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

September 28, 2011

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

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

Dear Sir or Madam:

By letter dated July 8, 2009, as supplemented by letters dated September 28, 2009, October 5, 2010, and July 28, 2011, 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 30 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

Enclosure:
As stated

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION (RAI)

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. ML091940176 (proprietary), ML091940177 (non-proprietary) and ML091940178 (non-proprietary), as supplemented by letters dated September 28, 2009, (ADAMS Accession Nos. ML092950460 (proprietary) and ML093020080 (non-proprietary)), October 5, 2010 (ADAMS Accession Nos. ML102910508 (proprietary), ML103080112 (non-proprietary), and ML103080113 (non-proprietary)) and July 28, 2011 (ADAMS Accession Nos. ML112200258 (proprietary), and ML112430437 (non-proprietary)), 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 4, ADAMS Accession No. ML11243A220 (non-proprietary). The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions:

CHAPTER 4 - CRITICALITY EVALUATION (CSDAB and SRXB)

4-1. In Section 6.2.8 of the SAR it describes the fuel assemblies sliding to the STC lid in the tip-over analysis. As this moves the fuel assemblies away from the neutron absorber in the cells, the NRC staff needs further information on this configuration. The NRC staff is concerned that a criticality event could add more heat than the analyzed heat load, resulting in pressurization and failure of the confinement boundary, as well as the hazard of a criticality event to workers in the area. Provide a criticality analysis for this configuration or redesign the STC internals (perhaps use a spacer) to ensure the fuel assemblies will not slide towards the STC lid during the postulated tip-over accident. If a criticality analysis is provided, it should take into consideration the fact that the upper part of the fuel assembly may be more reactive than the average of the entire fuel assembly.

This information is required to demonstrate that the system can withstand the postulated accidents and successfully maintain subcriticality, in compliance with General Design Criteria (GDC) 61, GDC 62, and the intent of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 72.122.

Note that additional Chapter 4 RAIs will follow in a subsequent letter.

Enclosure

CHAPTER 7 – SHIELDING DESIGN AND ALARA CONSIDERATIONS

7-1. Provide a dose rate evaluation for neutron source assemblies (NSAs) with the appropriate cobalt impurity level, modifying other aspects of the shielding evaluation as necessary. (CSDAB)

In response to RAI 7-9 in a letter dated July 28, 2011, the licensee provided cobalt impurity levels for the assembly hardware and non-fuel hardware that are based on hardware age and vendor data. The impurity level for NSAs doesn't appear to be consistent with the age and vendor data. Given the cooling time and the allowable burnup of the NSAs, this hardware would appear to have been manufactured at a time of higher impurity levels. Thus, the shielding evaluation should address NSAs with the higher impurity level or further justification should be provided to support using the lower impurity level for NSAs.

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).

7-2. Modify the shielding accident evaluations to address dose at distance from the HI-TRAC/STC system in a tipover that includes any uncovering of fuel in the STC. (CSDAB)

In response to RAI 7-13 in a letter dated July 28, 2011, the licensee modified the accident dose analysis to include aspects of the tipover scenario. However, the response does not appear to have addressed all the concerns expressed in that RAI. Given the void space in the STC, there may be a possibility for some of the fuel to become uncovered with the STC on its side. This condition could have significant dose consequences and should be evaluated. Any assumptions used in the analysis should be adequately justified.

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

7-3. Provide the neutron energy spectrum on the outer surface of the STC and the transfer cask at the measurement points specified in Technical Specification (TS) 5.4.6 that were used to determine the neutron dose rates at those points. (CSDAB)

TS 5.4.2 establishes requirements that the neutron plus gamma measured dose rates not to exceed 1400 mrem/hr on the top of the STC (with the lid in place) and 5 mrem/hr on the side of the transfer cask. In order to measure the neutron dose rate, the neutron energy spectrum must be approximated to ensure the detection equipment is appropriately calibrated for the neutron spectrum at the measurement points. Therefore, please provide the MCNP-based neutron energy spectrum at the measurement points specified in TS 5.4.6.

This information is needed to confirm compliance with 10 CFR 50.34, 10 CFR 20.1301(a) and (b), 20.1501(b), and the intent of 10 CFR 72.104.

CHAPTER 8 – MATERIALS EVALUATION, ACCEPTANCE TESTS and MAINTENANCE PROGRAM

Refer to RAI 8-4 from the licensee's letter dated July 28, 2011. Provide justification to reconcile the following discrepancies in applicable rules for construction between the American Society of

Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Division 1, Subsection ND and Subsection NC for the construction of the STC to ensure that the STC is constructed to acceptable quality standards:

8-1. The lateral expansion test results for the STC bolting material met the requirements for ASME Code Subsection ND, however, it did not meet the requirements stated in Subsection NC. Substantiate the justification provided previously by providing a numeric estimate of the lateral expansion value at the lowest service temperature. Include the basis and calculation for both the lowest service temperature and the estimated lateral expansion value. Alternatively, use bolting material meeting the acceptance requirements of ASME Code Subsection NC. (SMMB)

8-2. ASME Code Subsection NC requires full radiography of the weld joints. However, select radiography is required by ASME Code Subsection ND. Substantiate the justification provided previously for the limited radiographic testing (RT). Provide evidence that the portions of the confinement boundary welds that were not subject to RT will perform as expected. Include a discussion of weld repairs that resulted from RT over adjacent and similar joint configurations. Alternatively, perform full radiography of the weld joints as required by ASME Code Subsection NC. (SMMB)

Spent fuel canisters are normally constructed to ASME Code Subsection NB or NC. The licensee is proposing to construct the STC to ASME Code Subsection ND and has provided a table outlining the discrepancies between Subsection NC and ND. However, discrepancies regarding both brittle material behavior and weld quality were not adequately addressed.

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

8-3. Modify the acceptance tests descriptions to address the following items: (CSDAB)

   a. State in the first paragraph in SAR Section 8.4.1 that visual inspections and measurements will ensure the STC dimensions conform to the TS.
   b. Define the term "packaging" used in SAR Section 8.4.1.
   c. Define "significant" as used for the lead shielding acceptance testing in SAR Section 8.4.5.

RAI 8-3 in the NRC letter dated March 16, 2011, requested a change to Section 8.4.1 of the SAR to indicate that the STC dimensions would be inspected to ensure compliance with the TS. This aspect of that RAI was not completely addressed. The same section of the SAR also uses the term "packaging." This term is not defined anywhere in the SAR and it is unclear as to what constitutes the "packaging" that will be inspected. The STC lead shielding is found acceptable, per Section 8.4.5 of the SAR, if the gamma scan measurements don't vary significantly from the average gamma dose rate measurements, accounting for the presence of the STC ribs. The applicant should describe what constitutes a significant variation, especially in light of the dose rates at and near the STC ribs being rapidly varying from 11.4 to 4.4 rem/hr on contact while the average dose rate is about 3.3 rem/hr on contact. If comparison is made to some standard, such as a test block, then the acceptance criterion should be clarified to indicate this comparison and what constitutes acceptable shielding performance.

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).

CHAPTER 10 - OPERATING PROCEDURES

In a letter dated July 28, 2011, Entergy responded to NRC RAI TS-8 by stating that Surveillance Requirement (SR) 3.1.4.2 has been added to specify the pressure instrumentation to be used and that an ASME Code compliant pressure relief valve or rupture disc must be installed during the test. The proposed SR states, "Verify that an ASME code compliant pressure relief valve or rupture disc and two channels of pressure instrumentation with a range of at least 0.1 psia to 15 psia and calibrated to within 1% accuracy within the past 12 months are installed on the STC."

In a clarifying response, Entergy described that the STC lid is fitted with process connections which included valve quick-disconnect nipples (Snap-Tite model SHVN-12-12) and that, prior to installing the nuts that hold the STC lid in place, hoses with the mating valved quick-connect couplings are attached to the lid which ties them into the relief valves, pressure gauges, and further downstream, isolation valves used for the test.

10-1. The proposed arrangement has an isolation valve (quick disconnect) in the pressure relief line flow path. This configuration is inconsistent with the ASME Code. Describe how direct communication will be verified between the STC and the pressure relief valves and pressure instrumentation. (SBPB)

This information is needed to confirm compliance with GDC 61, GDC-63, and 10 CFR Part 50, Appendix B.

10-2. The pressure rise test is monitoring for a very small increase in pressure. However, any small leakage at fitting and pipe connections would adversely influence the test results. Describe how leakage will be prevented at the various valves and fittings to assure the pressure rise test results will not be invalidated. (SBPB)

This information is needed to confirm compliance with GDC 61, GDC-63, and 10 CFR Part 50, Appendix B.

10-3. Clarify the language in Chapter 10 of the SAR with regard to filling of the HI-TRAC neutron shield jacket and performance of the dose rate measurements as described below. (CSDAB)

The condition given in Section 10.1.3, Step 5 for filling the HI-TRAC neutron shield jacket is not clear. It should be modified to indicate that the shield jacket is filled, if not already filled. Additionally, the language regarding the dose rate measurements given in Section 10.2.3, Steps 18, 19, 23, and 24 should be clarified to indicate that the measurements are compared against dose rate limits and not expected dose rates. Also, the second aspect of the written evaluation described in Steps 19.c and 24.c is not clear. It seems this part should be written as: "...(2) if the higher dose rates are acceptable. If the higher dose rates are acceptable, fuel transfer can continue ..."

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.

TECHNICAL SPECIFICATIONS

TS-1. Modify proposed TS Appendix C, Part II, 5.4.2 and 5.4.6 to provide dose rate limits and associated measurements that are appropriate for the allowable loading configurations and are supported by the evaluations in the licensing report (SAR). (CSDAB)

In a letter dated July 28, 2011, in response to RAI TS-4 the applicant proposed dose rate limits and associated measurements for the transfer operations. The dose rate limits are for the STC lid and the side of the HI-TRAC. However, the proposed limit and measurements for the STC lid do not appear to be adequate for the operations. The limit for the STC lid and its measurement were derived from a configuration with the STC fully loaded and the air gap present in the STC. The description in the operations section of the SAR has the measurement performed with the STC filled with borated water. Also, given that the inner basket cells may be kept empty to comply with criticality requirements, it is not clear that the single measurement location at the STC lid radial center is sufficient. Additional measurement locations should be specified that include areas over the outer basket cells.

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

TS-2. Modify the following items in TS Appendix C, Part I (CSDAB):

a. Change the minimum outer STC shell thickness to ¾ inches in Section 1.0.
b. Ensure the minimum dimensions given for the HI-TRAC are consistent between the TS and the licensing drawing.
c. Modify 2.3.m to read "Manual crane operations for bare STC movements."

The shielding evaluations for the bare STC used ¾ inches as the outer shell thickness. Dose rates for the minimum thickness would be significantly higher and impact the evaluations in the Licensing Report. Since the STC has already been built and the as-built shell dimension is at ¾ inches no design change would be necessary for the actual STC. Additionally, consideration should be given to ensuring consistency between the licensing drawing and the TS. The HI-TRAC dimensions should also be consistent between the TS and the licensing drawing. It is not clear that this is the case. Changing 2.3.m to the suggested text captures more generically the off-normal operations, which would include a crane hang-up and any other operational event that requires personnel to be in close proximity to the STC when not in the HI-TRAC.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(a) and (c), 72.104, 72.106, and 72.126.

TS-3. Modify the following aspects of TS Appendix C, Part II, LCO 3.1.2 and Table 4.1.1-1 (CSDAB):

a. Include "and/or NON-FUEL HARDWARE" at the end of the NOTE and in SR 3.1.2.2 of the LCO (SR 3.1.2.2: "... prevents inserting fuel assemblies and/or NON-FUEL HARDWARE into cells 1, 2, 3, and 4 of the STC is installed.")
b. Change Table 3.1.2-2 to show the unanalyzed cooling times for the TPDs and NSAs as N/A, not allowed. Define N/A at the bottom of the Table. Capitalize non-fuel hardware and spell out ITTR in note (a).
c. Confirm the Hafnium Suppressors burnup and cooling time specifications are correct in Table 3.1.2-2.

d. Change note b to Table 3.1.2-2 to indicate that interpolation is not acceptable/applicable for TPDs for burnups greater than 90 GWd/MTU and cooling times greater than 15 years (similar to what is done for HI-STORM 100) as well as for NSAs, RCCAs and Hafnium suppressors.
e. Change LCO 3.1.2 a.1 and b.1 to state initial average enrichment, or change footnote (a) to Table 3.1.2-3 to indicate that initial enrichment is assembly average enrichment and the specification is a limit on the minimum average initial enrichment.
f. In Table 3.1.2-1, the last entry in the column for Configuration B is a dash. The action to take is not clear. Recommend replacing the dash with note (e), "Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel."
g. Clarify Table 4.1.1-1 to indicate that the guide tube material is also ZR.
h. Footnote (b) to Table 4.1.1-1 should be deleted as it is not necessary.

These changes are necessary to keep the proposed TS consistent with the evaluations performed in the SAR and to clarify the conditions and limits on allowable contents. With regard to the Hafnium Suppressors, it is not clear, based on staff's understanding of these suppressors and their specifications given in the HI-STORM 100 Certificate of Compliance TS, that the specifications listed in the proposed TS are appropriate.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(c), 72.104, 72.106, and 72.126.

TS-4. Remove from the proposed TS Appendix C, Part II the specification regarding the minimum restricted area size for the transfer operations (4.1.4.5 and 4.1.4.12). Instead, provide appropriate discussions in an RAI response and in the SAR regarding the radiological controls that will be exercised during the various stages of the operations. (CSDAB)

Based upon the further evaluations of the transfer operations, including design and operations changes, provided in response to the second round of RAIs and further discussions with the licensee, the NRC staff no longer finds a need for the TS to specify minimum restricted area sizes. Instead, the licensee should describe in the SAR, to an appropriate level of detail, the types of radiological controls to be used for the various stages of the operations to ensure compliance with 10 CFR 20.1301(b), providing some tie-in (or context) to the evaluation provided in the application. The applicant should also ensure that text in the application that referred to these technical specifications for minimum restricted area size are modified accordingly.

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

TS-5. TS 3.1.2 refers to Hafnium suppressors while TS 3.1.5 refers to Hafnium inserts. Select a standard term and ensure only that term is used in the TS. (LPL1-1)

September 28, 2011

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

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

Dear Sir or Madam:

By letter dated July 8, 2009, as supplemented by letters dated September 28, 2009, October 5, 2010, and July 28, 2011, 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 30 days of the date of this letter.

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:
As stated

cc w/encl: Distribution via Listserv

DISTRIBUTION:

RidsNrrDorlLpl1-1	RidsNrrPMJBoska	RidsOGCRp	LPL1-1 Reading File
RidsNrrLASLittle	RidsAcrsAcnw_MailCTR	RidsNrrDorlDpr	RidsRgn1MailCenter
RidsNrrDssSrxb	RidsNrrDssSpbp	RidsNrrDeEmcb	RidsNrrDraAadb
RidsNrrDlrRerb	RidsNrrDirslrib	RidsNrrDirsItsb	RidsNrrDciCsgb
RidsNrrDirslhpb	LLund, NRR	DWeaver, NMSS	JGoshen, NMSS
MRahimi, NMSS	DPstrak, NMSS	MWaters, NMSS	

ADAMS ACCESSION NO.: ML112650641 **Via email

OFFICE	NRR/DORL LPL1-1/PM	NRR/DORL LPL1-1/LA	NRR/DSS SBPB/BC	NMSS/DSFST LB/PM**	NMSS/DSFST SMMB/BC**
NAME	JBoska	SLittle	GCasto	JGoshen	DPstrak
DATE	9/26/11	9/26/11	9/27/11	9/26/11	9/27/11
OFFICE	NMSS/DSFST CSDAB/BC**	NMSS/DSFST LB/BC**	NRR/DORL LPL1-1/BC		
NAME	MRahimi	MWaters	NSalgado		
DATE	9/26/11	9/27/11	9/28/11		

OFFICIAL RECORD COPY