October 29, 2007

Mr. Michael A. Balduzzi
Sr. Vice President and COO
Entergy Nuclear Operations, Inc.
Regional Operations, NE
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3, LICENSE RENEWAL APPLICATION

Dear Mr. Balduzzi:

By letter dated April 23, 2007, as supplemented by letters dated May 3, 2007 and June 21, 2007, Entergy Nuclear Operations, Inc. submitted an application pursuant to 10 CFR Part 54, to renew the operating licenses for Indian Point Nuclear Generating Unit Nos. 2 and 3, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application (LRA) and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Mr. Robert Walpole, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1627 or e-mail KJG1@nrc.gov.

Sincerely,

/RA by Bo Pham for/
Kimberly Green, Safety Project Manager
License Renewal Branch B
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosure:
Requests for Additional Information

cc w/encl: See next page

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INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAI)

The U.S. Nuclear Regulatory Commission (NRC or staff) has reviewed the information related to the Reactor Vessel Surveillance Program and the Reactor Vessel Neutron Embrittlement Time-Limited Aging Analyses provided by the applicant in the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) LRA. The staff has identified that additional information is needed to complete the review as addressed below.

RAI B.1.1.32-1

The Reactor Vessel Surveillance Program is identified as consistent with the program described in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section XI.M31, "Reactor Vessel Surveillance," with enhancements. The enhancements are: (1) to withdraw and test a standby capsule to cover the peak reactor vessel fluence that is expected through the end of the period of extended operation; and (2) to revise procedures to require that tested and untested specimens from all capsules pulled from the reactor vessel be maintained in storage. Identify the lead factors for each standby capsule, the materials available to be tested in each capsule, and the date for capsule withdrawal to ensure that the fluence of the surveillance capsule will be equal or greater than the peak reactor vessel fluence through the end of the period of extended operation.

RAI 4.2.1-1

The Charpy Upper-Shelf Energy (USE) and Pressurized Thermal Shock analyses utilize the neutron fluence at 48 effective full power years (EFPY) to represent the neutron fluence for the reactor vessels at the end of the period of extended operation.

A) What were the EFPY achieved for each unit prior to the last refueling outage? What capacity factors and neutron flux were assumed for each unit from the last refueling outage to the end of the period of extended operation to result in 48 EFPY at the end of the period of extended operation? Explain why these capacity factors and neutron flux values are applicable for determining the neutron fluence for the reactor vessels at the end of the period of extended operation.

B) How will future capacity factors, neutron flux and neutron fluence values be monitored to ensure 48 EFPY values bound the actual conditions of the reactor vessels at the end of the period of extended operation?

RAI 4.2.2-1

Table 4.2-2 in the LRA indicates that the percentage drop in Charpy USE for plate B2803-3 is 21.3 percent at 48 EFPY. The percentage drop in Charpy USE for plate B2803-3 was determined using its surveillance data, in accordance with Position 2.2 of Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Provide the analysis that was used to determine the percentage drop in Charpy USE for plate B2803-3, include all surveillance data (unirradiated and irradiated Charpy USE and surveillance capsule neutron fluence) and references for the surveillance data.

Enclosure

RAI 4.2.2-2

According to Table 4.2-1 in the LRA, plates B2002-3 and B2003-1 in IP2 are projected to have a Charpy USE of 47.4 ft-lb and 49.8 ft-lb at 48 EFPY, respectively. This table also indicates that the material type was A302-B. Since the Charpy USE values are below 50 ft-lb, Section 4.2.2 indicates that the USE is acceptable in accordance with the Westinghouse 4-loop plant analysis described in WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation For Westinghouse Pressurized Water Reactors." WCAP-13587, Revision 1 was reviewed by the staff in a safety assessment in a letter dated April 21, 1994 to W. H. Rasin (Nuclear Management and Resources Council, NUMARC). This safety assessment indicates that licensees must confirm that the bounding plate used in the report has a lower J-R curve than any beltline material in the vessel. This safety assessment indicates that the A302-B material J-R curve methodology used in WCAP-13587, Revision 1 was unacceptable. However, the A302-B J-R curve methodology in a letter from D. B. Waters (Carolina Power and Light Company) dated December 21, 1993, includes appropriate adjustments for USE and temperature corrections and is acceptable to the staff.

A) Provide the J-R curve for the limiting IP2 plate, using corrections for material type used in the IP-2 reactor vessel, at 48 EFPY. Compare this curve to the J-R applied values in WCAP-13587, Revision 1 to demonstrate the analysis in the WCAP is bounding for the IP2 reactor vessel.

B) Table 3-1 in WCAP-13587, Revision 1 indicates the Westinghouse 4-loop plant J-R applied values are applicable for reactor vessels with a thickness of 8.5 inches and an inner radius of 86.5 inches and subject to Level A, B, C, and D conditions specified in Section 3.0 of the WCAP. Compare the wall thickness and inner radius of the IP2 reactor vessel at its beltline to the values used in the Westinghouse 4-loop plant analysis. Compare the Level A, B, C, and D conditions specified in Section 3.0 of the WCAP to the Level A, B, C and D conditions for IP2. Explain why the Westinghouse 4-loop plant analysis in WCAP-13587, Revision 1 is applicable to IP2.

RAI 4.2.5-1

A) Table 4.2-3 in the LRA indicates that the ΔRT_{NDT} value caused by irradiation for the intermediate shell axial welds and the lower shell axial welds in IP2 were determined using surveillance data reported in WCAP-15629, Revision 1, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation." This WCAP has surveillance data from IP2, IP3, and H.B. Robinson, Unit 2. The IP2 fluences were calculated using approved methodologies (WCAP-15557-R0, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," and WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves") that are based on RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, (This RG requires the use of ENDF/BVI for determining neutron cross-sections which are included in the BUGLE-96 cross-section file). In addition, there is excellent agreement between calculated and corresponding measured values. The IP3 capsule analyses also used approved methods and cross sections, thus, they are

acceptable. The H.B. Robinson calculations are reported in WCAP-14044, "Westinghouse Surveillance Capsule Neutron Fluence Re-evaluation," that was issued in 1994 (before the issuance of RG 1.190 and the availability of BUGLE-96 and ENDF/BVI). WCAP-15629, Revision 1 indicates that 15% was added to the values reported in WCAP-14044. Explain why 15% was added to the values reported in WCAP-14044. Provide neutron fluence values derived using a methodology that adheres to the guidance in RG 1.190. If the revised analysis results in a change in neutron fluence for the H.B. Robinson, Unit 2 surveillance capsules, provide the ΔRT_{NDT} value caused by irradiation and the RT_{PTS} value for the intermediate shell axial welds and the lower shell axial welds in IP2 and provide the surveillance data analysis required by 10 CFR 50.61(c)(2)(i).

B) Table 4.2-4 in the LRA indicates that the ΔRT_{NDT} caused by irradiation for the lower shell plate B2803-3 in IP3 was determined using surveillance data reported by the licensee's response to Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity." This surveillance data was reported in Attachment I to a September 4, 1998, letter from J. Knubel (New York Power Authority). As discussed in RAI 4.2.5-1A, the surveillance data from IP3 is also reported in WCAP-15629, Revision 1. The neutron fluence values for the IP3 surveillance capsule that are reported in WCAP-15629, Revision 1 and in the September 4, 1998, letter have different values. The applicant is requested to revise the PTS analyses using neutron fluence values for the surveillance capsules that are determined using the guidance in RG 1.190 and to provide the surveillance data analysis required by 10 CFR 50.61(c)(2)(I).

RAI 4.2.5-2

10 CFR 50.61(b)(4) indicates that each pressurized water nuclear power reactor for which the analysis required by PTS rule indicates that if there is no reasonably practicable flux reduction program to prevent the RT_{PTS} value from exceeding the PTS screening criteria based on the neutron fluence at the expiration date of the operating license, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criteria is allowed. The analysis must be submitted at least three years before the RT_{PTS} value is projected to exceed the PTS screening criteria.

Section 4.2.5 in the LRA indicates that the RT_{PTS} value for plate B2803-3 in IP3 will exceed the PTS screening criterion. Identify the flux reduction program initiated by the applicant to prevent the RT_{PTS} value for plate B2803-3 in IP3 from exceeding the PTS screening criterion. Based on the information provided in response to RAI 4.2.5-1(B) and RAI 4.2.1-1, identify when the RT_{PTS} value for plate B2803-3 in IP3 is projected to exceed the PTS screening criterion

Letter to MBalduzzi from KGreen, dated October 29, 2007

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