staff evaluated the materials performance of the new UVH overpack to ensure that it meets the requirements of 10 CFR Part 72. The UVH overpack is a MPC storage system designed to provide shielding and environmental protection of the MPC during storage. The principal objective of the new UVH overpack design is to prevent stress corrosion cracking of the welded stainless steel MPC. This is accomplished by its omission of inlet or outlet vents, creating an enclosed environment in the overpack.

14.1 Materials of Construction

As described in the application sections 1.IV.2.1.2.1 and the licensing drawings, the UVH overpack is comprised of a cylindrical structure consisting of inner and outer shells, a lid, and a baseplate that are fabricated from ASME Boiler and Pressure Vessel (B&PV) Code specification 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 ACI 318, 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 (ASME B&PV Code specification SA 540 B23 Class 3) and a metallic gasket to provide an enclosed environment in the overpack.

14.2 Drawings

The applicant provided new drawings in section 1.IV.5 of the application to incorporate the new UVH overpack. The drawings include a parts list that provides the material specification of each component, and they also provide the welding, examination, and coating 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.

14.3 Codes and Standards

The staff verified that the new UVH overpack uses the same ACI construction codes and ASME B&PV Code specifications for the 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 that states 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.

14.4 Welding

The new UVH 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 NF, Article NF-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.

14.5 Material Properties

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 UVH overpack. In the application supplement 4.IV, 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 and internal pressures 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.

14.6 Radiation Shielding Materials

The UVH overpack makes use of concrete and steel for gamma shielding and concrete for neutron shielding. The staff verified that the UVH 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.

14.7 Concrete and Reinforcing Steel

As described in the application section 1.IV.2.1.2.1, the new UVH overpack uses the same concrete as the previously approved casks, complying with the requirements in SAR appendix 1.D (Type II plain concrete (ASTM C150)). As described in the application table 1.D.1 of appendix 1.D, the UVH overpack follows the temperature requirements of ACI-349 (ACI). The staff reviewed the thermal analysis in the application supplement 4.IV to ensure theses concrete temperature requirements are met under the service conditions associated with the new UVH overpack. The staff finds the concrete design of the UVH overpack to be acceptable. Furthermore, the staff reviewed the clarifications made to the plain concrete specifications contained within the application appendix 1.D. The staff found no technical changes to the design and requirements of the concrete that would affect the applicability of the referenced codes and standards previously reviewed by the NRC. Therefore, the staff finds the concrete specifications for the HI-STORM 100 overpacks to be acceptable.

14.8 Bolt Applications

The new UVH overpack uses a set of equidistant anchor bolts to secure the cask lid to the cask body. The mounting bolts are fabricated from ASME SA 540 B23 Class 3, which has been previously evaluated by NRC staff for use in the HI-STORM 100 design. Therefore, the staff finds the applicants bolting materials to be acceptable.

14.9 Corrosion Resistance and Content Reactions

The staff reviewed the proposed 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 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.

14.10 Management of Aging Degradation

Holtec applied for a renewal of the HI-STORM 100 Cask System CoC in a letter dated January 31, 2020 (Holtec, 2020a) and provided an aging management evaluation in the renewal application. This amendment for the UVH overpack was proposed subsequent to the renewal application. The UVH overpack is essentially identical to the ventilated overpacks, except for the removal of the inlet and outlet air passages and that the closure lid installation on the cask body is now via a weather-resistant metallic gasket providing an enclosed environment in the overpack. In accordance with the guidance in NUREG-2215, the staff reviewed the amendment application and CoC renewal documentation to ensure that Holtec appropriately evaluated the need to revise (1) the scope of SSCs subject to aging management, (2) the aging management review (AMR) for applicable aging mechanisms and effects, and (3) the time-limited aging analyses (TLAAs) and aging management programs (AMPs) to ensure that they remain effective to manage the aging of all SSCs. The staff's review of the applicant's proposed aging management approach used the guidance in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," (NRC, 2016) and NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report." (NRC, 2019)

14.10.1 Scope of SSCs Requiring Aging Management

As described in NUREG-1927, a scoping evaluation is necessary to identify the SSCs requiring an aging management review (AMR). The objective of this scoping evaluation is to identify (1) SSCs that are classified as ITS and (2) SSCs that are classified as not ITS but, according to the design bases, their failure could prevent fulfillment of a function that is ITS. After the determination of in-scope SSCs, the SSCs are screened to identify and describe the subcomponents that support the SSC intended functions.

The staff reviewed the scoping results in section 2.2 of Holtec Report HI-2210316, "Aging Management Evaluation for the HI-STORM 100 Version UVH Dry Storage System," revision 0, to determine whether the applicant included all SSCs that are ITS and within the scope of renewal in approved design bases. The staff notes this amendment added one new SSC, the UVH overpack, and this SSC was found to be within the scope of renewal, as was the ventilated HI-STORM 100 overpack, for its ITS classification and for ensuring structural integrity, heat transfer, and shielding safety functions.

The staff reviewed table 2.2-3 of Holtec Report HI-2210316 to verify that the UVH overpack was screened to identify and describe the subcomponents with intended functions. The staff verified that it was screened in the same manner as the ventilated overpack with the exception of the removal of inlet and outlet vents that are not present on the UVH overpack and the addition of the new weather-resistant metallic gasket.

Based on its review, the staff finds that the applicant has identified the in-scope SSCs in a manner consistent with NUREG-1927, and therefore, the staff finds the scoping results to be acceptable. The applicant screened the in-scope SSCs to identify and describe the subcomponents that support the SSC intended functions.

14.10.2 Aging Management Review

The objective of the staff's evaluation of the applicant's AMR is to determine whether the applicant has adequately reviewed applicable materials, environments, and aging mechanisms and effects and proposed adequate aging management activities for in-scope SSCs. The AMR addresses aging mechanisms and effects that could adversely affect the ability of the SSCs and associated subcomponents to perform their intended functions during the period of extended operation.

The staff reviewed the AMR in section 3 of Holtec proprietary report HI-2210316, revision 0, "Aging Management Evaluation for the HI-STORM 100 Version UVH Dry Storage System." The staff notes that the materials identified are the same as the ventilated overpack (carbon steel and concrete), except for the addition of the new metallic seal. The environments identified are the same as those in the ventilated overpack (air-outdoor, embedded, and sheltered). The applicant revised their aging management evaluation in an RAI response to no longer credit the metallic seal with preventing the exposure of the internal cavity to outside air under all scenarios. These minor differences in materials and environments did not alter the overpack aging mechanisms requiring management from those identified in the renewal of the ventilated overpack (loss of material due to general, pitting, and crevice corrosion).

Based on its review, the staff finds that the applicant has considered the materials and specific environments to identify the aging mechanisms that could lead to loss of intended functions in a manner consistent with NUREG-1927, and therefore, the staff finds the AMR to be acceptable.

14.10.3 Time-Limited Aging Analyses

The applicant did not identify any TLAAs related to the addition of the UVH overpack. The staff reviewed the design basis documents and verified that the addition of the UVH overpack does not impact the TLAAs that were evaluated in the original renewal of the HI-STORM 100 Cask System design, nor does the UVH overpack introduce new TLAAs. Therefore, the staff finds the applicant's determination to be acceptable.

14.10.4 Aging Management Programs

The applicant revised its aging management evaluation in an RAI response to require the same visual inspections and use the same Aging Management Program (AMP) as the ventilated overpack. The staff reviewed this AMP in Holtec HI-STORM 100 FSAR revision 20 (Holtec, 2020b) to determine whether the applicant identified activities capable of effectively managing the effects of aging for the new UVH overpack.

The staff verified the technical adequacy of SAR table 9.A.1-2 to ensure the overpack AMP assessed the ten program elements of NUREG-1927 in sufficient detail to manage the previously discussed aging mechanisms. The staff also compared the AMP activities to the applicable NUREG-2214 example AMP, “Monitoring of Metallic Surfaces”, which recommends periodic visual inspections of external overpack surfaces to manage corrosion aging mechanisms. As discussed in section 14.10.2 above, the minor differences in materials and environments did not alter the overpack aging mechanisms requiring management from the ventilated overpack, thus the aging management activities were unchanged from the ventilated overpack. Per the above discussion, the staff finds the AMP to be adequate for managing the aging mechanisms identified per the AMR.

14.11 Evaluation Findings

F14.1 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.

F14.2 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.

F14.3 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.

F14.4 The applicant has met the requirements in 10 CFR 72.234(b). Quality assurance programs and control of special processes are demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

F14.5 The applicant’s AMR process is comprehensive in identifying the materials of construction and associated operating environmental conditions for those SSCs within the scope of renewal and has provided an acceptable summary of the information in the renewal application and the FSAR supplement.

F14.6 The applicant’s AMR process is comprehensive in identifying all pertinent aging mechanisms and effects applicable to the SSCs within the scope of renewal, and the applicant has provided an acceptable summary of the information in the renewal application and the FSAR supplement.

F14.7 The applicant has identified that the AMPs in the renewed CoC provide reasonable assurance that aging effects that are applicable to the new in-scope SSCs will be managed effectively during the period of extended operation, in accordance with 10 CFR 72.240(c)(3).

The staff concludes that the Holtec design adequately considers material properties, environmental degradation and other reactions, and material quality controls such that the design complies with 10 CFR Part 72. The staff also verified the AMR adequately identified the materials, environments, and aging effects of the in-scope SSCs. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

15.0 CONCLUSIONS

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

PC #1 Add a new UVH overpack, HI-STORM 100 UVH, which includes high-density concrete for shielding. The UVH overpack is to be used with the MPC-32M and MPC-68M.

PC #2 Include the ability to use the CFD analysis to evaluate site-specific fire accident scenario.

PC #3 Modify the vent and drain penetrations to include the option of a second port cover plate.

PC #4 Include the ability to use CFD analysis to evaluate site-specific burial under debris accident scenario.

PC #5 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 #6 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 #7 Replace the FQT in FSAR chapter 2 and the CoC, including the equation for calculation of the maximum allowable burnup as a function of the cooling time and cooling time-dependent coefficients, with simpler sets of burnup and cooling time limits. Reduce the minimum cooling time for PWR fuel from 2 years to 1 year.

PC #8 Revise FSAR appendix 1.D to enhance certain requirements, to add certain revised shielding assumptions following a significant thermal event, and to add critical characteristics for concrete employed in the HI-STORM 100 UVH system. Appendix 1.D is renamed to “Specification for Plain Concrete in the HI-STORM Family of Overpacks.”

Based on the statements and representations provided by the applicant in its amendment application, as supplemented, the changes described above to the HI-STORM 100 Cask System in Amendment No. 16 do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. Amendment No. 16 for the HI-STORM 100 Cask System should be approved.

Issued with Certificate of Compliance No. 1014, Amendment No. 16

on __________.

References

ACI. Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-85, American Concrete Institute, Detroit, Michigan. 1985.

ANSI, 1993. ANSI N14.6, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More," American National Standards Institute, Inc., Washington, DC, June 1993.

ANSI, 2014. ANSI N14.5-2014, “Radioactive Materials – Leakage Tests on Packages for Shipment,” American National Standards Institute, Inc., Washington, DC, June 19, 2014.

ASME, 2010. ASME International, Boiler and Pressure Vessel Code, 2010 edition.

DOE, 1992. “Characteristics of Potential Repository Wastes,” DOE/RW-0184-R1, Volume 1, July 1992

Holtec, 2020a. Letter from Holtec International to NRC, “Holtec International HI-STORM 100 Storage CoC Renewal Application.” January 31, 2020. This package contains 5 attachments, and Attachments 1, 3, and 4 are *Proprietary Information and Not Publicly Available*. ML20049A081.

Holtec, 2020b. Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System, revision 20, June 12, 2020. ML20167A018.

Holtec, 2021a. Letter from Holtec International to NRC, “HI-STORM 100 Multipurpose Canister Storage System Amendment 16 Request.” March 9, 2021. This package contains 36 attachments, and Attachments 9 and 11 through 20 are *Proprietary Information and Not Publicly Available*. ML21068A360.

Holtec, 2021b. Letter from Holtec International to NRC, “HI-STORM 100 Amendment 16 Responses to Requests for Supplemental Information.” August 11, 2021. This package contains 10 attachments, and Attachments 1, 5 through 9 are *Proprietary Information and Not Publicly Available*. ML21223A045.

Holtec, 2021c. Holtec International, Final Safety Analysis Report for the HI-STORM 100 Cask System, Revision 22. July 1, 2021. ML21221A329.

Holtec, 2022a. Letter from Holtec International to NRC, “HI-STORM 100 Amendment 16 Responses to Requests for Additional Information (Batches 1 and 2).” August 31, 2022. This package contains 8 attachments, and Attachments 1, 3, 5, 6, and 7 are *Proprietary Information and Not Publicly Available*. ML22243A269.

Holtec, 2022b. Letter from Holtec International to NRC, “HI-STORM 100 Amendment 16 Responses to Requests for Additional Information (Batch 2).” September 9, 2022. This package contains 9 attachments, and Attachments 1, 3, and 8 are *Proprietary Information and Not Publicly Available*. ML22249A347.

Holtec, 2022c. Letter from Holtec International to NRC, “HI-STORM 100 Amendment 16 Responses to Requests for Additional Information (Batch 3).” October 3, 2022. This package contains 3 attachments, and Attachment 1 is *Proprietary Information and Not Publicly Available*. ML22276A286.

Holtec, 2022d. Letter from Holtec International to NRC, "HI-STORM 100 Multipurpose Canister Storage System Amendment 19." February 17, 2022. ML22048C222.

Holtec, 2023a. Letter from Holtec International to NRC, "HI-STORM 100 Amendment 16 Responses to Requests for Additional Information (Batch 4)." January 4, 2023. This package contains 6 attachments, and Attachments 1, 3, and 5 are *Proprietary Information and Not Publicly Available*. ML23005A000.

Holtec, 2023b. Letter from Holtec International to NRC, "HI-STORM 100 Amendment 16 Responses to Requests for Additional Information (Batch 4) – Additional Supporting Document." January 5, 2023. ML23005A273.

Holtec, 2023c. Letter from Holtec International to NRC, "HI-STORM 100 Amendment 16 Responses to Requests for Additional Information (Batch 5)." January 13, 2023. This package contains 9 attachments, and Attachment 1 is *Proprietary Information and Not Publicly Available*. ML23013A337.

Holtec, 2023d. Letter from Holtec International to NRC, "HI-STORM 100 Amendment 16 RAI 3-3 Clarification." March 17, 2023. This package contains 5 attachments, and Attachments 1 and 4 are *Proprietary Information and Not Publicly Available*. ML23076A271.

Holtec, 2023e. Letter from Holtec International to NRC, "HI-STORM 100 Amendment 16 Responses to Second Round Requests for Additional Information." September 20, 2023. This package contains 4 attachments, and Attachment 3 is *Proprietary Information and Not Publicly Available*. ML23263B122.

NRC, 1998. "Engineering Drawings for 10 CFR Part 71 Package Approvals," NUREG/CR-5502, UCRL-ID-130438, Lawrence Livermore National Laboratory, May 1998.

NRC, 2003. "Cladding Considerations for the Transportation and Storage of Spent Fuel," ISG-11, revision 3. November 17, 2003. ML033230335.

NRC, 2016. "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel– Final Report," NUREG-1927, Revision 1, U.S. NRC, June 2016. ML16179A148.

NRC, 2019. "Managing Aging Processes in Storage (MAPS) Report," NUREG-2214, U.S. NRC, July 2019. ML19214A111.

NRC, 2020a. Letter from NRC to Nuclear Energy Institute, "Endorsement of Independent Spent Fuel Storage Installation License and Cask Certificate of Compliance Format, Content, and Selection Criteria-Graded Approach." January 8, 2020. ML19353D337.

NRC, 2020b. "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities," NUREG-2215, Final Report, U.S. NRC, April 2020. ML20121A190.

NRC, 2020c. Final Safety Evaluation Report, Docket No. 72-1032, Holtec International, HI-STORM Flood/Wind Multipurpose Canister Storage System, Certificate of Compliance No. 1032, Amendment No. 4. June 15, 2020. ML20155K745.

NRC, 2021. Final Safety Evaluation Report, Docket No. 72-1014, Holtec International, HI-STORM 100 Multi-purpose Canister Storage System, Certificate of Compliance No. 1014, Amendment No. 15. May 11, 2021. ML21118A871.

NRC, 2022. Letter from NRC to Holtec International, "Amendment No. 16 To Certificate of Compliance No. 1014 for the HI-STORM 100 Multipurpose Canister Storage System – Request for Additional Information Second Batch," August 5, 2022. ML22208A224.

NRC, 2023a. Conversation record, "Clarification call with Holtec on proposed change #4 for HI-STORM FW Amendment 7 application and proposed change #6 for HI-STORM 100 Amendment 16," January 20, 2023. ML23030B859.

NRC, 2023b. Final Safety Evaluation Report, Docket No. 72-1014, Holtec International, Certificate of Compliance No. 1014, HI-STORM 100 System, Amendment No. 17. Date December 5, 2023. ML23328A027.

Radulescu, 2017. G. Radulescu, et. al, "Shielding Analysis Capability of UNF-ST&DARDS," Nuclear Technology, 199:3, 276-288. 2017.