March 16, 2011

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 letters dated September 28, 2009, and October 5, 2010, 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 within 60 days of the date of this letter.

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

Enclosures:
1. RAI (non-proprietary)
2. RAI (proprietary)

cc w/encl1: Distribution via Listserv

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 letters dated September 28, 2009, (ADAMS Accession Nos. ML092950437 and ML093020080), and October 5, 2010 (ADAMS Accession Nos. ML102910511, ML103080112, and ML103080113) 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 chapters listed below refer to the safety analysis report (SAR) for the STC, HI-2094289, Revision 3, ADAMS Accession No. ML103080113. The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions:

CHAPTER 1 – GENERAL INFORMATION

1-1. Attachments 5 and 7 to your letter dated October 5, 2010, included a proposed Appendix C to the unit operating licenses, with Technical Specifications (TSs) for the spent fuel transfer operations. Proposed TS 3.1.3, “Shielded Transfer Canister (STC) Pressure Rise,” included a limiting condition for operation (LCO) of less than a 4.2 psi pressure increase over 24 hours. The associated proposed TS Bases were provided in Attachments 6 and 8, and the bases indicated that the LCO ensures that fuel assemblies selected for loading in the STC satisfy design basis limits because an analysis of the pressure rise for the design heat load of 9.6 kW was less than 4.2 psi. However, the licensing report (Enclosure 1 to letter dated October 5, 2010) includes additional analyses verifying that the STC design pressure would not be exceeded for various postulated accident conditions. (SBPB)

Paragraph (c)(2) of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, “Technical specifications,” requires in part that a LCO be established for a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. In addition, Paragraph (c)(3) of 10 CFR 50.36 requires the inclusion of Surveillance Requirements (SRs) related to test or inspection that assure that the LCOs will be met. Finally, Paragraph (c)(4) of 10 CFR 50.36 requires the inclusion in the TSs of design features of the facility that, if altered or modified, would have a significant effect on safety.

The NRC staff considers the STC pressure boundary, which is identified as a confinement boundary, to be a fission product boundary. The licensing design-basis analyses examine the effects of several postulated events with respect to challenges to the integrity of the fuel cladding and the STC pressure boundary. Accordingly, propose additional LCOs, SR, and Design Feature TSs necessary to satisfy the requirements of 10 CFR 50.36.

For example, address methods to verify that appropriate initial conditions have been established for STC overpressure protection (e.g., a steam bubble of appropriate volume exists within the STC). To ensure this steam space is formed, additional controls are needed. Proposed TS 4.1.4.6 should be converted to a TS SR and LCO. This SR would verify that the required volume of water is removed through the STC drain line following the application of steam pressure. Also, in the absence of an analysis showing the acceptability of an air-filled space rather than a steam-filled space, a TS SR needs to demonstrate that the space is filled with steam. One method would be to use the pressure change in the STC during the 24 hours following establishment of the steam space. LCO 3.1.3 could be modified to show that the pressure stays within an analyzed range over time (e.g., a response graph). The lower limit of the graph could be the zero heat load curve, and the upper limit could be the response with the design heat load. See RAI TS-8 for additional comments on LCO 3.1.3.

Design features essential for the assumed heat transfer capabilities necessary to prevent overpressure conditions (e.g., materials of construction [for thermal conductivity] and emissivity of outer surface) need to be described in TS. For example, TS section 4.1 could be modified to add a section (after Criticality) titled “4.1.3 Thermal Features.” The type of thermal features described here would be those critical to the heat transfer abilities of the STC and the HI-TRAC.

This information is needed to confirm compliance with 10 CFR 50.36.

1-2. Modify licensing drawing No. 6013, Sheet 2 of 4, to state that the lead thickness shown is a minimum thickness. (CSDAB)

The lead thickness for the STC in the referenced drawing is the minimum thickness allowed per the proposed TS, Appendix C, Part I, Section 1.0 description of the STC; thus, the drawing’s specification of the lead thickness should be consistent with that description and indicate that the stated dimension in the drawing is a minimum value.

This information is needed to confirm compliance with 10 CFR Part 50, Appendix B, and the intent of 10 CFR 72.104, 72.106, and 72.44(a).

CHAPTER 4 - CRITICALITY EVALUATION (CSDAB and SRXB)

4-1. Provide a list of the isotopes and a summary table to show the biases and uncertainties that are applied to the major actinides and the best estimate correction factors that are applied to minor actinides and fission products for which burnup credits are taken. The information provided in the revised SAR pages supplied with the previous RAI response does not adequately address this.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-2. Provide justification for the conclusion that 10000 histories, 80 skipped cycles, and 200 accumulated cycles are acceptable for both configuration A and B, the 8 fuel assembly basket in particular.

In general, the MCNP code is sensitive to the number of cycles skipped and the ksrc specifications for loosely coupled systems like the 8 fuel assembly basket configurations. For cases like this, the users often have to check the convergence of both the effective neutron multiplication factor (Keff) and the source term. The Shannon Entropy often is used to test the convergence of the eigenvalue calculations. It is not clear, however, how these control parameters are determined. The applicant is requested to provide justification for the selection of the values of these control parameters.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-3. Provide justification for the applicability of the selected critical experiments to the code benchmark for the system of the eight fuel assembly configuration.

From the selected benchmark critical experiments, it appears that none of them has a configuration similar to the configuration of the eight fuel assembly basket. The applicant is requested to provide justification for the applicability of the selected critical experiments for the eight fuel assembly basket.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-4. Review and provide a corrected term of relative burnup in reference to Table 4.5.3.

Table 4.5.3 contains two pairs of columns, one pair is relative burnup at zero burnup and the second pair is relative burnup at 45 gigawatt days per metric ton uranium (GWD/MTU). It appears that the term "relative burnup" should be "normalized power." The term "relative burnup" is not determined if there is no burnup; 0/0 is not defined. The applicant is requested to review and provide corrected information as necessary to accurately represent the data.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-5. Provide a justification of the burnup credit calculations submitted, or resubmit with lower bounding specific power.

In Section 4.7.1.2.1 of the SAR, the applicant states that the maximum value for all plants with Westinghouse 15x15 fuel assemblies is used as bounding assembly specific power. However, studies published in "NUREG/CR-6665, Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel" and "M. D. DeHart, Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996" indicate that using lower specific power produces conservative results for casks taking burnup credit for both actinides and fission products. To ensure criticality safety, lower specific power should be used for applications that take credit for both actinides and fission products. The applicant is requested to review and redo its burnup credit analyses using appropriate specific power.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-6. Demonstrate that the fuel assembly depletion code, CASMO-4, is capable of modeling both the poisoned portion and the non-poisoned portion of the wet annular burnable absorber (WABA) rods in a fuel assembly.

Page 4-25 of the revised SAR states that the length of the poisoned region of WABA rods varies from 120 to 128 inches. The top and bottom parts of a WABA rod are just cladding. This leaves the top and bottom parts of a fuel assembly unexposed to poison. The SAR further states: "Most of the calculations with WABAs take that into consideration, i.e., the depletion calculations for the top and bottom node are performed with the cladding of the WABA in place, but without any poison while the depletion calculations for the central part of the assembly contains both the WABA cladding and the absorber." It is not clear, however, how the CASMO-4 code models this three dimensional effect for the fuel assembly containing WABAs. The applicant is requested to demonstrate that the fuel assembly depletion code, CASMO-4, is capable of modeling both the poisoned portion and the non-poisoned portion of the WABA rods in a fuel assembly.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-7. Demonstrate that reactivity effect of the burnup gradient is negligible for the 12 fuel assembly loading configuration.

Page 4-29 of the revised SAR discusses the potential reactivity impact of burnup gradients across the fuel assemblies and further states: "However, since this is a highly unlikely configuration, and since even then the reactivity is small compared to the remaining safety margin (see Section 4.7.9), all design basis calculations are performed with a uniform planar burnup distribution." This assessment, however, may not be justified without a quantitative analysis of the system with consideration of the burnup gradient in the fuel assemblies. The staff is particularly concerned with the small system 12 fuel assembly shielded transfer cask. The applicant is requested to demonstrate that reactivity effect of the burnup gradient is negligible for the 12 fuel assembly loading configuration.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-8. Demonstrate that the chemical assay data used for code benchmarking are sufficient to cover fuel assemblies with burnup exceeding 50 GWD/MTU.

Page 4-32 of the revised SAR states: "Note that ISG-8 Rev. 2 recommends an upper limit for burnup credit of 50 GWD/MTU, based on an apparent lack of data above this value. However, the evaluation of the bias and bias uncertainties and isotopic correction factors from benchmarking have been performed using a statistical approach that accounts for the limited amount of data available and assigns higher uncertainties for parameters that are further away from (i.e. at higher burnups) the experimental data. Also, isotopic benchmarking includes data for fuel exceeding 50 GWD/MTU. An additional burnup limit does therefore not appear to be necessary and is therefore not applied here." The staff reviewed the relevant publications and found that the bias and bias uncertainties determined based on the comparisons of the calculated and experimental isotopic concentrations are valid only for the burnup range of the experimental data. Based on the staff's review, it appears that there is a very limited number of chemical assay data that have burnup in excess of 50 GWD/MTU and the statistical analyses using the very limited data points may not satisfy the criterion for obtaining meaningful results. In addition, it appears that the burnup values are the local burnup of samples rather than the fuel assembly average burnup that is used in fuel qualification calculation. The applicant is requested to demonstrate that the chemical assay data used for code benchmarking are sufficient to cover fuel assemblies with burnup exceeding 50 GWD/MTU. If the criticality analyses are limited to 50 GWD/MTU, revise TS LCO 3.1.2.a.2, 3.1.2.b.2, and Note b to TS Table 3.1.2-1 to show the revised limit.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-9. [deleted]

4-10. [proprietary]

4-11. [deleted]

4-12. Provide detailed information on and justification for the interpolation scheme used in determining the isotopic concentrations as a function of burnup.

Page 4-30 of the revised SAR states: "Since it is necessary to model the axial burnup distribution, a large number of isotopic compositions at irregular burnups are required. Given the significant number of criticality calculations and studies performed for the burnup credit evaluations, it would be impractical to perform CASMO-4 depletion calculations for each of these burnups. Instead, CASMO-4 runs are performed for fixed burnups at 2.5 GWD/MTU intervals (or less), and intermediate isotopic values are determined by linear-linear interpolation. [proprietary] "

From these statements, it is not clear what interpolation scheme was used in determining the isotopic concentrations at various burnup values. It is not clear whether the isotopic concentrations of the various isotopes of interest can be determined with this interpolation scheme. The applicant is requested to provide detailed information on and justification for the interpolation scheme used in determining the isotopic concentrations as a function of burnup.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

4-13. [proprietary]

4-14. [proprietary]

4-15. [proprietary]

4-16. [proprietary]

4-17. [proprietary]

4-18. [proprietary]

4-19. [proprietary]

4-20. [proprietary]

4-21. [deleted]

CHAPTER 5 - THERMAL-HYDRAULIC EVALUATION

5-1. Verify that all thermal properties used in the analyses properly cover the expected temperature range during normal, loading, off-normal and accident conditions. (TCB)

For example, Table 5.2.9 of the SAR includes thermal properties of steam which appear to be for superheated steam. Thermal evaluation results provided in the SAR do not indicate the presence of superheated steam. The use of these properties to calculate the effective thermal properties of the steam gap could overestimate heat transfer and could result in incorrectly calculated temperatures.

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

5-2. Justify the emissivity value of the water surface used to calculate the effective thermal conductivity of the steam gap region inside the STC. Provide also the reference where the water emissivity is obtained. (TCB)

The staff needs to verify realistic parameters are used to properly characterize regions represented by effective properties which would assure a realistic or conservative representation of the heat transfer characteristics of the system. Table 5.2.1 provides the thermo-physical property references for all materials used. Note 1 on this Table states the water emissivity is not reported, as radiation heat dissipation from these surfaces is conservatively neglected. However, water emissivity is used to calculate effective thermal conductivity of air and steam spaces.

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

5-3. Clarify why the SAR reports lower temperatures compared to the thermal evaluation included in the original application. (TCB)

The staff needs to verify how the updated thermal evaluation resulted in lower temperatures (e.g., if model conservatisms were removed, provide adequate justification, etc.)

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

5-4. Revise all thermal analyses to assure energy balance has sufficiently converged to provide assurance calculated temperatures are representative of all applied thermal loads. (TCB)

When performing analysis audits of some of the calculations, the staff noticed there is a heat imbalance of about 7% which indicates the calculation is not fully converged which could result in lower non-conservative temperatures.

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

5-5. Perform the transient pressure rise for at least 48 hours to make sure the pressure increase used to monitor for fuel misload is approaching steady state. (TCB)

Figure 5.3.2 of the SAR indicates a pressure increase of about 4.2 psi is expected for the first 24 hours. The staff needs to have additional assurance the pressure increase is converging to the steady state value which would indicate a safe onsite transfer may proceed.

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

5-6. Perform a transient pressure rise calculation for the case when a possible misload has occurred based on heat loads representative of the IP3 spent fuel fuel inventory to assure the pressure rise can be managed by the licensee's in-place operating control procedures. (TCB)

The transient pressure rise provided in the SAR applies only for design basis heat load. The staff needs to have assurance the licensee has the capability to implement corrective actions, if needed, in the case of an occurrence of a misload.

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

5-7. Review the thermal properties used to determine the thermal conductivity of saturated water vapor as a function of temperature. (TCB)

Page B-7 of Holtec Report HI-2084146 provides saturated vapor thermal conductivity as a function of temperature. Thermal conductivity values tabulated at 371 C and 372 C appear to correspond to saturated liquid water. This may increase the effective conductivity at these temperatures.

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

5-8. Clarify how the STC pressure rise would be controlled if, based on the thermal analysis for this configuration, it is predicted that water would be boiling under normal conditions of transfer. (TCB)

Based on auditing of some of the thermal calculations (see also RAI 5-4 above), the staff noticed there is not adequate convergence in the heat balance. For a properly converged solution (maximum temperatures reached, adequate heat balance), the licensee could predict higher temperatures with water reaching the boiling point.

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

5-9. Describe the code assessment conducted for the software models used to calculate the thermal response of the STC and HI-TRAC. (SBPB)

Section 5.3, “Thermal Evaluation of Fuel Transfer Operation,” of the SAR included a description of the three-dimensional modeling used for thermal evaluations. The discussion described the general modeling and some conservative assumptions included in the model. In addition, the response to NRC staff requests for additional information provided in Attachment 1 to the letter dated October 5, 2010, indicated that the models were used for similar evaluations of spent fuel stored in spent fuel pools. However, in this application the model was used to evaluate conditions involving natural convection heat transfer in air environments and to evaluate radiation heat transfer. Section 15.0.2, “Review of Transient and Accident Analysis Methods,” of the NRC Standard Review Plan (NUREG-0800) describes specific areas of review for models used for transient analyses. The review areas include code assessment, which the staff described as a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario.

Provide a code assessment for the accident scenarios involving natural convection heat transfer in air environments and radiation heat transfer that demonstrate the adequacy of the model for those scenarios.

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

5-10. Discuss the effect of the centering assembly on heat transfer from the STC to the HI-TRAC and then to the atmosphere during scenarios involving loss of water from the HI-TRAC annulus. (SBPB)

Holtec Report No. HI-2084146, “Thermal Hydraulic Analysis of IP3 Shielded Transfer Canister,” provided information about analyses of various scenarios evaluated for passive rejection of decay heat to the environment. Section 7.4 discussed the analysis of the simultaneous loss of water from the water jacket and HI-TRAC annulus, and mentioned the use of the Discrete Ordinates radiation heat transfer model in FLUENT. This analysis credited radiation and natural convection heat transfer within the annular space between the STC and the HI-TRAC.

The annular space between the STC and the HI-TRAC contains the HI-TRAC/STC centering assembly. The staff believes the centering assembly may adversely affect heat transfer during the scenario involving loss of water from the HI-TRAC annulus because it would interfere with internal radiation heat transfer and reduce internal natural circulation air flow. Describe in detail the internal natural circulation in air and radiation heat transfer models for the loss of water from the HI-TRAC annulus scenario. Specifically address how the effect of the centering assembly was incorporated in the models.

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

CHAPTER 6 - STRUCTURAL EVALUATION OF NORMAL AND ACCIDENT CONDITION LOADINGS

6-1. Provide the technical basis for ignoring the water inside the hollow aluminum tubes when modeling the tube assembly for mitigating side impact effects associated with the non-mechanistic tipover of loaded HI-TRAC cask analysis. (SMMB)

SAR Section 6.2.8, Rev. 3 states, "the tip-over of the transfer cask has been carried out without considering the fluid coupling (cushioning) effect of water inside the annular space around the STC and within the centering assembly tubes." It is not entirely clear why water inside of the hollow aluminum tubes is not included in the model, given that the water incompressibility may add significant stiffness to the tubes during a tipover event.

This information is required 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 and 10 CFR 72.122.

CHAPTER 7 – SHIELDING DESIGN AND ALARA CONSIDERATIONS

The proposed amendment seeks to perform a wet transfer of spent fuel from the IP3 SFP to the IP2 SFP using an STC. The STC functions as a transfer cask in these operations and may be considered a lightweight transfer cask since it is designed for a limited-capacity crane that cannot handle an approved transfer cask like those used for approved spent fuel dry storage systems. The STC presents some unique shielding and radiation protection considerations, with significantly higher dose rates for the proposed allowable contents. The STC must be used in conjunction with a HI-TRAC transfer cask, which provides additional shielding so that transfers between FSBs as well as preparation for the transfer and unloading of the STC can be done in a safe manner. This is a new, first-of-a-kind review for a system and operations of this kind.

Normally, transfer casks provide sufficient biological radiation shielding such that workers may safely be in the vicinity of the transfer cask. This does not appear to be the case with the STC design. The staff is not only concerned about occupational doses during normal, off-normal, and accident conditions, but also public doses. Since this is an amendment under 10 CFR Part 50 and the action is limited to the Indian Point Energy Center site, considerations are given with regard to the site features. Still, the staff must ensure that enough controls are in place to provide reasonable assurance that both the public and occupational dose limits in 10 CFR Parts 20, 50, and 100 (via compliance with the intent of Part 72 limits) are not exceeded.

The following RAIs are geared towards obtaining enough information so that the staff may make a determination regarding whether there is reasonable assurance that the STC may be used safely and in accordance with the regulations. The staff is particularly concerned about a potential off-normal event involving either the hang-up of the crane or a malfunction of the remote handling equipment (if used). Crane hang-ups are not uncommon, especially when cranes are loaded with weights approaching their capacity limits. Additionally, the staff is asking for clarification and/or additional information to ensure that the license, TS, and licensing report each contain the appropriate level of information needed to control the design basis for this unique transfer system.

7-1. Justify the homogenization of the fuel assembly with the moderator in the shielding model. (CSDAB)

Question 7-16 in the first round RAI dealt with accounting for neutron multiplication and its effect on dose rates with an appropriate geometry. Modeling assumptions regarding fissile material and moderator can influence the neutron multiplication of a system, with homogenized models under-predicting multiplication occurring in heterogeneous systems of fissile material and moderator. The applicant's response does not address this question and instead assumes a k-effective value to define multiplication without consideration of how k-effective differs between a heterogeneous and a homogeneous system of the same materials or how the different systems behave from a shielding perspective. Further, the response only considers the loaded HI-TRAC; it should also include the impact on dose rates for the loaded STC outside of the 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 10 CFR 72.104, 72.106(b) and 72.126(a).

7-2. Provide a dose rate evaluation for the STC with all shielding materials at their minimum thickness specified in the proposed TS, Appendix C, Part I, Section 1.0, addressing the impacts on the occupational and public doses. (CSDAB)

Based upon the sample input file provided as part of the RSI response, the current dose rate calculations are based upon nominal steel dimensions and minimum lead thickness. While not as strong as lead, steel is still a significant shield material. For example, the half-value thickness of steel for 1.0 MeV photons is about 1.5 cm and for 1.5 MeV photons is about 1.8 cm; thus, the difference in steel between a minimum thickness present and a nominal thickness present is nearly 1 half-value thickness at this gamma energy; indicating the potential for significant differences in dose rates. Given the high dose rates for the STC with the representative loading, the impact on dose rates could be significant and lead to higher estimated occupational and public doses and/or the need to further modify operations to reduce doses and keep them as low as reasonably achievable (ALARA).

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

7-3. Justify the changes to the dose rates reported in the tables in Section 7.4 of the STC licensing report, correcting the reported dose rates and dose evaluations as necessary. (CSDAB)

In response to staff's RAIs, the applicant modified the dose rates reported in the licensing report for the STC and HI-TRAC. However, the changes appear to be inconsistent. For example, the surface dose rates on the STC for the representative loading case are nearly double their previously reported values; however, a number of the dose rates reported at the axial surfaces and at distance from the axial and radial surfaces have either negligibly changed or have decreased compared to the previously reported values. Correct dose rate values should be provided, and changes in dose rates should be adequately explained and justified. The dose evaluations in the report should also be updated, as necessary, to account for the correct dose rates.

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

7-4. Provide dose evaluations for crane hang-up and other off-normal conditions occurring with a loaded STC outside the SFP and HI-TRAC in the FSBs (both IP2 and IP3), including the information described below, and provide adequately detailed operations descriptions in Chapter 10 for addressing these conditions. (CSDAB)

The STC function is like that of the transfer cask that many spent fuel storage systems use to load fuel from the pool and transfer it to the storage overpack. Modifications made to the licensing report in response to staff RAIs indicate that the STC dose rates (representative loading) are significantly higher than have been analyzed for nearly all spent fuel transfer devices currently approved under 10 CFR Part 72. Given the STC's high dose rates, off-normal events, such as crane hang-ups, may result in conditions that would not be encountered were the crane hang-up to occur with a standard transfer cask. Thus, the applicant should provide dose evaluations for personnel involved in performing operations to recover from a crane hang-up, including manual crane operations and crane repair. These evaluations should include descriptions of personnel actions during these operations, personnel numbers and locations relative to the STC, duration of each operation segment, and appropriate justification for each aspect of the evaluation. Additionally, the applicant should provide dose evaluations for other plant personnel (e.g., administrative staff, guards, plant technicians, etc.) and describe the impacts this event would have on operations/activities in site facilities adjacent to, or near the respective units' FSBs. Evaluations should also be performed for members of the public on-site and at the controlled area boundary, with adequately justified bases and assumptions. Chapter 10 of the licensing report should provide an adequately detailed description of these operations, with which the evaluations in Chapter 7 should be consistent.

This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a) and (b), GDC 61, and the intent of 10 CFR 72.104 and 72.126.

7-5. The following editorial errors were identified. Provide corrected SAR pages for review. (CSDAB)

a. Table 7.4.14 should list the regulatory accident dose limit units as mrem and not mrem/yr. Accident dose limits in 10 CFR 72.106(b) are given in terms of dose and not dose per year.
b. The first paragraph in Section 7.0 should be modified to clearly indicate that off-normal conditions are limited per 10 CFR 72.104 together with normal conditions. The current paragraph text appears to incorrectly indicate that off-normal conditions are limited with accident conditions by 10 CFR 72.106 limits.
c. Ensure appropriate terms are used for the respective dose evaluations. The current text indicates that site boundary dose rates are used to show compliance with the intent of 10 CFR Part 72 dose limits. Dose limits in 10 CFR Part 72 are for the controlled area boundary, as defined in that Part. While compliance with the intent of Part 72 limits is

attempted to demonstrate compliance with 10 CFR Part 50 and Part 20 limits, the term 'site boundary' is a 10 CFR Part 20 term. The relationship between the controlled area boundary and the site boundary can be elaborated to justify how demonstration of compliance with dose limits for the one equates to demonstration of compliance with the dose limits for the other.

This information is needed to ensure compliance with 10 CFR 20.1301(a), 10 CFR 50.36a and the intent of 10 CFR 72.104 and 72.106.

7-6. Provide an occupational dose assessment that captures all elements of the spent fuel transfer operations and include appropriate justification of assumptions regarding personnel numbers and locations relative to the STC, time durations, applicable dose rates, and adequacy of assessment detail and description accuracy. Additionally, an inspection found that during lifts of the STC, there is a 13/16" gap between the lid and the flange; the assessment should consider this gap and STC operations should be modified, as appropriate, to account for this gap. (CSDAB)

Accurate and adequate assessments of occupational dose are important to ensure appropriate considerations are taken in the development of detailed procedures and the planning of operations. Staff continues to have concerns regarding the occupational dose assessment provided in the licensing report. Staff's initial concerns (see first round RAI questions 7-17 and 7-18) were only partly addressed, and further questions have arisen due to changes made to the assessment and the much higher dose rate estimates for the loaded STC. First, it is not clear that the assessment includes all elements of the operations. A comparison with the assessments done for the HI-STORM 100 system's loading operations illustrates the level of detail and comprehensiveness that would be expected, allowing for differences between systems (e.g., bolted closure vs. welded closure). Additionally, new procedures have been introduced, such as the 24-hour pressure rise test. Second, there continue to be inconsistencies between the assessment's description of operation conditions and those analyzed in the shielding models and the descriptions in Chapter 10 of the licensing report. Third, dose rates for some operations appear to be inappropriate for the actual personnel locations and configurations as understood from the Chapter 10 descriptions. Examples include operations on the STC lid using dose rates from the HI-TRAC axial side. Finally, various changes to the numbers and locations of personnel relative to the operations have been made without explanation or basis. It is not clear how fewer people than previously stated are required for the same operations. It is also not clear how they are to perform their functions from greater distances. For example, operations for raising the STC from the SFP, such as surveying the STC lid dose rates and washing the STC and crane equipment with clean water appear to be performed from 10 meters distance from the STC. Also, some operations of a similar nature are done under different conditions (e.g., placing STC in HI-TRAC vs. moving STC from HI-TRAC into SFP).

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

7-7. Explain whether normal operations with the loaded STC by itself are done remotely, including how remote operations are performed, the kinds of equipment used, and the quality standards employed for such equipment. If normal conditions are dependent upon remote operations, an appropriate condition should be included in the proposed TS, Appendix C. (CSDAB)

With revised STC dose rates much higher than has been evaluated for normal spent fuel loading or unloading, for currently approved spent fuel storage systems and the revisions to worker positions relative to the STC outside the SFP and the HI-TRAC, it is not clear whether or not operations are expected to be done in a remote fashion such that under normal conditions personnel are not around the STC like they are for loading operations for approved spent fuel storage systems. If the evaluation and operations rely upon remote operations, the application should clearly indicate that is the case, providing a description of how remote operations are to be performed (e.g., optical guidance systems and remote crane maneuvering), an explanation of how some steps can be performed remotely, and the assurance of equipment reliability for such operations. Chapter 10 of the licensing report should also be modified to clearly indicate that operations are performed remotely. Additionally, a TS condition should be added to state that the STC is handled remotely when out of the SFP and the HI-TRAC in conjunction with appropriate ALARA practices and that equipment assigned appropriate quality standards for remote handling operations will be used for such operations, with consideration for redundancy of such equipment.

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

7-8. Perform an evaluation to demonstrate compliance with 10 CFR 20.1301(b) for STC movement between the SFP and the HI-TRAC for both loading and unloading operations, implementing additional controls to ensure compliance with this requirement, including appropriate TS, as necessary. (CSDAB)

It is not clear if the current evaluation addresses compliance with the limits of 10 CFR 20.1301(b) for the condition of the STC by itself, including for normal conditions. Given the high dose rates of the STC and the relatively close proximity of other buildings where members of the public may be located (as seen in Figure 7.4.2 of the licensing report) while loading or unloading operations are ongoing, it is not clear whether additional controls are needed to ensure compliance with 10 CFR 20.1301(b). Any assumptions regarding shielding provided by building materials and structures should be appropriately justified. Consideration should also be given to the need for additional conditions in the proposed TS, as necessary, to ensure appropriate controls are instituted to ensure compliance with 10 CFR 20.1301(b).

This information is needed to confirm compliance with 10 CFR 20.1301(b).

7-9. Provide further justification regarding the assumed cobalt impurity levels. (CSDAB)

While staff accepts that the dose evaluations for the public and the controlled area boundary use fuel with decay times appropriate for the assumed cobalt impurity levels (as stated in response to first round RAI 7-14), this does not hold true for the occupational dose assessments that use fuel that, based upon the decay time and the burnup, would have been manufactured in a time when cobalt was not controlled to limit its amount in assembly hardware as it is for more recently fabricated fuel (post 1989). Additionally, it is still not clear that Non-Fuel Hardware (NFH) impurity levels may not be higher as well since the combination of proposed burnup and cooling time limits for NFH indicate that these would also have been fabricated during the period before cobalt reduction efforts were begun. Thus, the applicant should provide justification for the impurity level assumed for the fuel contents used in the occupational dose evaluations and the NFH (in all evaluations) or modify the evaluations to account for higher cobalt levels, on par with what has been identified in literature as the cobalt levels found in assembly hardware from that time (pre-1989).

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 10 CFR 72.104 and 72.106(b).

7-10. Provide the basis for the division of off-normal conditions into two categories of time duration (8 hours and 30 days) and describe the kinds of events/conditions that fall into the two different categories, including adequate justification. (CSDAB)

It is not clear that the division of off-normal conditions into two categories of differing durations is appropriate or justified. It is also not clear what kinds of events or conditions would be considered to fall into one or the other of these categories nor why their classification as one or the other type of off-normal condition would be justified. Staff notes that assumptions regarding off-normal conditions may impact whether or not compliance with the intent of 10 CFR 72.104(a) can be demonstrated, even in cases where only a single transfer is considered as off-normal and the remaining transfers are normal; thus, evaluations to demonstrate this compliance should be modified, as necessary.

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

7-11. Justify the change in the hours used to determine the annual dose contribution to the controlled area boundary of the Independent Spent Fuel Storage Installation (ISFSI) and other site facilities or use the originally assumed hours. (CSDAB)

In its resubmittal of the licensing report along with the RAI responses, the applicant changed the number of hours per year used in the determination of the annual dose contribution from the ISFSI and other site facilities for evaluation of annual doses against the 72.104(a) limits. The hours were reduced to 192 from 500. While 192 hours is the total hours expected for the total number of spent fuel transfers anticipated in a given year (currently taken to be 24), the ISFSI and site facilities contribute to dose at the remaining times of the year when spent fuel transfers are not occurring. Based upon previous arguments, 500 hours seems to be a more justifiable time estimate to use for these facilities for 72.104(a) evaluations. As part of any justification, the applicant should also include dose evaluations for a controlled area boundary at a distance of 137 meters (based upon the distance assumed in the ISFSI 72.212 evaluation), calculating the ISFSI and site facilities' dose contributions assuming 500 hours per year and 24 spent fuel transfers. These evaluations would need to address both normal and off-normal conditions. Staff notes that the assumptions regarding exposure time may impact the ability to demonstrate compliance with the intent of 10 CFR 72.104(a), even in cases where only a single transfer is considered off-normal and the remaining transfers are normal; evaluations to demonstrate this compliance should be modified, as necessary.

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

7-12. Provide an evaluation of the impacts on dose rates around the loaded STC and the loaded HI-TRAC for contents including a neutron source assembly (NSA). (CSDAB)

The current evaluation relies upon the arguments used in the HI-STORM 100 FSAR for NSAs. However, the allowable loading configuration in a HI-STORM 100 is such that a basket cell with an NSA is always completely surrounded by basket cells loaded with fuel not containing NSAs on all sides. This is not the case in the STC basket. Furthermore, some NSAs have significantly long half-lives and source strengths similar to design-basis fuel assemblies. This was the basis for the restriction of only a single NSA loaded in the very center of the MPC. In the STC, no basket cell where a NSA may be loaded is completely surrounded by basket cells without NSAs. Thus, a more detailed evaluation of the impacts on dose rates around the loaded STC and the loaded HI-TRAC should be performed to show these impacts, including azimuthal variations to capture the areas around the STC and around the HI-TRAC where the outer basket cell contents do not shield the inner basket cells where NSAs are permitted. The evaluations should also address any potential impacts on the doses to personnel and members of the public.

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

7-13. Modify the accident dose evaluation to include the following configurations as a result of a tip over accident: (CSDAB)

a. dose resulting from exposure of the HI-TRAC base, side, and top, accounting for any areas of the STC basket that are no longer submerged in water as a result of the cavity water receding from the side of the STC now facing up, and
b. the STC off-center in the HI-TRAC as a result of the tip over accident and the crushing of the inner impact limiter/STC centering assembly, with the loaded HI-TRAC in the vertical orientation.

An evaluation of the tip over accident should address applicable shielding and dose rate conditions and evaluations. According to the structural evaluation, the STC centering assembly acts as an impact limiter and crushes during the tip over accident; however, it is not clear how much crush will occur. The shielding evaluation should consider the case where the STC is no longer centered in the HI-TRAC, shifting the STC from center to the extent that the structural evaluation shows the centering assembly will crush. The assumption of loss of the water jacket should also be used in these evaluations as it is for the current accident evaluation.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.106.

7-14. Provide an evaluation that demonstrates the bounding dose rates for a loaded STC containing the contents permitted by the proposed TS contents limits. (CSDAB)

A regionalized loading pattern is used for some dose evaluations in the application. Staff asked the applicant in the previous RAI (see question 7-15 of the first RAI) to justify the use of the selected source terms in the regionalized loading pattern and the bounding nature of the dose rates and dose estimates from these sources. The applicant discussed use of a uniform loading pattern to perform some of the evaluations, stating that it exceeds the limits of what is allowed by the proposed decay heat limits. The applicant's response does not address the issue, especially in light of the very high dose rates from the STC with the representative loading. Decay heat limits and dose rates do not correlate on a one-to-one basis given that different combinations of burnup, decay time and minimum enrichment can yield the same decay heat but quite different radiation source terms. For the proposed operations, the STC operates much like a transfer cask and can be considered a lightweight transfer cask, since it is the device that is used to load and unload fuel from/into the spent fuel pools. With the regionalized loading also resulting in very high dose rates on the STC, much higher than has been evaluated for all approved dry storage systems' transfer casks, it is important to understand the maximum dose rates that may be obtained for the allowable contents; even small relative dose rate variations can mean large changes in dose rates. Understanding the maximum dose rates that can occur during transfer operations will enable proper ALARA and operations planning as well as demonstration that transfer operations with all proposed contents will meet the regulations at all stages of the operations. Dose evaluations should be modified as necessary to account for the bounding dose rates.

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

7-15. Refer to RAI 7-1 from the previous RAI letter. Evaluate the effect on dose and potential effect on canister leakage rate assuming a 10% fuel rod breach for all off-normal conditions. (TCB)

Section 7.4.5 "Effluent Dose Evaluation" of Report HI-2094289 identifies the first off-normal condition as a breakdown of the cask transporter without HI-TRAC recoverability for 30 days, but only assumes 1% fuel rod breach. The percent of spent fuel postulated to fail for off-normal conditions is 10% as identified in Table 5-2 of NUREG-1536 Rev. 1 "Standard Review Plan for Spent fuel Dry Storage Systems at a General License Facility." There appears to be no justification for only assuming 1% fuel rod breach. The 10% release fraction identified in the standard review plan is a bounding value for off-normal conditions and is not meant to be reduced based on postulated specific scenarios. Similarly, we would not expect the 100% fuel rod breach for accident conditions to be reduced based on a postulated specific accident, but utilized as a bounding value.

10 CFR Part 50, Appendix B, Criterion III, Design Control states in part, that measures shall be established to assure that applicable regulatory requirements and the design basis….are correctly translated into specifications, drawings, procedures, and instructions. 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.

7-16. Refer to RAI 7-2 from the previous RAI letter. Justify or remove the statement at the end of the third paragraph in Section 7.4.5 "Effluent Dose Evaluation" of Report HI-2094289 that fines, volatiles and crud would remain entrapped within the water environment. If applicable, submit a revised Section 7.4.5 as part of your RAI response. (TCB)

Since the STC, the confinement boundary, is tested to a finite leak rate (i.e. not leaktight) and can be pressurized (refer to Figure 5.3.2 "24 Hour Pressure Rise Under Design Basis Conditions," a pathway exists for the potential release of radioactive material. No credit can be taken for the HI-TRAC since it is not part of the confinement boundary. Hence, making a statement that this contamination would remain within the water seems invalid.

10 CFR Part 50, Appendix B, Criterion III, Design Control states in part, that measures shall be established to assure that applicable regulatory requirements and the design basis….are correctly translated into specifications, drawings, procedures, and instructions. 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.

7-17. Refer to RAI 7-3 from the previous RAI letter. The response to NRC RAI 7-3 provided by letter dated October 5, 2010, included an unnumbered table on page 57 of Attachment 1 which summarizes inputs used to calculate the atmospheric dispersion factors (χ/Q values) used in the effluent dose evaluation. (AADB)

a. Please explain how the σ_y and σ_z values listed in the table on page 57 were derived from Figures 1 and 2 of Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." For example, were σ_y and σ_z computer generated or were they derived by approximation directly from the figures? Please provide additional detail, including computer generated summaries and/or annotated figures, which show how the calculations were made, particularly at 1 and 20 meters as these distances are less than the 100 meter minimum distance plotted in Figures 1 and 2.

b. Please explain the basis for why χ/Q values were calculated for a distance of 1 meter. Please provide additional detail supporting the use of the RG 1.145 methodology to determine χ/Q values considering that the intent of the guidance does not appear to be appropriate for this distance. RG 1.145, which provides guidance for calculation of χ/Q values applicable to typical exclusion area boundary and low population zone distances, implicitly assumes a minimum distance of 100 meters with regard to information provided in Figures 1 and 2. Further, NRC staff notes that other NRC documents such as RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," state that NRC guidance for the calculation of χ/Q values may not apply under certain conditions, such as at distances less than about 10 meters.

c. The application references Interim Staff Guidance (ISG)-5, "Confinement Evaluation," as the basis for the stability and wind speed inputs. ISG-5 states that the use of stability category D and a wind speed of 5 meters per second (m/s) are acceptable for the normal and off-normal case calculations, and stability category F and a wind speed of 1 m/s are acceptable for the accident case calculation. These are default values for dry cask storage systems. NRC staff notes that these values are based on a generic approximation of meteorological conditions for an average site in the United States. Please clarify whether the normal and off-normal cases are assumed to occur during 50 percentile and the accident case during 95 percentile meteorological conditions, respectively, as is typically assumed for other nuclear reactor effluent dose evaluations. Please confirm how the use of these default values is justified when compared with representative 50 and 95 percentile meteorological conditions at the Indian Point site.

d. Figure 7.4.2 of Holtec International Report HI-2094289 (ADAMS Accession Number ML103080113) is a site map showing the haul path and an exclusion area boundary (EAB), which the licensee has defined for use in the current license amendment request. This EAB does not appear to be the current Indian Point licensing basis EAB for either Unit 2 or Unit 3 defined by 10 CFR 50.2. Therefore, please explain the relationship of the EAB shown in Figure 7.4.2 and the current Indian Point licensing basis EABs. In addition, please provide the minimum distance between any point along the haul path and 1) the Indian Point current licensing basis EABs, and 2) the control room intakes.

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

CHAPTER 8 – MATERIALS EVALUATION, ACCEPTANCE TESTS and MAINTENANCE PROGRAM

8-1. Refer to RAI 8-1 from the previous RAI letter. Justify the lack of leak testing of the entire confinement boundary of the STC as well as describe what leak tests are done in the shop and the field. (TCB)

The original RAI 8-1 included the following which was not addressed in the response or in the referenced sections of Holtec Report HI-2094289: " 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-cm3/sec."

The applicant's response stated that "Section 8.4.4 had been revised to more clearly define the factory leakage test and the periodic leakage test of the lid gaskets that is performed during loading operations." However upon reviewing Section 8.4.4, no distinction is made between factory and periodic leak tests and no discussion is provided with regard to leak testing the entire 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 XI-Test Control.

8-2. Refer to RAI 8-3 from the previous RAI letter. Justify the response provided in amended Section 8.4.3. If applicable submit a revised Section 8.4.3 as part of your RAI response. The staff finds that the Code required pressure test (125% of design pressure) alone is not sufficient to pressure test the bottom flange area of the HI-TRAC since it neglects the weight of the fully loaded STC which sets on the bottom HI-TRAC flange. The applicant's submittal states that only the 125% of design pressure is required for the HI-TRAC pressure test, which is not correct. Also, indicate in this section the total test pressure needed to meet the Code required test including the weight of a fully loaded STC which weighs about 40 tons. As previously stated, a dead load equivalent to a fully loaded STC may be placed inside the HI-TRAC for the pressure test, with the weight of the HI-TRAC supported only by its trunnions, in lieu of increasing the pressure above 125% of the design pressure. (TCB)

The applicant's response to the previous RAI stated that changes were made to address this issue to Sections 10.1.2 "STC Preparation and Setup- HI-TRAC Inspections and Checklist" and 8.5.2 "Leakage Tests." However, the staff could find no such changes made relating to this issue. From reading the revised SAR sections, the staff is concerned that the applicant may not fully appreciate the purpose of the pressure test. The pressure test is a structural integrity test performed before use to assure proper fabrication of the HI-TRAC. It is not a leakage test, per se, but uses the absence of leaks as its acceptance criteria (as mandated by the Code) to ensure the as-constructed vessel's suitability for the loadings. Therefore, more than just the gasketed area of the pool lid needs to be checked for leaks.

The NRC staff also needs confirmation of the structural integrity testing planned for the STC. If it will only be by pressure test, demonstrate that the pressure test exceeds all deadweight loads with appropriate margin and that the STC will be supported only by its crane attachment points during the test.

10 CFR Part 50, Appendix B, Criterion XI-Test Control requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

8-3. Make the following modifications to provide additional clarity and consistency. (CSDAB)

a. Modify Section 8.4.1 to also state that the STC will be assembled in accordance with and verified to meet the TS requirements for the STC design. In addition to the licensing drawings, the TS contain design requirements with which the fabricated STC must comply.
b. Modify Section 8.4.5 to also state that the lead sheet will be layered so that the minimum total thickness meets the TS requirements for the STC design. In addition to the licensing drawings, the TS contain a design requirement on the minimum thickness of the STC lead shielding.
c. Modify Section 8.4.5 to include more of the response to RAI question 8-11 from the previous RAI letter, particularly that the layering of lead sheets where each layer is made of multiple sections will be done so that section edges in adjacent layers are offset to eliminate potential streaming paths. This aspect of fabrication is important with regard to radiation protection, and based on the response to RAI question 8-11, should have been included in the referenced section of the licensing report.
d. Describe the acceptance testing that will be used to ensure that areas packed with lead wool will perform in a manner comparable with the lead sheet for shielding purposes.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.44(c)(4) and 72.126(a).

8-4. Refer to RAI 8-8 from the previous RAI letter. Identify and reconcile the discrepancies in applicable rules for construction between ASME Code, Section III, Division 1, Subsection ND and Subsection NC for the construction of the Shielded Transfer Canister (STC) to ensure that the STC is constructed to acceptable quality standards. (SMMB)

Spent fuel canisters (i.e. the confinement boundary) are normally constructed to ASME Code Subsection NB or NC (reference NUREG-1536 Section 3.4.1). The applicant is proposing to construct the STC to Subsection ND, as indicated in Section 1.3.1 of HI-2094289, Rev. 3. This approach does not appear to provide the same degree of quality for a spent fuel storage or transfer canister.

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

CHAPTER 10 - OPERATING PROCEDURES

10-1. Regarding the operations descriptions, provide the following: (CSDAB)

a. Justification for removing water from the STC/HI-TRAC annulus at the currently described step in the unloading procedures. Considering the step in the loading procedures in which the annulus water level is raised (just after the STC lid bolting is tightened to the required levels) and the operations descriptions following the water drain down for the unloading operations, it seems more appropriate and in keeping with ALARA to not drain down the annulus water until just prior to loosening the STC lid bolts.
b. Modification of the operations descriptions to include the steps for installation and removal of the Bottom Missile Shield (BMS). The responses to the first round RAI questions 1-2 and 7-6, indicate that the BMS is to be installed with the HI-TRAC empty; however, Chapter 10 descriptions need to be updated to reflect these operations and not just a check that the BMS is installed at the time the loaded HI-TRAC is to be moved.
c. Modification of the operations descriptions to account for the 24-hour pressure rise check. Other evaluations, such as occupational dose estimates, should be updated as necessary.
d. Modification of Section 10.5.5 of the report to include descriptions of steps to be taken in the event gas sampling indicates damage to assemblies. These steps would include such things as whether or not fuel is off-loaded and how it is handled.
e. Inclusion of the step for filling the HI-TRAC neutron shield.
f. Descriptions of how the presence of a walkway crossing over the haul path is addressed. Clarify whether and how the EAB also covers this walkway.
g. Explanation of the kinds of delays envisioned for Section 10.5.2 and the configurations that are considered (e.g., whether the STC is always in the HI-TRAC or the STC may be outside of the HI-TRAC). Evaluations of these configurations, as necessary, should also be provided.

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

10-2. Regarding the dose rate measurements for the loaded STC and HI-TRAC, define and include the criterion/criteria used to determine when dose rates that exceed expected values may be acceptable and justify waiting to perform dose rate measurements on the HI-TRAC until the currently proposed step. (CSDAB)

The currently proposed operations descriptions for the dose rate measurements for the STC lid and the HI-TRAC side include performance of an evaluation to determine if higher dose rates are acceptable and fuel transfer can continue. However, it is not clear what criterion or criteria are used to make that determination. Such criteria should be defined and provided as part of the operations descriptions related to the measurement procedures. For example, a criterion used for determining the acceptability of the higher dose rates on the transfer cask or storage overpack in the TS Radiation Protection Program for the HI-STORM 100 system is a determination that the as-loaded MPC, considering its contents, the number of casks at the ISFSI, etc. will not cause the limits of 72.104 to be exceeded. Similarly, an appropriate criterion, or criteria, is needed for the currently proposed operations. Additionally, the basis for delaying the measurements on the HI-TRAC side until step 55 of Section 10.2.3, versus performing the measurements very shortly after the STC is placed in the HI-TRAC (e.g., after step 23 or 28), is not clear. If a problem arises that necessitates corrective actions, the various operations to prepare the STC and HI-TRAC for transfer will have to be undone with the current sequence of operations. This seems unnecessary and as well as to not meet the intent of ALARA. The dose rate measurements on the HI-TRAC side should be performed shortly after the STC is placed in the HI-TRAC.

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

10-3. In Table 10.0.1 of the SAR, "Operational Considerations," add a row for "Crane Hang-up or Loss of Power." This event is discussed on page 10-26, but is not listed in the table. Step 10.5.1.1.b should start with a requirement to perform radiation surveys to establish stay times for personnel, due to the high dose rates when the STC is suspended from the crane. (LPL1-1)

10-4. In Section 10.5.2 of the SAR, it states that water shall be circulated through the STC daily to insure that the STC internal cavity is filled. If the 24-hour pressure rise test is in progress, circulating water or venting will violate the test conditions. Add an exception to water circulation and venting for the pressure rise test. (LPL1-1)

TECHNICAL SPECIFICATIONS:

TS-1. Add a TS condition that the restricted area boundary for the transfer operation is a minimum of 20 meters (or the distance used in the evaluations, as modified in response to this RAI) from the haul path. (CSDAB)

The radiation protection evaluation relies upon a set minimum distance to separate members of the public on site from the transfer operations to show compliance with 10 CFR 20.1301(b). This distance is a significant parameter in the evaluation and should be appropriately controlled. The applicant refers to this distance, or the boundary at this distance, as the exclusion (area) boundary (see Section 7.4.6 of Report HI-2094289). This terminology does not seem to be correct for this particular activity. The terminology should be made consistent with 10 CFR Part 20, using the terminology defined in that part of the regulations for actions performed and controls that are set for purposes of radiation protection.

This information is needed to confirm compliance with 10 CFR 20.1301(b).

TS-2. Provide the following clarifications and modifications of the proposed TSs, Appendix C, Parts I and II. (CSDAB)

a. Clarify the apparent inconsistency between TS, Appendix C, Part II, 4.1.4.6 and SAR Section 10.2.3.37, modifying the appropriate location to give the correct tolerances on the STC water level.
b. Include the STC licensing report (SAR), Chapter 10 as part of the basis in Appendix C, Part I, Section 2.1 and SAR Chapter 8 as part of the basis in Appendix C, Part I, Section

2.2. These sections of the SAR form the basis for the operations, acceptance tests, and maintenance program directly related to the STC and the transfer operations.

c. Modify LCO 3.1.2.b (Appendix C, Part II) to include an item 4 that duplicates item 5 of LCO 3.1.2.a and adds that rod control cluster assemblies (RCCAs) and NSAs cannot be loaded in the configuration of LCO 3.1.2.b. Appropriate Non-Fuel Hardware (NFH) loading restrictions, supported by the licensing report evaluations, are needed for both loading configurations.
d. Clarify the meaning of Note 3 to LCO 3.1.4 (Appendix C, Part II) with regard to defining a given assembly's burnup, especially with respect to the proposed limits in LCO 3.1.2. It is not clear from this note that assemblies with burnup/exposure greater than 55 GWD/MTU may not be loaded in the STC (since the note recalculates the burnup if the assembly had hafnium inserts). The maximum allowable burnup in the TS should be supported by all the appropriate STC licensing report evaluations (e.g., shielding, etc.).
e. Add the minimum specifications of the HI-TRAC neutron shielding to the HI-TRAC description in TS, Appendix C, Part I, Section 1.0. The neutron shielding features are also an important aspect of the HI-TRAC.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(c), 72.104 and 72.126.

TS-3. Modify the proposed TS, Appendix C, Part I to: (CSDAB)

a. Include recovery from off-normal conditions (such as crane hang-up) in the operations listed in Section 2.1 and
b. Include manual crane operation and crane recovery/repair as part of Section 2.3.

The dose rates from the bare STC (analytical) are significant (even for the representative loading), much more so than has been seen for nearly all spent fuel loading operations to date. Therefore, staff is particularly concerned about a potential off-normal event such as the hang-up of the crane or a malfunction of the equipment allowing personnel to remain at significant distances from the STC during operations with the STC outside the pool and the HI-TRAC. Crane hang-ups are not uncommon, especially when the cranes are loaded with weights approaching their capacity. Thus, operations to recover from a crane hang-up with a loaded STC could provide significant occupational exposures. Procedures for such scenarios should, therefore, be developed beforehand and appropriate training provided. Inclusion of manual crane operation in the dry run will assure the licensee can effectively operate the crane manually in a high dose rate environment as well as inform the licensee's predictions of potential worker dose in the event of a crane hang-up or other off-normal event.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(c), 72.104, and 72.126.

TS-4. Provide a TS dose rate limit and measurement requirement for the top lid of the STC and side of the HI-TRAC. (CSDAB)

Requirements that establish dose rate limits and the necessary measurements are a feature of the TS associated with 10 CFR Part 72 dry storage loading operations (including the HI-STORM 100), which are similar to the loading operations performed for the proposed wet transfer system. TS dose rate limits and measurements provide assurance of correct contents loading, ensure operations and ALARA planning is still adequate/appropriate for a given loading operation, and ensure that conditions outside those assumed for the design and operations are identified and properly handled to assure protection of personnel and members of the public.

Considering that the dose rates for even the proposed contents are very substantial and that conditions of the kind noted herein could make them even more substantial, a TS should be established that provides dose rate limits for the STC lid and the HI-TRAC side, an appropriate measurement scheme for each limit, and the appropriate corrective actions for instances where the limits are exceeded. The limits should be derived from the analysis for the representative loading and should capture areas of the STC and HI-TRAC of significance to occupational and public dose. Given the similarities of the proposed system to dry storage transfer systems, staff anticipates that the limits and measurements will be established in a similar fashion as for dry storage transfer systems, with appropriate consideration for differences versus those systems and the applicant’s analyses serving as the basis.

If it is proposed that dose rates that are higher than the proposed limits may be evaluated and considered acceptable under appropriate circumstances, the process and criteria for finding higher dose rates acceptable and allowing continuance of operations should be included in the TS and appropriately justified. For example, for dry storage, higher dose rates trigger a verification of correct loading and, if the loading is correct, a determination of whether or not 10 CFR 72.104 limits can be met for the loaded MPC. Since this is a wet transfer between pools, a criterion (such as ensuring compliance with 10 CFR 20.1301(b)) in addition to 72.104 limit compliance, considering the number of transfers to be performed in a given year, may be appropriate.

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

TS-5. Modify the fuel specifications in proposed TS, Appendix C, Part II, Table 4.1.1-1 to accurately reflect the contents (as evaluated in the amendment application) that will be loaded into the STC, including the following changes. (CSDAB)

a. Include the fuel cladding material. The shielding evaluation only supports zirconium-based cladding; thus, this should be the listed cladding.
b. Change the fuel rod clad I.D. to be a maximum value. It is currently shown as a minimum value, which appears to be incorrect. A minimum cladding thickness should be established, which only can be done with the clad I.D. set at a maximum and the clad O.D. set as a minimum.
c. The correct maximum active fuel length should be given. The shielding evaluation, for example, indicates the maximum active fuel length is 144 inches.
d. The correct fuel assembly maximum length should be given. The current length (176.8 inches) given in the table does not physically fit in the STC, which the licensing drawings indicate has a cavity length of only 168 15/16 inches. Further, the inclusion of NFH extends the needed STC cavity length to be able to contain the proposed contents.
e. If the active length and assembly length are accurate, then the STC design should be revised and the evaluations modified accordingly to support the new design and the proposed contents.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.104.

TS-6. Propose TS limits on maximum burnup, minimum enrichment and minimum decay/cooling times to limit the radiation source term of the contents and provide supporting quantitative evaluations to justify the proposed limits. Otherwise, provide additional justification, including appropriate quantitative evaluations, that the currently proposed TS with respect to allowable STC contents are sufficient to limit the radiation source of those contents. (CSDAB)

The currently proposed operation is similar to loading operations for dry storage systems. TS for these systems, including the HI-STORM 100, define the allowable contents in terms of the maximum burnup, minimum enrichment and minimum decay time of the assemblies. That being the case, the applicant has proposed a very different approach to limit the contents. The applicant has provided some justification in response to the first round RAI question 10-5. However, no quantitative evaluation was provided to support the statements made in that response. While the applicant has included a source term that significantly exceeds the decay heat limits for comparison as part of the licensing report, this does not indicate the variation in dose rates that will exist between assemblies of different burnup, decay time, and minimum enrichment that have the same decay heat. Adequate justification would include evaluations that show how the dose rates vary for an adequate variety of assembly burnup, enrichment and decay time combinations that result in the same decay heats for the different STC basket regions.

Given that radiation source and decay heat do not correlate on a one-to-one basis and the significance of the STC dose rates for the ‘representative’ loading, it is important to ensure the TS properly define the allowable radiation source in the STC. A decay heat limit alone would allow for radiation sources of varying strengths that could result in dose rates that may be different enough to necessitate non-trivial changes to operation procedures and controls for the purposes of occupational and public radiation protection and significantly impact off-normal conditions and accident conditions exposures as well. Additionally, it is not clear how using a decay heat limit alone would be practically implemented, including independent verifications, considering that decay heat is a calculated value that is derived from an assembly’s enrichment, cooling/decay time and burnup.

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 10 CFR 72.104, 72.106, and 72.44(c)(1).

TS-7. Justify the lack of a TS surface contamination LCO for the STC. (CSDAB)

Operations include the immersion of the STC in the spent fuel pool (SFP). While the STC is washed down prior to and as it is removed from the SFP, this is to minimize contamination. This is the same kind of practice taken with the transfer cask for spent fuel loading operations for dry storage and there is normally a TS LCO limiting surface contamination (see TS 3.2.2 for the HI-STORM 100 system). The bases for that LCO state the LCO “allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination.” While the HI-TRAC for the currently proposed operations does not enter the SFP, the STC acts as a transfer cask for the loading and unloading operations in its use out of the HI-TRAC. Given this consideration and the leak rates allowed for the HI-TRAC, a contamination LCO may be appropriate.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.34, and the intent of 10 CFR 72.44(c)(1), 72.104, and 72.126(a).

TS-8. Revise the proposed TS 3.1.3 to ensure a correct measurement of the pressure rise in the STC after it is loaded with spent fuel. (LPL1-1)

Measurement of the pressure change in the STC cavity after the STC is loaded is used to verify that the decay heat load is within design parameters. The staff's understanding of the pressure rise monitored by proposed LCO 3.1.3 is that the pressure rise is the difference between the lowest observed absolute pressure in the STC and the absolute pressure in the STC at the end of the 24-hour period. That description of the pressure rise should be added to TS Bases 3.1.3. Also, since the staff expects the pressure in the STC to initially decrease after the water level is established, SR 3.1.3.1 should be revised to say "Once upon establishing required water level AND hourly thereafter." Taking at least 25 data points over the 24-hour period will allow more accurate checking of the pressure change. Also, SR 3.1.3.2 should be added to specify the pressure instrumentation to be used. Wording similar to the following is needed: "Verify that two channels of pressure instrumentation with a range of at least 1 psia to 75 psia, and calibrated to within 2% accuracy within the past 12 months, are installed on the STC." The Frequency of SR 3.1.3.2 could be "During performance of surveillance 3.1.3.1."

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-9. Throughout the TS, "non fuel hardware" is a defined term and should, therefore, be capitalized. (LPL1-1)

TS-10. SR 3.1.1.1 says to "Verify the STC boron concentration is within limit using two independent measurements." Please explain the independent measurements to be used. If they are not truly independent (such as titration and neutron absorption), then it may be better to describe them as "two separate measurements." (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-11. TS 4.1.4.3 says "LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains irradiated fuel only." Consider revising this to say "LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains no unirradiated fuel." This is more precise, since there typically are items other than irradiated fuel in the spent fuel pit. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-12. TS 4.1.4.8 says "TRANSFER OPERATIONS shall only be conducted when the HI-TRAC water level is within +0/-1 inch of the top of the STC lid and the water level has been independently verified." Consider revising this to say "TRANSFER OPERATIONS shall only be conducted when the HI-TRAC water level is within +0/-1 inch of the top of the STC lid prior to installing the HI-TRAC lid and the water level has been independently verified." It is obvious that as the water heats up in the HI-TRAC it will expand and will no longer be within the specified range. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-13. TS 5.2(iii) says “Four coupons will be tested at the end of each inter-unit fuel transfer campaign.” Since the duration of a campaign is not defined, please propose a time frame instead, such as every 2 years. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-14. The Note in SR 3.1.1. says that the surveillance is only required to be performed if the STC is submerged in water. Since the STC is submerged in water while it is in the HI-TRAC, the boron concentration would have to be checked every 48 hours. If the STC had to be left in the HI-TRAC for an extended period of time, for example, due to an equipment malfunction, it will be very difficult to get a sample for boron analysis. Recommend revising the wording to something similar to “This surveillance is only required to be performed if the STC is submerged in water with the STC lid not fastened…”. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-15. LCO 3.1.1 would allow the licensee to sample the boron in the STC to verify at least 2000 ppm, and then lower the STC into the IP3 spent fuel pit with the water in the spent fuel pit at 1000 ppm boron. This allows the possibility of diluting the STC boron. Either revise Appendix A, LCO 3.7.15, to require 2000 ppm in the IP3 spent fuel pit whenever fuel assemblies are in it, or add an LCO to Appendix C to verify 2000 ppm in the IP3 spent fuel pit prior to placing the STC in the spent fuel pit. Revise SAR section 1.4, and other sections as necessary, to reflect this change. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-16. TS 4.1.2.1 is insufficient to specify the criticality controls, as it does not specify the location or size of the Metamic panels. One method would be to incorporate a drawing by reference which shows that information. Also, TS 4.1.2.1.g refers to B-10 loading in the B_4C as greater than or equal to 18.4 %. This can be misinterpreted, as 18.4 weight percent of the B_4C is not B-10. Please revise it to specify the B-10 density in the B_4C in terms of an areal density (grams per square centimeter). (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

TS-17. One of the operating conditions of the STC is the possibility of leaving some fuel assembly locations open in order to allow the loading of fuel which does not meet the minimum burnup requirements in the remaining locations. This is, therefore, an initial condition of a design-basis accident (fuel misload). This should be controlled by a TS. Please revise the Note for LCO 3.1.2 to say that if one or more Type 1 fuel assemblies are in the STC, cells 1, 2, 3, and 4 must be empty, with cell blockers installed that prevent inserting fuel assemblies. Also propose a new SR 3.1.2.2 that verifies by visual inspection that cell blockers are installed on cells 1, 2, 3, and 4 prior to placing a Type 1 fuel assembly in the STC. Add the description of the cell blockers to the SAR. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

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

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2. RAI (proprietary)

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