January 13, 2004

Mr. Brian Gutherman
Licensing Manager
Holtec International
555 Lincoln Drive West
Marlton, NJ 08053

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE HOLTEC INTERNATIONAL HI-STORM 100 AMENDMENT 2 (TAC NO. L23657)

Dear Mr. Gutherman:

By letter dated March 4, 2002, Holtec International (Holtec) submitted an application to the United States Nuclear Regulatory Commission (NRC) to amend Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Cask System (License Amendment Request [LAR] 1014-2, Revision 0) in accordance with 10 CFR Part 72. The amendment proposes to: (a) revise the contents in accordance with various new thermal, confinement, criticality, and shielding review methodologies; (b) permit the inclusion of damaged fuel contents to the Multi-Purpose Canister (MPC) 32; and (c) permit the inclusion of intact, damaged fuel, and fuel debris contents to a new MPC-32F.

In response to the NRC letter dated August 15, 2002, in which you note omissions in your March 4, 2002, submittal, you provided Revision 1 to LAR 1014-2 on October 31, 2002. By letter dated May 1, 2003, the NRC sent a Request For Additional Information (RAI) regarding Revision 1 to LAR 1014-2 to which you responded by letter dated August 8, 2003. Your response to the staff's RAI included Revision 2 to LAR 1014-2 and supplemental calculations which significantly revised the existing technical analyses.

In connection with the staff's continued review effort, information identified in the enclosure to this letter is needed. This information should be provided by February 28, 2004. If you are unable to satisfy this schedule, you must notify us in writing at least 2 weeks in advance of your new submittal date, and include your reasons for the delay. The staff will then assess the impact of the new submittal date and notify you of a revised schedule. If additional information requested by this letter results in your making changes to the Final Safety Analysis Report (FSAR), revised FSAR pages should be submitted. Justification for any FSAR changes should also be included in your response.

If you have any questions regarding our review, you may contact me at (301) 415-1179. Please refer to Docket Number 72-1014 and TAC No. L23657 in future correspondence related to this review.

Sincerely,

/RA/

Christopher Regan, Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No.: 72-1014
TAC No.: L23657
Enclosure: Request for Additional Information

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Sincerely,

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Request For Additional Information 2
Holtec International HI-STORM 100 Amendment Request
License Amendment Request 1014-2, Revision 2, Docket 72-1014

By application dated March 4, 2002, as revised October 31, 2002, and August 8, 2003, Holtec International (Holtec) requested amendment to Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Cask System in accordance with 10 CFR Part 72. This Request for Additional Information (RAI) identifies additional information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the application. The requested information is listed by chapter number and title and, where possible, section number, in the applicant's safety analysis report. NUREG-1536, "Standard Review Plan For Dry Cask Storage Systems," was used by the staff in its review of the application.

Each individual RAI describes information needed by the staff for it to complete its review of the application and to determine whether the applicant has demonstrated compliance with the regulatory requirements.

Thermal

4-1 Justify the use of the turbulent flow regime option employed in the Multi-Purpose Canister (MPC) downcomer region.

Based on the FLUENT model results provided by the applicant for the MPC-68, a Reynolds number of about 400 (based on the maximum velocity of the gas) was calculated in that region indicating that the gas flow is laminar. The option used in the downcomer defaults to turbulent flow.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

4-2 Explain why radiation heat transfer was applied between the walls bounding the downcomer region.

This mode of heat transfer was accounted for when the effective thermal conductivities were calculated and used as input to the HI-STORM FLUENT model. Allowing radiation heat transfer in this region of the FLUENT model appears to credit radiation heat transfer twice.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

4-3 Provide a description of the equipment and/or procedure that is used during backfilling operations to pressurize the canister to the initial value that would support the thermal analysis presented in the Safety Analysis Report (SAR). In addition, provide a description of the method and instruments used to measure the initial pressure of the canister.

According to the proposed design, the operating pressure of the canister has a significant effect on the calculated peak cladding temperature. Based on staff sensitivity studies performed using the applicant's developed models, a pressure decrease of 7 psi resulted in a peak cladding temperature that exceeded the allowable limit.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

4-4 Justify the proposed linear relationship between the measured lid temperature and the peak clad temperature as stated in "Thermal Analysis Package for HI-STORM LAR 2 (NRC Docket No: 72-1014)," Rev. 2.

The proposed thermal testing assumes a linear relationship between the measured lid temperature and the peak cladding temperature. The above relationship is constructed as a lid-to-peak clad temperature differential. The constants involved in this relationship were obtained by performing calculations based on the HI-STORM 2D axisymmetric Porous Media FLUENT model. The temperature differential is assumed to depend on only one parameter "X," which is defined as the ratio of inner region to outer region fuel assembly decay heats. In reality, for the HI-STORM concept design, peak cladding temperature is a function of three different heat transfer mechanisms: conduction, radiation, and convection. It is also a function of the internal and external flow characteristics of the storage cask and, as such, a single temperature measurement may not be easily correlated using a single parameter. An acceptable thermal testing analysis should include, as a minimum, the following:

1) Measurement of internal temperatures (including peak fuel temperature) as well as external temperatures.

2) Use of prototypic geometry or appropriately scaled models.

3) Use of prototypic operating conditions (i.e., design heat loads and operating pressures).

4) External boundary conditions that correspond to actual storage conditions (i.e., no forced flow cooling or constant temperature boundary conditions).

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

4-5 Provide justification and quantification of the individual and total thermal margins embedded in the analytic models used to perform the thermal-hydraulic analysis presented in the SAR. Provide the code biases and uncertainties included in calculating the peak cladding temperature. Alternatively, provide a three-dimensional FLUENT model to justify the assumed two-dimensional perfect polar axisymmetrical FLUENT model used to predict the thermal behavior of the highly non-symmetric cross section of the HI-STORM 100 storage cask.

The thermal regime proposed for the HI-STORM design falls outside existing experimental data. Consequently, the staff needs assurances that the analytic results bound the expected HI-STORM 100 thermal-hydraulic conditions.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

4-6 Provide a detailed description of the equipment used to supply supplemental cooling to the MPC during on-site transport in the HI-TRAC transfer cask. Justify the availability and reliability of this system during normal conditions of transfer.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

4-7 Clarify which version of Interim Staff Guidance (ISG) -11 is adopted in the SAR thermal evaluation for Amendment 2 to the HI-STORM 100 CoC.

It is stated in Section 4.5.6.2 of the SAR that ISG-11, Revision 2 limits are adopted, but the thermal analysis presented is based on ISG-11, Revision 3 thermal limits. ISG-11, Revision 2 limits the maximum cladding temperature to 400°C for normal conditions of storage and for short-term operations including cask drying and backfilling. However, a cladding temperature limit of 570°C and hoop stress compliance requirement (ISG-11, Revision 3 criteria) are applied to Conditions 5 and 6 on Table 4.5.11 of the SAR for low burnup fuel.

If ISG-11, Revision 3 is applied, modify the SAR accordingly and provide a best estimate calculation of the cladding hoop stresses, in tabular form, for low burnup spent fuel rods for the fuel types included in Amendment 2 that will exceed the 400°C short-term temperature limit. This calculation should be done by selecting bounding End of Life pressure cases and applying bounding rod temperature assumptions for the fuel types that will exceed the 400°C short-term temperature limit. Further, the table should include bounding assumptions regarding fuel rod (assembly type), cladding dimensions, pressure, temperature, oxidation, and reduction in cladding wall due to oxidation.

A higher short-term temperature limit (greater than 400°C) may be used, for low-burnup fuel with approximately a 40-micron oxide layer, or less, if it can be shown by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi). This calculation should be performed at 350°C (the temperature for hydride precipitation for low burnup fuel) to ensure hydride reorientation does not occur.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Shielding

5-1 Provide a graphical representation of the burnup curve fitting methodology to demonstrate the derivation of the burnup equation proposed in Section 2.4 of the proposed Appendix B. The representation should include the following elements:

(a) Comparison of the fitted curve generated by GNUPLOT versus actual data points for heat load, enrichment, and burnup.

(b) Comparison of the adjusted curve with the “G” correction factor with respect to the fitted curve and data discussed above in item (a).

The representations should encompass the ranges of enrichment, burnups, and heat loads requested in the amendment.

This information is necessary to verify compliance with 10 CFR 72.236(d) and (f).

5-2 Provide the following information regarding GNUPLOT used to derive the burnup equation:

(a) The software reference version used to generate the fitted curve.

(b) A discussion of the curve fitting methodology that the program uses to determine equation coefficients.

(c) Benchmark and testing of the curve-fitting methodology that validates the program for source term applications.

This information is necessary to verify compliance with 10 CFR 72.236(d) and (f).

5-3 Provide a description of the methodology used to adjust the G constant in the equation. Clarify the reason why the adjustment is necessary after the equation is calculated with GNUPLOT.

This information is necessary to verify compliance with 10 CFR 72.236(d) and (f).

5-4 Revise Section 2.4 in the proposed Appendix B to clearly indicate that users must verify that each fuel assembly satisfies the decay heat limits specified in Sections 2.4.1 or 2.4.2.

This question is consistent with your responses to NRC RAI, dated August 8, 2003, questions 5-11 and 5-13, which indicate that the user is responsible for verifying heat limits.

This information is necessary to verify compliance with 10 CFR 72.236(a) and (f).

5-5 Justify whether a user must always apply equation 2.4.3 in the proposed Appendix B to determine allowable burnup and cooling times, even for fuel that contains non-fuel hardware.

It appears this could be accomplished by determining a “q” value for this equation by subtracting the non-fuel hardware heat load from the total allowable heat load derived in Section 2.4.2 of the proposed Appendix B.

This information is necessary to verify compliance with 10 CFR 72.236(d) and (f).

5-6 Justify the statement in your response to NRC RAI, dated August 8, 2003, question 5-9(c) that the burnup equation and associated coefficients are valid for fuel assemblies with enrichments below 0.7 wt%. Specify the types of depleted uranium fuel assemblies, including associated burnup and decay heats that are proposed for storage.

Staff confirmatory analyses appear to show the burnup equation may not be valid at all heat loads and at all enrichments below 0.7 wt%.

This information is necessary to verify compliance with 10 CFR 72.236(d) and (f).

Operating Procedures

8-1 Provide a detailed description of the necessary steps to prepare the MPC for transfer inside the HI-TRAC transfer cask for the case when additional supplemental cooling is needed.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

8-2 Clarify if there is a time limitation, before temperature limits are exceeded and action(s) by the user are required, in the event of Forced Helium Dehydration (FHD) failure.

Your response to NRC RAI, dated August 8, 2003, question 4-16, indicated that there is a provision, if necessary, for the connection of either air or water cooling to the MPC annulus. It is not clear if air or water cooling is necessary to maintain temperature limits in the event of FHD failure or if there are any time limitations before temperature limits are exceeded that would require the use of supplemental air or water cooling. Appropriate revision to procedures, TS, and/or Appendix B to the CoC may be necessary.

This information is necessary to verify compliance with 10 CFR 72.11.

Radiation Protection

10-1 Revise the proposed Technical Specification (TS) to include the upper bound dose limits for the storage overpack that are based on the shielding analysis in Chapter 5.

Your response to NRC RAI, dated August 8, 2003, question 10-3 indicates that, due to the conservative analytic assumptions, actual dose rates will never exceed the dose rates identified in Chapter 5. However, surface dose rates are a performance-based measure of the shielding system. The established limits provide a safety benefit because they help assure as low as reasonably achievable exposures and could help identify unexpected gross errors in the physical shielding system or the loaded contents, if exceeded. In addition, there does not appear to be an undue burden for the users to verify compliance with these limits. These measurements will be performed as part of a typical radiation protection program at reactor sites.

This information is necessary to determine compliance with 10 CFR 72.236(d) and 10 CFR Part 20.

10-2 Revise Section 10.3.2 of the SAR to reflect the justification for surveillance and maintenance exposures that are proposed in your response to NRC RAI, dated August 8, 2003, question 10-2.

The SAR analysis appears to be based on design-basis fuel loadings and assumed worker distances that are no longer considered in your response to RAI question 10-2.

This information is necessary to determine compliance with 10 CFR 72.236(d) and 10 CFR Part 20.

Accident Analysis

11-1 Provide a description and analysis of the postulated off-normal and accident events by including in the analysis the equipment used to provide supplemental cooling during transfer of the MPC in the HI-TRAC transfer cask. Provide limiting conditions of operations (LCOs) and appropriate revision to the TS for failure of the cooling system.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Operating Controls and Limits

12-1 Revise Technical Specifications, Section 3.2 to reflect the following addition or propose alternate language to address the following:

> <u>3.2.x Neutron Absorber Tests</u>
> Section 9.1.5.3 of the Holtec HI-STORM 100 FSAR, revision 2, dated August, 2003, is hereby incorporated by reference into the HI-STORM 100 Technical Specifications (TS). Minimum boron content for the neutron absorber shall meet the minimum requirements for each basket design specified (above/below) in this section of the Technical Specifications (TS).

The proposed TS change recognizes that neutron absorber materials are proprietary materials. As such, these materials are not subject to the uniform production and quality control standards that exist for ASME Code materials. Additionally, there is no reasonable manner in which to verify the performance of these materials during service. The function they perform is of high importance, eliminating the possibility of an inadvertent criticality. Consequently, the NRC staff believes the production and quality control methods and requirements for these materials should be formalized. In this manner, therefore, no changes to the materials production methods may occur unless such (proposed) changes are first subjected to an independent review.

This information is necessary to verify compliance with 10 CFR 72.11 and 72.236(d).