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

February 6, 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 2013-01 (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 or the applicant), 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).

As part of its review of Entergy's "Indian Point Energy Center Reactor Vessel Internals Inspection Plan," the staff has identified the need for additional information with respect to this plan and the associated Reactor Vessel Internals Inspection Program as described in the enclosure.

This request for additional information was discussed with Mr. Roger Waters, and a mutually agreeable date for Entergy's response is within 90 days from the date of this letter. If you have any questions, please contact me at 301-415-3873, or by e-mail at John.Daily@nrc.gov.

Sincerely,

John Daily

John Daily, Sr. 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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John Daily, Sr. 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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REQUESTS FOR ADDITIONAL INFORMATION, SET 2013-01
RELATED TO INDIAN POINT NUCLEAR GENERATING, UNITS 2 AND 3,
LICENSE RENEWAL APPLICATION

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

(Set 2013-01)

RAI 11a (RVI Program and combined effects of embrittlement)

Background
In request for additional information (RAI) 11, the staff requested additional information on the approach to be used for the plant-specific evaluation of the cast austenitic stainless steel (CASS) lower support column bodies. The applicant's response indicates it plans to use a screening approach using the screening criteria for thermal aging embrittlement susceptibility from the Nuclear Regulatory Commission (NRC) staff's May 19, 2000, letter (Reference 1). The applicant provided a table of the screening criteria based on chemistry, casting method, and delta ferrite content identical to Table 2 of Reference 1.

Issue
However, Reference 1 also recommends that to account for a potential synergistic effect on loss of fracture toughness due to the combined effects of thermal embrittlement (TE) and neutron irradiation embrittlement (IE), a component-specific assessment should be performed for components that will experience neutron fluence of $1x10^{17}$ neutrons per square centimeter (n/cm^2) or greater. Supplemental inspections would be recommended for those components that are potentially susceptible to TE and IE, that are also subject to significant tensile loadings under any normal operating or design basis condition. Per Table 4-6 of MRP-191 (Reference 2), the screening value of the neutron fluence for the lower support column bodies for Westinghouse-design reactor vessel internals (RVI) is $1x10^{22}$ to $5x10^{22}$ n/cm^2. This is significantly greater than the $1x10^{17}$ n/cm^2 threshold value provided in Reference 1.

Request
Describe how the effects of neutron fluence, with respect to a potential synergistic effect of TE and IE, will be addressed in the plant-specific evaluation of the lower support column bodies. The applicant should propose modifications of the aging management requirements for the lower support column bodies as necessary to address the concern with a potential synergistic effect.

The staff notes that for CASS, different neutron fluence thresholds, above which the synergistic effect of TE and IE must be considered, have been proposed in various industry documents. If a threshold value greater than $1x10^{17}$ n/cm^2 is used in the applicant's evaluation of the potential synergistic effect, a technical justification should be included for the threshold value chosen. The technical justification should include a description of the material test data used as the basis for the threshold including material type(s), thermal aging time and temperature, neutron fluence, type of reactor in which the irradiation was conducted, and relevant mechanical testing results.

ENCLOSURE

RAI 15a (RVI program and Fatigue)

Background

In its response to RAI 15, Question 1, by letter dated October 17, 2012 (Reference 3), the applicant revised its response to RAI 12 to indicate that it intends to use the RVI Program to manage the cracking-fatigue aging effect for RVI components that have a time-limited aging analysis (TLAA) that determined a cumulative usage factor (CUF). The applicant provided a list of the RVI components that have a CUF analysis, a table cross-indexing these components with the equivalent component name in MRP-227-A, along with the inspection requirements, and a justification for each component with a CUF that the inspection requirements are adequate to manage the cumulative fatigue damage aging effect.

Part 5 of Action Item 8 of the staff's final safety evaluation (SE) of MRP-227-A contains two requirements that must be fulfilled by licensees that intend to use the RVI Program to manage the cracking-fatigue aging effect for components with a TLAA for fatigue:

1. For those CUF analyses that are TLAAs, the applicant may use the pressurized-water reactor (PWR) Vessel Internals Program as the basis for accepting these 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 in the components during the period of extended operation.
2. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA.

Many of the RVI components with TLAA analyses for both Indian Point (IP) Units 2 and 3 (IP2 and IP3) are either "existing programs" or "no additional measures" components under MRP-227-A, which are inspected under the ASME Section XI, Inservice Inspection Program and are thus only subject to a VT-3 visual examination. Those components categorized as "expansion" may or may not be inspected under the RVI Program based on the findings of the RVI Inspection Program examinations of the linked "primary" component(s). Additionally, a VT-3 visual examination may not be adequate for all components for detecting fatigue cracking prior to the occurrence of structurally significant cracking, although the staff notes that VT-3 examination is used for some components that were determined to be primary components for fatigue (such as thermal shield flexures and baffle-edge bolts).

In general, a justification for the inspection periodicity was not provided in the response to RAI 15. The default inspection periodicity for most "primary" inspection category components in MRP-227-A is every ten years following the initial inspection.

All of the CUFs for RVI components provided in Tables 4.3-5 and 4.3-6 of the IP2 & IP3 license renewal application (LRA) are less than 1.0. However, these CUFs were determined without the application of an environmental correction factor (F_{en}) to account for the effects of the reactor coolant environment. However, it is reasonable to conclude that if the reactor coolant environment affects the fatigue usage of other components in the reactor coolant system and reactor pressure vessel, then it would affect the RVI components similarly. The F_{en} for the reactor pressure vessel (RPV) components and reactor coolant system given in Section 4.3.4 of the LRA range from 2.45 to 15.35. Application of F_{en} in this range could cause the CUF of some RVI components to exceed 1.0. This would affect the required periodicity of inspection. For a very high environmentally-adjusted CUF, even a 10-year inspection interval may not be adequate.

Issue

1. Most of the RVI components with a fatigue TLAA analysis are not "Primary" inspection category components under the RVI Program, thus may be subject to no inspection other than a VT-3 visual examination under the ASME Section XI, Inservice Inspection Program, since "Expansion" category component inspections are only triggered in the event of degradation of the linked "Primary" inspection category component(s).
2. The licensee did not justify the adequacy of the periodicity of the RVI Program inspections performed on RVI components that have fatigue TLAA analyses.
3. The staff considers the inspection techniques required by MRP-227-A for components in the "Primary," "Expansion," or "Existing Programs" categories, for which fatigue is a screened-in aging mechanism, adequate to detect cracking due to fatigue if the RVI Program is credited for managing a fatigue TLAA. However, those components that fall into the "No Additional Measures" category under MRP-227-A have no specified examination techniques, periodicity, coverage, and acceptance criteria in MRP-227-A.

Requested Information

1. For those RVI components having fatigue TLAA analyses for which the cumulative fatigue damage aging effect is to be managed by the RVI Inspection Program, but which are classified as "Expansion," "Existing Programs," or "No Additional Measures" inspection category components, provide a modification to the RVI Inspection Program to re-categorize these components as "Primary" inspection category components. If any such components are to remain in the "Expansion" category, provide a technical justification for potentially never inspecting these components. Discuss your plans to re-categorize these components as "Primary" inspection category components.
2. For those RVI components having fatigue TLAA analyses for which the cumulative fatigue damage aging effect is to be managed by the RVI Inspection Program, provide a quantitative justification that the periodicity of inspections for fatigue is adequate, either in terms of the calculated CUF (considering the effects of the environment on the CUF analysis), or by using a flaw tolerance approach.
3. For those RVI components having fatigue TLAA analyses for which the cumulative fatigue damage aging effect is to be managed by the RVI Inspection Program and which are classified as "No Additional Measures" components under MRP-227-A, identify the examination technique, coverage, periodicity, and acceptance criteria (i.e., provide the equivalent information to that provided in Tables 4-3 and 5-3 of MRP-227-A). In addition, provide the information requested in Parts 1 and 2 of this RAI for these components.

References

1. U.S. Nuclear Regulatory Commission Letter, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000 (NRC ADAMS Accession No. ML003717179)
2. MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," ADAMS Accession Number ML091910130
3. Letter from F. Dacimo to NRC dated October 17, 2012, Subject: Indian Point Nuclear Generating Unit Nos. 2 & 3 - Reply to Request for Additional Information Regarding the License Renewal Application. (ADAMS Accession No. ML12300A391)

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3, LICENSE RENEWAL APPLICATION, SET 2012-03 (TAC NOS. MD5407 AND MD5408)

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