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NL-12-074

May 21, 2012

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, DC 20555-0001

Subject: **Indian Point Nuclear Power Plant Units 2 and 3**
Draft Safety Evaluation for License Amendment Request Re: Inter-Unit Spent Fuel Transfer (TAC Nos. ME1671, ME1672, and L24299)
Indian Point Units 2 & 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

Reference:

1) NRC letter to Indian Point Vice President of Operations, 5/07/12, "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Draft Safety Evaluation For License Amendment Request Re: Inter-Unit Spent Fuel Transfer (TAC Nos. ME1671, ME1672, and L24299)".

Dear Sir or Madam:

By letter dated May 7, 2012 (Reference 1), the NRC transmitted to Entergy Nuclear Operations, Inc. (Entergy) a draft safety evaluation (SE) for the proposed amendment. As requested Entergy has reviewed the draft SER and has provided a markup illustrating Entergy's comments as Attachment 1 to this submittal. In addition, Entergy confirms that the only proprietary information contained in the SE is appropriately enclosed within double brackets.

There are no new regulatory commitments in this submittal.

In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated New York State official.

If you have any questions or require additional information, please contact me.

Sincerely,

RW/rw

A001
NRR

Attachment:

Attachment 1: Comments on Draft Safety Evaluation

cc: NRC Resident Inspector's Office
Mr. John Boska, Senior Project Manager, NRC NRR DORL
Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
Mr. William M. Dean, Regional Administrator, NRC Region 1
Mr. Francis J. Murray Jr., President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-12-074

Comments on Draft Safety Evaluation

Entergy Nuclear Operations, Inc.
Indian Point Units 2 and 3
Docket Nos. 50-247 and 50-286

Comment Summary Table

Comment	SE Page	Comment	Discussion
1	54	Delete "(a minimum for which is set in the proposed TS)".	Although the impact of NSAs has been analyzed, Entergy has determined that NSAs will not be transferred at this time. This determination is documented in submittals NL-12-007 and NL-12-047 and reflected in the final proposed TS. Markup provided.
2	15, 18, 30	These pages reference Revision 5 of the Licensing Report. Revision 6 was submitted via NL-12-047.	Markup not provided.
3	79	Replace "Pollack" by "Pollock" in References 1 and 2.	Markup not provided.

The neutron sources for the fuel assemblies are provided in Table 7.2.3 of the SAR. The neutron source spectra in that table are for the same burnup, enrichment and cooling time combinations for which gamma source spectra are provided in Table 7.2.1. In addition to the fuel neutron source, some types of NSAs may also be a significant source of neutrons as described previously. Pu-Be type NSAs will have a source strength similar to that of an assembly. The applicant used the typical initial Pu-Be source strength for the shielding analyses to account for the contribution of NSAs to dose rates. The staff finds this to be acceptable as there will be some decay of the source strength (the neutron production rate) over the course of the NSA's use in the reactor and its post-irradiation cooling time ~~(a minimum for which is set in the proposed TS)~~; therefore, sources with initial strengths that are somewhat above the typical initial source strength will have strengths that are similar to that assumed in the analysis.

3.7.3 Shielding Model Specification

The applicant described the shielding model, including the configurations of the shielding and the source and the material properties, primarily in Section 7.3 of the SAR. Other sections in Chapter 7 of that report contain additional information, including loading patterns. The staff reviewed this information in consideration of the other evaluations, including the structural, thermal, and materials areas, and the descriptions of the transfer operations, as discussed in the following sections.

3.7.3.1 Configuration of the Shielding and Source

3.7.3.1.1 Source Configuration

The configuration of the source is based upon the proposed loading restrictions in the TS. TS LCO 3.1.2, together with TS Figure 3.1.2-1 and TS Tables 3.1.2-2 and 3.1.2-3, defines restrictions on the allowable contents per STC basket cell. For fuel assemblies, the basket is divided into two zones or regions, the central four cells and the remaining outer eight cells. The TS defines six loading configurations for assemblies. The sixth configuration was added to allow for assemblies with inconel grid assemblies in the active fuel zone to be loaded in any basket cell; the analyses for the other five configurations assume these assemblies are only allowed in the inner four cells. Dose rates were calculated for these configurations at 1 meter distance from the STC surfaces and the HI-TRAC surfaces. The configurations yielding bounding dose rates, configurations 3 and 4 (see Table 7.1.1 of the SAR and TS Table 3.1.2-3), were used for subsequent calculations. Configuration 4 was bounding for all surfaces of the STC. For the HI-TRAC, configuration 3 was bounding for the radial surface and configuration 4 was bounding for the top and bottom surfaces; thus, both configurations are used to determine the bounding HI-TRAC dose rates. All analyses are for intact fuel assemblies, with the fuel retaining its basic configuration for all normal, off-normal and accident conditions; only intact fuel assemblies are allowed to be transferred in the STC. The TSs provide the definition of intact fuel assemblies (see proposed operating license Appendix C, Part II, Section 1.1).

For the NFH configurations, an STC (and HI-TRAC) containing assemblies with BPRAs was found to be generally bounding for dose rates around the side and above the top. The configuration with RCCAs replacing the BPRAs in the inner four basket compartments was generally bounding for dose rates below the base. In some cases other configurations resulted in higher dose rates; however, the differences were negligible and/or were localized (e.g., at the STC steel radial ribs). Thus, dose analyses were based on loading configurations with BPRAs, or BPRAs and RCCAs as appropriate.