ENT000661
Submitted: August 10, 2015

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

July 26, 2013

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3, LICENSE RENEWAL APPLICATION, SET 2013-04 (TAC NOS. MD5407 AND MD5408)

Dear Sir or Madam:

By letter dated April 23, 2007, as supplemented by letters dated May 3, 2007, and June 21, 2007, Entergy Nuclear Operations, Inc. (Entergy), submitted an application pursuant to Title 10 of the *Code of Federal Regulations* 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 documented its findings in the Safety Evaluation Report (SER) related to the license renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, which was issued August 2009 and supplemented August 30, 2011 (SER Supplement 1).

By letter dated July 14, 2010, Entergy amended its license renewal application (LRA) and submitted the Indian Point Nuclear Generating Unit Nos. 2 and 3 Reactor Vessel Internals Program. By letter dated September 28, 2011, as supplemented by letter dated February 17, 2012, Entergy submitted the "Indian Point Energy Center Reactor Vessel Internals Inspection Plan" (RVI Inspection Plan). The RVI Inspection Plan, as supplemented, was intended to be consistent with the Nuclear Regulatory Commission (NRC)-approved Electric Power Research Institute (EPRI) topical report, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (MRP-227-A), and to partially fulfill Commitment 30 associated with the IP LRA.

By letter dated May 15, 2012, the staff requested additional information regarding the RVI Program and RVI Inspection Plan. Entergy provided a partial response to RAI 11 and 12 in a letter dated June 14, 2012. The staff issued follow-up RAI 11-A and 15-A by letter dated February 6, 2013. Entergy responded to the staff's RAI by letter dated May 7, 2013. Separately, by letter dated November 20, 2012, Entergy provided its response to the staff's RAI 16. Based on its review of these responses, the staff has determined it requires additional information related to the Entergy's responses to RAI 11-A, 15-A, and 16, as detailed in the enclosure.

This RAI was discussed with Mr. Roger Waters, and a mutually agreeable date for Entergy's response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1627, or by e-mail at Kimberly.Green@nrc.gov

Sincerely,

Kimberly Green

Kimberly Green, Sr. Mechanical Engineer
Aging Management of Plant Systems Branch
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosure:
As stated

cc: Listserv

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/RA/

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As stated

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ADAMS Accession No.: ML13204A199 *concurred via email

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DATE	07/25/13	07/26/13	07/26/13	07/26/13

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REQUEST FOR ADDITIONAL INFORMATION, SET 2013-04
RELATED TO INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION

REACTOR VESSEL INTERNALS PROGRAM AND INSPECTION PLAN
DOCKET NOS. 50-247 AND 50-286

RAI 11-B

The response to RAI 11-A, by letter dated May 7, 2013 (Ref. 1), describes the functionality analysis approach for the evaluation of the IP2 and IP3 lower support columns in support of Applicant/Licensee Action Item 7 from MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines."

1) The response states, in part, that based on the lack of any documented history of fracture in the lower core support columns, it will be assumed that only a limited number of columns could actually contain flaws of significant size. Provide a more detailed basis for the number of columns that will be assumed to contain flaws, including a description of any relevant operating experience or research supporting the assumed incidence of cracking in the columns. The basis for the number of cracked columns should address flaws due to any screened-in aging mechanism for the columns, in addition to fabrication defects.

2) The response states, in part, that since the effects of embrittlement are only significant in the presence of pre-existing flaws (e.g. from the casting process) and tensile stresses capable of propagating these flaws, the screening analysis will identify regions of individual columns where thermal and irradiation effects could give rise to embrittled materials and would also be subjected to significant tensile stresses under design and service loadings. Define what is meant by "significant tensile stresses" – is there a specific numerical value of stress considered to be a threshold of significance?

3) Provide a general description of the fabrication of the IP2 and IP3 lower support columns, including:

    a. the grade of cast stainless steel used (e.g. CF-8)

    b. the approximate location relative to the lower core plate of the weld joining the upper (cast) portion of the column (the column cap) to the lower portion of the column.

4) Provide a summary of the most recent American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI Inservice Inspection of the lower support columns at IP2 and IP3, including the dates of the inspections, coverage obtained (including a specific description of the coverage limitations on the columns), and the size, location and orientation of any recordable or rejectable indications.

5) MRP-227-A, Section 4.2.7, requires the plant-specific analysis for Applicant/Licensee Action Item 7 demonstrating that the lower support column bodies (expansion components) will maintain their functionality during the period of extended operation to be submitted along with an applicant/licensee's submittal to apply the approved version

of MRP-227. This analysis was not provided with the applicant's submittal of the Reactor Vessel Internals (RVI) Inspection Plan for IP2 and IP3. Entergy later made a commitment to submit the analyses prior to the start of the period of extended operation (PEO) for both units.

However, Entergy's May 7, 2013, letter proposed a revision to Commitment 47 changing the date for the submittal of the analysis for IP2 until March 1, 2015. A delay of this nature would jeopardize satisfactory completion of the staff's review of the analysis prior to the refueling outage in 2016 when the initial inspections of the MPR-227-A primary components are scheduled for IP2. The staff estimates that it will need at least 18 months to review the analysis once it is submitted. The staff would expect applicants/licensees (of Westinghouse plants) to inspect the lower support column bodies during the initial inspections if a plant-specific analysis showed that the expansion components could not maintain their intended function during the PEO, or if the staff could not review and approve the analysis prior to the initial inspections of the primary components.

In the absence of an NRC-approved plant-specific analysis for the lower support column bodies, please explain how these components will maintain their intended function during the PEO.

RAI 15-B

The revised response to RAI 15, provided in Reference 1, states that the term "Class 1" was inadvertently included in the response to RAI 12, and that the phrase "ASME Code Class 1 fatigue evaluations for reactor vessel internals" is changed to read "ASME Code Subsection NG fatigue evaluation for reactor vessel internals." However, the markups to License Renewal Application (LRA) Sections A.2.2.2.1 and A.3.2.2.1 containing the proposed content for the Updated Final Safety Analysis Report (UFSAR) supplement related to metal fatigue list the reactor vessel internals fatigue time-limited aging analysis under "Class 1 Metal Fatigue." The staff requests that Entergy correct this apparent inconsistency in LRA Sections A.2.2.2.1 and A.3.2.2.1. The staff also requests that Entergy add the commitment to complete the revised fatigue cumulative usage factor analyses accounting for environmental effects (Commitment 49 from the May 7, 2013 letter) to LRA Sections A.2.2.2.1 and A.3.2.2.1.

RAI 16-A

The response to RAI 16, by letter dated November 20, 2012 (Ref. 2), addressed the remaining life prediction for the IP2 split pins and provided the estimated replacement schedule for the split pins. Also in the response to RAI 16, Entergy stated that if the [split pin] replacement is not implemented as currently scheduled in 2016, it will provide the NRC staff with a detailed inspection plan, including inspection methods, inspection coverage, and inspection frequency, by March 2015. The staff requests that Entergy add a commitment to provide the NRC staff with a detailed inspection plan for the IP2 split pins, including inspection methods, inspection coverage, and inspection frequency, by March 2015, if the planned replacement of the IP2 split pins is not to be implemented in 2016. LRA Sections A.2.1.41 and A.3.1.41 containing the proposed UFSAR supplement content for the IP2 and IP3 Reactor Vessel Internals Aging Management Activities should be revised to include the new commitment.

RAI 17

Appendix A to MRP-227-A indicates that failures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. A recent metallurgical analysis of bolts removed from this plant confirmed that the bolts cracked due to primary water stress corrosion cracking (PWSCC). Appendix A to MRP-227-A indicates that most of the failures of Alloy X-750 material have occurred in material with heat treatment condition AH1, while Alloy X-750 given the high temperature heat treatment (HTH) has proved more resistant to PWSCC.

The only aging mechanism requiring management by MRP-227-A for the clevis insert bolts is wear. The clevis insert bolts are categorized as an "Existing Programs" component under MRP-227-A, with the ASME Code, Section XI Inservice Inspection program credited for managing aging due to wear only. The ASME Code, Section XI specifies a VT-3 visual inspection for the clevis insert bolts which may not be adequate to detect cracking before it results in bolt failure.

The staff requests that Entergy modify the MRP-227-A inspection requirement for the clevis insert bolts as necessary to manage the effects of PWSCC for the IP2 and IP3 bolts. If the inspection requirement is not modified, the staff requests that Entergy provide a technical justification for the adequacy of the existing inspection requirement to manage PWSCC.

References:

1) Indian Point Nuclear Generating, Units 2 & 3 - Reply to Request for Additional Information Regarding the License Renewal Application, May 7, 2013, (ADAMS Accession No. ML13142A202).

2) Indian Point, Units 2 and 3 - Reply to Request for Additional Information Regarding the License Renewal Application, November 20, 2012, (ADAMS Accession No. ML12340A154).

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1 AH = Hot rolled "equalized" at 1625 °F (885 °C) followed by 20 hours at 1300 °F (704 °C)