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

April 20, 2010

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
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SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT APPLICATION FOR INTER-UNIT SPENT FUEL TRANSFER (TAC NOS. ME1671, ME1672, AND L24299)

Dear Sir or Madam:

By letter dated July 8, 2009, as supplemented by letter dated September 28, 2009, Entergy Nuclear Operations, Inc. (Entergy) submitted an application for proposed license amendments for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) that would allow the transfer of spent fuel from the IP3 spent fuel pool (SFP) to the IP2 SFP using a newly-designed transfer canister.

The Nuclear Regulatory Commission staff is reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). The Entergy staff stated that a response to the RAI would be provided by October 5, 2010.

Please contact me at (301) 415-2901 if you have any questions on this issue.

Sincerely,

John P. Boska

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosure:
RAI

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING SPENT FUEL TRANSFER

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3

DOCKET NOS. 50-247 AND 50-286

By letter dated July 8, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091940177 and ML091940178), and supplemented by letter dated September 28, 2009, (ADAMS Accession Nos. ML092950437 and ML093020080), Entergy Nuclear Operations, Inc. (Entergy or the licensee), submitted a license amendment request for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3). The proposed changes are requested to provide the necessary controls and permission required for Entergy to move spent fuel from the IP3 spent fuel pool (SFP) to the IP2 SFP using a newly designed shielded transfer canister (STC), which is placed inside a HI-TRAC 100D cask for outdoor transport. The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions:

CHAPTER 1 – GENERAL INFORMATION

1-1. Revise Holtec International document HI-2094289 to incorporate the information and analyses provided in the response to the NRC request for supplemental information and the responses to this request for additional information (RAI). (TCB, CSDAB)

HI-2094289 should serve as the primary document that integrates the supporting calculations and supporting information together and provides the regulatory bases for this license amendment. The application needs to capture the most updated information regarding the system design, evaluations and operations in a single document. That document is HI-2094289. As the application now exists, the information is scattered among several documents and, with the response to staff's RAIs, HI-2094289 contains information that is now outdated, incomplete, or not in agreement with information in the supporting documentation submitted as part of a previous response. This leads to confusion as to what information is correct and is to be used as the basis for the application.

This information is needed to confirm compliance with 10 CFR 50.90.

1-2. Clarify if the HI-TRAC 100D has been changed through the change authority of 10 CFR 72.48. (TCB)

The supplemental response 8b indicates that the system used is based on the HI-STORM 100D "as licensed by the NRC" with modification to the lids. It is not clear if this configuration includes other configuration changes not directly licensed by NRC. Several portions of the amendment application refer to the HI-TRAC 100D as the licensing basis, in part, for evaluating the structural and confinement integrity of the wet transfer system.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in General Design Criterion (GDC) 1 and 61.

1-3. Clarify the water level in the HI-TRAC cask cavity with the STC present. (CSDAB)

Descriptions of the annulus water level in the loaded HI-TRAC are not consistent in the inter-unit transfer report (the report). Chapter 1 of the report indicates the water level will be 10 inches below the transfer cask lid. The shielding model is consistent with this description; however, the Operations Chapter of the same report indicates that the transfer cask is filled so that the water level will be 3 inches below the top of the STC flange. Based on the dimensions provided in the report drawings, this water level means there is an air gap of more than 18 inches. This water is relied upon for various functions (e.g., shielding), and the application evaluations should appropriately account for the correct level of the annulus water.

This information is needed to confirm compliance with 10 CFR 50.90 and 50.34a(c) and the intent of 10 CFR 72.104 and 72.126(a).

1-4. Provide clarification regarding the STC special lifting device and attachment (refer to page 1-7)(SMMB).

Section 1.3.1 states that "The STC has two lift points which will attach to the overhead cranes at IP3 and IP2 through the STC lid and a lifting device. The STC lifting points are designed in accordance with NUREG-0612 [C.A] for critical loads. The lid attaches using threaded studs and nuts."

Figures 1.4.6, 1.4.7, 1.4.10, 1.4.11, 1.4.13 show that a below-the-hook special lifting device is attached to the STC using 4 threaded studs and nuts. It appears that a "single" load path, through the two STC trunnions, is used to support the loaded STC.

Clarify the second paragraph description in Subsection 1.3.1 and revise, as appropriate, the Item iii summary of the evaluation criteria for the STC lifting points, including the threaded stud engagement and STC lift trunnions for critical loads.

This information is required by the staff to assess compliance with GDC-1, GDC-2, GDC-4, GDC-61. There is guidance in NUREG-0800 15.7.5, RG 1.183, NUREG-0612 5.1.6,; NUREG-0800 9.1.5; NUREG-0612 (Appendix C); NUREG-0554; ASME NOG-1 (2004), and 10 CFR 71.122.

CHAPTER 3 – PRINCIPAL DESIGN CRITERIA AND APPLICABLE ACCIDENTS

3-1. Revise the engineering drawings to include the dimensions of seals and sealing surfaces, as well as surface finish specifications. Specify the selection criteria used to determine that the seal sizing is adequate for the service conditions and accident events considered in the safety analyses report. (TCB)

These characteristics are important to assure the seal will perform as intended during operations and accident events.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

3-2. Perform tip-over analyses of the HI-TRAC cask with the STC inside to demonstrate the potential consequences of this event. (TCB)

The analyses can be based on a non-mechanistic tipover over the lower corner onto a receiving surface (which bounds the characteristics of the haul path) from a position of balance, with no initial velocity. The analyses should also address confinement integrity, thermal-hydraulic response, gaseous releases from fuel rods, which also affect the internal pressure, and variations in external dose rate and effects on STC emissions during such an event. Although the applicant has identified the crane, airpad, and vertical cask transport systems as components that preclude tipover, analyses are needed to provide defense-in-depth of the safety of the cask transfer system. Additional guidance on acceptable methods of tip-over analyses is provided in NUREG-1536.

If the assessment demonstrates the STC cannot maintain confinement under such an event, the STC should be redesigned such that confinement is maintained. The consequence assessment will provide insights on the level of confidence (including rigor of license condition design controls and operational controls) that is needed to assure that tipover will be precluded (i.e. low likelihood), as currently maintained by the applicant.

This information is needed to demonstrate that the system can withstand the worst-case loads and successfully preclude an unacceptable release of radioactive materials to the environment, in compliance with GDC 61.

3-3. Analysis of Accidents (SBPB)

In accordance with 10 CFR 50.90 and 10 CFR 50.34(b)(4), the application for a license to support fuel transfer shall include an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Furthermore, the evaluation shall include a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," contains guidance on the evaluation of postulated accidents. This guidance states that the effects of anticipated process disturbances and postulated component failures should be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

GDC 61 of Appendix A to 10 CFR Part 50 specifies that the fuel storage and handling systems shall be designed to prevent significant reduction in fuel storage coolant inventory under accident conditions. In addition, GDC 63 specifies that appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Table 1.2, "Failure Modes and Effects Analysis," and Table 1.3, "Accident/Initiating Events and the Resultant Effects," in Attachment 1 to the supplemental letter dated September 28, 2009, provided analysis of various equipment failures and initiating events. The potential failure modes presented in Table 1.2 were described as being either:

i. ruled out by defense-in-depth operational measures, or
ii. detected and corrected before the loaded cask leaves the Part 50 structure.

Consequently, with the exception of a misloaded fuel assembly, the consequences of accidents listed in Table 1.3 were not evaluated. Instead, operational measures were credited to prevent or provide early detection and correction of conditions beyond design bounds, and potential accidents resulting from improper performance of operational measures were not addressed.

The NRC staff has little confidence that essentially concurrent operating measures would rule-out these events because operating experience indicates that omission or improper performance of one or more concurrent procedurally-directed operational measures has commonly occurred. Accordingly, provide analyses demonstrating the design and operating measures are sufficiently robust to prevent a significant reduction in coolant inventory under conditions that could result from anticipated omission(s) or improper performance, such as establishment of an inadequate HI-TRAC annulus water inventory, failure to add any water to the HI-TRAC annulus, and failure to establish any air gap in either the STC or the HI-TRAC. Consider the following examples of potential approaches:

- Provide a human reliability analysis consistent with the guidelines of Regulatory Position 1.2 of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," that demonstrates a very low frequency (e.g., less than 1 in 1 million operating years) of omission(s) or improper performance of actions that could result in a significant reduction in coolant inventory.
- Provide thermal-hydraulic analyses demonstrating that omissions or improper performance of critical actions would not result in a significant reduction in coolant inventory.
- Modify the design of the HI-TRAC and/or STC to include instrumentation that would, independent from the initial establishment of conditions in the HI-TRAC and STC, ensure appropriate safety actions would be implemented to preserve adequate coolant inventory and residual heat removal in the event of omissions or inadequate operator performance that failed to establish proper initial conditions.
- Modify the design of the HI-TRAC and/or STC to include design features that inherently protect against conditions that could result in a significant reduction in coolant inventory or loss of residual heat removal (e.g., use of dished heads that inherently provide the necessary air space for overpressure protection when installed and provide for enhanced heat transfer between the STC and HI-TRAC). Provide necessary supporting analyses for the design change addressing thermal-hydraulic, shielding, and mechanical protection issues.

3-4. Justify the tornado missile event for the HI-STORM 100 system, as bounding for the IP Fuel Transfer (refer to Table 1.3 in Entergy's Response with Supplemental Information (RSI), dated September 28, 2009.)(SMMB).

Staff notes that the applicant bounds the loaded STC and HI-TRAC missile events from the HI-STORM 100 FSAR and NRC Safety Evaluation Reports as indicated in paragraph (b)(3) of Revision 7 in the HI-STORM 100 10 CFR 72.212 Evaluation Report. However, the original MPC32 and HI-TRAC did not have a large annulus filled with water.

The intermediate missile strike considered in the HI-STORM FSAR (Revision 4 of Report HI-2002444, Section 3.4.8.2.1) will penetrate through the Outer Transfer Lid Door. Consider the new configuration of the STC/HI-TRAC system compared to the MPC/HI-TRAC system and the implication of a breach of the lid door.

Staff notes that the applicant considers that, "ensuring that the annulus water is not lost is a central objective in the system design," as indicated in RSI response 1.b. Analyze the effect on an immobilized STC/HI-TRAC system with a breached outer transfer lid door, and the consequent loss of annulus water if the STC is immobilized outdoors for 30 days due to a transport problem.

This information is required by the staff to assess compliance with GDC-61 and the intent of 10 CFR 72.104, 72.122.

CHAPTER 4 - CRITICALITY EVALUATION (SRXB and CSDAB)

4-1. In HI-2084176 Table 5.7 the METAMIC Boron-10 (B-10) Areal Density (AD) is determined to be 0.032 gm/cm2. How does the STC surveillance program ensure the METAMIC B-10 Areal Density doesn't drop below this value? What are the STC surveillance program's acceptance criteria for the METAMIC B-10 AD? What are the tolerances associated with those acceptance criteria? How are those items covered in the STC criticality analysis?

4-2. HI-2094289 does not appear to consider items such as fuel rod storage baskets (FRSB) or 'dummy' fuel assemblies. Please identify how such items are modeled in the STC or are precluded from being placed in the STC.

4-3. HI-2084176 Section 7.1 addressed the reactivity effect of different fuel assembly types. Please provide the following information concerning the analysis of different fuel assembly types:

a) Table 5.1 describes the design basis fuel assembly. Provide the distance and tolerance from the bottom of the fuel assembly to the bottom of the active fuel. If this distance varies with fuel design, provide design-specific information.
b) Table 7.7 provides information concerning the reactivity effect of different fuel assembly types. However, it is not clear what is being presented in that table. Provide a fuller description of the information in that table; include the description of the simulations used to derive the table.

4-4. Staff guidance is to use the most reactive fuel (Reference 3). NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (Reference 4) provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. NUREG/CR-6665 is focused on criticality analysis in storage and transportation casks and should therefore be an appropriate reference for the licensee's STC. The basic premise is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power, and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. With respect to the IP3 core operating parameters used in the depletion analysis, provide the following information:

a) HI-2084176 Table 5.2 provides the core operating parameters used for CASMO depletion calculations. Were analyses performed to show that these parameters are conservative and bounding?
   i) Note that consideration should be given to the full range of parameters experienced by all fuel currently stored and for fuel that will be stored in the future. Consider too that parameters that lead to spectral hardening and increased plutonium production also reduce depletion of thermal neutron absorbing fission products. It should not just be assumed that anything that hardens the spectrum is conservative.
b) IP3 UFSAR Table 3.2-4 indicates that the Cycle 1 Hot Channel had a core exit temperature of 635.7 °F. IP3 UFSAR Table 3.2-4 also indicates that the Cycle 16 average core exit temperature was higher than that for Cycle 1, indicating that the Hot Channel for Cycle 16 would have had a higher core exit temperature that that used in the licensee's analysis. Identify a bounding hot channel core exit temperature and use that as the core operating temperature for the depletion portion of the analysis.
c) HI-2084176 Table 5.2 indicates a constant soluble boron concentration of 900 PPM was used during the CASMO depletion calculations. Provide the cycle average soluble boron concentration for current and past cycles.
d) For the remaining four parameters discussed in NUREG/CR-6665 provide the ranges of operating parameters affecting the CASMO depletion calculations and provide better justification for bounding values selected.
e) Describe how these parameters will be controlled for future cycles and actions to be taken if a future cycle deviates from the assumption that the parameters used in this analysis are limiting.

4-5. HI-2084176 Section 2, first paragraph discusses how the lumped fission products are validated. The method described is not "validation." The cross-code comparison utilizing the same cross-sections does not tell us anything about the potential composition and cross-section errors and associated biases introduced by the modeling of lumped fission products and use of lumped fission product cross sections. The "5% of the reactivity decrement" suggested by the Kopp memo does not cover modeling simplifications and approximations such as use of lumped fission products. Please provide the following information concerning the use of the lumped fission products:

a) What is the worth of the lumped fission products in the fuel storage racks?
b) What fission products are included in the lumped fission products?
c) What are the cross sections for each lumped fission product? What is the basis for the cross section for each lumped fission product? How do the cross sections of the lumped fission products respond to changes in temperature and spectral hardening?
d) What are the decay constants for each lumped fission product?

e) Are there any neutron absorbers represented in the lumped fission products? What are the cross sections for those neutron absorbers? What are the decay constants for those neutron absorbers?
f) Are there any neutron sources represented in the lumped fission products? What are the source terms? What are the decay constants for those neutron sources?
g) Assumption #4 states, "A conservative cooling time of 0 hours is used along with setting the xenon concentration to zero for all CASMO-4 calculations in the fuel basket models. No credit is therefore taken for the significant cooling time of the fuel assemblies." How does modeling of Lumped Fission Products affect this assumption?
h) Why is isotopic inventory at zero cooling time acceptable for the criticality calculations, given the fact that short-lived fission products will decay out in a few days to a few years?

4-6. The text in Section 7.3, "Isotopic Compositions," describes modeling approximations related to calculation and use of burned fuel compositions. Provide additional detail in this section to clearly state the conditions used in the calculation of fuel compositions as a function of initial-enrichment and fuel burnup.

i) From the text in Section 7.3, it appears that assembly average compositions are used rather than pin-by-pin compositions. This minimizes the impact of reactivity control inserts such as WABA rods. Provide additional detail describing and justifying the use of modeling approximations associated with modeling burned fuel. Where appropriate, include biases and bias uncertainties associated with the modeling simplifications and approximations.

4-7. Section 7.6 of Holtec Report HI-2084176 indicates CASMO-4 was used to calculate the reactivity uncertainties in Table 7.9.

a) Explain what modeling constructs were employed that allowed CASMO-4 to model the asymmetric STC storage cells and distributed axial burnup profiles of the depleted fuel assemblies.
b) Did these CASMO-4 model constructs affect the results?
c) Where appropriate, include biases and uncertainties associated with CASMO-4 model constructs.

4-8. HI-2084176 Assumption # 2 indicates that minor structural components of the fuel assembly such as spacer grids are not modeled because they have a negligible effect on reactivity. Assumption #2 indicates there is a discussion of this in Section 7.9; however, the staff could not find such a discussion in Section 7.9. Provide a justification for not modeling spacer grids.

4-9. HI-2084176 Assumption 7, Section 7.1 addressed the reactivity effect of fuel assembly reactivity control devices, including Hafnium Inserts. Please provide the following information regarding the reactivity control devices:

a) Provide justification for the way these reactivity control devices were modeled throughout fuel assembly life, both in the reactor during depletion simulations and in the STC during criticality calculations.
b) Section 7.1 provided a discussion of fuel assembly reactivity control devices. Confirm that integral (e.g. integral fuel burnable absorbers (IFBAs)) and non-integral (e.g., wet

annular burnable absorbers (WABAs), rod cluster control assemblies (RCCAs), etc.) reactivity control devices are modeled during depletion and removed prior to restart in rack geometry. Verify that the reactivity control devices were removed from the model before the restart in the STC.

c) It appears that IFBA and WABA were not modeled during depletion as being present in the same assembly. Does some mechanical feature or technical specification prevent them from being in the same assembly? Provide justification for not modeling both in the same assembly.
d) It does not appear that depletion with control rods present was considered. Frequently, second- and third-cycle fuel is placed under control rods, some of which may be used to control reactor power. Thus, a realistic fuel depletion scenario could include first cycle depletion with burnable absorbers and second cycle depletion with partially inserted control rods. Some plants have also included part-length absorbers rods in some peripheral locations to reduce neutron flux to reactor vessel welds. In low-leakage loading patterns, these peripheral locations frequently hold fuel that is being used for a third cycle.
e) Tables 7.5 and 7.6 provide information concerning the reactivity effect of fuel assembly reactivity control devices. However, it is not clear what is being presented in those tables. Provide a fuller description of the information in those tables; include the description of the simulations used to derive the tables.

4-10. HI-2084176 Section 7.4 addresses uncertainty in depletion calculations. Please provide the following information regarding the reactivity control devices:

a) It is unclear where the Depletion Uncertainty in Table 7.1 comes from. Using the description in Section 7.4 and the information in Table 7.5, the staff did not calculate the same value. Provide the simulations used to derive the Depletion Uncertainty. Explain how they differ from the simulations used to derive Tables 7.5, 7.6, and 7.7.

4-11. During irradiation in light-water reactors the fuel assemblies undergo physical changes associated with irradiation and residence time in an operating reactor. Some of those changes are clad thinning due to fuel rod growth, fuel densification, collapse of the pellet/cladding gas gap in the fuel rod, and crud build up on the outside surface of the fuel rod. In SFP criticality analyses, fuel has been modeled as fresh and clean fuel. As the fuel undergoes extended burnup and residence in an operating reactor modeling it as fresh and clean becomes even more of an approximation. There is no discussion of how these changes were addressed in HI-2084176. There is no basis in Reference 3 to suggest that the current guidance for the depletion uncertainty provides coverage of changes in fuel geometry during irradiation. Uncertainty in k_{eff} due to random fuel geometry changes should be handled as variations in the parametric ranges of the analysis and included along with the other uncertainties. Variation in k_{eff} due to anticipated geometry changes during irradiation should be handled as a bias. The fuel geometry changes as it is irradiated in an operating reactor. Those physical changes include phenomena such as clad creep, clad thinning, pellet densification, and others. Confirm that the impacts of fuel geometry changes during irradiation are properly handled.

4-12. HI-2084176 Section 7.6 addresses uncertainties from manufacturing tolerances. Please provide the following information regarding the uncertainties from manufacturing tolerances:

a) Where the reactivity worth of an uncertainty is calculated using MCNP, the reactivity worth should also include allowance for the Monte Carlo uncertainty in the calculation of the reactivity worth. Revise the analysis to properly incorporate k_{eff} uncertainties calculated using MCNP.
b) Table 7.9 appears to be incomplete. It appears to be missing columns for two cases: 4.95 w/o ^{235}U with zero burnup and 600 PPM of soluble boron and 4.95 w/o ^{235}U with 40 GWD/MTU burnup and 600 PPM of soluble boron. Provide the information for those cases.
c) There is virtually no information provided about the computer simulations performed to obtain the data that was used to create Table 7.9. It isn't indicated which STC storage configuration the simulations considered. Provide a full description of the computer simulations performed in support of Table 7.9.
d) How were the manufacturing tolerances of integral (e.g. IFBA) and non-integral (e.g., WABA, WDR, RCCAs, etc.) reactivity control devices treated? Provide a justification for the treatment of the manufacturing tolerances of integral (e.g. IFBA) and non-integral (e.g., WABA, WDR, RCCAs, etc.) reactivity control devices.

4-13. HI-2084176 Sections 7.7 and 7.9.1 discussed temperature and water density effects. Please provide the following information regarding the temperature and water density effects:

a) For the temperature bias clarify the source of the bias, the method by which the bias is calculated, and the justification for using CASMO-4 to estimate the bias. Confirm that the results shown in Table 7.8 reflect changes in both water density and temperature.
b) As with Table 7.9, Table 7.8 appears to be incomplete. It appears to be missing columns for two cases: 4.95 w/o ^{235}U with zero burnup and 600 PPM of soluble boron and 4.95 w/o ^{235}U with 40 GWD/MTU burnup and 600 PPM of soluble boron. Provide the information for those cases.
c) Were any sensitivity studies performed at low moderator densities corresponding to steam blankets or firefighting materials? If not perform the sensitivity studies to determine whether there is a reactivity concern at very low moderator densities.

4-14. HI-2084176 Sections 7.8 discusses the calculation of maximum k_{eff} for normal conditions. Please provide the following information regarding the calculation of maximum k_{eff} for normal conditions:

a) Table 7.1 of Holtec Report HI-2084176 is a summation of the biases and uncertainties associated with a fuel assembly with 4.95 w/o enriched U235 and 38.0 GWD/MTU of burnup. That is one entry on Table 7.3. Provide a summation of the biases and uncertainties for the other entries on Table 7.3.
b) Table 7.1 is a summation of the biases and uncertainties associated with a fuel assembly with 4.95 w/o enriched ^{235}U with 38.0 GWD/MTU of burnup. The Reference k_{eff} in Table 7.1 is 0.9118. It is unclear how this value is derived. Tables 7.5, 7.6, and 7.7 indicate the 0.9118 will be greatly exceeded for 4.95 w/o enriched ^{235}U with 38.0 GWD/MTU of burnup. Provide the details of the computer simulations used to arrive at the Reference k_{eff} in Table 7.1. Compare and contrast those computer simulations with those used to produce Tables 7.5, 7.6, and 7.7. Provide this information for all the other entries on Table 7.3.

4-15. HI-2084176 Sections 7.9 discusses the abnormal and accident conditions. Please provide the following information regarding the abnormal and accident conditions:

a) Section 7.9.2 indicates that a drop event of the STC would only result in a negligible change in reactivity. Justify why a drop event need not consider the loss of borated moderator and replacement with unborated fresh water from the HI-TRAC annulus.
b) Table 7.11 of Holtec Report HI-2084176 is a summation of the biases and uncertainties associated with a fuel assembly with 4.95 w/o enriched U235 and 38.0 GWD/MTU of burnup with 600 PPM of soluble boron. That is one entry on Table 7.3. Provide a summation of the biases and uncertainties for the other entries on Table 7.3.
c) Table 7.11 is a summation of the biases and uncertainties associated with a fuel assembly with 4.95 w/o enriched ^{235}U with 38.0 GWD/MTU of burnup with 600 PPM of soluble boron. The Reference k_{eff} in Table 7.11 is 0.9189. It is unclear how this value is derived. Provide the details of the computer simulations used to arrive at the Reference k_{eff} in Table 7.11. Compare and contrast those computer simulations with those used to produce Tables 7.5, 7.6, and 7.7. Provide this information for all the other entries on Table 7.3.
d) Table 7.11 is a summation of the biases and uncertainties associated with a fuel assembly with 4.95 w/o enriched ^{235}U with 38.0 GWD/MTU of burnup with 600 PPM of soluble boron. The Depletion Uncertainty in Table 7.11 is 0.0067. It is unclear how this value is derived. Provide the details of the computer simulations used to arrive at the Depletion Uncertainty in Table 7.11. Compare and contrast those computer simulations with those used to produce the Depletion Uncertainty in Table 7.1. Provide this information for all the entries in Table 7.3.

4-16. It appears that the purpose of HI-2084176 Tables 7.5 and 7.6 is to produce an estimate for the bias introduced by not modeling the integral (e.g. IFBA) and non-integral (e.g., WABA, WDR, RCCAs, etc.) reactivity control devices during the depletion calculations. Since the storage racks do not take credit for the presence of integral and non-integral reactivity control devices, it would seem that these tables should show only the impact of the presence of integral and non-integral reactivity control devices, during the depletion calculations. All of the results should have been restarts in the rack geometry with integral and non-integral reactivity control devices removed. Consequently all "Ref." values and associated k_∞ values at zero burnup should have been the same. With respect to the modeling of the integral and non-integral reactivity control devices during the depletion calculations, provide the following information:

i) Describe the simulations that were performed to derive Tables 7.5 and 7.6, especially the modeling of the integral and non-integral reactivity control devices.
ii) Justify the modeling of the integral and non-integral reactivity control devices.

4-17. HI-2084176 Section 7.9 discusses the analyses performed for abnormal and accident conditions. If the abnormal/accident condition results in an unborated k_{eff} greater than 0.95, soluble boron is credited to ensure k_{eff} remains below 0.95. With respect to the abnormal/accident condition analysis provides the following information:

a) The limiting abnormal/accident condition identified is the misloading of one fuel assembly into either storage configuration. Justify why the misloading of one fuel assembly is limiting.
b) Section 4.4.6.6 of Attachment 1 to Reference 1 indicates that the IP3 required SFP minimum required soluble boron is 1000 PPM while the IP2 required SFP minimum

required soluble boron is 2000 PPM and that the STC is therefore a potential dilution source for IP2 SFP. The discussion mentions administrative controls that will be used to ensure the STC has a minimum soluble boron concentration of 2000 PPM. Rather than rely on administrative controls add a surveillance requirement to the proposed IP3 TS 3.7.18 to ensure the STC soluble boron concentration is above 2000 PPM before being removed from the IP3 SFP.

4-18. Either in Section 4 or in Appendix A to Section 4, the following validation issues should be addressed:

i) State the ranges of parameters that the safety analysis fits within. For example, the minimum and maximum values for ^{235}U enrichment, EALF, soluble boron concentrations, Pu content and composition, etc.

4-19. Describe other features present in the safety analysis models that require validation.

4-20. State the ranges of parameters covered by the critical experiments used in the validation study.

4-21. The staff in its acceptance review requested the applicant to provide justification for the applicability of the selected critical experiments to the code benchmark and USL calculation of the Indian Point fuel transfer cask criticality calculation. In its response, the applicant states that Holtec has performed comprehensive validation calculations in the context of taking burnup credit for Holtec's transportation cask, HI-STAR 100. The applicant advised the staff to read the SAR for HI-STAR 100 cask design. The staff reviewed the SAR for HI-STAR 100 cask design and found the methodology acceptable for the purpose of MCNP code benchmark. However, the applicant has not demonstrated that the results for HI-STAR 100 cask design are applicable to and adequate for the Indian Point spent fuel transfer cask given the facts: (1) the Indian Point transfer cask design takes burnup credit for all isotopes that CASMO keeps track of, except Xenon-135, at zero cooling time whereas the HI-STAR 100 uses only some selected actinides and fission products with minimum of 5 years cooling time; and (2) the HI-STAR 100 applicant analyzed individually the reactivity effects of each isotope via benchmark for all of the isotopes for which burnup credit was claimed, however, there is no such analysis for any of the isotopes in the Indian Point fuel cask design. If burnup credit is desired for a specific isotope, the code benchmark for criticality should be performed to quantitatively determine the reactivity worth and the associated bias and uncertainties. For these reasons, the uncertainties and bias established for HI-STAR 100 may not be applicable. The applicant is requested to provide justification for the applicability of the benchmarks performed for HI-STAR 100 design and provide bias and uncertainty analyses for each of the isotopes for which burnup credits are taken. Provide justification for the applicability of the selected critical experiments to the code benchmark and USL calculation of the Indian Point fuel transfer cask criticality calculation. In addition, in its response to RAI 3b, the applicant discussed, though not sufficiently, benchmarks performed for HI-STAR 100. These discussions should be part of the Safety Analysis Report for the Indian Point fuel transfer cask. It seems appropriate to add the response to RSI 3.b in order to make the SAR complete and consistent. Discuss the applicability of the validation study to the safety analysis models.

4-22. Discuss gaps within the parametric range covered by the validation and, if necessary, additional margin adopted to cover interpolation.

4-23. Discuss extrapolation beyond the parameter range covered by the validation study and, if necessary, additional margin adopted to cover extrapolation.

4-24. Discuss validation gaps (e.g., fission product validation) and, if appropriate, additional margin adopted to cover validation gaps.

4-25. HI-2084176 Appendix A Sections A.2 and A.3 include discussion of comparisons between MCNP and KENO results. From Section A.2: "Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias." These two methods are not independent. They both use data derived from the same cross section measurements and, in part, from the same ENDF/B-V evaluation. Consequently, they include the same nuclear data measurement errors and evaluation errors and would be expected to respond similarly to these errors. The comparisons provided serve only to confirm that both codes respond to the same data, erroneous or not, in the same way. The conclusions in Section A.2 and A.3 on enrichment and B-10 biases based on the KENO/MCNP comparison should not be given any credit.

a) Revise the text in Sections A.2 and A.3 to remove use of code-to-code comparisons.
b) Describe how this affects the analysis.

4-26. HI-2084176 Appendix A Section A.4.1 includes the following statement: "The tendency toward over-prediction at close spacing means that the rack calculations may be slightly more conservative than otherwise." This also means the critical experiment calculations are too high, which is non-conservative. The rack calculations are not "more conservative" because the computational bias also includes this trend.

i) Revise the text to remove claims that the tendency toward over-prediction at close spacing is conservative.
ii) Quantify this potential non-conservatism.

4-27. Section 4.8 of Holtec Report HI-2094289 discusses the acceptability of storing IP3 fuel in the IP2 spent fuel pit. In order for the staff to evaluate whether it is acceptable to store the IP3 SNF in the IP2 SFP provide the following information:

a) A comprehensive comparison of the design of the IP3 fuel assemblies, IP2 fuel assemblies, and the fuel assembly used in the IP2 SFP criticality analysis of record. Include grid strips.
b) A comprehensive comparison of the IP3 reactor and IP2 reactor operating parameters, such as maximum core exit temperature and cycle average soluble boron concentrations, with the values used in the IP2 SFP criticality analysis of record. Include nominal and maximum values.
c) A comprehensive comparison of the IP3 reactor and IP2 reactor use of integral (e.g. IFBA) and non-integral (e.g., WABA, WDR, RCCAs, etc.) reactivity control devices with the values used in the IP2 SFP criticality analysis of record. Include nominal and maximum values.
d) A comprehensive comparison of the IP3 reactor and IP2 reactor axial burnup profiles with the axial burnup profiles used in the IP2 SFP criticality analysis of record. Include nominal and maximum values.

e) Items a through d above are not intended to be an all encompassing list of items the licensee should consider in evaluating the storage of IP3 SNF in the IP2 SFP. The licensee should also include any additional items they self identify.
f) Compare and contrast the above information with regard to criticality in the IP2 SFP.

4-28. In Holtec Report No. HI-2084176 Appendix B, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask," the applicant cites the use of CASMO to determine the Δk_{eff} effect caused by variations of depletion parameters and manufacture tolerances. However, the explanation does not seem to have established a basis for why the difference in k_∞ calculated by the lattice physics code CASMO can be applied directly as quantitative uncertainties and biases of k_{eff} calculated by the criticality code MCNP. In particular, given the fact that the neutron spectrum in a finite system is much softer comparing to that of an infinite system because the fast neutrons in a finite system have a higher leakage probability in comparison with that of an infinite medium, the Δk_{eff} calculated for an infinite system may be not appropriate for use as an adjustment for correcting the effects of the variations of the fuel assembly design parameters. The applicant is requested to demonstrate that the Δk_∞ calculated based on infinite lattice can be applied directly as uncertainties and biases of k_{eff}. In addition, the CASMO code is a discrete ordinate method code whereas MCNP is a Monte Carlo method code. These two codes use different approaches to solve the neutron transport problem and use different schemes of processing cross sections (the former uses fine group cross section library and the latter uses continuous energy cross section library) as well as different basic nuclear data. Therefore, the biases and uncertainties in these two codes may come from different sources, i.e., truncation error caused by discretizing ordinates versus statistical bias associated with the random sampling process. As such, the bias and uncertainties from a discrete ordinate code may not be of the same nature. The applicant is requested to demonstrate applicability of the proposed approach for addressing the code bias and uncertainty question. Demonstrate that the difference in k_∞ caused by variations of fuel assembly nuclear characters, such as depletion parameters and manufacture tolerances, can be used as a basis for determining the k_{eff} difference in criticality calculations.

4-29. On page 5 of the revised Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask," the applicant states: "Benchmark calculations for CASMO-4 for fresh fuel, presented in Appendix B, indicate a very small bias and a bias uncertainty of ± 0.0035 ...." However, the modeled configuration does not seem to be a critical experiment. The applicant is requested to provide justification for benchmark experiments used to determine the bias and uncertainty of the computer codes. Also address the uncertainty in the k_{eff} value derived for any such critical system used as benchmark. In addition, the information presented in Table 7.2 seems to indicate that the bias and uncertainty values are from calculations of different code, i.e., the bias is from MCNP and the bias uncertainty is from CASMO. The applicant is requested to provide explanation for these data and justification why the bias and bias uncertainty from different code calculations can be combined and used as bias and uncertainty of criticality calculations.

a) Provide benchmark calculations to demonstrate that the bias is small for CASMO-4 benchmark with fresh fuel.
b) Provide justifications for the applicability of the benchmark calculation as presented in Table 7.2 in the Appendix B of the revised Holtec Report No HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask."

4-30. In Section B.2 of Appendix B of revised Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask" the applicant discusses the effect and implementation of the bias and uncertainty determined for CASMO-4. However, the conclusions therein do not appear justified because (1) in principle, bias may be canceled out, but uncertainties must be propagated; and (2) using error cancellation does not seem to be a reliable means to achieve reliable results in scientific calculations. The applicant needs to demonstrate that the biases in CASMO calculations are always small or moderate for the system it models. Therefore, the staff requests the licensee to demonstrate that the uncertainties in the criticality calculations can always be cancelled out.

4-31. The applicant provides discussion for code to code comparison as a basis for justifying the use of CASMO-4 code for determining the bias uncertainty of k_{eff} calculation for spent fuel cask. It is a well established and understood principle (refer to ANS 8.1 and NUREG-1617) that code to code comparison is not acceptable as a basis for quantifying the bias and uncertainty of a code. Qualitative comparison is not acceptable either. Provide justification for the discussion on code comparison on pages B-2, B-5, and B-6 of the revised Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask."

4-32. On page B-2 of the revised Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask," the applicant states that benchmarking calculations performed by the developer of the CASMO code were used to determine the bias and bias uncertainty of the code. However, it was not clear how the code was benchmarked by the developer and how the applicability of these benchmarks for this specific application was justified. The applicant is requested to provide benchmark calculations that can demonstrate that the bias and bias uncertainty from the code benchmark can be used for the Indian Point spent fuel transfer cask criticality safety evaluation. Provide a CASMO benchmark calculation that can demonstrate that the bias and bias uncertainty from the code benchmark can be used for the Indian Point spent fuel transfer cask system.

4-33. The applicant, on page B-5 of the revised Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask," states: "The results are summarized in Table B.1. The average k_∞ value is 1.0003, indicating a small bias of -0.0003. The standard deviation is 0.0012. The K-factor for a single sided tolerance limit for 44 experiments is 2.1 for the 95%/95% level. The bias uncertainty at the 95%/95% level is therefore 0.0025. For additional conservatism, a value of 0.0035 is used in the design basis calculations." The applicant further provides information on the selected benchmark experiments in Table B.1. However, the exact configurations of these critical experiments were not clear. It is also not clear how the CASMO-4 code could produce a k_∞ for systems with finite dimensions. Finally, information on the applicability of these benchmark results to the Indian Point spent transfer cask system was not provided. The applicant is requested to provide detailed explanations for the information presented in Table B.1, with respects to the above discussions. Provide detailed explanations for the information presented in Table B.1.

References

1. Entergy Nuclear Operations, Incorporated, letter NL-09-076, J. E. Pollock, Site Vice President, to USNRC document control desk, re: Indian Point Nuclear Power Plant Units 2 and 3, Application for Unit 2 Operating License Condition Change and Units 2 and 3

Technical Specification Changes to Add Inter-Unit Spent Fuel Transfer Requirements, July 8, 2009. (ADAMS ML091940177)
2. Entergy Nuclear Operations, Incorporated, letter NL-09-100, J. E. Pollock, Site Vice President, to USNRC document control desk, re: Indian Point Nuclear Power Plant Units 2 and 3, Response to Request for Supplemental Information Regarding the Spent Fuel Transfer License Amendment Request (TAC Nos. ME1671, ME1672, and L24299), September 28, 2009. (ADAMS ML092950437)
3. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998. (ADAMS ML003728001)
4. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (ADAMS ML003688150)

CHAPTER 5 - THERMAL-HYDRAULIC EVALUATION

5-1. Provide all input and output files used to perform the thermal-hydraulic analyses of the system during normal and accident conditions. (TCB)

The staff needs to verify that the thermal hydraulic system is appropriately analyzed and modeled, in order to verify the acceptability of the temperatures and pressures reported in the application. The files should include, in part, the calculations used to support supplemental responses 2b and 2g. This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-2. Justify and explain how the water level under the STC lid is operationally maintained one inch above the basket. (TCB)

The thermal-hydraulic analysis of the system states the water level under the STC lid must be maintained above the basket to allow convective water motion around the basket. The calculation states water level above the fuel basket inside the STC is 1 inch. Convective heat transfer appears to be the main heat rejection feature for the system to maintain temperatures and pressure below allowable limits. Therefore, the staff needs to have assurance the water level assumed in the thermal-hydraulic analysis is reliably maintained for all operational modes.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-3. Justify why ignoring the volume of grid spacers, top fittings, and bottom fittings would be conservative for the calculation of pressure rise on the STC lid. (TCB)

The application states the volume of grid spacers, top fittings, and bottom fittings are ignored, and that this maximizes the calculated volume of water inside the STC and, hence, is conservative for the calculation of pressure rise on the STC lid. It appears that ignoring this volume would result in having a larger volume of water which would decrease the bulk water temperature and result in a lower pressure. This assumption is used in other calculations, as presented in the application. The applicant should revise the calculation package as appropriate to address the realistic volume of water expected in the cask system.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-4. Justify the initial temperatures used in the thermal-hydraulic analyses of the system for both normal conditions of transfer and accident conditions. (TCB)

The assumed initial temperatures appear to be a critical parameter for subsequent calculations of predicted temperatures and pressure during normal transfer and accident conditions. These predictions are directly depending on the initial assumed value which would imply the thermal-hydraulic analysis is not bounding.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-5. Clarify if the correlation used to calculate the Nusselt number on Appendix G of Holtec Report No. HI-2084146 is applicable to rod array (as the spent fuel assembly case). The staff needs to have assurance an applicable correlation is used in the thermal-hydraulic calculations. (TCB)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-6. Review the heat transfer correlation used to obtain the heat transfer coefficient for the evaluation of the STC without the HI-TRAC. Page I-3 of Holtec Report HI-2084146 includes a correlation to calculate a heat transfer coefficient. The temperature difference is expressed to the power of 1/3 and subsequent use of the correlation is inconsistent with this value. (TCB)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-7. Provide a detailed description and validation of the misload temperature measurement system including any diagrams and detailed operating procedures. (TCB)

The applicant should specify exact radial and axial measurement locations, instrumentation, accuracy and calibration of equipment, validation testing, and operating steps. The applicant should also identify components that are safety related. In addition, the applicant should specify the reliability of this system, and provide any benchmark data for use of this type of system in nuclear industrial applications.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 1, 61, 62, and 10 CFR 50.68.

5-8. Clarify how the temperature probes/system is calibrated to cover the temperature range for fuel misload with very high decay heat. (TCB)

Supplemental response 2b states that the temperature probes/system shall be calibrated and shall be capable of accurately measuring temperature changes in the predicted range. It appears that the predicted range was obtained for the case when no fuel misload has occurred as shown on Figure 2.2 of the reference letter and therefore, will not include the range when a case of fuel misload has indeed occurred. Therefore, the system may not cover the broader range. Additional transient analyses should be performed to predict a range for the fuel misload case.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-9. Revise the operating procedures to specify the exact location (both axial and radial) of the temperature probes in the STC. Ensure the location corresponds to the axial position with the maximum power peaking factor. (TCB)

Supplemental response 2b states that the operational procedures include a step to place calibrated probes at the designated locations. However, the location in the TSC is not specified. Temperature probes located at an elevation corresponding to the maximum peaking factor would react faster to large local decay heat power variations such in the case of a fuel misload event.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-10. Provide the increase in cavity pressures for the misload analyses summarized in Response 2g. Based on pressure and temperature design limits, specify the time to failure of the STC and HI-TRAC seals for this accident condition, similar to the information provided in response to 2b. (TCB)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

5-11. Clarify if other flammable materials (e.g. hydraulic fluids, rubber) are present on the VCT that could contribute to a large duration fire. (TCB)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 3 and 61.

5-12. Protection Against Overpressure (SBPB)

Table 1.3, "Accident/Initiating Events and the Resultant Effects," in Attachment 1 to the supplemental letter dated September 28, 2009, states that the HI-TRAC and STC are ASME Code compliant pressure vessels in discussing the effects of various postulated events. However, the design criteria applied to the HI-TRAC and STC do not include pressure relieving capability as specified in the ASME B&PV Code, Section III, Division I, Subsection ND.

To improve the staff's understanding of the degree of overpressure protection inherent in the design, describe the degree of conformance with Article UG-140, "Overpressure Protection by System Design," from Section VIII of the ASME B&PV Code – 2009b, Division I.

5-13. Perform an analysis to show the effect of a simultaneous loss of jacket water and annulus water from the HI-TRAC, and show the long-term (30 day) effect on a loaded STC. Include the impact on dose rates for personnel in the vicinity. (LPL1-1)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 2, 60, and 61.

CHAPTER 6 - STRUCTURAL EVALUATION OF NORMAL AND ACCIDENT CONDITION LOADINGS

6-1. Provide/justify the lack of fatigue analysis for the pool lid top plate and pool lid bolts (refer to page 6-17)(SMMB).

Report HI-2084118, Appendix C, evaluates the structural adequacy of the HI-TRAC under the design internal pressure and the lid lifting points. Report HI-2084118 indicates that the pool lid will undergo a maximum bending stress of 24,380 pounds per square inch (psi), whereas the allowable is 30,000psi. The pool lid bolts will undergo a maximum stress of 18,380 psi, whereas the allowable is 25,000 psi (thread engagement safety factor is 1.5). The pool lid will undergo a bending stress of 30,190 psi during 3 times lifted load (per Regulatory Guide 3.61 Section 3.4.3), whereas the allowable is 32,500. These values are also included in Section 6.2.3.4.

Considering the low (less than 1.5) safety factors, provide a fatigue evaluation, or justify the lack of one, for the pool lid top plate and pool lid bolts considering the loss of allowable stresses over the intended service life.

This information is required by the staff to assess compliance with GDC-61 and the intent of 10 CFR 72.122.

6-2. Provide clarification regarding the normal, off-normal, and extreme environmental conditions (refer to page 6-1)(SMMB).

Chapter 6 appears to evaluate the STC and HI-TRAC 100D structural performance only for normal and accident conditions. The loading conditions, which are consistent with regulatory requirements and staff guidance, must clearly be delineated and categorized for proper application of structural and stress acceptance criteria.

Revise/supplement the Section 6.0 statement, "The objective of structural analyses is to ensure that the integrity of the STC and HI-TRAC is maintained under normal, off-normal, and extreme environmental conditions as well as all credible accident events," by noting the off-normal and extreme environmental conditions are not defined in this SAR chapter.

This information is required by the staff to assess compliance with GDC-4, GDC-61, and the intent of 10 CFR 72.122.

6-3. Revise the reference to "Environmental" load (refer to page 6-3)(SMMB).

Section 6.1.2 states, "Principal design criteria for the design basis, normal condition, and the accident/environmental loads are discussed in Sections 3.2 and 3.3. In this section, the loads, load combinations, and the required structural performance of the STC and the HI-TRAC under the various loading events are presented."

However, Section 3.3 does not exist in the report.

Remove misleading words and revise, as appropriate, the first sentence of Subsection 6.1.2 to read similar to: "Principal design criteria for the design basis, normal condition, and accident loads are discussed in Section 3.2."

This information is required by the staff to assess compliance with GDC-4, GDC-61, and the intent of 10 CFR 71.122.

6-4. Clarify the lift load design criteria (refer to page 6-5)(SMMB).

Section 6.1.2.2 states, "However, for conservatism the primary stresses in STC and the HI-TRAC components in the lift load path are set to meet the smaller of 1/10 of the material ultimate strength and 1/6 of the material yield strength under a normal handling condition."

The subject statement should be consistent with the Item 1 description of SAR Subsection 6.2.3, which refers to ANSI N14.6 as the basis for the proposed stress limits for the lifting load path evaluation.

Revise this statement, as appropriate, to recognize that ANSI N14.6 is implemented for the lifting load path evaluation.

Section 8.5.3.3 states, "All special lifting devices shall be maintained and inspected in accordance with ANSI N14.6." NUREG-1567 and ANSI N14.6 require preservice testing of 150 percent of the maximum service load if a dual-load path is employed or at 300 percent of the service load if a single-load path is used.

In Section 9.1.2.1 of the HI-STORM FSAR (Revision 4 of Report HI-2002444, Section 3.4.8.2.1), the applicant has committed to the 300 percent acceptance test.

Provide evidence that this condition is met for the IP Fuel Transfer.

This information is required by the staff to assess compliance with GDC-61, and the guidance in NUREG-0800 15.7.5, RG 1.183, and the intent of 10 CFR 72.122.

6-5. Clarify/Revise the Allowable Stress material properties (refer to pages 6-7 and 6-8)(SMMB).

Tables 6.1.3 and 6.1.4 in Section 6.1.2.2 illustrates the Maximum Allowable Stress and Primary Stress for SA-564, 630 and SA-193, B7 bolt materials.

Primary stresses must be defined in stress categories to facilitate bolt structural evaluation. The tables seem to suggest that only the membrane stress is evaluated for the closure bolts. Therefore, it is unclear as to whether shear and bending stresses must also be considered in the closure bolt evaluation.

Revise Tables 6.1.3 and 6.1.4 to also recognize stress allowables for the complete primary stress categories, including membrane and bending stresses and bending and shear stress interaction equations, for the STC and HI-TRAC 100D closure bolts evaluation.

This information is required by the staff to assess compliance with GDC-61, and the intent of 10 CFR 72.122.

6-6. Clarify the "prestressed" state of the closure bolts (refer to pages 6-11 and 6-12)(SMMB).

Section 6.2.1.1 states, "The bolts shall be prestressed to the corresponding maximum resultant load."

Clarify the statement and provide an evaluation of bolt structural integrity, including fatigue damage and thread overstress failure modes, as related to the applied bolt torque and load induced tensile, bending, and shear stresses (see the structural RAI regarding the fatigue for the STC closure bolts).

This information is required by the staff to assess compliance with GDC-61 and the intent of 10 CFR 72.122.

6-7. Verify the Design Basis Earthquake (DBE) is applicable to loading/unloading zones of the IP Fuel Transfer (SMMB).

Section 6.2.6 and 6.2.7 discusses the seismic stability of the loaded VCT and HI-TRAC, as well as the seismic stability of the STC in the fuel pool.

It is unclear if the DBE is amplified at the location of loading/unloading of the STC into the HI-TRAC and the HI-TRAC onto the VCT during the IP Fuel Transfer.

Define and evaluate the applicable seismic stability loading and unloading conditions for the loading/securing a loaded STC and HI-TRAC onto the VCT, as well as the loading/securing of the STC into the HI-TRAC.

This information is required by the staff to assess compliance with GDC-1, GDC-2 and GDC-4, and the guidance in NUREG-0800 9.1.5; NUREG-0612 (Appendix C); NUREG-0554; ASME NOG-1 (2004), and the intent of 10 CFR 72.122.

CHAPTER 7 – SHIELDING DESIGN AND ALARA CONSIDERATIONS

7-1. Provide the calculation that justifies the leak rate value used during testing for the STC (confinement boundary), employing the methodology from ANSI 14.5 for normal, off-normal and accident conditions. The percent of spent fuel postulated to fail for these conditions typically is 1%, 10%, and 100%, respectively. (TCB)

This methodology should identify the complete source term associated with the fuel and crud, determination of the A2 for this mixture, and finally calculation of a referenced leak rate. Also, refer to NUREG/CR-6487, "Containment Analysis for Type B Packages Used to Transport Various Contents," for description of methodology to determine a leak rate test value. Also refer to NRC's Division of Spent Fuel Storage and Transportation (SFST), Interim Staff Guidance (ISG) - 5, Revision 1. It does not appear credit can be provided by the HI-TRAC cask because it is not part of the confinement boundary.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and 10 CFR Part 50, Appendix B, Criterion III, Design Control. Also, ASME Code, Section III, NCA-4000, Article 4134.3(a) references ASME NQA-1, where in

Supplement 3S-1, Section 3.1, it states that design analyses such as physics, stress, hydraulic, and accident shall be performed in a planned, controlled, and correct manner.

7-2. Evaluate the source terms for spent fuel fines and crud in the calculation for determining the referenced leak rate and effluent dose evaluation. (TCB)

The applicant in Section 7.4.6 "Effluent Dose Evaluation" states that only gases were considered. This assumption does not appear to be conservative in that discharge from a failure of the spent fuel would result in discharge of fines and volatiles, in addition to gases, into the STC water. These fines could remain suspended in the water due to the vibration associated with movement of the STC and could be available for release. Since the STC water could be pressurized and the temperature above 212 °F (100 °C), that would supply the energy to expel the fines past the confinement boundary.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and 10 CFR Part 50, Appendix B, Criterion III, Design Control. Also, ASME Code, Section III, NCA-4000, Article 4134.3(a) references ASME NQA-1, where in Supplement 3S-1, Section 3.1, it states that design analyses such as physics, stress, hydraulic, and accident shall be performed in a planned, controlled, and correct manner.

7-3. Section 4.7.5.3, "Effluent Dose Evaluation," of Attachment 1 to the July 8, 2009, letter from Entergy (ADAMS Accession No. ML091940177) states that the atmospheric dispersion factor (χ/Q value) used in the dose assessment is analyzed for a distance of 100 meters, based upon RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Section 4.7.5.3 further states that this distance is also consistent with 10 CFR Part 72 and that use of the χ/Q value resulted in dose values "so low that this utilized distance is sufficient for the current case." Table 7.4.4, "Analysis Inputs for Effluent Dose Release Analysis," of Enclosure 2 to the July 8, 2009 letter lists the χ/Q value as 8.0×10^{-3} s/m^3. (AADB)

a. NRC staff notes that RG 1.145 does not recommend a specific distance (e.g., 100 meters) for which a χ/Q value should be calculated. Further, while 10 CFR 72.106, "Controlled area of an ISFSI or MRS," states that the nearest boundary of the controlled area must be at least 100 meters, it does not state that a distance of 100 meters is necessarily adequate. Therefore, was the distance of 100 meters chosen because it bounds the minimum source to receptor distance for all potential receptors? Please provide a list of the receptor locations and minimum distances of separation between the fuel and receptor locations for each limiting effluent dose receptor (e.g., control room air intake/operator, exclusion area boundary). Provide a scaled site drawing showing true north, which highlights the source and receptor locations, including the transit route of the fuel, from which distance and direction inputs can be approximated. Also, please provide the scale of the drawing.

b. RG 1.145 provides a methodology for calculation of χ/Q values based on meteorological data representing 1 hour averages, which are then used as a basis for determining χ/Q values for other, longer, time periods. Table 7.4.5 of Enclosure 2 to the July 8, 2009, letter provides a summary of estimated doses for three time periods, namely, 8 hours, 30 days, and an instantaneous release. For what time period was the χ/Q value of 8.0×10^{-3} s/m^3 calculated? Was this χ/Q value the result of an initial calculation or is it generated from another χ/Q value? What are the bases for use of this single χ/Q value? Were calculations made for other χ/Q values and the value of 8.0×10^{-3} s/m^3 found to be limiting for all receptors and time periods applicable to this license amendment request?

c. Please provide a description of the methodology, assumptions, and inputs used to calculate all of the χ/Q values that were assessed to result in selection of the value of 8.0×10^{-3} s/m^3. What are the bases of selection for each input? If a computer code was used to perform the calculations, please provide an electronic copy of input files (e.g., meteorological data) sufficient to permit verification of the χ/Q values.

7-4. In Section 7.4.6 of the license amendment request, entitled Effluent Dose Evaluation, the licensee asserts that "Doses from submersion in the plume are neglected because they are shown in [K.A] [HI-2002444, Latest Revision, "Final Safety Analysis Report for the HISTORM 100 Cask System", USNRC Docket 72-1014] to be small compared to inhalation doses." (AADB)

The NRC staff has performed confirmatory dose consequence analyses to verify the dose values shown in LAR Table 7.4.5, "Dose from Effluent Release at 100 Meters." Based on the NRC staff's preliminary results, it appears that the dose contribution from Krypton 85 (Kr-85) was not included in the reported dose values. While the contribution from Kr-85 may not be limiting in the accident analyses, it may not necessarily be negligible, especially in the evaluation of the total effective dose equivalent (TEDE) for the instantaneous accident release.

Please provide additional information describing the basis of the statement that "Doses from submersion in the plume are neglected because they are shown in [K.A] to be small compared to inhalation doses." Please include a numerical analysis showing the calculated submersion dose and an explanation of why its inclusion in the doses shown in Table 7.4.5 is not necessary.

7-5. Evaluate the azimuthal variation of dose rates around the loaded STC and the loaded HI-TRAC and describe how fuel transfer operations (as described in Report HI-2094289) account for this variation to keep doses ALARA. (CSDAB)

The STC basket is designed with an inner and an outer region of fuel cells. The outer cells are for fuel with less decay heat. However, the outer cells do not completely shield the inner cells; thus, dose rates around the loaded STC and around the loaded HI-TRAC can have significant variations. The applicant should provide an evaluation of the dose rate variations due to this regionalized loading configuration. The evaluation should describe the azimuthal dose rate variations at the STC and HI-TRAC surfaces and anticipated worker distances from the cask. The applicant should also describe how operations with the loaded STC and the loaded HI-TRAC consider this variation in maintaining occupational doses ALARA.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104(b) and 72.126(a).

7-6. Modify Chapter 10, "Operating Procedures," and Chapter 7, "Shielding Design and ALARA Considerations," in Report HI-2094289 to include operations involving the new Bottom Missile Shield (BMS) and dose evaluations of those operations. (CSDAB)

In response to NRC's Request for Supplemental Information (RSI) questions 1.a and 1.c (Section A of the request), the applicant introduced a BMS to protect the HI-TRAC's pool lid-to-bottom flange seal from a tornado missile. However, it is not clear how this shield is to be used, such as when it is installed and when it is removed from the HI-TRAC during the fuel transfer operations. The occupational dose evaluations need to be updated to account for operations with the BMS, as appropriate (e.g., for installation of the BMS on a loaded HI-TRAC).

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104(b) and 72.126(a).

7-7. Clarify the specifications of the STC's allowable non-fuel hardware (NFH) contents and loading restrictions, providing appropriate supporting evaluations. (CSDAB)

The application indicates that NFH may be transferred with the fuel in the STC, and a shielding evaluation including analysis for burnable poison rod assemblies (BPRAs) was provided as part of the response to staff's RSI. However, the specifications and limits for the allowable NFH remain unclear. The applicant needs to provide the following information: the types of NFH to be transferred, their radiation source terms (including source strength) and the basis for these specifications, the allowable numbers of the NFH types, the acceptable STC basket cells, and the burnup and decay time specifications.

Additionally, it is not clear that the shielding evaluation with BPRAs is bounding for a loaded STC (or HI-TRAC) containing NFH. The applicant should provide an evaluation with the bounding NFH source and justify the evaluation to be bounding, considering the NFH specifications described in the preceding paragraph and the STC configuration. For example, HI-STORM 100 shielding analyses have indicated that axial power shaping rods (APSRs) or control rod assemblies (CRAs) can have significantly higher source terms than the analyzed BPRAs. While limiting placement of these NFH types to the inner basket cells provides some shielding, similar to what was for MPCs, the STC basket configuration has less shielding of its inner cells than the MPCs due to fewer basket cells and outer cells not completely surrounding inner cells (see also RAI 7-1). This is particularly the case for APSRs. Additionally, neutron source assemblies need to be addressed if they too are to be transferred.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 20.1301(b), 10 CFR 50.90, and 50.34a(c), and the intent of 72.104, 72.106(b) and 72.126(a).

7-8. Specify the following regarding the response to the RSI (question 3.a of Section B of the request) that discusses the dose rate measurements taken upon removal of the STC from the spent fuel pool and their use (i.e., their comparison to predetermined, expected values and performance of further evaluations if the measurement results exceed these values). (CSDAB)

a. the number of measurements and relative locations of the measurements, including their basis. The measurements should be sufficient in number and appropriate in location to serve the purpose for which they are used.

b. the predetermined, expected values that will be used in making these comparisons and the values' basis. The values may be based upon the actual loaded contents for a particular transfer operations or the evaluation with design basis contents. In any case, the values should be based upon a loaded STC in the same configuration as that for which measurements will be performed.

c. the actions that will be taken if the expected dose rate values are exceeded (e.g., by 25%, 50%, 100%, etc.) and at what values the STC will be unloaded and the contents re-loaded to lower dose rates to within the expected values.

Similar information should also be given regarding the measurements taken on the HI-TRAC that are described in the operating procedures (Section 10.2.3, Step 33 of HI-2094289).

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90, 50.34a(c) and the intent of 72.104.

7-9. Include dose rate measurements on the STC side as part of the measurements described in the inter-unit transfer operating procedures and RSI response, considering the information in RAI 7-8 in determining what measurements will be necessary. (CSDAB)

It is not clear that measurements on the STC lid alone are sufficient to fulfill the purposes of the dose rate measurements described in the operating procedures. For example, some misloaded contents (e.g., some NFH) may not be detectable at the STC lid because their source is distributed in the lower areas of the cask. Additionally, transfer operations personnel will be present around the side of the cask (when in the HI-TRAC and nearby when the STC is raised from the spent fuel pool) as well as around the STC top during the various transfer operations. Similarly, appropriate measurements should be performed on the loaded HI-TRAC as part of the measurements described in the operating procedures (Section 10.2.3, Step 33 of HI-2094289).

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104.

7-10. Provide dose evaluations that clearly demonstrate compliance with the regulatory dose limits for members of the public. (CSDAB)

These evaluations should include all direct radiation and effluent contributions (i.e., from transfer operations and site facilities, including routine plant operations and the ISFSI). It is not clear that this is done for all the evaluations (e.g., the 20.1301(a) and (b) evaluations). The evaluations should also include off-normal conditions or anticipated occurrences for transfer operations. The applicant should also describe the haul path and site features such as the ISFSI, the site boundary, the Part 20 controlled area and restricted area, the controlled area defined for 72.104 evaluation purposes, the ISFSI's Part 72 controlled area boundary, and the minimum distances from the haul path to these various boundaries, providing a site map to illustrate these and other relevant features. Realistic occupancy factors should be used in determining the maximum doses to real members of the public. The bases for these factors should be described and consider any member of the public that may reside, work, or recreate in or near the areas addressed for the various evaluations. Estimates of effluent contributions should be determined using appropriate, conservative parameters for the distances from the loaded STC, loaded HI-TRAC, and site facilities. For example, estimates based upon parameters for distances of 100 meters are not appropriate for distances such as at the haul path's restricted area boundary.

Estimated doses for these evaluations should be based upon the appropriate dosimetry concepts and dose factors for the respective regulatory limits. For example, the estimated annual doses to the highest exposed real members of the public in the general environment (outside the site boundary), and the annual doses to on-site members of the public located within either the 10 CFR Part 20 controlled area or restricted areas should be based on ICRP-2 dosimetry concepts and dose factors (e.g., RG 1.109). Doses determined for members of the public in the general environment outside the site boundary (in accordance with 10 CFR 20.1301(e)) should be estimated for the whole body, thyroid, and other (highest) organ. In addition, estimated annual doses to on-site members of the public in the controlled and restricted areas should be based on total effective dose equivalent (in accordance with 10 CFR 20.1301(a) and (b)). For this calculation, dose factors provided in the Environmental Protection Agency's (EPA's) Federal Guidance Report No. 11 are acceptable.

Staff notes that some information regarding dose estimates was supplied in the application and in supplemental information submitted in response to the acceptance review letter. However, it is not clear if and how this information is used or its applicability for demonstrating compliance with dose regulations. Staff also notes that SFST's ISG-13 provides useful guidance for 10 CFR 72.104 evaluation purposes.

This information is needed to confirm compliance with 10 CFR 20.1301(a), 20.1301(b), 20.1301(e), 10 CFR 50.90, and 50.34a(c) and the intent of 72.104 and 72.126(d).

7-11. Include the effluent contribution to the occupational dose estimates for the transfer operations. (CSDAB).

Occupational dose is derived from direct radiation and effluents, or releases, from the loaded STC/HI-TRAC. The effluent contribution should be determined based upon parameters appropriate for the transfer operations personnel locations.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104.

7-12. Update all affected tables in the shielding evaluation and supporting documents to account for NFH dose rate/dose contributions. (CSDAB)

In response to staff's RSI, the applicant modified its evaluation to address NFH; however, some of the evaluation results, including occupational dose estimates, were not modified. The applicant should ensure that all affected items are appropriately modified to account for NFH.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104(a) and 72.106(b).

7-13. Modify the dose analysis for off-normal events to account for breaching of 10% of the fuel rods. (CSDAB)

As described in the RAI on the fraction of rods assumed breached, the off-normal conditions should be analyzed for 10% of the fuel rods breaching. Dose estimates for off-normal conditions should be modified to account for this fraction of rods being breached.

This information is needed to confirm compliance with 10 CFR 50.90 and 50.34a(c) and the intent of 72.104(a).

7-14. Justify the assembly hardware's cobalt impurity level assumed for analyses with older fuel (fuel with cooling times approximately 20 years or more) used for the dose evaluations. (CSDAB)

The applicant assumes a cobalt-59 impurity level of 1 g/kg for fuel assembly hardware, consistent with the HI-STORM 100 FSAR analyses. For the HI-STORM 100, this was accepted because the design basis calculations used assemblies cooled for only a few years and therefore would have been made after efforts were begun to reduce cobalt impurity levels in assembly hardware materials. Additionally, evaluations showed that the cobalt source strengths of older assemblies' with higher impurity levels were approximately equivalent to that of younger fuel, given the older assemblies' longer decay time. The current application, however, uses older fuel (decay times of 20 or more years) for some of the dose evaluations. Using this lower impurity level would be non-conservative for those dose evaluations. Based upon PNL-6906, Vol. 1, older assemblies could have as much as an impurity level of 2.2 g/kg in the steel hardware. This higher impurity level could have a significant effect on dose rates, particularly in the top, upper cask side, and bottom areas of the cask. The shielding evaluation should use appropriate cobalt impurity levels in the assembly hardware and adequately justify the selected impurity levels. This justification could include any available information regarding measurements of the composition of the assembly hardware used by the IP3 assembly manufacturer(s). The applicant should also consider whether the appropriate impurity level is used for the NFH source determination, providing appropriate justification.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104 and 72.106(b).

7-15. Justify use of the selected source terms in the regionalized loading scheme and the bounding nature of the dose rates and dose estimates using these sources. (CSDAB)

The application should provide an evaluation of the bounding dose rates and provide bounding dose estimates to demonstrate the ability to comply with the regulatory requirements for shielding and radiation protection for the design of the inter-unit transfer equipment and the allowable contents. A regionalized loading pattern is used for some dose evaluations in the application. It is not clear that the selected sources for these evaluations are bounding for the proposed allowable contents. Evaluations with the bounding sources provide an indication of the dose rates (and thus the doses) that can occur during transfer operations and enable proper ALARA planning as well as provide for demonstration that transfer operations with all proposed allowable contents will meet the regulations. Staff also notes that information available to it regarding the IP3 discharged fuel indicates there are a significant number of assemblies with enrichments below the minimum enrichments assumed for their respective burnups in the evaluation. The evaluation and justification should account for these lower enrichments for their respective burnup ranges.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104.

7-16. Justify the homogenization of the fuel assembly with the moderator in the shielding model. (CSDAB)

While staff has accepted the practice of homogenizing the fuel assemblies in previous dry storage cask applications, the evaluations for regulatory compliance in those applications have relied upon design basis calculations with the cask interior dry. The current application relies upon calculations with the STC basket flooded with moderator. It is not clear that homogenization, in the case of the current application, is appropriate when considering that subcritical multiplication, and thus the neutron dose rates, may be significantly underestimated. The neutron dose rates are a significant portion of the overall dose rates from the STC and HI-TRAC. Justification should include an evaluation of the impacts on source terms and dose rates.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104 and 72.106(b).

7-17. Justify the dose estimates for the transfer operations personnel considering the non-conservatisms in the shielding models for the STC and HI-TRAC. (CSDAB)

The applicant uses the dose rates from around the radial surface of the STC and the HI-TRAC to provide estimates of occupational exposures, including for operations around the STC and HI-TRAC top surfaces. However, it is not clear that these dose rates are appropriate or conservative for the operations around the top surfaces, especially in light of the actual cask configuration, which differs from the model due to various non-conservative simplifying assumptions. For example, the current models have the lead shielding in both the STC and HI-TRAC extending from their bases to their lids, replacing the steel of their top flanges. The HI-TRAC is also modeled with the neutron shield extending from its base to its lid. Additionally, the loaded HI-TRAC is modeled with the lid in place and extending out to cover the neutron shield and the annulus water at a height that is about 9 inches greater than that described in the package operations chapter of the inter-unit transfer report (HI-2094289). In some operations, the HI-TRAC lid will not be in place. Thus, due to axial differences in the shielding configuration, it is not clear that axial mid-height dose rates are conservative for all operations. The applicant should also justify its conclusion that the non-conservatisms do not affect dose rates at distance.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104(b).

7-18. Clarify the following items: (CSDAB)

a. the reported surface dose rates (STC and HI-TRAC) are from surface detectors or point detectors, explaining the basis if from the former
b. the 72.106(b) evaluation dose rates use the same source as the 72.104(a) evaluation, explaining the basis for using a different source if not the same
c. the dose condition description for the operations steps in Table 7.4.6 of the inter-unit transfer report (the report), the description is unclear and/or appears to be inconsistent with operations descriptions and shielding models
d. the correct values and title for Table 7.4.2 of the report; the title doesn't match the loading pattern description given for this table in Section 7.4.4 of the report, and some values are inconsistent with the comparable table in the supporting calculation package
e. the correct correlation between dose rate tables and operation steps for dose estimates in Table 7.4.6 or the report and its companion table in the supporting calculation package; doses for some steps appear to be described as being derived

from an incorrect table per the footnote on the companion table in the calculation package.

f. The dose rate and dose calculations at distance include contributions from the cask (STC or HI-TRAC) lid as well as its side; calculations at distance should include all appropriate contributions

This information is needed to confirm compliance with 10 CFR 20.1101(b), 1301(a) and (b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104 and 72.106(b).

7-19. Modify the application to address the potential for streaming through the vent and drain ports on the STC. (CSDAB)

Spent fuel canisters usually have some kind of feature to reduce radiation streaming through vent and drain ports. The technical drawings do not show such a feature for the STC; therefore, the vent and drain ports represent streaming paths through the STC lid. The applicant should address streaming through these ports as part of the shielding and radiation protection evaluation and make appropriate modifications to operations descriptions (e.g., including ALARA warnings and cautions for operations involving and occurring around the vent and drain ports).

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104(b).

CHAPTER 8 – MATERIALS EVALUATION, ACCEPTANCE TESTS and MAINTENANCE PROGRAM

8-1. Justify the testing performed on the entire STC confinement boundary. (TCB)

The entire confinement boundary should be pressure tested in accordance with ASME Section III, Subsection NB or NC requirements. The acceptance criteria for the ASME pressure test would be no visual leakage after the pressure had been maintained for a minimum of 10 minutes.

Additionally, the entire confinement boundary should be leak tested in accordance with the guidance of ANSI 14.5-1997 to verify compliance with the design leak tightness as determined in RAI 7-1, above. This leak testing should be performed initially at the fabrication facility and periodically (within 12 months prior to each use) to ensure that the containment leak tightness has not deteriorated over time. The leak testing done at the time of loading fuel is usually less stringent and is done to ensure that the gaskets are properly seated and the containment has been assembled properly, and typically is checked to be at least 1E-3 ref-cm^3/sec.

It isn't clear from the application whether or not the seals are replaced after every use, but it should be made clear that if the seals are replaced then the leak test should confirm that the design leak tightness as determined in RAI 7-1, above, is in agreement with the guidance of ANSI 14.5-1997.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and 10 CFR Part 50, Appendix B, Criterion IX- Test Control.

8-2. Revise the design to specify bolted closure plates with O-Rings (or equivalent confinement capability) and associated testing procedures for the vent and drain valves to ensure a positive testable closure for this portion of the confinement boundary. (TCB)

Valves are not typically considered part of the confinement boundary given the potential for leakage. In addition, the current design configuration with protruding valves above the closure lid could result in damage from errors handling the STC. In addition, it is not clear from the information provided whether these valves are tested to meet design leak tightness requirements.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and 10 CFR Part 50, Appendix B, Criterion IX- Test Control.

8-3. Justify and provide details of the pressure testing and leak testing performed on the entire HI-TRAC pressure fluid boundary. (TCB)

The HI-TRAC should be pressure tested before its use, as an ASME Code vessel. Because the HI-TRAC has a bolted bottom flange that is subjected to the deadload of the STC and its contents during transfer, the pressure test should replicate this normal configuration. Besides performing the pressure test to the ASME Code required factor of the HI-TRAC's design pressure, this test pressure should be increased by an amount equivalent to the maximum weight of the STC and its payload, and tested in the transfer orientation to apply this maximum loading to the HI-TRAC's bottom flange. As currently described in HI-2094289, it appears only the top seal of the HI-TRAC is leak tested. The bottom seal needs to be similarly tested. The entire fluid boundary should be checked for leakage after each loading.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and 10 CFR Part 50, Appendix B, Criterion IX- Test Control.

8-4. Provide information on the specific material compounds and manufacturing data that validates the elastomeric seals will perform under the cyclic operating conditions specified for the STC and HI-TRAC system. Provide quantified parameters for the critical performance characteristics of the seal that are qualitatively specified in Section 8.2, and demonstrate that selected seal(s) meet these critical performance characteristics. (TCB)

HI-209428 generally states that silicone, neoprene, and similar elastomers may be used. Supplemental response 6m also states that silicone will be used and the temperature limit is 248 °F with no basis for this determination to be found in the generic description of the seals. A specification of material compounds and parameters is required to ensure seal integrity during repetitive use during normal and off-normal conditions, as well as potential accident events. The critical performance characteristics should include time-dependent temperature conditions with stress-relaxation effects, as well as other chemical, physical, and nuclear impacts on the confinement seals.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

8-5. Clarify the conditions that may affect sealing capability and require replacement as specified in Section 8.5 of the maintenance program for the seals. Clarify the replacement frequency for the seals and verify that the procedures are consistent with the validation of the seal performance as discussed in the RAI 8-4 above. (TCB)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 1 and 61.

8-6. The September 28, 2009, supplemental letter states that periodic surveillance of the Metamic in the STC is unnecessary. However, all of the plants cited in your letter as precedents for wet storage application of Metamic are committed to surveillance programs that include neutron attenuation testing on a pre-defined schedule. Diablo Canyon is the only exception because their temporary cask is only licensed for three operating cycles. Since our initial review of Metamic in 2003 (see the NRC Safety Evaluation, ADAMS Accession No. ML031681432), the NRC staff has required applicants to perform periodic surveillance of their Metamic. Since the STC will experience a unique service environment including wet, dry, hot and cold conditions the staff concludes that periodic surveillance of the Metamic to ensure it is capable of performing its intended function is even more important than in the currently licensed SFP applications. The effects of the cyclic environment on the Metamic in the STC over a long service life are unknown. Given that the Metamic in the STC is clad in stainless steel, visual inspections will not be possible, nor would they provide indication of the neutron attenuation capabilities of the Metamic. For this reason, the staff feels that a periodic surveillance program using in-situ neutron attenuation testing of a representative sample of the Metamic panels in the STC would provide reasonable assurance that the Metamic maintained the ability to perform its intended function over the service life of the cask, including the prevention of neutron streaming. Considering the above discussion, please provide your plans for ensuring the neutron attenuation capability of Metamic over the service life of the STC. If a surveillance program will be employed, please provide the proposed interval between samples, the number of samples taken during each surveillance, the specific test method used and the acceptance criteria. (CSGB)

8-7. Justify the lack of fatigue analysis and service life of 240 cycles for the STC closure bolts (refer to page 8-13)(SMMB).

RSI response (Attachment 1 to NL-09-100, dated September 28, 2009) 6.b states that, "The resulting cyclic loading produces stresses that are well below the endurance limit of the canister's materials, and therefore, will not lead to a fatigue failure in the STC. However, the applicant does not quantify the allowable stresses of the STC closure bolts after 240 cycles, or indicate the tracking process of accounting for this time limit.

Report HI-2084118, appendix B evaluates the STC base plate, the shell, the closure lid, and associated connections under the design internal pressure and lifting scenarios. Report HI-2084118 indicates that the bolts undergo a maximum stress of 24,380 psi, whereas the allowable is 28,000 psi, under the design pressure and the loads experienced during lifting.

Provide an evaluation of STC closure bolt structural integrity, including fatigue damage and thread overstress failure modes (tensile, bending, and shear), as related to the applied bolt torque and load.

This information is required by the staff to assess compliance with GDC-61 and the intent of 10 CFR 72.122.

8-8. Justify the use of ASME Code Subsection ND for the design and testing of the STC instead of Subsection NB (SMMB).

Spent fuel canisters are normally designed and tested to Code Subsection NB or NC. The STC is designed to Subsection ND. This approach does not appear to provide the same degree of quality for a spent fuel storage or transportation canister.

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 51 and 61, and the intent of 10 CFR Part 72.122(a).

8-9. Related to the previous question, show how the brittle fracture performance of the steels selected for the STC shell will provide an equivalent level of quality or safety as for austenitic stainless steels designed under subsection NB (SMMB).

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 51 and 61, and the intent of 10 CFR Part 72.122(a).

8-10. Discuss the types and frequency of STC and HI-TRAC interior inspections. (SMMB)

Section 8.5.3 discusses STC and HI-TRAC exterior inspections but is silent regarding the possible degradation and inspection of the STC and HI-TRAC interiors.

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 1 and 61, and the intent of 10 CFR Part 72.122(a).

8-11. Describe how the lead sheet shielding material of the STC is installed to avoid gaps at the edge of the sheets which could result in radiation streaming. (SMMB)

It is not clear from the drawings or SAR description how the butt joints of the lead sheets are arranged or how the shielding at the STC bottom and top to sidewall corners is arranged to avoid gaps in the shield.

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61, and the intent of 10 CFR 72.126.

CHAPTER 10 - OPERATING PROCEDURES

TECHNICAL SPECIFICATIONS

The NRC staff's approach is that normally there is a Certificate of Compliance (CoC) for a spent fuel cask system, and the CoC cannot be revised without prior NRC approval. As this licensing action is being done under Part 50, there is no CoC. Therefore, all the cask system requirements which would normally be located in the CoC must now be located in the Part 50 license (which includes Technical Specifications). The staff is willing to discuss proposals on how to accomplish this.

10-1. State how the proposed TSs meet the regulatory requirements of 10 CFR 50.36. (ITSB)

The TSs are derived from the plant safety analyses. Proposed revisions to TS must provide continued assurance that all assumptions, restrictions, and requirements of plant safety analyses will be met.

The proposed new TSs, Unit 2 TS 3.7.15, "Shielded Transfer Canister (STC) Unloading," and Unit 3 TS 3.7.18, "Spent fuel Assembly Transfer" contain no restrictions on STC water level, water boron concentration, air gap, pressure, or temperature.

10-2. Explain how all assumptions and requirements of all the safety analyses will be met with no restrictions on STC water level, boron concentration, air gap, pressure, or temperature. (ITSB)

10-3. If all assumptions and requirements of all the safety analyses cannot be met without adding restrictions to the TSs, state how the additional Limiting Conditions for Operation, Actions, and Surveillance Requirements meet the regulatory requirements of 10 CFR 50.36. (ITSB)

10-4. Clarify that the decay heat limits include the decay heat from both the fuel assembly and any non-fuel hardware that may be loaded with the assembly in a given basket cell, modifying the technical specifications accordingly. (CSDAB)

It is not clear that the TS limits appropriately account for decay heat from the non-fuel hardware (NFH) that may be loaded with the fuel assemblies. If the decay heat from NFH is not included, the acceptability of this condition should be justified. If the limit is meant to include the NFH contribution, the TSs should be modified to explicitly indicate this requirement.

This information is needed to confirm compliance with 10 CFR 50.90 and the intent of 72.44(c)(1).

10-5. Justify that the proposed TSs with respect to the allowable STC contents are sufficient to limit the radiation source of those contents. (CSDAB)

The currently proposed TSs provide only a single maximum burnup, a single minimum decay/cooling time, and a maximum decay heat for each basket region. Thus, it appears that a decay heat limit alone is relied upon to control the radiation source for shielding. It is not clear that a decay heat limit by itself is sufficient to control the source for shielding. Different combinations of the minimum enrichment, maximum burnup and minimum cooling time parameters may generate the same decay heat but result in different shielding source terms. Thus, the applicant should provide an appropriate evaluation to justify that the proposed limits are sufficient to control the shielding source term (and hence dose rates). This evaluation should consider the enrichments, burnups and decay times of the IP3 pool inventory, providing sufficient detail to support the basis of the evaluation including any assumptions. Instead of this evaluation, the applicant could modify the technical specifications to include minimum cooling time limits for different combinations of minimum enrichment and maximum burnup and provide supporting evaluations for the proposed limits. Staff notes that information available to it regarding the IP3 discharged fuel indicates there are a significant number of assemblies with enrichments below the minimum enrichments assumed for their respective burnups in the evaluation; thus, the applicant should account for these lower enrichments for the various burnup ranges in proposing and justifying the content limits. The applicant should also justify the lack of specifications regarding axial blankets for assemblies (including blanket lengths) and other parameters important for shielding in the TSs for the allowable STC contents.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104, 72.106, and 72.44(c)(1).

10-6. Modify the TSs to include limits on the non-fuel hardware (NFH) that may be transferred in the STC. (CSDAB)

The applicant proposes to transfer NFH with spent fuel. The TSs should be modified to provide appropriate limits on NFH, including: the types of NFH and the number of each NFH type that may be transferred, the allowable STC basket locations for each NFH type, the minimum decay time (i.e., minimum post-irradiation cooling time), and the maximum burnup. These limits should be based upon the descriptions and evaluations of NFH in the application (see also RAI 7-3) and are important to include in the TSs to ensure that the operations are within the envelope of the analysis.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.104 and 72.44(c)(1).

10-7. Modify the TSs to include a description of the STC, including information similar to that found in a cask certificate of compliance. (CSDAB)

The applicant has proposed to transfer IP3 spent fuel to the IP2 SFP with the STC. The STC evaluations are based upon it having the design features described in the application. Changes to some STC design features could have a significant impact on the system performance. There is no 10 CFR Part 72 CoC for the STC. Thus, the TSs should include a description of the STC. This description should include features such as the configuration of the STC (e.g., it is a multi-walled (steel/lead/steel) canister for wet transfers of spent fuel from the IP3 SFP to the IP2 SFP) and the minimum thickness of lead and steel relied upon for shielding. Other features that may need to be added to the specifications include the minimum water heights in the STC and in the HI-TRAC annulus as well as the use of the Bottom Missile Shield for the inter-unit transfer. The applicant should propose a description of the STC for inclusion in the TSs and justify that this description adequately captures the STC's important design features.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 72.44(c)(4), 72.104, 72.106(b), and 72.126(a).

10-8. Provide TS language which describes the critical constituents of Metamic. Also, provide language which controls/outlines the acceptance testing that will be employed to ensure the quality of Metamic production batches. (SMMB)

Previous Holtec licenses for Parts 72 and 71 have included TS items which control the critical aspects of the manufacture and testing of Metamic production batches. Replication of previously approved Metamic TS requirements would be appropriate for the STC.

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 62, and the intent of 10 CFR 72.124(b).

10-9. Provide TS language to limit STC movements to when the ambient temperature is in a range that will ensure ductile behavior of the materials involved (for example, at or above 0 degrees F). (SMMB)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61, and the intent of 10 CFR 72.122(b).

April 20, 2010

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT APPLICATION FOR INTER-UNIT SPENT FUEL TRANSFER (TAC NOS. ME1671, ME1672, AND L24299)

Dear Sir or Madam:

By letter dated July 8, 2009, as supplemented by letter dated September 28, 2009, Entergy Nuclear Operations, Inc. (Entergy) submitted an application for proposed license amendments for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) that would allow the transfer of spent fuel from the IP3 spent fuel pool (SFP) to the IP2 SFP using a newly-designed transfer canister.

The Nuclear Regulatory Commission staff is reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). The Entergy staff stated that a response to the RAI would be provided by October 5, 2010.

Please contact me at (301) 415-2901 if you have any questions on this issue.

Sincerely,
/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286
Enclosure:
RAI
cc w/encl: Distribution via Listserv
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See next page
ADAMS ACCESSION NO.: ML101020486 *See associated memo **Via email

OFFICE	NRR/DORL LPL1-1/PM	NRR/DORL LPL1-1/LA	NRR/DSS SRXB/BC**	NRR/DSS SBPB/BC**	NRR/DCI CSGB/BC*
NAME	JBoska	SLittle	AUlses	GCasto	RTaylor
DATE	4/19/10	4/14/10	4/16/10	4/15/10	2/2/10
OFFICE	NRR/DRA AADB/BC*	NRR/DIRS ITSB/BC*	NMSS/DSFST LB/PM**	NMSS/DSFST TCB/BC**	NMSS/DSFST SMMB/BC**
NAME	TTate	RElliott	JGoshen	MRahimi	CRegan
DATE	2/18/10	2/16/10	4/16/10	4/16/10	4/18/10
OFFICE	NMSS/DSFST CSDAB/BC**	NMSS/DSFST LB/BC**	NRR/DORL LPL1-1/BC		
NAME	LCampbell	EBenner	NSalgado		
DATE	4/14/10	4/16/10	4/20/10		

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INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT APPLICATION FOR INTER-UNIT SPENT FUEL TRANSFER (TAC NOS. ME1671, ME1672, AND L24299)

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