UNITED STATES
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PRELIMINARY SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HOLTEC INTERNATIONAL
HI-STORM 100
MULTIPURPOSE CANISTER STORAGE SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 11

SUMMARY

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (staff) review and evaluation of the request to amend Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Multipurpose Canister (MPC) Storage System. Holtec International (Holtec) submitted the request to the NRC by letter dated January 29, 2016 (Holtec, 2016a), and supplemented on February 16, 2016 (Holtec, 2016b), June 6, 2016 (Holtec, 2016c), December 22, 2016 (Holtec, 2016d), April 22, 2016 (Holtec, 2016e; modified request), September 8, 2017 (Holtec, 2017a), November 10, 2017 (Holtec, 2017b), and December 21, 2017 (Holtec, 2017c). The amendment request proposes the following changes:

1. Increase the per-storage location weight limit for cells authorized for damaged fuel container (DFC) in MPC-68, MPC-68FF, and MPC-68M in HI-STORM 100 storage system.
2. Change surveillance requirements for cask with certain heat load as specified in the Technical Specifications (TS).
3. Allow the storage of higher average initial enrichment wt.% U-235 fuel with low enriched CRUD-induced localized corrosion (CILC) fuel.
4. Increase the enrichment limit for 10x10G boiling water reactor (BWR) fuel assembly from 4.6 wt.% U-235 to 4.75 wt.% U-235.
5. Change the minimum soluble boron concentration limits for the 17x17A pressurized water reactor (PWR) fuel assemblies in MPC-32.
6. Increase the burnup limit to accommodate non-fuel hardware (NFH), including neutron source assembly (NSA), in combination with other control components.
7. Add thoria rods/canister as contents for the MPC-68M.
8. Add a second permissible composition for thoria rods for all MPC-68 models. The new thoria rod composition is made of 98.5 wt% ThO_2 and 1.5% UO_2. The maximum enrichment of U-235 in UO_2 is 93.5 wt%.

The applicant also made the following editorial changes:

1. Clarify heat load limit and drying method in Apendix A, Table 3-1.
2. Include NUREG-0612 as a basis for stress limits.
3. Remove manufacturer's tolerance in Appendix B, Tables 2.1-2 and 2.1-3.
4. Clarify dose evaluation for stainless steel replacement and dummy rods in Appendix B, Tables 2.1-2 and 2.1-3.

Although the applicant originally proposed additional changes, including removing the requirement of pressure test and helium leaking test and removing the burnup calculation in CoC Appendix B, Section 2.4.3, the applicant withdrew these proposed changes in letters dated April 22, 2016 (Holtec, 2016e), and September 8, 2017 (Holtec, 2017a), respectively. Therefore, the staff did not review those proposals.

This amended CoC, when codified through rulemaking, will be denoted as Amendment No. 11 to CoC No. 1014.

In performing the review and evaluation of the proposed amendment, the staff followed the guidance in NUREG-1536, Revision 1, "Standard Review Plan for Dry Cask Storage Systems at a General License Facility," July 2010 (NRC, 2010). The staff's evaluation is based on a review of Holtec's application and supplemental information and 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 changes requested in Amendment No. 11 and did not reassess previous revisions of the final safety analysis report (FSAR) nor previous amendments to the CoC.

1.0 GENERAL INFORMATION EVALUATION

The applicant did not propose any changes that affect the staff's general information evaluation provided in the previous SERs for CoC No. 1014, Amendments Nos. 0 through 10. Therefore, the staff determined that a new evaluation was not required.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

The applicant did not propose any changes that affect the staff's principal design criteria evaluation provided in the previous SERs for CoC No. 1014, Amendments Nos. 0 through 10. Therefore, the staff determined that a new evaluation was not required.

3.0 STRUCTURAL EVALUATION

The staff reviewed the changes proposed in Amendment No. 11 to the HI-STORM 100 storage system to ensure that the applicant performed adequate structural analyses to demonstrate the system would be acceptable under normal and off-normal operations, accident conditions, and natural phenomena events. In conducting this evaluation, the staff seeks reasonable assurance that the cask system will maintain confinement, subcriticality, radiation shielding, and retrievability or recovery of the fuel, as applicable, under all credible loads for normal and off-normal conditions accidents, and natural phenomenon events.

There is one proposed change that require staff's structural evaluation:

(1) Proposed Change #1 - Increase the storage location weight limit from 730 lbs. to 830 lbs. for cells authorized for DFCs in the MPC-68, MPC-68FF and MPC-68M in the HI-STORM 100 System.

3.1 Revised Fuel Assembly Weights

The applicant performed a structural analysis with the proposed fuel weight of 830 lbs., as shown in the proposed SAR Table 2.1.5, to demonstrate the structural adequacy of the fuel basket in the HI-STORM 100 cask. The method of analysis was identical to the method used to qualify the MPC-68 enclosure vessel and fuel basket for the previous HISTORM 100 FSARs, and was previously reviewed and accepted by the staff. The applicant used the finite element code ANSYS to carry out a 2-D quasistatic analysis of the MPC-68 fuel basket under a 45-g lateral impact load for two basket orientations (i.e., 0 and 45 degrees).

The staff reviewed the applicant's structural analysis and its results, and found that the results, as documented in Holtec's proprietary Supplement No. 66, Revision 0 (Holtec, 2016b Enclosure 1), show the minimum safety factor is 1.02, which is above the ASME Boiler and Pressure Vessel Code allowable safety factor of 1.0. Thus, the staff found the proposed new fuel assembly weight of 830 lbs. acceptable.

In support of this amendment, the applicant proposed to change the lifting acceptance criterion from one-sixth to one-third of the material yield strength. This is acceptable because one-third of the material yield strength is consistent with the guidance in Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask," (NRC, 2014a).

3.2 Evaluation Findings

F3.1 The SAR adequately describes all SSCs that are important to safety, providing drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

F3.2 The applicant has met the requirements of 10 CFR Part 72.236(b). The SSCs important to safety are designed to accommodate the combined loads of normal or off-normal operating conditions and accidents or natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask 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 found to be within limits of applicable codes, standards, and specifications.

F3.3 The applicant has met the requirements of 10 CFR Part 72.236(c) for maintaining subcritical conditions. The structural design and fabrication of the dry cask storage system includes structural margins of safety for those SSCs important to nuclear criticality safety. The applicant has demonstrated adequate structural safety for the handling, packaging, transfer, and storage under normal, off-normal, and accident conditions.

F3.4 The applicant has met the requirements of 10 CFR 72.236(l), "Specific Requirements for Spent Fuel Storage Cask Approval." The design analysis and submitted bases for evaluation acceptably demonstrate that the cask and other systems important to safety will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The applicant has met the specific requirements of 10 CFR 72.236(g) and (h) as they apply to the structural design for spent fuel storage cask approval. The cask system structural design acceptably provides for the following required provisions:
a. Storage of the Spent Fuel for the 20 year term specified in the certificate.
b. Compatibility with Wet or Dry Loading and Unloading Facilities.

Based on the review of the applicant's description, proposed design criteria, appropriate use of material properties and adequate structural analysis of the relevant structures, systems and components, the staff concluded that the SSCs of the HI-STORM 100 storage system are in compliance with 10 CFR Part 72 regulations. The evaluation of the structural properties provides reasonable assurance that the HI-STORM 100 storage system will allow safe storage of spent nuclear fuel (SNF) for the licensed term.

4.0 THERMAL EVALUATION

4.1 Review Objective

The staff reviewed the changes proposed in Amendment No. 11 to the HI-STORM 100 storage system to ensure that the applicant had performed adequate thermal evaluation to ensure that the cask and fuel material temperatures will remain within the allowable values or criteria for normal, off-normal, and accident conditions. Staff's review and evaluation include confirmation that the temperatures of the fuel cladding will be maintained throughout the storage period to protect the cladding against degradation that could lead to gross rupture.

There are five proposed changes that require staff's thermal evaluation:

(1) Proposed Change #2 – Change surveillance requirements for casks with certain heat load as specified in the TS.
(2) Proposed Change #3 – Add the storage of higher average initial enrichment wt.% U-235 fuel with low enriched CILC fuel.
(3) Proposed Change #6 – Increase the burnup limit to accommodate NFH consisting of NSA in combination with other control components.
(4) Proposed Change #7 – Add thoria rods/canister as contents for the MPC-68M.
(5) Proposed Change #8 – Add a second permissible composition for thoria rods for all MPC-68 models.

4.1 Surveillance Frequency

The applicant proposed requested changes to the Technical Specification (TS), Appendix A, Section 3.1.2, surveillance requirement (SR) 3.1.2 and Table 3-5. The TS surveillance requirement is to verify overpack inlets and outlets are free of blockage from solid debris or floodwater. The TS surveillance requirement is also for overpacks with installed temperature monitoring equipment to verify that the difference between the average overpack air outlet temperature and ISFSI ambient temperature is ≤ 155°F for overpacks containing PWR MPCs, and ≤ 137°F for overpacks containing BWR MPCs. The TS surveillance frequency requirement for SR 3.1.2 was 24 hours and the applicant proposed to change the requirement to 30 days in the TS, Appendix A, Section 3.1.2 for overpacks containing MPCs with heat loads that are less than or equal to 18 kW for MPC-68 (including MPC-68F/FF/M) and MPC-24 (including

MPC-24E/EF) and 16 kW for MPC-32 (including MPC-32F) at the time of inspection. The applicant did not propose changing the TS surveillance frequency requirement for SR 3.1.2 from 24 hours for overpacks containing MPCs with heat loads greater than 18 kW. Similarly, the applicant requested to change the TS completion time to perform the required action to restore the HI-STORM 100 cask heat removal system to operable status (should the heat removal system be deemed to be "inoperable") from 8 hours to 24 hours for overpacks containing MPCs with heat loads less than or equal to 18 kW (MPC-68 and MPC-24) and 16 kW (MPC-32), respectively, at the time of entering the condition. The applicant did not propose changing the TS completion time to perform the required action to restore the spent fuel storage cask heat removal system to operable status from 8 hours for overpacks containing MPCs with heat loads greater than 18 kW (MPC-68 and MPC-24) and 16 kW (MPC-32), respectively, at the time of entering the condition.

In support of those requested changes the applicant performed a steady-state thermal analysis of the HI-STORM 100 with 100% blocked inlet vents with the MPC-68 basket (bounds MPC-68 variants and MPC-24 variants [Holtec 2016b Enclosure 2]) for a threshold decay heat of 18 kW at 80°F ambient conditions and the MPC-32 baskets for a threshold decay heat of 16 kW at 80°F ambient conditions. The applicant described this analysis in Section 4.6.2.4 of the application and provided the maximum fuel and component temperatures in Table 4.6.9 of the application. The reported temperatures illustrated in Table 4.6.9 of the application were below accident conditions temperature limits, with the majority of the components, including fuel cladding. Specifically, the reported calculated fuel cladding temperature during accident conditions was below the normal condition limit of 400°C (752°F) and significantly below the accident limit of 570°C (1,058°F), consistent with the provision of Spent Fuel Storage and Transportation (SFST)-Interim Staff Guidance (ISG)-11, Revision 3 (NRC, 2003).

Based on the original amendment application, the staff performed a confirmatory steady-state thermal analysis of the HI-STORM 100 with 100% blocked inlet vents with the MPC-32 and MPC-68 baskets at the threshold decay heat of 19 kW at 80°F ambient temperature and confirmed that the maximum fuel and component temperatures were below accident conditions temperature limits, but some components were not below normal conditions limits. The results of the staff's confirmatory analysis also showed the HI-STORM 100 with the MPC-32 basket had fuel and component reported temperatures that were bounding compared to the HI-STORM 100 with the MPC-68 basket. Therefore, the staff disagreed with the applicant that the thermal analysis of the HI-STORM 100 with the MPC-68 basket is essentially the same as the MPC-32 or bounds the other MPC types. Subsequently, in response to requests for additional information (RAI) (Holtec, 2017a), the applicant provided thermal models for the MPC-68 with a threshold decay heat load of 18 kW and the MPC-32 with a threshold decay heat load of 16 kW. Staff's review of these models found that overall the models were acceptable for demonstrating component integrity under the threshold thermal load. The staff reached the conclusion because the review of the models showed reasonable modeling choices and the results of the modeling illustrated that the component temperatures were significantly below the accident condition allowable temperatures. The one exception to this conclusion is discussed in the next section.

4.1.1 Computational Fluid Dynamics Modeling Code Versions

The applicant indicated that it performed a sensitivity study for the MPC-68 that evaluated reported component temperature results when run in FLUENT 6.3.26 and ANSYS FLUENT 14.5.7. The applicant's results indicated that the reported temperatures for all components either remained unchanged or decreased by no more than 3.6°F (2°C).

The staff performed a convergence evaluation on the revised models submitted for the MPC-68 and the MPC-32 canisters. The results confirmed that with a restart analysis of the MPC-68 model using either FLUENT 6.3.26 or ANSYS FLUENT 14.5.7, no marked changes were observed for peak cladding temperatures. The staff performed a restart analysis with the MPC-32 model using ANSYS FLUENT 18.2 and found that beyond 22,000 iterations, where the applicant terminated the evaluation, the peak cladding temperature did not remain constant. This indicated a possible convergence problem.

A restart analysis of FLUENT 6.3.26 for the MPC-32 did not indicate any deviations for peak cladding temperature. The NRC recognized that ANSYS FLUENT version 18.2 was employed for MPC-32 rather than ANSYS FLUENT 14.5.7; however, there is a reasonable expectation that version 18.2 should function in a similar way to version 14.5.7 when merely restarting the analysis. Since the MPC-68 convergence evaluation demonstrated consistency between code versions, it is reasonable to assume that the same should be true for the MPC-32 case. Given this fact and that the peak cladding temperatures are still significantly below the accident condition temperature limits (as well as most normal condition temperature limits), the staff finds it acceptable to use ANSYS FLUENT 14.5.7 for the MPC-32 model for this amendment. For future licensing actions, the staff may restrict the use of ANSYS FLUENT 14.5.7 for the MPC-32 model until the anomaly is satisfactorily explained.

4.1.2 Concurrent Events

a. Pressure (Fuel Rod Rupture)

The applicant evaluated fuel rod rupture concurrently with 100% blocked vents at the threshold decay heat(s) and found that the reported temperatures of the fuel cladding are all below the normal condition temperature limit of 400°C (752°F), consistent with SFST-ISG-11, Revision 3. Thus, the applicant stated that there is no credible event that will cause the fuel rods to rupture, thereby increasing MPC cavity pressure.

The staff reviewed the applicant's analysis and noted that in nearly all the cases where 1% fuel rod rupture was assumed in the pressure calculations, the MPC cavity pressure exceeded the normal condition pressure limit by up to 10%. However, each reported pressure was significantly below the accident condition limit of 200 psig. Because this concurrent event scenario is defined as an accident condition, and the pressure limit for accident condition is not exceeded, the staff concludes that there is reasonable assurance that the MPC confinement function remains intact under accident conditions.

b. Off-Normal Ambient

The applicant evaluated an off-normal ambient temperature of 100°F, which corresponds to a 20°F temperature increase over normal condition ambient temperatures. The staff notes that the results illustrated in FSAR Table 4.6.12 demonstrate that a significant margin exists for this temperature excursion from the fuel cladding off-normal temperature limit of 570°C (1,058°F). Because there is a significant margin in reported results, the staff concludes that there is reasonable assurance that fuel cladding remains intact during an off-normal ambient temperature excursion.

c. Fire

The applicant evaluated a concurrent fire accident with 100% blocked vents and asserted that the fire event for the maximum design basis heat load is significantly more thermally challenging than the temperatures produced by decay heat alone. The reported temperatures for the lower threshold decay heat loads (16 kW and 18 kW) are below normal condition temperature limits for peak cladding temperature, and thus below the maximum design basis decay heat used for previous approved fire accident analysis. Therefore, the staff determined that the previously approved maximum design basis heat load fire temperatures would bound a fire for the 100% blocked vent case at a threshold decay heat load (16 kW and 18 kW).

d. Extreme Environmental Temperature

The applicant evaluated an extreme ambient temperature of 125°F, which corresponds to a 45°F temperature increase over normal condition ambient temperatures. The staff notes that the results illustrated in Table 4.6.13 demonstrate that a significant margin exists for this temperature excursion from the fuel cladding accident temperature limit of 570°C (1,058°F). Because there is a significant margin in reported results, the staff concludes that there is reasonable assurance that fuel cladding remains intact during an Extreme Environmental ambient temperature excursion.

e. Burial Under Debris

The applicant evaluated burial under debris concurrently with 100% blocked vents at the threshold decay heat load. Based on the reported temperatures for the 100% blocked vent case with the maximum design basis heat load, the staff determined that the evaluation for the previously approved maximum design basis heat load remains bounding.

As described above, the staff found that the TS changes in Appendix A, Section 3.1.2 are acceptable based on the reported fuel and component temperatures for the HI-STORM 100 with 100% blocked inlet vents with the MPC-68 at the threshold decay heat limit of 18 kW, the MPC-32 at the threshold decay heat limit of 16 kW, and the MPC-24 at the threshold decay heat limit of 18 kW.

4.2 CRUD-Induced Localized Corrosion

The applicant requested to store low enriched CILC BWR fuel with normal fuel. The applicant stated CILC BWR fuel does not require placement in DFCs and that the fuel complies with the intact fuel assembly heat load limits specified in Appendix B, Table 2.4-1 for the MPC-68M. Because the low enriched CILC BWR fuel must comply with the intact fuel assembly heat load limits, the staff concludes that inclusion of the low enriched CILC BWR fuel does not present a deviation from the safety conclusions drawn for the evaluation of the MPC 68M.

4.3 Non-Fuel Hardware

The applicant requested to increase the burnup limit to accommodate non-fuel hardware consisting of neutron source assemblies in combination with other control components. CoC Appendix B, Section 2.4.4 for the HI-STORM 100 cask requires that users must account for the decay heat from both the fuel assembly and any non-fuel hardware to ensure the decay heat emitted by all contents in a storage location does not exceed the decay heat limit. Because the decay heat must remain bounding with the inclusion of the non-fuel hardware as specified in CoC Appendix B, Section 2.4.4, the staff found that the inclusion of non-fuel hardware will not present a deviation from the safety conclusions drawn in Sections 4.4, 4.5, and 4.6 of the FSAR.

4.4 Thoria Rod Canisters

The applicant requested the addition of one Dresden Unit 1 thoria rod canister as contents for the MPC-68M, as well as a second composition for thoria rods for all MPC-68 models. The NRC has previously approved the maximum allowable decay heat limit for a thoria rod canister in MPC-68F/68/68FF of 115 Watts as described in CoC Appendix B, Table 2.1-1, items II.A.7.d and III.A.3.d. The maximum allowable decay heat limit for damaged fuel and fuel debris stored in a canister in an MPC-68/68FF/68M is 393 watts, which is described in CoC Appendix B, Table 2.4-1. The applicant proposed a maximum decay heat for the thoria rod canister in MPC-68M of 115 watts, which is described in CoC Appendix B, Table 2.1-1, item VI.A.3.d. Because the maximum allowable decay heat for the thoria rod canister is less than the maximum allowable decay heat for damaged fuel and fuel debris in an MPC-68/68FF/68M and the NRC has previously approved the maximum allowable decay heat for a thoria rod canister in MPC-68F/68/68FF, the staff found the reported fuel and component temperatures for an MPC-68M that includes the thoria rod canister are bounded by the previously calculated temperatures in Sections 4.4, 4.5, 4.6, and Supplement 4.III of the FSAR, and therefore acceptable.

The current approved thoria rod composition is 98.2 wt.% thorium oxide (ThO_2), 1.8 wt.% uranium dioxide (UO_2) with an enrichment of 93.5 wt.% U-235. The applicant stated the second composition for thoria rods (98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5 wt.% U-235) has no impact on the hoop stress, hydride reorientation, or cladding properties. As discussed in Section 8.3 of this SER, ThO_2 offers some advantages. Therefore, the staff found that the reported fuel and component temperatures for an MPC-68/68M/68FF/68F that includes the second composition for thoria rod canister are bounded by the previously calculated temperatures in Sections 4.4, 4.5, 4.6, and Supplement 4.III of the FSAR, and therefore acceptable.

4.5 Alternate Computational Method for Site-Specific Conditions

The applicant also included changes in the FSAR updating the FLUENT thermal models described in Chapter 4 of the FSAR to evaluate: 1. the time for water within the MPC to boil using site-specific conditions, 2. the HI-STORM site-specific fire accident event, and 3. the HI-TRAC site-specific fire accident event.

The FLUENT thermal models described in Sections 4.5.1, 4.III.5.1, 4.4, 4.III.4, 4.5, and 4.III.5 of the FSAR have not changed in this amendment.

The applicant identified the following computational modeling additions:

(1) An alternate method using the FLUENT thermal model described in FSAR Section 4.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions.
(2) An alternate method using the FLUENT thermal model described in FSAR Section 4.III.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions.
(3) An alternate method using the FLUENT thermal model described in FSAR Section 4.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM Flood/Wind (FW) FSAR.
(4) An alternate method using the FLUENT thermal model described in FSAR Section 4.III.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in FSAR Section 4.6.
(5) An alternate method using the FLUENT thermal model described in FSAR Section 4.5 can be adopted to evaluate HI-TRAC site-specific fire accident event.
(6) An alternate method using the FLUENT thermal model described in FSAR Section 4.III.5 can be adopted to evaluate HI-TRAC site-specific fire accident event.

The applicant provided Tables 4.5.10, 4.6.10, and 4.6.11 that described modeling steps to be used with existing the FLUENT thermal models referenced above in addition to relevant acceptance criteria for used for site-specific conditions. The staff reviewed Tables 4.5.10, 4.6.10, and 4.6.11 of the application and found the modeling steps consistent with previously used modeling approaches therefore the revised models are acceptable for use with the site-specific conditions described in these tables.

4.6 Evaluation Findings

F4.1 The staff has reasonable assurance that the structures, systems, and components (SSCs) important to safety are described in sufficient detail in Chapter 4 of the SAR to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

F4.2 The staff has reasonable assurance that the HI-STORM 100 is designed with a heat removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The staff has reasonable assurance that the spent fuel cladding is protected against degradation leading to gross ruptures by maintaining the cladding temperature below maximum allowable limits in a helium gas environment in the cask cavity under normal, off-normal, and accident storage conditions. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.4 The staff concluded that the thermal design of the HI-STORM 100 is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the HI-STORM 100 will allow safe storage of spent fuel for the licensed life. This finding is reached based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5.0 CONFINEMENT EVALUATION

The staff reviewed the changes proposed in Amendment No. 11 to the HI-STORM 100 storage system to ensure that the applicant had performed adequate confinement evaluation to ensure that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross rupture.

There are two proposed changes that require staff's confinement evaluation:

(1) Proposed Change #2 – Change surveillance requirements for casks with certain heat load as specified in the TS.
(2) Proposed Change #3 – Add the storage of higher average initial enrichment wt.% U-235 fuel with low enriched CILC fuel.

Staff did not review the initial proposed changes to revise (1) the helium leak testing of the confinement boundary welds, or (2) the requirement to hydrostatically test the MPC. The applicant withdrew these two proposed changes in a letter dated April 22, 2016 (Holtec, 2016e).

5.1 Surveillance Frequency

The staff reviewed the request to change the TS, Appendix A, Section 3.1.2, surveillance frequency requirement, SR 3.1.2. The confinement boundary of MPC lid includes the MPC port cover plates, MPC closure ring, and MPC baseplate. The temperatures of these components are provided in Tables 4.6.9, 4.6.12, and 4.6.13 of the application, and illustrate that the confinement components are significantly below accident conditions limits, while only being marginally above normal condition limits listed in FSAR Table 2.2.3, Design Temperatures.

In Section 4.6.2.4 of the application, the applicant provided the calculated MPC pressures, which peaked at 110.6 psig, based on the average gas temperature from the steady-state thermal analysis of the HI-STORM 100 with 30-day 100% blocked inlet vents accident with the MPC-68 basket for a threshold decay heat of 18 kW. This pressure exceeds the normal design pressure, 100 psig, but does not exceed the accident design pressure, 200 psig.

The staff found that the TS changes in Appendix A, Section 3.1.2 are acceptable based on the reported fuel and component temperatures for the HI-STORM 100 with 100% blocked inlet vents for the MPC-68 and MPC-24 (and variants described in the TS) at the threshold decay heat limit of 18 kW and the MPC-32 (and variants described in the TS) at the threshold decay heat limit of 16 kW.

5.2 Mixture of CLIC Fuel and Normal Fuel

The applicant requested to store low enriched CILC BWR fuel with normal fuel. The applicant stated CILC BWR fuel does not require placement in DFCs which staff confirmed in the HI-STORM FW MPC, CoC No. 1032, Amendment No. 1 (NRC, 2014b), where an identical request to store CILC fuel was approved. Therefore, the staff found that 10 CFR 72.122(h)(1) is met.

5.3 Evaluation Findings

F5.1 The design of the HI-STORM 100 adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4, "Thermal Evaluation" of the SER discusses the relevant temperature considerations.

F5.2 The staff concluded that the design of the confinement system of the HI-STORM 100 is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the HI-STORM 100 will allow safe storage of spent fuel. This finding is reached based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis, and accepted engineering practices.

6.0 SHIELDING EVALUATION

The staff reviewed the changes proposed in Amendment No. 11 to the HI-STORM 100 storage system to ensure that the radiation shielding features are sufficient to meet the operational dose requirements of 10 CFR 72.104 and 72.106 in accordance with 10 CFR 72.236(d). The staff evaluated the shielding performance of the dry cask storage system under normal and off-normal operations, accident conditions, and natural phenomena events as they related to the proposed changes below.

The applicant proposed three changes that impact the shielding design of the HI-STORM 100 storage system:

(1) Proposed Change #6 – Permit loading of combination of NFH and other components (including NSA), and maintain the total design basis source term to increased burnup and cooling time.
(2) Proposed Change #7 – Add irradiated thoria rod design in thoria rod canister as contents for the MPC-68M.
(3) Proposed Change #8 – Add a new thoria rod type that has a material composition that is different from the current allowable thoria rod for all MPC-68M models.

6.1 Increase Burnup Limit

The applicant requested to increase the burnup limit in order to accommodate NFH contents, including NSA, in combination with other control component (such as burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), or rod cluster component assemlbies (RCCAs)). The applicant proposed to increase the burnup limit for these NFHs. In order to maintain the Co-60 activity limit at or below 895 curies, the corresponding increase in cooling time is made. Holtec initially analyzed the burnup limit based on Co-60 activity limit in HI-STORM 100 Amendment No. 2 (NRC, 2005), where the Co-60 activity limits for the BPRAs were chosen as 895 curies.

The staff reviewed Holtec's proprietary shielding calculation package HI-951322 (Holtec, 2016b Enclosure 3) for the proposed new burnup limit and associated cooling time increases. The applicant calculated burnup, cooling time, and Co-60 activity NFH to determine acceptability of the new contents for storage. In Amendment No. 2, the applicant chose the allowable burnup based on linear interpolation of the data in "Final BPRA Data" to determine minimum cooling time (13.1 years) that produces an activity of 895 curies of Co-60 with a maximum burnup of

60,000 MWD/MTU. The corresponding decay heat was also calculated by linear interpolation. In Amendment No. 11, an extension on the interpolated burnup and cooling times from Amendment No. 2 is proposed. The applicant used linear data interpolation to determine the burnup and cooling time combination that produces an activity of 895 curies of Co-60 with a maximum burnup of 360,000 MWD/MTU.

The applicant performed ORIGEN-S calculations using fluxes taken from SAS2H calculations to irradiate the steel/inconel material. The staff reviewed the inputs and outputs of these calculations and found them to be acceptable. The staff increased burnup in conjunction with the increased cooling time listed in CoC Appendix B Table 2.1-8 and FSAR Table 2.1.25 to keep the source terms the same as previously approved. Because the source terms are kept the same, the dose at the controlled area boundary and dose rates around the transfer canister will remain unchanged. Therefore, the system with the new content continues to meet the regulatory limits established in 10 CFR 72.104 and 72.106. On this basis, the staff determined the proposed change to be acceptable with respect to shielding design.

The staff reviewed the method used by the applicant to demonstrate that the required cooling time for the increased burnup (maximum burnup of 360,000 MWD/MTU) will produce the same amount of Co-60 activity (895 curies). The staff finds the method used by the applicant to determine the minimum cooling time acceptable because previous studies have demonstrated that the gamma source has a linear relationship with burnup. Therefore, using linear interpolation to determine the required cooling time for the maximum allowable burnup is acceptable for NFH (including NSA) in combination with non-fuel components, such as BPRAs, TPDs, or RCCAs, as shown in the Final BPRA Data table. The staff found the proposed contents and the method of determining required cooling time for a given burnup using linear interpolation to be acceptable.

6.2 Thoria Rods

The applicant proposed to add a new thoria rod design to be stored in a thoria canister as contents for the MPC-68M. According to the applicant, the reason for this proposed change is to expand the allowable contents for the MPC-68M to include the thoria rods/canister, which are already approved contents for the other MPC-68 models. The new thoria rod design has the same geometric characteristics as the previously approved thoria rod design but has slightly different material composition. Specifically, the new thoria rod is made of 98.5 wt% ThO_2 and 1.5 wt% of UO_2 at 93.5 wt% U-235 enrichment.

To verify whether thoria rods can be stored in the MPC-68M, the applicant performed source term and dose rate calculations for thoria rods. Source term and dose rate calculations were provided in the proprietary shielding calculation package HI-951322 (Holtec, 2016b Enclosure 3). The applicant made two separate source term calculations. The first one used SAS2H with an entire ThO_2 fuel assembly and the second calculation was with an entire UO_2 fuel assembly. The applicant compared the spectra from both ThO_2 and UO_2 fuels with the spectra of the design basis 6x6 fuel assembly, and the results showed that the estimated gamma and neutron dose rate from the design basis 6x6 fuel are greater than the estimated dose rate from thoria rods canister. Only one thoria rod canister and 18 thoria rods per canister will be allowed to be stored in the MCP-68M. The cooling time used in the analysis was up to 20 years.

The applicant presents its source term calculation results in Tables 5.2.37 and 5.2.38 for gamma and neutron sources, respectively. The results show that with the same burnup and cooling time, the new thoria rod design has a slightly lower gamma source and slightly higher neutron source in comparison with the previously approved thoria rod design.

The staff reviewed the applicant's source term and shielding calculations and finds that the assumptions are conservative for calculating the dose at the controlled area boundary. In particular, the staff notes that the applicant used 18 years cooling in its source term calculation for the thoria rods whereas the required cooling time is 20 years. The required 20 year cooling time assures that the thalium-208, which is a daughter product of U-232 in the thoria rods, has passed its peak radiation level. The staff finds that this is a conservatism in source term and subsequent dose and dose rate calculations.

Based on the conservatisms of the depletion parameters used and potential uncertainties in fuel burnup, the staff finds, with reasonable assurance, that the cask design with the new contents continues to meet the regulatory requirements of 10 CFR 72.236(d).

6.3 Stainless Steel Replacement/Dummy Rods Dose Rate Evaluation

In Section 5.4.10 of the SAR, the applicant stated that the analyses are performed for the design basis burnup, enrichment, and cooling time combinations listed for the tables in Section 5.1 of the SAR. A shorter cooling time was assumed for all assemblies, specifically those on the periphery of the basket. To represent a realistic loading configuration, a regionalized loading was selected by the applicant where shorter cooled and lower burned assemblies would be placed on the periphery for ALARA purposes. The applicant also stated that for the dose rate effect with fuel assemblies containing dummy rods and ALARA considerations, assemblies with steel rods would be allocated and loaded in the cells located in the inner areas of the basket, not on the periphery. According to the applicant, in this case, the radiation from the steel rods would be shielded by the outer assemblies that do not contain any dummy rods, which would significantly reduce or even eliminate any impact on external dose rates.

The staff performed source terms calculations to confirm the applicant's analysis using SCALE6.2/TRITON depletion code. The staff analyzed two different inputs, one containing only UO_2 fuel rods and the second input containing stainless steel rods replacing some fuel rods. The comparison between these two configurations shows a decreased in the gamma source terms. The staff concluded that fuel assemblies containing irradiated replacement or dummy rods are acceptable for storage in the HI-STORM 100 Cask System. The staff also found that any number of fuel rods in an assembly can be replaced by irradiated or unirradiated steel or Zirconia rods. This is because unirradiated steel rods are not activated and do not produce any radiation, and the irradiated zirconia rods have a very low neutron absorption cross section and thus low activation and no significant radioactivity.

The applicant used MCNP 4A computer code in performing shielding calculations. The applicant used the 1977 version of ANSI/ANS 6.1.1 standard for flux-to-dose conversion factor. The staff finds that the computer code and flux-to-dose conversion factor meet the acceptance criteria as recommended in NUREG-1536, Revision 1.

The staff has reviewed the changes requested in this amendment to the HI-STORM 100 Cask System that impact the shielding evaluation. Based on the statements and representations in the application, as supplemented, and Revision 13 of the SAR, the staff concludes that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 72.104 and 72.106 in accordance with 72.236(d) when loaded under the proposed conditions in Appendix B, Table 2.1-1:

(1) Only 18 thoria rods are allowed per MPC-68 or MPC-68M canister.
(2) The thoria rods must be contained in the thoria rod canister and one canister per MPC-68 or MPC-68M canister.

6.4 Evaluation Findings

F6.1 The staff confirmed that the applicant's burnup calculation produced a maximum of 895 curies of Co-60. Therefore, non-fuel hardware which contain a NSA is bound by the non-fuel component such as BPRAs, TPDs, or RCCAs shown in the Final BPRA Data table, and staff found them to be acceptable to store NFH in combination with other control components.

F6.2 The staff found that the applicant sufficiently describes the shielding methodology for the thoria rod/canister. Radiation shielding features of the HI-STORM 100 Cask System are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F6.3 The staff concluded that the design of the radiation protection system of the HI-STORM 100 Cask System can be operated in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM 100 Cask System will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

7.0 CRITICALITY EVALUATION

The staff reviewed the changes proposed in Amendment No. 11 to the HI-STORM 100 storage system to ensure that the applicant had performed adequate criticality evaluation to demonstrate the system will remain subcritical under all credible normal, off-normal and accident conditions encountered during handling, packaging, transfer and storage. The staff's review involved ensuring that the requested changes meet the regulatory requirements of 10 CFR 72.124(a), 72.124(b), 72.236(c), 72.236(g) and the acceptance criteria listed in Section 7 of NUREG-1536.

There are five proposed changes that require staff's criticality evaluation:

(1) Proposed Change #4 – Increase the maximum allowable initial uranium enrichment limit for the 10x10G BWR fuel assembly, in the MPC-68M, from 4.6 wt.% U-235 to 4.75 wt.%U-235.
(2) Proposed Change #7 – Add thoria rods/canister (98.2 wt.% ThO_2, 1.8 wt.% UO_2) as contents for the MPC-68M.
(3) Proposed Change #8 – Add a second permissible composition for thoria rods (98.5 wt.% ThO_2, 1.5 wt.% UO_2) for all MPC-68 models.
(4) Proposed Change #3 – Modify Note 19 of Table 2.1-3 in Appendix B to the CoC and Note 9 of Table 2.III.3 allowing for the storage of higher average initial enrichment wt.% U-235 fuel along with low enriched channeled undamaged fuel.
(5) Proposed Change #5 – Add new minimum soluble boron concentration limits for the 17x17A PWR fuel assemblies for the MPC-32.

7.1 Criticality Design Criteria and Features

The HI-STORM 100 system has three basket designs to accommodate distinct fuel characteristics. The MPC-24 is designed to contain a maximum of 24 PWR fuel assemblies, the MPC-32 contains a maximum of 32 PWR fuel assemblies, and the MPC-68 contains a maximum of 68 BWR fuel assemblies. The external diameters of all the MPCs are identical and can use the same storage and transfer overpack. The MPC-24 design uses flux traps for criticality control while the MPC-32 and MPC-68 designs rely on single neutron absorber plates between each assembly. The MPC-32 additionally relies on soluble boron within the MPC during wet fuel loading and unloading.

7.2 Fuel Specification

The fuel parameters important to criticality safety include an increase in the maximum planar average initial enrichment limit for the 10x10G BWR fuel assembly from 4.6 wt.% U-235 to 4.75 wt.% U-235. Tables 6.III.1.1, 6.III.2.2, 6.III.3.1, 6.III.4.2, 6.III.4.4, and 6.III.4.7 of the amendment request (Holtec, 2016a Attachment 5) list the reactivity effects of increasing the enrichment limit from 4.6 wt.% U-235 to 4.75 wt.% U-235 for the 10x10G fuel assembly. Although the increase in U-235 enrichment resulted in a more positive reactivity effect, the change is still bounded by the design basis.

The applicant also requested that, when loading low enriched channeled undamaged fuel assemblies (CILC fuel enriched up to 3.3 wt.% U-235) into the MPC-68M, all other undamaged fuel assemblies in the MPC be limited to the maximum planar average initial enrichments as specified in Appendix B Table 2.1-3 of the CoC rather than 3.3 wt.% U-235. The applicant's criticality analysis for this change is discussed in the subsequent section.

In addition, the applicant added a second thoria rod composition (98.5 wt.% ThO_2, 1.5 wt.% UO_2) for all the MPC-68 models and expanded the previous allowable thoria content (98.2 wt.% ThO_2, 1.8 wt.% UO_2) to the MPC-68M. The MPC-68 canister models can store a single thoria rod canister, which contains up to 18 thoria rods. In the initial analysis, the applicant modeled the thoria rods as unirradiated and did not include an isotopic depletion and criticality evaluation to account for the production of fissile material due to irradiation (thorium absorbs neutrons to produce U-233, which is fissile and has a higher neutron yield than U-235 or Pu-239). To address this issue, the applicant submitted supplemental information in a letter dated December 22, 2016 (Holtec, 2016d). Although only a single thoria rod canister is qualified for storage in the MPC-68 models, the applicant's December 22, 2016 letter conservatively modeled the thoria canister in every basket cell (68 thoria rod canisters). In addition, the applicant used an upper bound value for the U-233 content that may be produced as a result of the depletion of the U-235 in the initial UO_2 fuel. All the results showed the proposed changes are bounded by the design basis and applicable to both thoria compositions.

Staff reviewed the referenced material and found the applicant's argument acceptable, and concluded that the applicant's thoria fuel rod evaluation provides reasonable assurance of criticality safety.

7.3 Model Specifications

The MPC basket and fuel assembly models used for the criticality safety analysis were the same as those which have been previously reviewed and approved by the NRC staff. The applicant performed full three-dimensional calculations using the Monte Carlo N-Particle code, MCNP4a, and the ENDF/B-V continuous energy cross-section library.

The applicant evaluated the 10x10G BWR fuel assembly with the increased 4.75 wt.% U-235 enrichment using the fuel and basket dimensions within the manufacturing tolerance limits that produce the maximum k_{eff}. The results are reported in FSAR Table 6.III.3.1.

The applicant performed the calculations modeling bare fuel rods without damaged fuel canisters at low enrichment (3.3 wt.% U-235) in arrays of varying sizes occupying all cells within the MPC-68M. The array sizes considered and their corresponding k_{eff} results are listed in FSAR Table 6.III.4.9.

FSAR Table 6.III.4.10 lists the results of the criticality analysis given the proposed consequences of allowing for the storage of higher average initial enrichment wt.% U-235 fuel along with low enriched channeled undamaged fuel (Note 19 of Table 2.1-3 in Appendix B of the CoC and Note 9 of Table 2.III.3 in the FSAR proposed changes). The criticality analysis conservatively modeled normal undamaged fuel mixed with low enriched channeled BWR fuel modeled as a bare fuel rod of varying array sizes. In both tables, the system remains subcritical with the worst case scenario, including consideration of fuel and basket fabrication tolerances and other possible uncertainties and biases in the model and computer code. The models are bounded by the referenced undamaged fuel assembly reactivity, at its maximum planar average initial enrichment in all cells, listed in FSAR Table 6.III.4.2. Therefore, the staff found the changes to Note 19 of Table 2.1-3 in Appendix B of the CoC and Note 9 of Table 2.III.3 in the FSAR are acceptable.

7.3.1 Material Properties

The minimum soluble boron concentration for the PWR MPC during loading and unloading operations are listed in limiting condition for operations (LCO) 3.3.1 of the TS. The applicant has revised this LCO to reduce the minimum required soluble boron for the 17x17A array. In the case of 17x17A intact fuel assemblies, the boron concentration with a maximum initial enrichment of 4.1 wt% U-235 was decreased from 1,900 ppmb to 1,600 ppmb. In the case of 17x17A intact fuel assemblies, with a maximum initial enrichment of 5.0 wt.% U-235, the soluble boron concentration was decreased from 2,600 ppmb to 2,200 ppmb. Furthermore, in the case of one or more damaged 17x17A fuel assemblies or fuel debris, the boron concentration with a maximum initial enrichment of 4.1 wt.% and 5.0 wt.% were each decreased by 300 ppmb to 1,800 ppmb and 2,600 ppmb, respectively. The calculated k_{eff} values as a result of the revised minimum soluble boron concentrations for the 17x17A PWR assembly were updated in FSAR Tables 6.1.5, 6.1.6, 6.1.12, 6.3.2, 6.4.10, 6.4.11, 6.4.14, and 6.C.1. The decrease in minimum required soluble boron resulted in a more positive reactivity effect; however, the results are still bounded by the design basis analysis. Therefore, the staff found the proposed decrease in boron concentrations is acceptable.

7.4 Evaluation Findings

Based on the above statements, the staff has the following evaluation findings with respect to the criticality analysis:

F7.1 Structures, systems, and components important to criticality safety are described in sufficient detail in Chapters 1, 2 and 6 of the HI-STORM 100 MCP Storage System SAR to enable an evaluation of their effectiveness.

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

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

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

The staff concluded that the criticality design features for the HI-STORM 100 are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the HI-STORM 100 will allow safe storage of spent fuel. These findings are reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

8.0 MATERIALS EVALUATION

The staff reviewed the changes proposed in Amendment No. 11 to the HI-STORM 100 storage system to ensure that the applicant had performed adequate materials evaluation to ensure adequate material performance of components important to safety under normal, off-normal, and accident conditions.

There are five proposed changes that require staff's materials evaluation:

(1) Proposed Change #2 – Certain component materials performance under 100% vent blockage event for 30-day of surveillance frequency.
(2) Proposed Change #2 – Re-evaluation of aluminum-based alloy components for Metamic-HT neutron absorber and aluminum shims.
(3) Proposed Changes #7 and 8 – MPC-68M Thoria Rods (ThO_2 and UO_2).
(4) Proposed Change #3 – Mixture of Low Enriched CRUD-Induced Localized Corrosion (CILC) and Normal Fuel.
(5) Proposed Changes #3, 7, and 8 – Fuel performance for thoria fuel, CILC fuel, and intact and undamaged fuel.
(6) Proposed Changes #3, 7, and 8 – Operational safety and materials performance during the drying process.

8.1 Materials Performance under 100% Vent Blockage for 30-Day Surveillance Frequency

The applicant proposed to change the current 24 hour surveillance frequency for MPCs with threshold heat loads that are less than or equal to 18 kW for MPC-68 (including F/FF/M) and MPC-24 (including E/EF) and 16 kW for MPC-32 (including F) at the time of inspection, to a 30 day surveillance frequency. To analyze this change, the applicant assumed an accident condition of 100% vent blockage event and assessed steady state maximum HI-STORM temperatures for 30 days. As discussed in Section 4.1 of this SER, MPC-68 bounds MPC-68 variants and MPC-24 variants, and MPC-32 bounds MPC-32 variants. In a response to RAI (Holtec, 2017a), the applicant provided the steady-state temperatures for MPC-68. The staff's evaluations of the applicant's calculated steady-state component temperatures for a 30-day 100% vent blockage accident are discussed below.

8.1.1 Fuel Cladding

The applicant proposed to change surveillance requirements for casks with certain heat load from 24 hours to 30 days. The staff requested the applicant to provide information on materials performance for high burnup fuel in the event of 100% vent blockage for 30 days to ensure each component would continue to adequately perform its safety functions after exposure to elevated temperatures during the extended vent blockage. The applicant provided the design temperature limits in SAR Table 2.2.3 for 30-day accident condition and steady state maximum temperature and pressure at threshold heat load in FSAR Table 4.6.9. The applicant limited MPCs to the threshold heat loads to avoid exceeding the 752°F (400°C) cladding temperature limit.

The staff used SFST-ISG-11, Revision 3 to evaluate the applicant calculated maximum cladding temperature. SFST-ISG-11, Revision 3 is applicable to the transport and storage of spent fuel, and provides the criteria for no changes to the analyzed fuel configuration as a function of temperature and time.

The SFST-ISG-11, Revision 3, states:

- For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 752°F (400°C) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad).
- For off-normal and accident conditions, the maximum cladding temperature should not exceed 1,058°F (570°C).

The staff considered the creep test results of irradiated zircaloy-4 rod (Einziger and Kohli, 1984; Einziger et al., 1982) when establishing these maximum allowable cladding temperatures. No cladding rupture was observed for test times of 30 and 73 days for low burnup. More recent data also showed that high burnup fuel cladding has low creep rates without rupture (Ito et al., 2004; Hirose et al., 2013). The staff found that the applicant's proposed threshold heat loading for cladding temperature is acceptable, with the support of data available for high burnup fuel.

8.1.2 Concrete and Polymer

The applicant analyzed concrete performance over time, including the extended surveillance time, for the following scenarios: creep, shrinkage, coefficient of thermal expansion, thermal cycling, spalling, dehydration, chemically bound water, physically bound water, concrete composition for the HI-STORM system, dose rate effects, and compressive strength impacts.

The primary function of the plain concrete in the HI-STORM overpack is shielding against gamma and neutron radiation. Concrete in the HI-STORM 100 overpack is considered as a structural member only for missile impact evaluations; the applicant considered a 50% reduction in concrete compressive strength (Carette and Malhora, 1983) to account for degradation in strength when exposed to elevated temperatures caused by various scenarios listed above.

The reduction in shielding with rise in temperature of the concrete and the surrounding SSCs is primarily due to vaporization of volatiles, including the contained moisture present in the concrete. The American Concrete institute (ACI) allowed concrete surface thermal exposure temperature limit is 300°F (150°C) (ACI 349.3R). Temperatures higher than 300˚F may be allowed for concrete surface if tests are provided to evaluate the reduction in strength, and this reduction is applied to design allowable. At the maximum local temperature experienced by concrete presented in SAR Table 4.6.9, the volume of concrete affected by the temperature gradient above 300°F (150°C) is about 20%. Therefore, the applicant concluded that the bulk of the concrete is unaffected under the 30-day 100% vent blockage event.

The staff evaluated all cited references on concrete performance and conducted confirmatory analyses of the references. Based on these evaluations and analyses, the staff determined the applicant's assessment of concrete performance is acceptable because the bulk of the concrete changes are within design limit.

The staff evaluated the polymer, e.g., Holtite, at the design temperatures 365°F and 399°F (185°C and 204°C) in SAR Table 1.B.3, and found that this temperature range bounds the polymer temperature under off-normal and accident conditions (neutron shield, SAR Table 2.2.3).

8.1.3 Steel

The applicant uses carbon steels (e.g., overpack shell) and stainless steels (e.g., MPC). Carbon steels, per FSAR Table 3.3.2 and ASME Boiler and Pressure Vessel Code Section II Part D, are acceptable for use below 750°F (399°C). The applicant presented the maximum temperatures of overpack shell components (carbon steel) for 30-day 100% vent blockage condition in Table 4.6.9, which were significantly below 750°F (399°C). The austenitic stainless steels are acceptable for use up to 800°F (427°C) per ASME Boiler and Pressure Vessel Code Section II Part D.

As stated in NUREG-2214, Managing Aging Processes in Storage (MAPS) Report, Draft (NRC, 2017), the ferrite present in austenitic stainless-steel welds can transform by spinodal decomposition to form Fe-rich alpha and Cr-rich alpha prime phases, and further aging can produce an intermetallic G-phase during extended exposure to temperatures between 572°F and 752°F (300°C and 400°C). The staff notes that the maximum temperatures of stainless steel components during and after the 30-day vent blockage accident presented in FSAR Table 4.6.9 are below 572°F (300°C). Also, the delta ferrite composition of austenitic stainless steel welds is typically 4 to 15 percent, which is not enough to get significant embrittlement of the stainless steel welds. Therefore, the staff concluded that degradation of austenitic stainless-steel material properties is not credible in the event of 100% vent blockage accident for 30 days.

Based on the applicant's information on the materials performance under 100% vent blockage event for 30-day surveillance frequency, the staff has reasonable assurance the 30-day surveillance frequency would continue to provide adequate safety and meet the requirements of 10 CFR 72.122 and 2.236(f).

8.2 Re-evaluation of Aluminum-based Alloy Components: Metamic-HT Neutron Absorber and Aluminum Shims

The applicant stated that the property changes of Metamic-HT and aluminum shims due to exposure to elevated temperature will not affect the ability of various components to fulfill their safety functions in the event of 100% vent blockage for 30 days. The temperature limits for all components during the 30-day 100% vent blockage event are provided in Table 2.2.3 and the maximum temperatures experienced by components are provided in Table 4.6.9. The applicant indicated that under these temperature limits, both Metamic-HT and aluminum shims would retain properties and functionality during the 100% vent blockage event. The simultaneous vent blockage and other accidents are not taken as credible.

The applicant reassessed Metamic-HT and aluminum shims performance based on existing and new test data and available literature data as described below.

8.2.1 Metamic-HT

Metamic-HT is a neutron absorber with structural function, and the applicant has used it in storage and transportation of SNF. Metamic-HT is a metal matrix composite consisting of an aluminum matrix reinforced by nano-particles of alumina and superfine particles of boron carbide. The applicant derived the fracture toughness of Metamic-HT from the Charpy impact energy measurements. The derivation is based on the correlation of fracture toughness and Charpy impact energy, which was developed for steels from literature. The staff assessed the fracture toughness based on an energy balance, along with available analogue literature data analyses.

The applicant further provided actual measured fracture toughness values, using an ASTM standard practice, over the range of potential operating temperatures. Based on these measurements, the applicant also determined the minimum unstable crack size at each temperature. The staff evaluated both fracture toughness values and minimum unstable crack size at each temperature as discussed below.

- The applicant provided standard deviations of yield stress from the latest version of the proprietary document, “Metamic-HT Qualification Sourcebook” (Holtec, 2017b Attachment 7). The applicant’s formula for calculating the minimum unstable crack size is dependent on the stress demand. The staff found the applicant’s minimum unstable crack size acceptable because stress demand, which is lower than the yield stress, produces more conservative results.

- The applicant used the most critical orientation for a flaw or crack in the manufactured Metamic-HT panel, and noted that under normal storage conditions the fuel basket only supports its own dead weight inducing small compressive stresses in the Metamic-HT panels. The staff determined that the dead weight would not exceed fracture toughness and the applicant’s explanation of the effect of crack orientation is acceptable.

- Previously, the applicant used the conservative 1/16-inch crack size in the Metamic-HT Qualification Sourcebook, and the corresponding safety factor is above 1.0. The latest version of the Metamic-HT Qualification Sourcebook (Holtec, 2017b Attachment 7), which integrates the Metamic-HT fracture toughness measurements, uses the more precise value of 1/32-inch for the maximum undetectable crack size.

Because the applicant is relying upon industry accepted guidance in using 1/32-inch, the staff found the applicant's use of 1/32-inch crack size acceptable.

- The ASTM E1820-15a provides a J-integral (i.e., energy) vs. crack growth resistance (J-R) curve, which allows for evaluation of meaningful testing data at elevated temperatures, e.g., greater than 200°C (392°F). ASTM E1820-15a also states that the strain energy release rate is not influenced significantly by events within the plastic zone if the plastic zone is relatively small and accounted for. Therefore, the staff found the applicant's fracture toughness measurements acceptable with the strain energy release rate at elevated temperature without significant influence by plasticity due to small size of plastic zone.

- The applicant provided fracture toughness measurements for the elastic stress regime below the minimum guaranteed value for the yield stress by extrapolation. The maximum induced primary stress (axial plus bending) in the compact specimen during tensile loading remains below the materials yield stress at 400°C (752°F). Because the staff has previously determined that the below yield stress was acceptable, the staff found the applicant's more conservative/bounding fracture toughness measurements for elastic stress regime acceptable not including plasticity.

- The applicant provided the potential geometric reconfigurations for the non-mechanistic tip-over events and the effect of plastic deformation. According to the applicant, the Metamic-HT fuel baskets do not experience any gross plastic deformation, and the primary stresses in the fuel basket panel remain elastic during the non-mechanistic tip-over and 9-meter drop events. As a result, the applicant contends that the potential localized effect of causing plastic straining would be very limited. Because the basket will be mainly subject to the elastic stress and such stress would be limited, the staff determined that the applicant's analysis of fuel basket panel is adequate.

8.2.2 Aluminum Shims

The applicant assumed the aluminum alloy to be effective for the short duration dynamic loading from the tip-over accident. Aluminum alloy, such as Alloy 2219 used by the applicant, is a precipitation-hardened alloy. The applicant provided the yield stress and tensile stress for aluminum shims at elevated temperature approximately 260°C - 270°C (500°F – 518°F). The applicant provided data that demonstrated that the mechanical properties of precipitation hardened aluminum alloys as a function of time at temperature due to over aging. Based on the staff's independent literature data review (Aluminum Association, 2003), the staff determined that the data provided by the applicant was sufficient to conclude that the mechanical properties of the precipitation hardened aluminum alloy is acceptable for its intended function.

8.3 MPC-68M Thoria Rods (ThO_2 and UO_2)

The applicant proposed to add thoria rods as contents for the MPC-68M and add a second composition for the thoria rod for all MPC-68 models. The applicant stated, from a material perspective, thorium dioxide have:

- higher thermal conductivity (lower operating temperature)

- considerably higher melting point
- chemical stability (unlike uranium dioxide, thorium dioxide does not oxidize in the presence of water/oxygen)
- less thermal expansion than uranium dioxide

The applicant's proposed addition of thoria rods as contents is in compliance with all cladding functional requirements (SFST-ISG-1, Revision 2 and SFST-ISG- 2, Revision 2), especially for the thermal limits as discussed in Section 4.4 of this SER. Therefore, the staff found MPC-68M with the proposed heat load limit of 0.115 kW is acceptable for thermal limits.

8.4 Mixture of Low Enriched CRUD-Induced Localized Corrosion (CILC) and Normal Fuel

The applicant proposed to load low enriched CILC fuel and normal fuel together. The applicant stated that the thermal analyses under normal, off-normal, and accident events remain applicable to CILC fuel because the CILC fuel is undamaged fuel. The staff evaluated the applicant's data and the confirmation that CILC fuel does not have grossly breached spent fuel rods and, therefore, considered as undamaged fuel according to CoC Appendix A, Section 1.1. To ensure that users of the HI-STORM 100 system are aware of the definition of undamaged fuel as related to CILC fuel, the applicant has added a caution note to Section 8.1.4 to alert users. With the definition of "no gross breach" and the applicant's caution note, the staff determined that the materials performance of CILC fuel is acceptable.

8.5 Intact Fuel and Undamaged Fuel

The applicant changed intact fuel to undamaged fuel in functional requirements at several places in the SAR, based on the statement in SFST-ISG-1, Revision 2 (NRC, 2007) that:

> Intact SNF is any fuel that can fulfill all fuel-specific and system-related functions, and that is not breached. Note that all intact SNF is undamaged, but not all undamaged fuel is intact, since under most situations, breached spent fuel rods that are not grossly breached will be considered undamaged.

Both intact and undamaged fuels are required to meet all fuel-specific or system-related functions. In this case, the change from intact fuel to undamaged fuel is needed specifically for loading CILC fuel with normal fuel. Because CILC fuel meets the requirements of undamaged fuel and the applicant has added the caution note (see Section 8.4 of this SER), the staff determined the change from intact fuel to undamaged fuel is acceptable.

8.6 Drying Process

The applicant used an accepted drying process in previous SAR and amendments for all loaded canisters. In this amendment, the applicant proposed to load thoria rods and CILC rods. Both added thoria rods and CILC rods meet the definition of undamaged fuel for the current loading requirements. In addition, the applicant has added a caution note regarding the use of CILC rods (see Section 8.4 of this SER). Therefore, the staff determined that the applicant's use of the previously approved drying process is acceptable.

8.7 Evaluation Findings

F8.1 The staff concludes that the material properties of the structures, systems, and components of the HI-STORM 100 Cask System remain in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the material properties provides reasonable assurance that the cask will allow safe storage of spent nuclear fuel for the licensed life. This finding is reached based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

9.0 OPERATING PROCEDURES EVALUATION

The applicant did not propose any changes that affect the staff's operating procedures evaluation provided in the previous SERs for CoC No. 1014, Amendments No. 1 through 10. Therefore, the staff determined that a new evaluation was not required.

10.0 ACCEPTANCE TESTS AND MAINTANANCE PROGRAM EVALUATION

The applicant did not propose any changes that affect the staff's acceptance tests and maintenance program evaluation provided in the previous SERs for CoC No. 1014, Amendments No. 1 through 10. Therefore, the staff determined that a new evaluation was not required.

11.0 RADIATION PROTECTION EVALUATION

The applicant did not propose any changes that affect the staff's radiation protection evaluation provided in the previous SERs for CoC No. 1014, Amendments No. 1 through 10. Therefore, the staff determined that a new evaluation was not required.

12.0 ACCIDENT ANALYSES EVALUATION

The only proposed change that requires staff's accident analysis evaluation is the change in surveillance frequency requirement in SR 3.1.2 from 24 hours to 30 days for overpacks containing MPCs with heat loads that are less than or equal to 18 kW (MPC-68 and MPC-24) and 16 kW (MPC-32) at the time of inspection. The applicant performed a steady-state thermal analysis of the HI-STORM 100 with 100% blocked inlet vents with the MPC-68 basket for a threshold decay heat of 18 kW at 80°F ambient conditions and the MPC-32 baskets for a threshold decay heat of 16 kW at 80°F ambient conditions. The applicant also analyzed 100% vent blockage concurrently with other events, such as rod rupture and fire. As discussed in Sections 4.1 and 8.1 of this SER, the staff found the surveillance frequency change in Appendix A, Section 3.1.2, acceptable.

13.0 TECHNICAL SPECIFICATIONS AND OPERATING CONTROL AND LIMITS EVALUATION

13.1 Appendix A, LCO 3.1.2 and Table 3-5—Change Completion Time for Required Actions

The applicant proposed "the completion time for actions to restore spent fuel system cask heat removal system to operable" for the Proposed Change #2 of 30-day surveillance frequency for casks with certain heat loads, and tabulate such information in a new Table 3-5. As discussed in Section 4.1 of this SER, the applicant analyzed the HI-STORM 100 with 100% blocked inlet vents with the MPC-68 at the threshold decay heat limit of 18 kW, the MPC-32 at the threshold decay heat limit of 16 kW, and the MPC-24 at the threshold decay heat limit of 18 kW. The reported fuel and component temperatures in SAR Tables 4.6-9, 4.6-12, and 4.6-13 are below the normal condition temperature limit of 400°C (752°F), which is consistent with SFST-ISG-11, Revision 3. Therefore, the staff determined the TS surveillance frequency change presented in Appendix A, Section 3.1.2, Table 3-5 acceptable.

13.2 Appendix A, LCO 3.3.1—Change Boron Concentration

The applicant's Proposed Change #5 reduces the minimum required soluble boron concentration for the 17x17A PWR array in LCO 3.3.1. As discussed in Section 7.3.1 of this SER, the applicant calculated k_{eff} values as a result of the revised minimum soluble boron concentrations. The staff reviewed the results and noted that the decrease in minimum required soluble boron resulted in a more positive reactivity effect. However, because the results are bounded by the design basis analysis, the staff determined that the proposed decrease in boron concentrations is acceptable.

13.3 Appendix A, Table 3-1—Cavity Drying Limits

The applicant made Editorial Change #1 to Table 3-1 to clarify the heat load limit and drying methods. Since the editorial changes do not change the intent of the table, the staff determined the changes acceptable.

13.4 Appendix A, Section 5.5 and Appendix B, Section 3.4—Add NUREG-0612 as a Basis for Stress Limits

In addition to ANSI N14.6, the applicant proposed Editorial Change #2 to add NUREG-0612 as another basis for stress limits in Appendix A, Section 5.5 and Appendix B, Section 3.4. Since NUREG-0612 is an NRC accepted guidance, the staff found this addition acceptable.

13.5 Appendix B, Table 2.1-1—Addition of Thoria Rods to MPC-68M as Contents

In the Proposed Change #7, the applicant proposed to add thoria rods/canister as contents for MPC-68M. In Sections 4.4, 6.2, 7.2, and 8.3 of this SER, the staff evaluated the addition of thoria rods to MPC-68M as contents. From thermal perspective, the reported fuel and component temperatures for an MPC-68M that includes the thoria rod canister are bounded by the NRC previously approved maximum allowable decay heat for a thoria rod canister in other MPC-68 models. With respect to shielding, the total shielding provided by the MPC-68M has not significantly changed from the shielding provided by MPC-68 which was previously evaluated by the staff. In addition, the estimated gamma and neutrons dose rates from the design basis 6x6 fuel are greater than the estimated dose rate from thoria rods canister. The applicant performed criticality analysis by modeling thoria rods in all basket cells. The result showed that the proposed addition of thoria rods to MPC-68M as contents is bounded by the design basis. The applicant also stated that, from material perspective, thorium dioxide offers several advantages, such as higher thermal conductivity, lower operating temperature, higher melting point, and less thermal expansion than uranium dioxide. As explained in Sections 4.4, 6.2, 7.2, and 8.3 of this SER, the staff determined that the addition of thoria rods to MPC-68M as content based on staff's evaluations in thermal, shielding, criticality, and material areas acceptable.

13.6 Appendix B, Table 2.1-1—Addition of New Thoria Rod Composition

In Proposed Change #8, the applicant proposed to add a second permissible composition for thoria rods for all MPC-68 models. In Sections 4.4, 6.2, 7.2, and 8.3 of this SER, the staff evaluated the second composition (98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5 wt.% U-235) for thoria rods for all MPC-68 models. The current approved thoria rod composition is 98.2 wt.% ThO_2, 1.8 wt.% UO_2 with an enrichment of 93.5 wt.% U-235. Thermally, the reported fuel and component temperatures for the second composition for thoria rod canister are bounded by the previously calculated temperatures. Comparing with the currently approved thoria composition, the source term calculations for the new composition of thoria rods has shown to have a negligible impact on dose rates. The applicant performed criticality analysis by modeling thoria rods in all basket cells. The result showed that the proposed new composition is bounded by the design basis. The applicant also stated that, from material perspective, thorium dioxide offers several advantages, such as higher thermal conductivity, lower operating temperature, higher melting point, and less thermal expansion than uranium dioxide. The staff found the addition of second composition of thoria rods to all MPC-68 models acceptable based on staff's evaluations in thermal, shielding, criticality, and material areas.

13.7 Appendix B, Table 2.1-1—Fuel Assembly Weight Increase

The applicant's Proposed Change #1 increases the maximum allowable weight of fuel assembly from 730 lbs. to 830 lbs. As discussed in Section 3.2 of this SER, with the proposed increase in fuel assembly weight, the total cask weight will be 20,823 lbs., which is less than the bounding weight used for the lifting lug analysis previously approved by the NRC. The staff found that the increase in the weight of the fuel assemblies from previously allowed 730 lbs. to 830 lbs. is acceptable because the requested change was bounded by the previously analyzed fuel assembly weight.

13.8 Appendix B, Tables 2.1-2 and 2.1-3—Notes

The current Note 3 to Appendix B, Table 2.1-2 has the statement that "the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer's tolerances." Note 3 to Appendix B, Table 2.1-3 has similar statement for increase up to 1.5 percent. The applicant made the Editorial Change #3 to remove "to account for manufacturer's tolerances" to avoid confusion during implementation. The applicant indicated that many users may have small variations in uranium weight that are within 1.5 or 2 percent, but are not noted as manufacturer's tolerances in the record. Because removal of the phrase does not change its meaning, the staff has determined that the proposed deletion is acceptable.

In the Proposed Editorial Change #4, the applicant also proposed to add Note 11 to Table 2.1-2 and Note 20 to Table 2.1-3. Both notes would allow any number of fuel rods in an assembly to be replaced by irradiated or unirradiated steel or zirconia rods. As discussed in Section 6.3 of this SER, staff's evaluation demonstrated that this replacement does not cause significant changes in dose rates. Therefore, the staff found the addition of Note 11 to Table 2.1-2 and Note 20 to Table 2.1-3 in Appendix B is acceptable.

13.9 Appendix B, Table 2.1-3—Fuel Assembly Enrichment Increase

The applicant's Proposed Change #4 increases the maximum initial planar-average enrichment for the 10x10G fuel assembly class from 4.6 to 4.75 weight percent U-235 in Appendix B, Table 2.1-3. As discussed in Section 7.2 of this SER, this change in maximum initial planar-average enrichment increases the k_{eff} of the cask when loaded with this fuel, but is still bounded by the design basis fuel assembly design. Because this requested change is bounded by the design basis fuel assembly design, the staff determined that the higher initial planar-average enrichment for 10x10G fuel acceptable.

Additionally, the applicant's Proposed Change #3 increases the maximum initial planar-average enrichment of undamaged fuel loaded with low-enriched channeled undamaged fuel (CILC fuel enriched up to 3.3 weight percent ^{235}U) in the MPC-68M. Such undamaged fuel is limited to the maximum planar-average initial enrichments as specified in Appendix B, Table 2.1-3 of the CoC. As discussed in Section 7.3, the applicant's criticality analysis of low-enriched channeled undamaged fuel at 3.3 weight percent U-235 mixed with undamaged fuel at its maximum allowable enrichment specified in Appendix B, Table 2.1-3 of the CoC demonstrates that this system is less reactive than when the cask is fully loaded with undamaged fuel at its maximum allowable enrichment specified in Appendix B, Table 2.1-3 of the CoC. Therefore, the staff determined that the increased initial assembly-average enrichment for undamaged fuel loaded with low-enriched channeled undamaged fuel is acceptable.

13.10 Appendix B, Table 2.1-8—Burnup Limit Increase

In Proposed Change #6, the applicant proposed to increase the burnup limit to accommodate non-fuel hardware consisting of NSA in combination with other control components. As discussed in Section 6.1 of this SER, the staff performed decay heat calculations and determined that the non-fuel hardware which contains a NSA is bounded by the non-fuel component, such as BPRAs, TPDs, or RCCAs. Therefore, the staff determined that the increase in burnup limit is acceptable for non-fuel hardware to store NFH in combination with other control components.

14.0 QUALITY ASSURANCE EVALUATION

The applicant did not propose any changes that affect the staff's quality assurance evaluation provided for CoC No. 1014, Amendments No. 1 through 10. Therefore, the staff determined that a new evaluation was not required.

15.0 CONCLUSIONS

Based on its review of the amendment request to CoC No. 1014, Amendment No. 11, the staff has determined that there is reasonable assurance that: (1) the activities authorized by the amended certificate can be conducted without endangering the health and safety of the public, and (2) these activities will be conducted in compliance with the applicable regulations of 10 CFR Part 72.

November 28, 2018

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SUBJECT: PRELIMINARY SAFETY EVALUATION REPORT, DOCKET NO. 72-1014, HOLTEC INTERNATIONAL HI-STORM 100 MULTIPURPOSE CANISTER STORAGE SYSTEM, CERTIFICATE OF COMPLIANCE NO. 1014, AMENDMENT NO. 11. DOCUMENT DATE: November 28, 2018

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