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

May 15, 2012

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

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 related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 which was issued in August 2009 and supplemented on August 30, 2011 (SER Supplement 1). Since the issuance of the Safety Evaluation Report and Supplement 1 thereto, the staff has identified the need for additional information with respect to certain aging management programs based on lessons learned from past license renewal applications (LRAs) and recent industry operating experience. Additionally, the staff has identified issues that need additional clarification for the LRA. Therefore, the staff requests additional information as described in the enclosure.

The staff's planned issuance of this request for additional information was discussed with Mr. Robert Walpole, 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 301-415-3733, or via e-mail Robert.Kuntz@nrc.gov.

Sincerely,

Robert F. Kuntz, Senior Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286
Enclosure:
As stated
cc w/encl: Listserv

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION

RAI's Related to License Renewal Application Amendment No. 9 (Ref. 1)

RAI 1

On page 3 of license renewal application (LRA) Amendment 9 (Ref. 1), it is stated that Table 2.3.1-2-IP2 and Table 2.3.1-2-IP3 list the mechanical components subject to aging management review and component intended functions for the reactor vessel internals. However, Table 2.3.1-2-IP3 (the table for Indian Point Nuclear Generating Unit No. 3 (IP3)), is missing, and the table for Indian Point Nuclear Generating Unit No. 2 (IP2) listing the components subject to aging management review is numbered Table 2.3.1-4-IP2. Provide Table 2.3.1-2-IP3 and correct the numbering of the table for IP2.

RAI 2

LRA Sections 3.1.2.2.6, 3.1.2.2.9, 3.1.2.2.15, and 3.1.2.2.17, provided in LRA Amendment 9 refer to MRP-227. For consistency with the revised LRA Section B.1.42 submitted by letter dated February 17, 2012, the staff requests that the applicant revise the LRA sections listed above to update the reference to MRP-227-A.

RAI 3

The applicant addressed the further evaluation criteria in Section 3.1.2.2.12 of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Rev. 1 (SRP-LR) by stating (in the "Discussion" column of Table 3.1.1 Item 3.1.1-30) that cracking will be managed by the Water Chemistry Control Program (Primary and Secondary) and either the Reactor Vessel Internals (RVI) Program or the Inservice Inspection (ISI) Program. Crediting the ISI Program for managing cracking is inconsistent with LRA Tables 3.1.2-2-IP2 and 3.1.2-2-IP3, in which the components aligned with Table 3.1.1 Item 3.1.1-30 only credit the Water Chemistry Control – Primary and Secondary Program and the RVI Program for aging management. Further, LRA Amendment 9 does not include a revised LRA Section 3.1.2.2.12. In addition, the use of the Inservice Inspection Program (ISI) Aging Management Program (AMP) is not consistent with the NUREG-1801, "Generic Aging Lessons Learned Report", Revision 1 (GALL Report, Rev. 1), Table 1, Item 30 for this line item or the recommendations of SRP-LR Section 3.1.2.2.12.

The staff therefore requests the following information:

1. Correct the inconsistency between Table 3.1.1 Item 3.1.1-30 and the associated line items in Tables 3.1.2-2-IP2 and 3.1.2-2-IP3.

2. Provide a markup to LRA Section 3.1.2.2.12 consistent with the changes in LRA Table 3.1.1 provided in LRA Amendment 9.

3. If the ISI Program is being used as the AMP to manage cracking for certain RVI components aligned with Table 3.1.1 Item 3.1.1-30, justify the use of the ISI Program rather than the RVI Program for managing aging of the affected components, and make

all the necessary conforming changes to Table 3.1.1, Table 3.1.2-2-IP2, and Table 3.1.2-2-IP3.

RAI's Related to Reactor Vessel Internals Program

RAI 4

NUREG-1801, "Generic Aging Lessons Learned Report," Revision 2 (GALL Report, Rev. 2), Section XI.M16A, recommends, under the "Monitoring and Trending" program element, using the methods of the latest Nuclear Regulatory Commission (NRC)-approved version of Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227, Section 6 for monitoring, recording, evaluating and trending the data from the program inspection results. MRP-227 Section 6 includes recommendations for flaw depth sizing and crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications.

However, in the staff's final safety evaluation (SE) on MRP-227, Revision 0 (Ref. 2), the staff noted that in a request for additional information (RAI) response, Electric Power Research Institute (EPRI) stated that topical report WCAP-17096-NP is the document that will be used as the framework to develop those generic and plant-specific evaluations triggered by findings in the RVI examinations, and observed that the NRC staff is currently reviewing WCAP-17096-NP, Revision 2. Therefore, the staff requests that the applicant clarify whether the Indian Point Energy Center (IPEC) RVI Program will use the guidance of WCAP-17096-NP, Rev. 2 (Ref. 3) for evaluating the acceptability of relevant conditions found by the inspections conducted under the RVI Inspection Plan.

RAI 5

For baffle-former bolts, MRP-227-A, Table 5-3 states that the examination acceptance criteria for the ultrasonic test (UT) shall be established as part of the examination technical justification. "Materials Reliability Program: Inspection Standard for PWR Internals," (MRP-228)(Ref. 4) provides additional guidance on preparation of technical justifications (TJs). However, the IPEC RVI Program does not indicate whether a TJ has been or will be developed for the baffle-former bolts. Therefore, the staff requests the applicant submit a TJ for the IP2 and IP3 baffle-former bolts.

RAI's Related to Reactor Vessel Internals Inspection Plan (Ref. 6)

RAI 6

Applicant/Licensee Action Item 1 from the staff's final SE on MRP-227, Revision 0 requires that applicants/licensees submit an evaluation that demonstrates that their plant is bounded by the assumptions regarding plant design and operating history that were made in the failure modes, effects and consequences analyses (FMECA) and functionality analyses for reactors of their design.

The applicant's response to Applicant/Licensee Action Item 1 in the RVI inspection plan addresses the core loading assumptions (switch to a low-leakage core) and operational (base loaded plant) aspects of design and operation that are mentioned in MRP-227-A, Section 2.4.

An additional assumption listed in Section 2.4 of MRP-227-A is that there have been no design changes to the RVI beyond those identified in general industry guidance or recommended by the original vendors. Section 2.4 of MRP-227-A indicated that these assumptions are considered to conservatively represent any U.S. Pressurized Water Reactor operating plant provided that these three assumptions are met, given the information on design and operation known to the MRP as of May 2007.

MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," documents the screening for susceptibility to aging effects, the FMECA results, and the categorization and ranking of the RVI components. In addition to the assumptions listed in Section 2.4 of MRP-227-A, MRP-191 documents additional assumptions that were used. In particular, neutron fluence range, temperature, and material grade for each generic component of the Westinghouse design internals were used for input to the screening process. These values were determined based on an "expert elicitation" process. Stress values were not explicitly tabulated, but were recorded as either above the stress threshold (>30 ksi) or not based on the expert interviews.

MRP-232, Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals," reported more specific stress, temperature and neutron fluence values based on finite element analyses for selected high consequence of failure components identified in MRP-191.

MRP-227-A did not verify that the values of fluence, temperature, stress, and material, documented in MRP-191 and MRP-232 were bounding for all individual plants, and in fact MRP-227-A states, "These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter."

Each plant should have access to design information enabling verification that the material for each RVI component is bounded by the design assumptions of the MRP. In this context, the staff requests the following information:

1) To provide reasonable assurance that the RVI components are bounded by assumptions in the FMECA and functionality analyses supporting the development of MRP-227-A, the applicant is requested to respond to either 2.a or 2.b of this RAI:

2.a)

Provide the plant-specific values of neutron fluence (n/cm^2, E>1.0 MeV), temperature, stress, and materials for a sample of RVI components. The components selected should represent a range of neutron fluences, and temperatures. This information should identify whether the stress is greater or less than 30 ksi. Values of neutron fluence and temperature may be estimated or analytical values. The values should be the peak values of each parameter for each component (e.g., peak end-of-life value for fluence). Provide the method used to estimate the values, or describe the analysis method. An acceptable sample of components is:

i) Lower Core Plate
ii) Core Barrel Flange
iii) Barrel-Former Bolts

iv) Upper Core Barrel Welds
v) Lower Core Barrel Welds
vi) Upper Core Plate Alignment Pins

2.b) If the sample verification approach in Part (a) is not used, describe the process used to verify that the RVI components at IP2 and IP3 are bounded by the assumptions regarding the neutron fluence, temperature, stress values, and materials that were made for each component in the FMECA and functionality analyses supporting the development of MRP-227-A.

3) If there are any components at IP2 or IP3 not bounded by assumptions regarding neutron fluence, temperature, stress or material used in the development of MRP-227-A, describe how the differences were addressed in the plant-specific RVI Inspection Plan. The staff requests that the applicant, as a part of its demonstration, discuss whether there would be any changes to the screening, categorization, FMECA process and functionality analyses if the plant-specific variables (the neutron fluence, temperature, stress values, plant-specific operating experience, and materials) are used. This evaluation should address whether additional aging mechanisms would become applicable to the component.

4) For any non-bounded components, determine if any changes to the inspection requirements of MRP-227-A are needed. Provide plant-specific inspection requirements or an alternate aging management program, as appropriate. If no changes to the inspection requirements are proposed, provide a justification for the adequacy of the existing MRP-227-A inspections for the unbounded components.

5) Identify all design changes to the IP2 and IP3 RVI, and describe (1) if any of these are beyond those identified in general industry guidance or recommended by the original vendors, and (2) if any of the design changes were implemented after May 2007. Assess the impact of these design changes on the recommendations of the RVI Inspection Plan. Provide plant-specific inspection requirements if necessary for the affected components.

RAI 7

The staff reviewed the applicant's response to Applicant/Licensee Action Item 2 from the NRC staff's final SE on MRP-227, Revision 0. In Section 3.6 of the RVI Inspection Plan (Ref. 5), the applicant stated that it reviewed the information in Table 4-4 of MRP-191 and determined that this table contains all the RVI components that are within the scope of license renewal and that this is shown in Table 5-7. The staff notes that Table 5-1 contains a cross-index between the component designations in Entergy Letter NL-10-063 (Amendment 9 to the LRA, Ref. 1) and the component names as designated in MRP-191, Table 4-4 (Ref. 6). All the IPEC component designations correlate with an equivalent component designation in MRP-191 (Ref. 7), Table 44 with the exception of the Lower Internals Assembly – Column Cap.

The staff therefore requests that the applicant verify that the Lower Internals Assembly – Column Cap would be subject to the same inspection requirements that are applied to the lower support assembly, lower support column bodies (cast) in MRP-227-A, Table 4-6. If not, provide plant-specific aging management requirements for the Lower Internals Assembly – Column Cap.

RAI 8

The staff requests the following information related to the applicant's response to Applicant/Licensee Action Item 3 from the NRC staff's final SE on MRP-227, Revision 0.

1. Provide more detail on the operating experience for cold-worked type 316 split pins to support the prediction that split pins of this material will last until the end of the period of extended operation (PEO) for IP3.

2. Describe the inspection schedule, methods, and basis for replacement split pins at IP3. If no inspections are planned, provide a justification for not inspecting the split pins.

3. Describe the criteria for the replacement split pin material and design for IP2.

4. Describe the inspection strategy for the replacement IP2 split pins during the PEO.

RAI 9

The applicant's response to Applicant/Licensee Action Item 5 from Revision 1 of the staff's final SE on MRP-227, states in part that the acceptance criteria will ensure the remaining compressible height of the spring shall provide hold down forces within the IPEC design tolerance. If a plant specific acceptance criterion is not developed for the hold down spring, IPEC will replace the spring in lieu of performing the first required physical measurement.

MRP-227-A, Table 4-3, calls for direct measurement of the hold-down spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years

The staff requires clarification of how the applicant will determine whether the first set of measurements could be extrapolated to demonstrate acceptable spring functionality through 60 years. Therefore, the staff requests the following information:

1. Provide the specific acceptance criteria for spring height and/or hold down force from the IP2/IP3 licensing basis.

2. Describe the procedure by which the remaining hold down forces will be projected to end-of-life based on one measurement. Address whether the decrease in spring height or hold-down force is assumed to occur linearly over time or via some other function of time.

3. What results of the first spring measurements would indicate a need for successive measurements?

RAI 10

The applicant's response to Applicant/Licensee Action Item 7 indicates that the plant-specific analysis to demonstrate functionality of the lower support column bodies during the period of extended operation will be submitted to the NRC prior to the PEO. In the aging management review tables submitted in LRA Amendment 9, the applicant credits the "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program" for managing loss of fracture toughness of the lower core support column bodies, as well as several other CASS components. NUREG-1930 indicates that the staff determined this program was consistent with the Generic Aging Lessons Learned Report, Revision 1, Aging Management Program (AMP) XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program." Per GALL, Rev. 1, Section XI.M13, the "Thermal Aging and Neutron Irradiation Embrittlement of CASS Program" generally requires supplemental visual inspections (equivalent to an EVT-1) for CASS RVI components that are either susceptible to thermal aging based on chemistry and other manufacturing parameters, or receive a neutron fluence $\geq 1x10^{17}$ n/cm^2, unless it can be demonstrated that the stresses on the component are either compressive or low in magnitude if tensile. The RVI Program is credited with managing cracking of the core support column bodies and other CASS components. Under the RVI Program, the core support column bodies are expansion components that would be subject to an EVT-1 visual examination for cracking due to irradiation assisted stress corrosion cracking if cracking were found in the associated primary component.

The staff requests the following information:

Since both the plant-specific analysis and Thermal Aging and Neutron Irradiation Embrittlement of CASS Program could both potentially involve screening for thermal or neutron irradiation embrittlement, stress analyses, and flaw tolerance evaluations, and both the RVI Program and Thermal Aging and Neutron Irradiation Embrittlement of CASS Program could potentially require inspections, discuss the relationship of the two programs and the plant-specific analysis.

RAI 11

In response to Applicant/Licensee Action Item 7, the applicant stated that the plant-specific analyses to demonstrate the lower support column bodies will maintain their functionality during the period of extended operation will consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement. The analyses will be consistent with the IP2/IP3 licensing basis and the need to maintain the functionality of the lower support column bodies under all licensing basis conditions of operations.

The staff requests the following additional information:

1) Section 3.3.7 of Revision 1 of the staff's final SE on MRP-227, Revision 0 lists three possible options for the type of plant-specific analysis used to fulfill the requirements of this action item. The three approaches are 1) functionality analyses of the set of like components, 2) component-specific flaw tolerance evaluations, or 3) a screening approach demonstrating that the CASS Components are not susceptible to thermal embrittlement, neutron embrittlement, or the combined effects of both. Discuss which of these approaches will be used and why.

2) Describe the acceptance criteria for the plant-specific analysis results that are derived from the IP2/IP3 licensing basis.

3) Since the applicant stated that the analysis of the core support columns will be submitted prior to the period of extended operation for IP2 and IP3, the staff requests the applicant submit a letter documenting this as a formal licensing commitment.

RAI 12

Background

In its letter dated February 17, 2012, the applicant provided the response to Applicant/Licensee Action Item 8 of the Staff SE of MRP-227-A. The applicant stated that the RVI AMP description has been revised to be consistent with MRP-227-A, and the applicant's response to Applicant/Licensee Action Item 8 does not request any deviations from the guidance provided in MRP-227-A. The staff noted that Applicant/Licensee Action Item 8 also addresses cumulative usage factor (CUF) analyses that are time-limited aging analyses (TLAAs).

The applicant's response does not address LRA Section 4.3.1.2, which provides the applicant's TLAA and associated CUF values for the IP2 and IP3 RVI. The staff noted that in Amendment 3 to the LRA dated March 24, 2008, (ADAMS Accession No. ML081070255), the applicant amended LRA Section 4.3.1.2 to state that "fatigue on the reactor vessel internals will be managed by the Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii) for both IP2 and IP3."

Issue

The staff noted that Applicant/Licensee Action Item 8 indicates that RVI Program may be used as the basis for accepting CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking during the period of extended operation. Applicant/Licensee Action Item 8 also indicates that the Fatigue Monitoring Program may be used as the basis for accepting CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii), in which case the evaluation requirements of ASME Code Section III, Section NG are to be satisfied.

It is not clear to the staff whether the applicant will use (a) its RVI Program, (b) its Fatigue Monitoring Program, or (c) a combination of both programs to manage RVI fatigue during the period of extended operation.

Request

Identify the aging management program that is used to manage fatigue of the reactor vessel internals:

1) If the RVI Program will be used:
    a. Verify that each RVI component with a CUF value will be periodically inspected for fatigue-induced cracking during the period of extended operation.
    b. For each component to be inspected for fatigue-induced cracking:
        i. Identify the examination method(s).

ii. Provide the inspection periodicity, including the initial inspection timing and timing of subsequent examinations.

iii. Justify that the periodicity of the inspections for each RVI component is adequate.

2) If the Fatigue Monitoring Program will be used, verify that the requirements of ASME Code Section III, Subsections NG-2160 and NG-3121, as delineated in Applicant/Licensee Action Item 8, will be satisfied.

References

1. Letter from Fred Dacimo, Entergy, to NRC dated July 14, 2010, Subject: Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64 (ADAMS Accession No. ML102010102)

2. Letter from Robert Nelson, NRC, to Neil Wilmshurst, EPRI dated December 16, 2011; Subject: Revision 1 of the Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680) (ADAMS Accession No. ML11308A770)

3. Reactor Internals Acceptance Criteria Methodology and Data Requirements, WCAP-17096-NP, Rev. 2, Westinghouse Non-Proprietary Class 3 Report, December 2009, ADAMS Accession No. ML1014601570

4. Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228) 1016609 Final Report, July 2009 Electric Power Research Institute, Palo Alto, CA (EPRI Product No. 1016609) (ADAMS Accession No. ML092120573)

5. Indian Point Energy Center Revised Reactor Vessel Internals Inspection Plan Compliant with MRP-227-A. Attachment 2 to Entergy Letter NL-12-037, Letter from Fred Dacimo to NRC dated February 17, 2012, Subject: License Renewal Application - Revised Reactor Vessel Internals Program and Inspection Plan Compliant with MRP-227-A, Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286–License Nos. DPR-26 and DPR-64 (ADAMS Accession No. ML1206A312)

6. MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," ADAMS Accession No. ML091910130

7. NUREG-1930, Volume 2, "Safety Evaluation Report Related to The License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Dockets No. 50-247 and 50-286, November 30, 2009 (ADAMS Accession No. ML093170671)

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

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The staff's planned issuance of this request for additional information was discussed with Mr. Robert Walpole, 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 301-415-3733, or via e-mail Robert.Kuntz@nrc.gov.

Sincerely,

/RA/
Robert F. Kuntz, Senior Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosure:
As stated

cc w/encl: Listserv

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