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

August 8, 2017

Ms. Kimberly Manzione
Licensing Manager
Holtec International
Holtec Technology Campus
One Holtec Blvd.
Camden, NJ 08104

SUBJECT: AMENDMENT NO. 11 TO CERTIFICATE OF COMPLIANCE NO. 1014 FOR THE HI-STORM 100 MULTIPURPOSE CANISTER STORAGE SYSTEM – SECOND REQUEST FOR ADDITIONAL INFORMATION

Dear Ms. Manzione:

By letter dated January 29, 2016 (Agencywide Document Access and Management System (ADAMS) Accession No. ML16029A528), Holtec International (Holtec) submitted an amendment request to the U.S. Nuclear Regulatory Commission (NRC) to revise Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Multipurpose Canister Storage System. Holtec supplemented it on February 16, 2016 (ADAMS Accession No. ML16069A246), and June 6, 2016 (ADAMS Accession No. ML16159A344), and modified it on April 22, 2016 (ADAMS Accession No. ML16113A394). NRC issued a request for additional information (RAI) on November 17, 2016 (ADAMS Accession No. ML16323A118), and Holtec responded to some of the RAIs in a letter dated December 22, 2016 (ADAMS Accession No. ML17005A236).

The NRC staff reviewed the application and Holtec's December 22, 2016, responses to the RAIs, and determined the need for the second RAI in the enclosure to this letter. The staff held a public meeting with Holtec on June 7, 2017, to discuss the staff's position and concerns on the application. Based on Holtec's responses to the first RAI and the discussion at the public meeting, the staff revised some RAIs, added a number of RAIs, and deleted RAIs that are no longer needed in the enclosed second RAI.

We request that you provide the responses to these RAIs within 30 days from the date of this letter. If you are unable to meet this deadline, please notify us in writing, at least one week in advance, of your new submittal date and the reasons for the delay. The staff will then assess the impact of the new submittal date and notify you of a revised review schedule.

Please reference Docket No. 72-1014 and CAC No. L25087 in future correspondence related to this licensing action. If you have any questions, please contact me at 301-415-1018.

Sincerely,

/RA/

Yen-Ju Chen, Sr. Project Manager
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No.: 72-1014
CAC No.: L25087

Enclosure:
HI-STORM 100 Amendment No. 11, 2nd RAI

SUBJECT: AMENDMENT NO. 11 TO CERTIFICATE OF COMPLIANCE NO. 1014 FOR THE HI-STORM 100 MULTIPURPOSE CANISTER STORAGE SYSTEM – SECOND REQUEST FOR ADDITIONAL INFORMATION, DOCUMENT DATE: AUGUST 8, 2017

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Second Request for Additional Information
Docket No. 72-1014
Holtec International
HI-STORM 100 Multipurpose Canister Storage System
Certificate of Compliance No. 1014
Amendment No. 11

By letter dated January 29, 2016 (Agencywide Document Access and Management System (ADAMS) Accession No. ML16029A528), Holtec International (Holtec) submitted an amendment request to the U.S. Nuclear Regulatory Commission (NRC) to revise Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Multipurpose Canister Storage System. Holtec supplemented it on February 16, 2016 (ADAMS Accession No. ML16069A246), and June 6, 2016 (ADAMS Accession No. ML16159A344), and modified it on April 22, 2016 (ADAMS Accession No. ML16113A394). NRC issued a request for additional information (RAI) on November 17, 2016, and Holtec partially responded to the RAI on December 22, 2016.

The staff identified additional information needed in connection with its review of the application as provided in the second RAI discussed below. The staff in its review of the application used NUREG-1536, Revision 1, "Standard Review Plan for Dry Cask Storage Systems at a General License Facility." Each question describes information needed by the staff to complete its review of the application and to determine whether the applicant has demonstrated compliance with regulatory requirements in Part 72 of Title 10 of the *Code of Federal Regulations* (10 CFR).

Chapter 4 Thermal Evaluation

The following information is needed to determine compliance with 10 CFR 72.236(f).

RAI 4-2 (Followup) Revise the Technical Specification (TS), Appendix A, 3.1.2 to address the note in Table 4.6.8 of the application.

Table 4.6.8 of the application provides the multipurpose canister (MPC) threshold decay heat equal to 19 kW, and notes, "The heat load at any storage location in the basket must be less than or equal to the threshold heat load tabulated herein divided by the number of storage locations." The TS, Appendix A, Section 3.1.2 only addresses the concept of the MPC threshold decay heat equal to 19 kW, and does not capture the concept in the note. The note could be more thermally limiting than the threshold decay heat equal to 19 kW.

Based on Holtec's response to this RAI, the decay heat thresholds only apply to the MPC-24, MPC-32, and MPC-68, and do not apply to the MPC-24E, MPC-24EF, MPC-32F, MPC-68F, MPC-68FF, and MPC-68M; this should be clearly captured in the TS. In addition, any different threshold decay heat values (e.g., 16 kW for the MPC-32 which is lower than 19 kW for the MPC-68 and MPC-24) should not be captured in a footnote, but should be clearly part of the surveillance frequency and completion time to minimize the potential for applying a higher decay heat threshold of 19 kW to the MPC-32.

RAI 4-3 (Revised) Provide the following component maximum through-thickness average temperatures in Table 4.6.9 of the safety analysis report (SAR):

(a) MPC lid (including port cover plates).
(b) MPC closure ring.
(c) MPC baseplate.
(d) Overpack lid top plate.
(e) Overpack steel structure (excluding overpack lid top and bottom plates).
(f) Provide confirmation that the maximum through-thickness average innermost overpack concrete temperature for each overpack concrete component has been provided in Table 4.6.9 of the application, or provide in Table 4.6.9 of the application the maximum through-thickness average innermost overpack concrete temperature for each overpack concrete component.

The maximum through-thickness average temperatures of these components were not provided in Table 4.6.9 of the application, therefore it is not clear to the staff whether these components are below their associated design temperatures.

RAI 4-4 (Deleted)

RAI 4-5 (Revised) Provide an evaluation which demonstrates that the MPC-32 and MPC-68 are safe at normal, off normal, and accident conditions for the design basis heat load with 100% blocked vents. At a minimum, this should include a revision to the MPC-68 thermal analysis provided to include the proper solar insolation values referenced in the final safety analysis report (FSAR), if this thermal analysis will be relied upon to demonstrate safety.

The MPC-68 was identified as the bounding thermal case (19kW design basis heat load) for this amendment. It was not sufficiently demonstrated in the analyses and discussion provided that the MPC-68 was the bounding case. Furthermore, analyses results and conclusions drawn from those results were contradictory to other information presented by the applicant regarding the bounding thermal configuration.

Modeling features such as solar insolation, dissimilar decay heat, geometric configurations, and model convergence should all be considered in the evaluation. If a qualitative approach is used in part to demonstrate the safety of a bounding configuration, the evaluation should be sufficiently detailed and technically sufficient in order for the NRC to make a safety determination.

RAI 4-6 (Deleted)

RAI 4-9 (Deleted)

RAI 4-10 (Revised) Demonstrate that the differential thermal expansion for the 100% blocked vent steady-state thermal analysis at the threshold decay heat in Section 4.6.2.4 of the amendment request is bounded by the results in Table 4.4.10 of the FSAR.

The differential thermal expansion of the 100% blocked vent steady-state thermal analysis at the threshold decay heat in Section 4.6.2.4 of the application has not been addressed. During the 30 day period where no passive cooling can occur, component temperatures could be significantly different than those reported for the maximum design basis heat load for the system that had been previously evaluated. A comparison of the component temperatures for both the threshold decay heat for this amendment and the component temperatures for the maximum design basis heat load would confirm whether previous calculations were bounding.

RAI 4-11 (Deleted)

RAI 4-12 (Deleted)

RAI 4-13 (Deleted)

RAI 4-14 (Revised) Provide and completely describe in Section 4.6.2.4 of the application the following regarding the FLUENT model submitted in support of the 100% blocked vent steady-state condition:

(a) Identify the version of the FLUENT code used to obtain results for the FLUENT model for the 100% blocked vent steady-state condition that are reported in Section 4.6.2.4 of the application.
(b) Provide results convergence testing for the model, to demonstrate that the peak component temperatures reported in Section 4.6.2.4 of the application are approaching a true steady-state asymptote, within an acceptable range of convergence.

The FLUENT model submitted in support of the 100% blocked vent steady-state condition does not appear to have been run to an appropriate level of convergence. Staff's evaluations of the FLUENT model and solution indicate that at 19,000 iterations, the temperature field is not converged to within acceptable limits. No explanation or discussion of additional convergence testing was provided in Section 4.6.2.4 of the application that addresses how it was determined the model was verified to be converged and that the peak component temperatures (particularly the fuel peak cladding temperature) are approaching a true steady-state asymptote.

Chapter 5 Shielding Evaluation

RAI 5-1 Include in the TS the range of burnup, enrichment, cooling times, UO_2 mass, and specific power corresponding to the radiation source terms and dose rates for which the storage system is designed, if the equations for calculating burnup limits as a function of cooling time for ZR clad fuel and Tables 2.4.3 and 2.4.4 as described in Section 2.4.3 of the CoC No. 1014 are proposed to be deleted.

In Sections 5.2.5.1 and 5.2.5.2 of the FSAR, the applicant referenced Section 5.2.5.3 to state that the "allowable burnup limits in Section 2.1.9 of the FSAR were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes." The applicant also stated that "design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 14x14A array class (PWR) and 9x9G array class (BWR) were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other PWR and BWR array classes." Also, the applicant stated that "**this approach assures that the calculated source terms and dose rates will be conservative.**" However, the applicant has proposed to delete the equation for calculating burnup limits as a function of cooling time for ZR clad fuel and Tables 2.4.3 and 2.4.4 as described in Section 2.4.3 of the CoC No. 1014 without providing in the TS the range of burnup, minimum enrichment, cooling times, UO_2 mass, and specific power corresponding to the radiation source terms and dose rates for which the storage system is designed. Provide a justification for the removal of the burnup equation and Tables 2.4.3 and 2.4.4 considering that the burnup calculation is an important method used by the General Licensees to ensure compliance with the requirements in the CoC for heat load, burnup, and enrichment.

This information is needed to determine compliance with 10 CFR 72.236(d).

Chapter 8 Materials Evaluation

RAI 8-1 (Revised) Demonstrate that material property changes due to exposure to elevated temperatures will not affect the ability of various components to fulfill their functions in the event of 100% vent blockage for 30 days. Consider structural and shielding functions, as appropriate, for fuel cladding, concrete overpacks, various steel components, aluminum basket shims and the Metamic-HT basket.

The response to RSI 4-7 states that the 100% blocked vent condition falls under the accident temperature limits. The applicant did not provide information on materials performance in the event of 100% vent blockage for 30 days. It is unclear whether materials, as discussed below, for each component, will continue to adequately perform their safety functions when exposed to elevated temperatures during the extended vent blockage.

<u>Fuel Cladding</u>

NRC interim staff guidance (ISG)-11, Revision 3, states "…for low burnup fuel, a higher short term temperature limit may be used, if the applicant can show by calculation the best estimate of stress." This statement applies to low burnup fuel. HI-STORM 100 Amendment No. 11 proposes to store high burnup fuel and integral fuel burnable absorber fuel (with high gas pressure in the cladding gap). The primary cladding degradation mechanism of concern over a potential 30-day blockage event is creep. Very limited data (e.g., Ito, et al., 2004) are available on creep behavior of high burnup fuel cladding although simulated unirradiated cladding were studied with more increasing hydrogen concentration (Koimbaiah, et al., 2014; Mozzani, et al., 2014). The application does not include an assessment of this degradation mechanism.

<u>Concrete</u>

American Concrete Institute (ACI) 349.3R limits the temperature of concrete to 150˚C (ACI; NRC, 1996) for structural purposes. Creep and thermal expansion at elevated temperature may also affect the shielding function of concrete. Considering the potential 30-day vent blockage, the applicant should include the requirements in the SAR, such as Section 2.3.

Carbon Steel and Stainless Steel Components

Mechanical properties of steel and stainless steel components vary with temperature. American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code Section II, Part D and literature data are available on the mechanical properties of steels at various temperatures, as shown in SAR Section 2.2.

In some cases, elevated temperatures may cause a change in microstructure, which has the potential to degrade material properties. For example, the ferrite present in austenitic stainless steel welds can transform to a more brittle intermetallic phase when exposed to temperatures greater than 300˚C (Chandra, 2012).

Aluminum Basket Shims and Metamic-HT Basket

Elevated temperature exposure is known to reduce the strength of aluminum, in many cases, due to microstructural changes (Ferrell, 1995). ASME B&PV Code Section II, Part D provides data on the mechanical properties of aluminum alloys at various temperatures.

RAIs 8-3 and 8-4 addressed potential degradation of mechanical properties of aluminum basket shims and the Metamic-HT basket under normal conditions. The same evaluation should be performed for the 30-day vent blockage condition.

Therefore, as discussed above, the staff requests the applicant provide justification that potential elevated temperature, due to the 30-day surveillance interval and one-day recovery, will not degrade the capability of materials to support the important-to-safety functions of the storage system components.

This information is needed to determine compliance with the requirements of 10 CFR 72.236(b), 72.236(c), 72.236(d), and 72.236(l).

RAI 8-3 (New) Provide updated information on appropriate fracture toughness for Metamic-HT basket to support structural reviews for potential crack propagation due to non-mechanistic tip over.

The staff tracked back previous amendment applications on the information on fracture toughness for Metamic-HT. Based on Charpy impact energy correlations for steels, the fracture toughness, K_{1C}, was estimated to be

$$K_{1C} = 30 \text{ ksi in}^{1/2}$$

The staff has questions on the validity of this fracture toughness value:

(a) There is no direct measurements of Metamic-HT fracture toughness. The above equation uses one of several candidate correlations (Hetrzberg, 1995) for steels.
(b) Fracture toughness data for commonly used aluminum metal matrix composites (MMCs) with ceramic particles, an analog for Metamic-HT, could be lower than this K_{1C} value.

(c) Fracture toughness of aluminum MMCs may also depend on other factors including variations in composition and microstructure, potential aging such as hardening, and valid measurement methods with ductility increase at a higher temperature (approximately 300˚C).

The staff used information in Holtec's latest version of Metamic-HT Qualification Sourcebook (Holtec International, 2014). Given peak stresses and a 1/16-inch flaw size, the Sourcebook calculated the possible maximum stress intensity factors. With the factors, the Sourcebook then calculated Charpy impact energies from one of the correlations of fracture toughness and Charpy impact energy for steels. The calculated Charpy impact energies, 3.22 ft-lb and 2.71 ft-lb, are below the minimum guaranteed value of 4 ft-lb and below the minimum measured value of 7 ft-lb. Both the calculated stress intensity factor and the calculated Charpy impact energy are conservative (i.e., safer) using bounding values of stress and flaw size and using the ductile/brittle transition temperature range.

The staff requests that the applicant review updated information on Metamic-HT material properties (e.g., composition and microstructure) and evaluate available fracture toughness of aluminum MMCs especially at elevated temperature. The staff requests that the applicant provide justifications on the current fracture toughness values used.

This information is needed to determine compliance with the requirements of 10 CFR 72.236(b).

RAI 8-4 (New) Provide information on potential strength degradation of aluminum basket shims by thermal over-aging of precipitation-hardened microstructure.

The application addressed the use of aluminum alloy basket shims primarily in thermal performance. The applicant assumes aluminum alloy to be effective for the short duration dynamic loading from the tip-over accident. Aluminum alloy, such as Alloy 2219, used by Holtec is precipitation-hardened alloy. The application shows that the shims temperature could be as high as 260°C (500°F) under normal conditions (FSAR Table 3.III.3 and Table 4.III.3). Literature data shows that over-aging and accompanying strength degradation could occur between 210 – 240˚C in a few hours (for Alloy 2219 in Rafi Raza et al., 2011).

It is unclear to the staff whether the structural analysis adequately accounts for potential degradation of strength of aluminum alloy for prolonged conditions including normal conditions as discussed in HI-STAR SAR Section 2.2 (Holtec International, 2017). The staff requests that the applicant (i) provide justification that the current tip-over analysis in the design basis is valid, (ii) revise the analysis to adequately account for the degradation of aluminum alloy strength, or (iii) state that the type of Alloy 2219 (e.g., 2219-O) is in the annealed conditions which would not be subject to degradation of strength due to over-aging.

This information is needed to determine compliance with the requirements of 10 CFR 72.236(b), 72.236(c), 72.236(d), and 72.236(l).

References

American Concrete Institute (ACI). ACI 349.3R "Evaluation of Existing Nuclear Safety-Related Concrete Structures."

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