Request for Supplemental Information

Docket No. 72-1014
Certificate of Compliance No. 1014
Amendment No. 15 to the HI-STORM 100 Multipurpose Canister Storage System

Chapter 4 - Thermal Evaluation

RSI 4-1

Provide detailed examples of sizing calculations for the Dry Ice Jacket and the HI-DRIP auxiliary cooling systems.

Section 1.2.1.7 of the final safety analysis report (FSAR) states that the dimensions of the chiller and the amount of dry ice will depend on the rate of heat extraction required and the duration of the short-term operation and therefore, must be custom sized for each application.

Regarding the HI-DRIP auxiliary cooling system, Section 4.II.5.3 of the FSAR states that sizing of HI-DRIP for a typical scenario is archived in the Holtec Calculation Package (Holtec International Thermal Calculation Package HI-2043317 and did not find any information related to sizing calculations of this auxiliary cooling system.

Detailed sizing calculations should be provided for design basis conditions. The staff needs detailed sizing calculations of these two cooling systems to determine adequate cooling is provided for the period of time that the systems are used and thus verify no thermal limits are exceeded.

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

Holtec Response:

Detailed sizing calculations for the Dry Ice Jacket (DIJ) are performed and documented in the thermal calculation package HI-2043317, Revision 36, Appendix N (Attachment 11 of Letter 5014878). The methodology outlined therein can be adopted on a site-specific basis to size the DIJ and to ensure all acceptance criteria specified in the FSAR are satisfied.

Similarly, detailed methodology for sizing and determining the specifications of the HI-DRIP cooling system are also added in the thermal calculation package HI-2043317, Revision 36, Appendix N. The methodology outlined therein can be adopted to size the HI-DRIP components and operating parameters on a site-specific or cask-specific basis.

Chapter - Confinement Evaluation

RSI 5-1

Provide the following confinement information and update the application as described in the responses to the HI-STORM 100 Amendment No. 13 requests for supplemental information (RSI) that were provided as part of the HI-STORM 100 Amendment No. 15.

Verify that the statement, “Condition D and SR 3.1.1.3 are not applicable to casks that were

loaded to Amendment 2 through 7," does not appear, or is removed from page 3.1.1-1 of HI-STORM 100 proposed Technical specification Appendices A and C. This is requested because Condition D and SR 3.1.1.1 are applicable to Amendment Nos. 2 through 7, and the HI-STORM 100S Version E Cask is not applicable to Amendment Nos. 2 through 7.

Also remove the statement, "MPCs that were loaded under CoC Amendment No 7 and prior amendments are subject to the requirements of those amendments, which may differ, " from page 2-179 of the application because the statement should not be applicable to vent and drain port cover plate welds, or proposed change No. 10.

This information is needed to determine compliance with 10 CFR 72.236(d), and (f), and 72.244.

Holtec Response:

The statements related to Amendments 2 and 7 are removed from Appendix C, since those amendments are not related to the Version E addressed by Appendix C.

There are 3 different helium leak testing requirements impacted by the proposed change:

1) Fabrication leak testing of the shell to baseplate weld, which was originally an FSAR required test. This test was removed under 72.48 for a period of time. As a result of NRC EA-09-190 this test was restored for all canisters regardless of amendment as of July 1, 2009, and was also added as a CoC Condition in Amendment 8. Therefore, this test impacts canisters loaded to Amendments 2 through 7 fabricated prior to July 1, 2009
2) Base metal helium testing of the MPC lid, which was added as a CoC condition in Amendment 8. Therefore, this test impacts canisters loaded to Amendments 2 through 7.
3) Base metal helium testing of the vent and drain port cover plates, which was added as a CoC condition in Amendment 8. Therefore, this test impacts canisters loaded to Amendments 2 through 7.

Based on these three items, the wording in Appendix A, LCO 3.1.1 Notes section was changed to specify that only the MPC helium leak rate limit for the base metal of the port and cover plates, specified in Condition D and SR 3.1.1.3, is not applicable to casks that were loaded to Amendments 2 through 7. This portion of the note is needed because helium leak rate testing of the base metal of the MPC vent and drain port cover plates is not a condition for Amendments 2 through 7 (item 3 above). Condition D did not exist prior to Amendment 8. Therefore, sites that would like to upgrade canisters to Amendment 15 may not be able to demonstrate their compliance with Condition D of LCO 3.1.1. Therefore, it is important to maintain the statement about Condition D in the notes of Appendix A, LCO 3.1.1. However, the listed date only applies to the fabrication leak testing of the shell to baseplate weld (item 1 above), so that portion of the note is removed.

In Chapter 2, the statement, "MPCs that were loaded under CoC Amendment No 7 and prior amendments are subject to the requirements of those amendments, which may differ. "was removed from Table 2.2.15 on page 2-179, because this table addresses the welds and not the base metal described in item 3 above.

Chapter 5 - Shielding Evaluation

RSI 6-1

Update the information that justifies the source term(s) assumed within the shielding evaluation

are appropriate to represent the equation that defines burnup and cooling time for the MPC-32M.

Burnup and cooling time specifications for the MPC-32M contain an approach that has been presented to the NRC under review of Amendment 4 of the HI-STORM FW (Docket No. 72-1032). This is a new approach that uses an enrichment that bounds 99% of the fuel population from a database and an equation that calculates cooling time as a function of burnup. The staff discussed this approach with Holtec during a teleconference on March 7, 2019 (Conversation Record: “Discuss Holtec’s proposed approach in response to NRC’s second round of request for additional information (RAI) for HI-STORM Flood/wind Amendment No. 4, " March 7, 2019, ADAMS Ascension No. ML 19072A166), and at a public meeting n March 19, 2019 Meeting (Memorandum from Y. Chen to C. Reagan, “Summary of March 19, 2019 Meeting with Holtec International to Discuss Proposed Response to the Second Round of Request for Additional Information for Certificate of Compliance No. 1032 for HI_STORM Flood/Wind, Amendment No. 4,” April 4, 2019, ADAMS Accession No. ML 19093A048). During these meetings with Holtec, the staff shared its request for information to justify the methodology. Holtec stated that it would provide the information that the staff requested and has updated the HI-STORM FW Amendments 4 and 5 applications with additional information; however, the application for HI-STORM 100 Amendment 15 does not include the requested information. The staff requests that the applicant update the information related to the burnup and cooling time equation method and provide the following information:

a) The applicant needs to provide the reference for the database used to generate Figure 5.II.2-2 from the safety analysis report (SAR) and the minimum enrichment Table 5.II.2.4 from the SAR. The applicant should justify that the database is appropriate, and it is using the most recent available data.
b) The applicant needs to provide a discussion of how it determined that the enrichments in Table 5.II.2.4 are bounding for 99% of discharged PWR spent nuclear fuel.
c) For higher burnup assemblies (> 55GWd/MTU), there is less data available for establishing an enrichment value that bounds 99% of the discharged fuel population. These are also the assemblies whose source term (and ultimately dose/dose rate) is most sensitive to enrichment (see Figures 12 and 13 of NUREG/CR-6716, “Recommendations on fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks”). The applicant needs to justify that the selected enrichments for these higher burnup assemblies are bounding for 99% of the discharged PWR spent nuclear fuel population.
d) The applicant needs to provide dose and dose rate results for the cooling time/burnup points along the curves specified in Table 2.1-4 of CoC Appendix D (also in Table 2.II.1.6 of the SAR) to demonstrate that it found the bounding cooling time/burnup combination along the curve for use in these evaluations.
e) For the regionalized loading patterns that contain unique cooling time and burnup correlation for each region corresponding to each decay heat, the applicant needs to provide a detailed discussion of th procedure it used to determine the maximum dose and dose rates.

This information is needed for the staff to evaluate the appropriateness of the technical specification dose rate limits, as well as to be able to evaluate if the shielding features ensure that the DSS design and operations facilitate licensee compliance with the requirements prescribed in 10 CFR Part 20, Subpart C, “Occupational Dose Limits,” and 10 CFR 72.126(a) and the capability of the cask system to meet the requirements of 10 CFR 72.104 and 72.106, including the dose limits, as required by 10 CFR 72.236(d).

Holtec Response:

In order to justify that the source term(s) assumed within the shielding evaluation are appropriate to represent the equation that defines burnup and cooling time for MPC-32M, the additional details have been presented as well as the additional studies have been performed to support the conclusions made.

a) At the time of initial evaluations for Amendment No. 15, only the Nuclear Fuel Data Survey (Form RW-859 from 2002) was used to estimate the lower bound enrichment (LBE) values for shielding evaluation of the HI-STORM 100 system with MPC-32M.

   The most recent Nuclear Fuel Data Survey with a large fuel population (Form GC-859 from 2013) has been combined with the RW-859 data in order to increase the statistics and confidence that the determined LBEs bound the assumed fraction of the real fuel population. Particularly, since the RW-859 dataset includes the data on the spent nuclear fuel assemblies shipped to away-from reactor (off-site) facilities, which are not provided in GC-859, the RW-859 dataset is additionally screened and all unique data records (which are not present in GC-859) have been added to the GC-859 data.

   The final set of PWR data provide a sufficient statistics to determine a reasonably accurate LBE, while for the burnup groups with less than 200 assemblies, the smallest enrichment, which eventually bounds 100% of the fuel assembly population, is used. The analysis and determined LBEs for various fractions of fuel assemblies are documented in HI-2188480, Revision 0. The reviewer is referred to this report for the additional details.

   The shielding calculations have been revised using the updated set of the LBE values (99%) from HI-2188480 and the results have been updated in Supplement 5.II and HI-2188253.

b) For a given fuel type, the fuel enrichments are distributed over different burnup range bins (0-5, 5-10 … 70-75 GWd/mtU). For instance, for the burnup group of 5-10 GWd/mtU, the data array includes the enrichments for the fuel assemblies with the burnup from 5,000 MWd/mtU to 9,999 MWd/mtU. Then, in each burnup group, the array of enrichments is sorted from low to high, and the array index that precedes a selected fraction of the population is determined. The fuel enrichment under this array position represents the LBE that conservatively bounds a selected fraction of the fuel assembly population. A discussion has been also added to Supplement 5.II and Subsection A.4.2 of HI-2188253.

c) For higher burnup assemblies (> 55 GWd/mtU), the following data is considered in HI-2188480:

   - 55,000 - 60,000 MWd/mtU – 1814 assemblies;
   - 60,000 - 65,000 MWd/mtU – 193 assemblies;
   - 65,000 - 70,000 MWd/mtU – 27 assemblies.

   The number of assemblies in the 55-60 GWd/mtU burnup group (1814 items) is considered sufficient to establish a reliable LBE that bounds 99% of the discharged spent nuclear fuel population. For the other two burnup groups, where there is fewer data available, the lowest enrichment in a bin that actually bounds 100% of the population is used as the LBE value. It should be noted that the available population of the discharged fuel assemblies may potentially contain the rare outliers, such as low-enriched assembly with atypically high burnup, so by using the LBE that covers 100% of the fuel population, such potential outliers are not rejected and conservatively become the design basis

enrichment that describes the more common fuel assemblies.

Furthermore, in order to assess the effect of enrichment for higher burnup assemblies on the dose rates around the Version E cask with MPC-32M, the sensitivity studies have been performed and documented in Subsection B.5.2 of HI-2188253. Specifically, the LBE value has been additionally reduced by ~25%, as the following:

- 60,000 - 65,000 MWd/mtU – from 3.9 wt% to 2.9 wt% ^{235}U;
- 65,000 - 70,000 MWd/mtU – from 4.2% wt% to 3.1 wt% ^{235}U.

The results of the sensitivity studies documented in HI-2188253 show that the effect of lower enrichment on adjacent and 1 m dose rates for the Version E cask with MPC-32M is relatively small.

d) As discussed in Section 5.II.1, a number of burnup, enrichment and cooling time combinations along the curve have been selected for each basket loading region (see Table 5.II.2.5) and analyzed in order to determine the bounding loading pattern for each dose location that results to the maximum total dose rate. To demonstrate that the bounding bu/enr/ct combinations have been in fact used in all basket loading regions for each dose rate location and confirm that the maximum dose rates have been determined, the total dose rates are additionally provided in Table B.3.6 of HI-2188253 for all considered bu/enr/ct combinations in each loading region separately. In summary, the dose rate results presented in HI-2188253 demonstrate that the bounding burnup, enrichment and cooling time combinations along the loading curve have been used in the design basis calculations presented in Tables 5.II.1.1 and 5.II.1.2 of the FSAR. It should be noted that generally, a dose rate variation with burnup in Table B.3.6 is smooth, which provides additional assurance that a number of analyzed bu/enr/ct combinations along the loading curve is sufficient to validate the loading curve and demonstrate that the reasonably highest dose rates are determined. Additional information is provided in Subsection B.3.2.1 of HI-2188253.

e) The additional details of the procedure used to determine the maximum dose rates for the regionalized loading patterns have been added to Section 5.II.1 of the FSAR and Section B.3.2 of HI-2188253.

RSI 6-2

Provide the minimum lead and water jacket thickness for the HI-TRAC MS, and provide normal, off-normal, and accident condition dose evaluations for the HI-TRAC MS with these minimum thicknesses.

Based on the discussion in Section 1.II.2.3 of the SAR and Drawing No. 11381, the HI-TRAC MS has variable lead and water jacket thicknesses. Supplement 5.II of the SAR does not include information on the lead thickness or water jacket thickness used ot perform the dose rate evaluations; however it does include statements (such as the following) that lead the staff to conclude that it did not use the minimum thickness to evaluate the dose/dose rates:

Section 5.II.4 of the SAR states: ”… the shielding analysis of Version MS under the accident condition is not necessary for the reference Version MS and the expected results are bounded by the analysis of reference HI-TRAC 100. Nonetheless, the additional site-specific shielding evaluations shall be performed to confirm the shielding performance of Version MS, if the lead thickness of the customized Version MS cask is less than the lead thickness of the reference

100-ton HI-TRAC, analyzed in the main body of Chapter 5."

Although the staff may have accepted nominal dimensions used in Chapter 5 for previous amendments, the nominal in those cases are with respect to a tolerance on the order of a few millimeters rather than a variable lead thickness. The difference between minimum and nominal lead for the HI-TRAC MS with a variable lead shield could be significant. In section 1.ii.2.3 of the SAR, the applicant compares the HI-TRAC MS to the HI-TRAC VW (approved for use in the HI-T+STROM FW system, Docket No. 72-1032) by stating that the HI-TRAC MS "… is a smaller diameter counterpart of HI-TRAC VW Version V in HI-STORM FW FSAR." The staff notes that the difference in nominal versus minimum lead in the HI-TRAC VW is greater than 1 inch. The amount of lead needed to attenuate 1 MeV gammas by half is roughly 9mm (0.35 inches). Thus, a reduction in lead of more than 1 inch would have a significant effect on the gamma dose rates.

Although site specific evaluations are needed and used to demonstrate compliance with 10 CFR 72.104 (per 0 CFR 72.212(b)95)) at the general licensee's site, 10 CFR 72.236(d) requires that the applicant for a CoC ensure that radiation shielding features of the spent fuel storage cask system are sufficient for compliance with 10 CFR 72.104 and 72.106.

This information is necessary to determine compliance with 72.236(d).

Holtec Response:

The minimum lead and water jacket thicknesses of the HI-TRAC MS overpack have been added to Table 3.II.2.3. The shielding evaluations of HI-TRAC MS with the minimum thicknesses under the normal and accident conditions have been performed and documented in Section C.3 of HI-2188253 and Supplement 5.II of the HI-STORM 100 FSAR.

RSI 6-3

Provide an evaluation demonstrating that the MPC-32M with stainless steel clad fuel assemblies meets the regulatory dose requirements.

CoC Appendix D Table 2.1-1 Section V.A.2a allows the storage of stainless steel clad assemblies within the MPC-32M; however, there is no corresponding analysis of stainless steel clad assemblies within Chapter 5.II of the SAR demonstrating that storage of stainless steel clad assemblies in the MPC-32M meets regulatory dose limits.

This information is necessary to determine compliance with 10 CFR 72.236(b) and 10 CFR 72.158.

Holtec Response:

The stainless steel clad (SS-clad) fuel properties that have been already qualified for loading into the MPC-32 basket are provided in Chapter 2 of the FSAR. The same burnup and cooling time requirements are applied for loading of stainless steel clad fuel assemblies into MPC-32M (see Table 2.II.1.1 of the FSAR) without any changes. Comparing the gamma sources in all 7 energy groups for the bounding SS-clad fuel with those provided in Table 5.II.2.1 of the FSAR for the design basis zircaloy clad fuel shows that they are bounded with a significant margin. Hence the

dose rates for the HI STORM 100 System with MPC-32M and the allowable stainless steel clad fuel are less than those of design basis zircaloy clad fuel, and no further analyses are needed for stainless steel clad fuel in MPC-32M. A discussion has been added to Supplement 5.II of the HI-STORM 100 FSAR and Section D.3 of HI-2188253.

Chapter 6 - Criticality Evaluation

RSI 7-1

Revise the application to provide Reference 6.A.19, HI-2033039, "CRITICAL EXPERIMENT BENCHMARK," Revision.

Appendix 6.A of the SAR, "Benchmark Calculations", refers to Reference 6.A.19, HI-2033039, "CRITICAL EXPERIMENT BENCHMARK", Revision 5, for detailed discussion of the MCNP5-1.5 benchmarking analysis. This document supports the benchmarking analysis for the previously approved HI-STORM 100 MPCs. This document has been updated to include experiments relevant to the partial gadolinium credit requested in this amendment for BWR fuel in the MPC-68M canister. This information is needed for the staff to confirm that the MCNP5-1.51 code used for the partial gadolinium credit calculations is properly validated for this purpose.

This information is necessary to compliance with the criticality safety requirements in 10 CFR 72.124 and 72.236(c).

Holtec Response:

Reference 6.A.19, HI-2033039, "CRITICAL EXPERIMENT BENCHMARK," Revision 5 has been provided as Attachment 5 to letter 5014878.

Chapter 9 - Acceptance Criteria and Maintenance Program Evaluation

RSI 10-1

Based on sizing calculations, including appropriate thermal tests that demonstrate the HI-DRIP and Dry ice jacket auxiliary cooling systems will function as designed.

Section 9.II of the FSAR does not include appropriate acceptance thermal test that demonstrate the performance of these systems. The test should demonstrate systems design criteria are met. Thermal performance tests are needed to determine the adequacy of these auxiliary systems to provide the necessary cooling to prevent exceeding any thermal limits.

This information is necessary to verify the requirements of 10 CFR 72.162, 72.234(a), and 72.236(f).

Holtec Response:

Chapter 9 of the main FSAR has been revised to include the thermal testing requirements of the HI-DRIP and Dry Ice Jacket Supplemental Cooling Systems. See Sections 9.1.6.1 and 9.1.62 of Chapter 9PR17.

Chapter 10 – Radiation Protection

RSI 11-1

Update SAR Chapter 10 to include the HI-TRAC MS.

Chapter 10: Radiation Protection, from the SAR was not updated to account for the operations of the HI-TRAC MS. Throughout this chapter there are many considerations that warrant inclusion of the HI-TRAC MS especially considering variable shielding options for this transfer cask.

Section 10.II of the SAR states: "… additional auxiliary/temporary shielding, such as listed in Table 10.1.1, may be used to further reduce the dose rates around the HI-TRAC when performing short term operations." Table 10.1.2 of the SAR states requirements for using auxiliary and temporary shields for the other HI-TRAC designs but needs to be updated to include requirements for the HI-TRAC MS. The dose evaluations need to be performed without additional auxiliary/temporary shielding. If additional auxiliary/temporary shielding is assumed in the dose assessments, these should be listed as required.

Table 10.3.1b and Table 10.4.1 of the SAR need to be updated to consider HI-TRAC MS when using minimum lead and water jacket thicknesses.

This information is needed for the staff to evaluate the capability of the cask system to facilitate the control and limiting of occupational exposures consistent with the requirements in 10 CFR Part 20 and 10 CFR 72.126(a), including maintaining exposures ALARA, and to evaluate the capability of the cask system to meet the requirements in 10 CFR 72.104 and 106, including the dose limits, as required by 10 CFR 72.236(d).

Holtec Response:

SAR Supplement 10.II has been updated to include the occupational dose assessment during the loading operations of HI-STORM 100S Version E using the HI-TRAC Version MS transfer cask as well as the collective dose assessments. Specifically, Table 10.II.1.1 has been added to outline the auxiliary/temporary shield requirements for Version MS. Table 10.II.3.1 provides a summary of the dose assessment for the Version E loading operations using the Version MS transfer cask. Tables 10.II.4.1 and 10.II.4.2 provide the dose results at a distance from the ISFSI and HI-TRAC transfer cask, respectively. All evaluations have been made considering the minimum lead and water jacket thicknesses for the HI-TRAC Version MS cask.

O-6-1

Tables 5.II.4.1 and 5.II.4.4 are used to compare dose rate performance between the HI-STORM 100 overpacks and the HI-TRAC transfer casks, respectively. The applicant needs to justify why the source terms selected to perform these comparisons are representative of the variety of spent fuel contents that may be loaded and stored in the transfer casks and overpacks or provide

additional comparisons with source terms that have different gamma/neutron contributions. The shielding performance for the different transfer casks and overpacks may change depending on the source term's contribution from gammas and neutrons, i.e., whether the source is neutron or gamma dominated. In other words, the shielding performance may be similar for a gamma dominated source term if both cask features are especially good at shielding gammas; however, for a neutron dominated source, depending on the cask features, the shielding performance may not be similar. In addition, the applicant needs to consider any features that would affect the positioning of the contents and consequently the shielding performance of the Version E overpack and the HI-TRAC MS as compared to the other overpacks and transfer casks.

Holtec Response:

The design basis burnup and cooling time combinations, presented in Section 5.1 as the bounding combinations for all acceptable uniform and regionalized loading burnup levels and cooling times for MPC-32 and MPC-68, were used for demonstration of the shielding performance of the HI-STORM 100 overpacks and the HI-TRAC transfer casks. In order to enhance a comparison of the cask's shielding performance, the additional evaluations with the source terms that have different gamma/neutron contributions have been performed.

The results of the calculations presented in Tables B.2.3 and C.2.5 of HI-2188253 confirm the original conclusion, i.e. that the shielding performance of the Version E and Version MS overpacks is improved in comparison with the reference casks for different source terms with various combinations of gamma/neutron contributions.

Contents Positioning:

The positioning of the contents inside the MPC-32 and MPC-68M cavities, which were used for evaluation of the shielding performance, are identical between Version E and Version MS as compared to the reference overpacks. The minimum MPC elevations in the Version E and Version MS overpacks, based on the drawings listed in Section 1.II.5 of the FSAR, were assumed, and the minimum cavity heights of the overpacks are set greater than the MPC height by fixed amounts (see Table 3.II.2.1 of the FSAR). This makes MPC closer to the inlet/outlet vents of Version E and results in higher dose rates at the vents. Therefore, the calculational models of Version E and Version MS are conservative and/or equivalent to the models of the reference overpacks, hence they are representative for evaluation of the shielding performance.

It should be noted that for both MPC-32M and MPC-68M baskets, considered in the Version E and Version MS overpacks, no elevation of the fuel assemblies above the MPC baseplate was assumed. Any additional fuel spacer below or above fuel assemblies in MPC would increase the MPC/overpack cavity and, consequently, the fuel-to-vent distance will be increased as well. Therefore, the bounding models of the Version E overpack and Version MS transfer cask with respect to the contents positioning have been analyzed in the FSAR.

O-6-2

The shielding performance for damaged fuel adopts the conclusions from Section 5.4.2.2 of the SAR. Additional justification is needed to show how these conclusions remain applicable for the HI-STORM 100S Version E and the HI-TRAC MS. The damaged fuel analysis for PWR fuel is based on the MPC-24 with 4 damaged fuel locations while the MPC-32M is allowed for 16 damaged fuel locations within CoC Appendix D, Table 2.1-1.V.B. The 16 locations where damaged fuel are allowed are periphery versus localized increase of the 4 damaged assemblies

within the MPC-24. Also, due to its variable shield thicknesses, the HI-TRAC MS can have less shielding than the HI-TRAC 100, which is what the damaged fuel analysis in Section 5.4.2.2 of the SAR is based on. In addition, NUREG/CR-7203, “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,” shows a significant increase in dose rates near storage cask air vents due to reconfiguration. This report also shows reconfiguration can have a significant impact on controlled area boundary doses.

Holtec Response:

To illustrate that the potential effect of storage of damaged fuel and/or fuel debris on the dose rate is not significant, the shielding evaluations of HI-TRAC MS with MPC-32M with a maximum quantity of damaged fuel locations have been performed in accordance with the analysis in Paragraph 5.4.2.2 of the HI-STORM 100 FSAR. Specifically, the analysis consists of modeling the fuel assemblies in the 16 peripheral damaged fuel locations with a fuel density that is twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. Also, a flat axial power distribution is used. The analysis is performed for Version MS under the normal conditions with the minimum lead and water jacket thicknesses and documented in Section C.4 of HI-2188253 and Supplement 5.II of the HI-STORM 100 FSAR.

O-10-1

In Supplement 9.II, “Acceptance Criteria and maintenance Program,” of the application, the word “toto,” should be revised to clarify the meaning of the sentence.

Holtec Response:

In Supplement 9.II, the word “toto” was corrected to “total” on page 9.II-1.