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

September 4, 2009

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 - SUPPLEMENTAL INFORMATION NEEDED FOR LICENSE AMENDMENT REQUEST FOR SPENT FUEL TRANSFER (TAC NOS. ME1671, ME1672, AND L24299)

Dear Sir or Madam:

By letter dated July 8, 2009, 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 amendment would license a new spent fuel transfer canister and would allow spent fuel to be transferred from the IP3 spent fuel pool to the IP2 spent fuel pool in preparation for further transfer to the Independent Spent Fuel Storage Installation (ISFSI) already located on the site. The purpose of this letter is to provide the results of the Nuclear Regulatory Commission (NRC) staff's acceptance review of this amendment request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to commence its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

Consistent with Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), an amendment to the license (including the Technical Specifications) must fully describe the changes requested, and following as far as applicable, the form prescribed for original applications. Section 50.34 of 10 CFR addresses the content of technical information required. This section stipulates that the submittal address the design and operating characteristics, unusual or novel design features, and principal safety considerations.

The NRC staff has reviewed your application and concluded that the information delineated in the enclosure to this letter is necessary to enable the NRC staff to commence its detailed review and make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment.

In order to make the application complete, the NRC staff requests that Entergy supplement the application to address the information requested in the enclosure by October 1, 2009. This will enable the NRC staff to commence its detailed technical review. If the information responsive to

the NRC staff's request is not received by the above date, the application will not be accepted for review pursuant to 10 CFR 2.101, and the NRC staff will cease its review activities associated with the application. If the application is subsequently accepted for review, you will be advised of any further information needed to support the NRC staff's detailed technical review by separate correspondence. The information requested and associated time frame in this letter were discussed with Roger Waters of your staff on September 2, 2009.

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:
As stated

cc w/encl: Distribution via Listserv

NRC STAFF ACCEPTANCE REVIEW COMMENTS

REGARDING LICENSE AMENDMENT REQUEST FOR 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 No. ML091940176, Entergy Nuclear Operations, Inc. (Entergy or the licenseee) submitted a license amendment request (LAR) for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3). The proposed amendment would license a new shielded transfer canister (STC) and would allow spent fuel to be transferred from the IP3 spent fuel pool to the IP2 spent fuel pool in preparation for further transfer to the Independent Spent Fuel Storage Installation (ISFSI) already located on the site. The NRC staff has reviewed your application for acceptance and concluded that it did not provide technical information in sufficient detail to enable the NRC staff to commence its detailed review and make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment. The following items need to be addressed.

A. Information Needed to Complete the Acceptance Review

1. Safety Functions of Major Components

a. The LAR should completely delineate the performance requirements for the STC and HI-TRAC cask and include a failure modes and effects analysis to demonstrate that evaluations have been performed to show that safety functions will be accomplished for design basis events and other credible failures. At a minimum, the performance requirements should describe how each major component contributes to the safety functions described in Title 10 of the *Code of Federal Regulations* (10 CFR) Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," GDC 61 and GDC 62 (i.e., shielding, confinement, decay heat removal, and criticality prevention). Refer to NUREG-0800, Standard Review Plan, for additional guidance, especially Chapter 15. For example, the application should contain a sufficiently broad spectrum of accidents and initiating events including the hazard and events further addressed below. Also, for accident analysis acceptance criteria, the design should be robust enough that all the postulated accidents produce about the same level of risk.

b. Section 1.3.2 of Enclosure 1 to the LAR states that the function of the HI-TRAC is to retain its contents under normal, off-normal, and accident conditions. That section also states that the HI-TRAC is designed to:

i. Provide maximum shielding to the plant personnel engaged in conducting "short-term operations" pertaining to inter-unit transfer.

ii. Provide protection to the STC and the spent fuel against extreme environmental phenomena loads, such as tornado missiles, during short-term operations.
iii. Serve as the container equipped with the appropriate lifting devices in full design compliance with NUREG-0612, Section 5.1.6.(3) and ANSI N14.6 to lift, move, and handle the STC, as required, to perform the short-term operations.

However, the LAR does not indicate if the annular water volume inside the HI-TRAC performs shielding or heat transfer safety functions. Consequently, the LAR did not include a thermal or shielding evaluation of the effect of the loss of the annular water volume in the HI-TRAC. Only the loss of the external jacket water was considered.

c. The referenced tornado missile analysis is incomplete because the safety functions performed by the HI-TRAC are different. Specifically, the referenced tornado analysis evaluated the HI-TRAC for penetration and deformation of the HI-TRAC shell and lid to demonstrate that the canister would not be penetrated and the canister would be retrievable; the analysis does not demonstrate that the HI-TRAC bolted lid connections would retain the annular water volume following a missile impact.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.34 and GDCs 2, 61, and 62.

2. <u>Thermal (Heat Removal) and Containment (HI-TRAC and STC) Analyses</u>

a. Control of the STC internal temperature and pressure is based predominantly on factors managed by administrative controls (procedures) rather than design measures. Procedures control the heat load within the STC, the water volume within the STC, and the water volume in the annular space between the STC and the HI-TRAC. The heat transfer to the environment is significantly affected by the water volumes in both the STC and the HI-TRAC. The pressure response is highly dependent on the administratively controlled establishment of an air volume within the STC and HI-TRAC. In a water-solid condition within either the STC or HI-TRAC, the pressure change resulting from a small change in temperature could be significant.

Given the dependence on administrative controls, the licensee has not evaluated the temperature and pressure response resulting from: (1) the thermal misloading of a fuel assembly; (2) the failure to establish adequate water volumes in the STC and HI-TRAC, or (3) the failure to establish an adequate air volume in the STC and HI-TRAC. As further specified below, this information is needed to evaluate the potential consequences and the prerequisite reliability of the administrative controls that is needed in preventing or mitigating these events. (Also, see item 1a above)

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

b. The application needs to contain a thermal analysis for misload of fuel assemblies that exceed specified decay heat limits. Misloads are credible events based on industry operational events in spent fuel pools and dry storage casks. The impact of a thermal misload on the proposed STC containment pressure boundaries and potential consequences are not analyzed.

The application indicates misloads will be detected by measuring and identifying "large differences in temperature" between STC cells. However, the application did not provide any information on the equipment, techniques, sensitivity, nor testing and validation of this method, which is needed to assure reliable identification of fuel assemblies, which have a potential wide range of decay heat.

The thermal misload analyses needs to consider the type of misloads that are credible, including the maximum possible decay heat, which could result in immediate heat-up problems during STS closure and handling operations; and moderate heat loads above the limits which could result in delayed heat-up problems during HI-TRAC transfer. The misload analyses should determine thermal safety margins with respect to the specified design pressure limits, and specify the potential consequence of exceeding the pressure limit. If a method for detecting thermal misloads is credited to mitigate the consequences, then the application should provide sufficient information on the equipment and validation of the method to demonstrate the reliability in preventing misloads is sufficient.

This information is needed for staff's review to ensure compliance with the criteria contained in GDC-61 and the intent of 10 CFR 72.128.

c. The design needs to provide a positive means of establishing the required air volume in the STC and the air volume in the HI-TRAC. Failure to establish an adequate water volume in the HI-TRAC or an adequate air gap in either the STC or HI-TRAC appears to be credible because any one of these results would require the incorrect performance of just one procedural step. Alternatively, the application needs to analyze failure to maintain an appropriate air volume as a credible accident. The use of an air volume appears to be critical in ensuring thermal-hydraulic performance during normal and accident conditions. The application indicates that performance of a pump will determine the air volume in the STC by loss of suction. However, based on operational experience, it appears that administrative errors, human factors, or undetected equipment malfunction in the pump system could result in not establishing the required air volume in the containment system.

This information is needed in order for the staff to proceed to perform a technical review of the spent fuel transfer system in accordance with GDC 61 and the intent of 10 CFR 72.4, 72.122, and 72.128.

d. The application needs to provide design information on the gasket sealing systems, including the gasket dimensions, elastomeric seal material, and justification that the seal will perform as required during normal and accident conditions. This information does not appear to be present in the application, and is necessary to determine the confinement will function during use.

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

e. The application needs to justify that the fire parameters used in the fire analyses bound those expected for a potential fire affecting the casks. Although the use of Part 71 values is appropriate for general certification of transportation packages, the application needs to verify that the actual HI-TRAC/STC physical characteristics and fuel fire characteristics for the analyzed accident are appropriate. Also, provide a list with the location of any significant permanent fire hazards located close enough to the haul path to affect the transporter and cask.

If a fire should occur during transport, the application should specify the expected fire temperatures at the cask surfaces and the duration of the fire.

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

f. The application in the acceptance test and maintenance program needs to specify leak testing requirements for the entire confinement boundaries of both the STC and HI-TRAC. The leak testing requirements should clearly specify the frequency and allowable leakage rates to ensure the protocols are generally consistent with the requirements for design, fabrication, periodic, and maintenance tests specified in American National Standards Institute (ANSI) standard N14.5.

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

g. The application should provide a thermal analysis for the loaded STC and consider a misloaded recently irradiated fuel assembly in the event there is a crane malfunction while moving the STC from the spent fuel pool to the HI-TRAC. The duration of the analysis should consider the time needed to repair the crane or use manual crane overrides to return the STC to the spent fuel pool. If the water in the STC reaches the boiling point, the application should show if vents could restrict steam release and result in pressurization of the STC.

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

h. The application needs to include the calculation package(s) for the thermal-hydraulic analyses of the system during normal and accident conditions. The package should provide design inputs, modeling assumptions, and evaluation of calculation values. 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.

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

3. <u>Criticality Analyses</u>

a. Provide code validation for using the CASMO-4 code, including input data and results, for fuel assembly isotopic concentration for the spent fuel assemblies to be transferred.

On page 4-16, the Safety Analysis Report indicates that CASMO-4, version 2.05.14, was used in determining the isotopic concentrations of the spent fuel assemblies to be transferred from IP3 to the IP2 spent fuel pool. However, the staff was unable to find any detailed information regarding the validation of the CASMO-4 code for spent fuels that have been discharged from the core and cooled for various times.

In accordance with ANS 8.1, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors," Chapter 9 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" and NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," computer codes that are used to calculate effective multiplication factor,

keff, and the isotopic concentration of the spent fuel must be validated. The Entergy application, however, does not provide any detailed information concerning the validation of the CASMO-4 code for spent fuel isotopic concentration analysis. The licensee is requested to provide supplementary information for the CASMO-4 computer code that is used in the application for determination of isotopic concentration of the spent fuels to be transferred from IP3 to the IP2 spent fuel pool using the STC.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.68.

b. Provide justification for the applicability of the selected critical experiments to the code benchmark and upper safety limit (USL) calculation of the Indian Point STC criticality calculation. Provide, if necessary, an updated USL with additional applicable benchmark experiments included.

Table 4.A.1 provides information on the selected critical experiments that were used in the code benchmark and USL calculation for the Indian Point STC criticality safety evaluation. However, it appears that these critical experiments may not be adequate for this purpose because the fuel compositions from these experiments are very different from that of the IP3 spent fuels. In general, the IP3 spent fuels contain various low quantities of fissile materials such as U-235 and Pu-239 and many actinides and fission products. Therefore, it may be inadequate to use fresh fuel and mixed-oxide (MOX) fuel critical experiments to determine the code performance in terms of bias and uncertainties. The licensee is requested to provide supplemental information that can justify the applicability of these experiments for the spent fuel STC.

If the current set of benchmark experiments needs to be augmented by additional benchmark calculations, provide updated critical experiments as necessary.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.68 and NUREG-0800.

c. Metamic neutron absorber plates are credited in the criticality analysis. However the qualification of Metamic for its safety-related use in the STC is not provided. Therefore, the staff requests the results of the qualification of Metamic for its safety-related use in the STC. The licensee shall provide the service conditions and design requirements identified for the life of the STC. The licensee shall provide the neutron absorber material qualification performed. The qualification must include neutronic as well as mechanical aspects of the Metamic neutron absorber plates necessary for them to perform their safety-related function in the STC. If reliance is placed on precedents, those precedents should be explicitly identified and differences between those precedents and the current application must be identified and justified as to why the precedent remains valid. There may be a need for a periodic surveillance of these plates to check for degradation.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.68.

4. Transport Roadway Analysis and Cask Tipover Accident

In a letter dated June 11, 2009, ADAMS Accession No. ML091520167, the NRC staff set forth draft fuel cask evaluation criteria. One of these criteria was an analysis for a cask tipover event. In the LAR, the licensee stated that cask tipover was not a credible accident, as the roadway would be qualified for the load, and the vertical cask transporter (VCT) has a redundant drop protection feature. However, there was no roadway analysis provided, just a commitment to perform one before the first cask movement. In order to meet the requirements of GDC 4 (environmental conditions and dynamic effects of missiles (i.e. dropped load) on affected structures, systems, and components (SSCs) that are important to safety) and GDC 61 (adequate safety of the transfer cask with the STC loaded with spent fuel under normal and postulated accident conditions) during the heavy load movement along the haul path, please provide the appropriate analyses in accordance with NUREG-0612.

Please provide the following for staff review: (i) the proposed haul path for the heavy load movement; and (ii) the evaluation of all affected important-to-safety (includes safety-related) SSCs along and adjacent to the haul path that demonstrate that these SSCs are capable of withstanding applicable loads and load combinations resulting from the heavy load movement, including those required by GDC 2 and GDC 4, with an acceptable margin of safety. Also, (i) establish that the potential for an accident condition along the haul path is minimized by providing analyses, load test details and/or operating experience that demonstrate that the VCT is of a single-failure-proof design and the entire transport roadway (haul path) has adequate strength capacity under applicable design loads and load combinations during the proposed heavy load movement of spent fuel; or (ii) postulate a worst-case accident condition (e.g. cask tipover) along the haul path and demonstrate that the consequences are acceptable.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDCs 2, 4, and 61.

5. Crane Design

The licensee proposed a new commitment related to the crane design in Attachment 6 to the LAR. The commitment states:

> The IP3 crane will be upgraded to a single failure proof crane meeting the intent of NUREG-0554 through the use of ASME [American Society of Mechanical Engineers] NOG-1-2004 as the governing design code.

The commitment is unclear because NUREG-0554 and ASME Standard NOG-1, 2004, have been accepted by the NRC staff only as design standards for entirely new cranes or entirely new portions of upgraded cranes. Appendix C to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines that supplement or provide alternatives to NUREG-0554 criteria for upgrades of existing cranes, particularly those related to unreplaced portions of upgraded cranes. However, when material properties and weld configurations for unreplaced structures can be reliably determined, the seismic and dynamic structural analysis methods presented in ASME NOG-1, 2004, may be used for the entire upgraded crane.

The licensee must clarify if the crane design upgrade itself is part of the LAR or will be implemented without NRC staff review, pursuant to the provisions of 10 CFR 50.59. If the crane

upgrade itself is part of the LAR, the licensee should provide a matrix listing the guidelines of NUREG-0554, as modified by the guidelines of Appendix C to NUREG-0612, and a brief description of how the intent of the guideline would be satisfied. Otherwise, the licensee should commit to the guidelines of Appendix C to NUREG-0612 and NUREG-0554, except that the criteria of ASME NOG-1, 2004, may be employed as an acceptable alternative to the NUREG-0554 criteria. Commitment to only the intent of an NRC-approved methodology is unacceptable when implementing the methodology, pursuant to the provisions of 10 CFR 50.59, without NRC staff review.

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

6. STC Materials and ASME Boiler and Pressure Vessel Code (ASME Code) Requirements

a. Holtec Licensing Report HI-2094289 (the submittal) identified the 2004 Edition of Section III of the ASME Code, Subsection ND, for stress limits. The submittal is silent on the edition and addenda of Section III of the ASME Code applicable to the fabrication, testing, and inspection of the STC. Provide the ASME Code, Section III, edition and addenda that apply to these activities.

b. The STC is capable of being used for multiple cycles over its planned life. Provide a statement that identifies the planned life duration of the STC.

c. Table 8.2.1 of the submittal provides cycle duration (8 hours to several days) for fuel transfer operations. For postulated accident conditions, paragraph 3.2.3.2(h) of the submittal used 30 days for dose calculations. Provide the limiting duration for an abnormal fuel transfer operation cycle and postulate your action if the duration limit is approached.

d. Table 8.2.1 of the submittal lists Carboguard 890 coatings for the STC surfaces. Although Section 8.3 states that coatings on carbon steel do not react with borated water, no information is provided to support this statement. The coating applied to the STC's interior surface is in contact with water for several days during the transfer of the spent fuel assemblies, and the coating may be in contact with water for a longer duration as a result of abnormal conditions. Provide technical data on the effects of abnormal time (using the answer to the question above) and temperature (Table 3.1.1 of the submittal) and radiation conditions on the interior coating integrity.

e. The spent fuel cladding must be protected during storage against degradation. The Carboguard 890 product data sheet only identifies the generic ingredients. The product data sheet does not identify the residual and tramp elements that are in the generic ingredients, if any. These residual and tramp elements in the STC interior surface coating could potentially be detrimental to the fuel cladding. Provide technical data on leaching of elements (such as halogens) detrimental to the fuel basket and cladding as a result of abnormal time and temperatures exposure. If the data indicate elevated levels of detrimental elements, provide your mitigating action.

f. Paragraph 8.5.1 of the submittal states that periodic structural or pressure tests are not required to verify continuing performance. The statement is without supporting technical data. The loading and unloading process may have an accumulative effect on the STC's weld integrity and interior coating integrity. Provide the non-destructive evaluation (NDE) methods, inspection

frequency, and acceptance criteria for verifying weld and interior coating integrity is maintained over the planned life duration of the STC.

g. Paragraph 8.5.2 of the submittal states that the STC seal will be tested prior to each fuel transfer. Although paragraph 8.4.4 and paragraph 10.2.3, procedure step 23 contains leakage test criteria, this information was not referenced in Paragraph 8.5.2. In Paragraph 8.5.2, provide a reference or description of the leakage test process; identify the inspection method, and provide inspection acceptance criteria.

h. The STC has a lead shield sandwiched between the inner and outer carbon steel shell. The lead shield is used to reduce dosage in the surrounding area. For the lead shield, provide the NDE method and/or manufacturing process used to verify material soundness and shielding effectiveness. An example of fabricating lead shielding and testing for effectiveness is in Section 9.1.5 of the HI-STORM 100 Final Safety Analysis Report (FSAR).

i. Paragraph 8.4.5 of the submittal states that certain ferritic steels in the STC are tested in order to assure that these materials are not subject to brittle fracture failures. The STC references ASME specifications. These specifications provide an option for the purchaser to request brittle fracture testing (impact testing). Identify the specific steel items used in STC that were impact tested (such as the reference numbers on the drawings from Section 1.5 of the submittal) and state the temperature acceptance requirement.

j. Table 5.4.2 of the submittal lists the STC internal pressure as 53.5 psig in the analysis of the loss of the HI-TRAC jacket water. Paragraph 2.2.3 states that the STC is qualified to withstand a normal internal pressure of 50 psig, and Table 3.2.1 has an accident pressure limit of 65 psig. Paragraph 10.2.3, Procedure Step 23 leak test for the STC is at 55 +5/-0 psig per American National Standards Institute (ANSI) standard N14.5, Section A.5.7. ASME Code, Section III, NB-6000 and ANSI N14.5 have a hold test pressure requirement based on system design. Provide the specific internal test pressure that will be used to ensure that the STC will perform its safety function for any credible abnormal condition and satisfy NB-6000 test pressure requirements.

k. Paragraph 8.4.2 of the submittal referenced the drawing in Section 1.5 and applicable codes and standards for weld examinations. The drawings in Section 1.5 do not contain STC weld examination criteria. The STC welds perform containment functions and can experience pressure boundary conditions. Provide the examination methods, acceptance criteria, and volume (methods can be ASME Code, Section III, Subsection NB construction and preservice, or Draft NUREG-1536, Revision 1A, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility)." Volume can be expressed in sketches of the weld cross section area, heat affected zone, and base metal.

l. Section 8.2 of the submittal is void of information on weld material, although weld specifications are included in the references. Provide a discussion on the welding material and on the application of the referenced welding specifications (show the weld specification for specific locations on the applicable drawings).

m. Table 3.1.1 of the submittal has a temperature rating for the STC seal of 248 °F. The closed and sealed STC is placed in the HI-TRAC. All of the HI-TRAC temperatures in Table 3.1.1 exceed the 248 °F rating for the STC seals. Provide a method for monitoring STC seal

temperature, data supporting seal effectiveness at the higher HI-TRAC temperatures, or other ways of mitigating STC seal temperatures exceeding 248 °F.

n. In Section 4.7.6 of the submittal it states, "During manufacturing there is a potential for minor damage to the neutron absorber panels from welding the sheathing to the cell walls. Such damage, up to an area equivalent to 1 inch diameter per panel, was considered in Holtec's HI-STAR Transport SAR [K.C], Section 6.4.11, for various baskets similar to the STC basket, and was found to be acceptable. This condition is therefore also acceptable for the STC, without any further calculations." However, the STC will have a much different operating regime/history than the HI-STAR Transport Cask. Provide an evaluation of the welding damage given the intended operating regime/history of the STC.

All of the above information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDCs 1, 4, and 61 and the intent of 10 CFR 72.24, 72.122, 72.140, and 72.234.

7. Shielding Analyses

a. Modify the shielding evaluation to address the non-fuel hardware that is to be transported with the IP3 assemblies in the STC.

Some sections (e.g., Criticality) of the application enclosure evaluating the fuel transfer indicate that non-fuel hardware (e.g., BPRA, CEA, APSR, WABA, etc.) will be moved with the fuel assemblies. However, this hardware was not accounted for in the shielding evaluation. The evaluation should include the parameters important for shielding, such as the types of non-fuel hardware to be transported, the material specifications and assumed impurity levels (including the basis for the assumption), the maximum equivalent burnup and minimum cooling time for these items covered by the shielding evaluation, and the impacts on dose rates.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).

b. Provide the following fuel specifications: maximum uranium mass loading of the assembly types and the size/length of axial blankets.

The shielding evaluation should address these fuel specifications. The maximum mass loading is a main driver in determining the assembly type that is used as the design basis assembly for shielding evaluations. Axial blankets of sufficient length can have a noticeable effect on source term and thus dose rate. The shielding evaluation should appropriately address these fuel characteristics.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).

c. Provide acceptance tests for the STC shielding to ensure the as-fabricated shielding features will perform as designed.

The application (see enclosures 1 and 2) does not describe any acceptance testing of the design features relied upon for shielding in the STC. One of those features is the lead between

the steel inner and outer shells of the STC body. Also, it is not clear how this lead is placed in the STC (i.e., poured or installing of pre-cast sections). The fabrication of the lead shielding should be described, including how development of voids or gaps in the shielding will be precluded. The staff notes that descriptions regarding fabrication of the shielding and testing of its effectiveness are described for the HI-TRAC transfer casks in the HI-STORM 100 FSAR and these fabrication descriptions and effectiveness tests have been found to be acceptable in the licensing activities for the HI-STORM 100 system. Also, there are no acceptance tests described for the Metamic neutron absorber plates. While a significant criticality safety design feature, the licensee's shielding analysis also relies upon the Metamic plates for the shielding design. Thus, the application should describe an acceptance testing program for ensuring the plates perform as designed. Or, the licensee could quantify the effect of the Metamic plates on dose rates and show the effect of the plates is negligible and, therefore, for the purposes of the shielding design only, an acceptance testing program is not needed.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104.

d. Provide a conservative evaluation of off-normal conditions and the resulting doses.

The application does not provide an evaluation of off-normal conditions and the doses that would result from such scenarios. Instead, the application leaves this evaluation for later. Since the application is for use of the STC at a single site, Indian Point, the possible scenarios for off-normal conditions should be relatively easy to postulate and a reasonably conservative evaluation can be performed. Further, the licensee is applying the regulatory criteria from 10 CFR 72.104 to the shielding and radiation protection evaluation. Those criteria apply to normal and anticipated occurrences (i.e., off-normal conditions). Thus, to verify that the design meets these criteria, the evaluation needs to address off-normal conditions. The evaluation should include appropriate descriptions of the conditions assumed, including time duration, with appropriate justification, as well as dose estimates (from both direct radiation and effluent release) to show that the 72.104 criteria are met when off-normal conditions are considered.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).

e. Include in the shielding evaluation the contributions from the other "uranium fuel cycle operations within the region" as well as the number of fuel transfers that can be performed to demonstrate compliance with the criteria of 10 CFR 72.104(a).

The criteria of 10 CFR 72.104(a) includes the contributions from other fuel cycle operations in the region in demonstrating compliance with the 25 mrem annual dose limit. For Indian Point, these operations include the reactor units and associated facilities as well as the operating ISFSI. The evaluation does not include the contributions from these facilities. In addition, the criteria in 10 CFR 72.104(a) are annual dose limits. The evaluation only considers a single transfer under normal conditions. It is anticipated that multiple transfers will occur each year. Thus, an evaluation against the 72.104(a) criteria needs to consider the number of fuel transfers per year as well as anticipated occurrences (see the previous question). The evaluation should clearly show that the contributions of direct radiation and effluents are included in the analysis. Depending upon this evaluation's results, fuel transfers may need to be limited to a set number per year.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).

f. The application needs to include the calculation package(s)for the radiological release analyses during normal and accident conditions. The staff needs to verify that the system is appropriately analyzed and modeled, in order to verify the acceptability of the radiation doses reported in the application.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).

8. Miscellaneous

a. The NRC staff requests that the following documents be submitted, as they are needed for the NRC review:

i. [L.I.] Holtec International Report HI-2084118, "Shielded Transfer Canister Structural Calculation Package" Latest Revision
ii. [U.C.] Holtec International Report HI-2084118, Revision 1, "STC Structural Calculation Package"
iii. [U.D.] Holtec International Report HI-2094345, Revision 0, "Analysis of a Postulated HI-TRAC 100D Drop Accident During Spent Fuel Wet Transfer Operation"
iv. [U.B] IPEC HI-STORM 100 Cask System 72.212 Evaluation Report.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.212.

b. The application needs to update each reference to the HI-STORM FSAR. Each reference needs to include a specific citation to the FSAR section, revision number, and date. The safety bases of the application appear to rely in part on the safety bases of design and analytical information in the FSAR. The staff needs to verify that the specific safety bases are applicable to this application and whether the application relies on information changed in the HI-STORM FSAR under the 72.48 process. To facilitate the review, the licensee should provide a table identifying each FSAR reference, applicable regulatory requirements, applicable industry/code requirements, and a description of how compliance with each is attained.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.34.

c. Please include a compliance matrix which identifies the regulatory requirements that apply to this LAR and the specific acceptance criteria for each SAR chapter that is used to demonstrate compliance with the regulatory requirement. This can be a reference to the pages in the LAR.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.34 and the intent of other criteria in 10 CFR Part 72.

B. Questions That May Be Addressed Now Or In Future Staff Requests

1. Technical Specification (TS) Changes

a. Please explain how the proposed TS changes meet the requirements of 10 CFR 50.36. Also, it appears that IP3 should have a TS for 2000 ppm boron in the spent fuel pool rather than relying on a statement in the updated FSAR to provide the 2000 ppm requirement.

b. There are typographical errors on IP3 TS Bases pages B3.7.18-1 and B3.7.18-2.

2. Administrative Items

a. The information contained in Section 4.5.3 and Table 4.5.3 should not be marked as proprietary information. It is copied from NUREG/CR-6801, which is a publicly available document.

3. ALARA

a. Describe the actions performed in the event that measured dose rates exceed the calculated (or "expected") dose rates.

Operations descriptions in the Enclosures include operations No. 18 on page 10-9 and No. 33 on page 10-10. These operations are dose rate measurements to check dose rates against calculated (or "expected") dose rates. The operations should also describe actions to be taken in the event that measured dose rates exceed the calculated values.

b. Clarify that the estimated occupational exposures estimated in Section 7.4.7 of the enclosures include the dose from effluents.

Occupational exposures result from direct radiation from the STC as well as effluents from the STC with the leak rate defined in the application. It is not clear that the occupational exposure estimates include potential contributions from both of these sources.

c. Provide a sample shielding calculation input file.

A sample input file can help with the review of the shielding analysis. A sample input file allows the reviewer to quickly understand the model and alleviate the need for questions regarding parts of the analysis that are not clear in the application's description of the analysis model, thus speeding up the review.

d. Provide information to demonstrate that compliance will be achieved with the radiation dose limits for individual members of the public as required in 10 CFR 20.1301(a)(1), (a)(2), (b), and (e).

1. Such information may include a map of the proposed transfer route from IP3 to IP2 with approximate nearest distances to Indian Point's controlled area and unrestricted area boundaries.

2. If members of the public are allowed in the controlled area and/or the restricted area, provide information to demonstrate that compliance will be achieved with the 100 mrem dose limit in 20.1301(a)(1) as required in 20.1301(b).

3. Provide a dose assessment to demonstrate that the dose limit of 2 mrem in an hour in the unrestricted area will be met as required in 20.1301(a)(2).

4. Provide a dose assessment to demonstrate that compliance will be achieved with the EPA generally applicable environmental radiation standard for real individuals in the unrestricted area as required in 10 CFR 20.1301(e).

All the information above is needed to ensure compliance with the criteria in GDC 61, 10 CFR Part 20, and the intent of 10 CFR 72.104.

4. Transport Analyses

Please provide information on the structural capacities of the air pads, low profile transporter and VCT and how they compare with the maximum loads placed on them during the proposed spent fuel transfer. Also, provide the maximum height above the ground at which the HI-TRAC 100D transfer cask may be carried on the VCT during transport along the haul path.

This information is needed to ensure compliance with the criteria in GDCs 2, 4, and 61.

the NRC staff's request is not received by the above date, the application will not be accepted for review pursuant to 10 CFR 2.101, and the NRC staff will cease its review activities associated with the application. If the application is subsequently accepted for review, you will be advised of any further information needed to support the NRC staff's detailed technical review by separate correspondence. The information requested and associated time frame in this letter were discussed with Roger Waters of your staff on September 2, 2009.

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:
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INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - SUPPLEMENTAL INFORMATION NEEDED FOR LICENSE AMENDMENT REQUEST FOR SPENT FUEL TRANSFER (TAC NOS. ME1671, ME1672 AND L24299)

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