Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 254-2055

Fred Dacimo
Vice President
Operations License Renewal

NL-12-134

September 28, 2012

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

SUBJECT: Reply to Request for Additional Information Regarding the License Renewal Application
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCE:
1. NRC letter, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application," dated May 15, 2012
2. Entergy letter (NL-12-089)," Reply to Request for Additional Information Regarding the License Renewal Application," dated June 14, 2012

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment 1, a reply to the additional information requested in Reference 1 pertaining to NRC review of the License Renewal Application (LRA) for Indian Point 2 and Indian Point 3. The reply provided in this transmittal addresses RAIs 6, 7, 9, 10 and 11 on the Reactor Vessel Internals (RVI) Program. Responses to RAIs 1-5, 8, and 12 were provided in Reference 2.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-254-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 28, 2012

Sincerely,

Patrick W. Conroy acting for Fred Dacimo

FRD/rw

A128
NRR

Attachment: 1. Reply to NRC Request for Additional Information Regarding the License Renewal Application

cc: Mr. William Dean, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. Dave Wrona, NRC Branch Chief, Engineering Review Branch I
Mr. Robert F. Kuntz, NRC Sr. Project Manager, Division of License Renewal
Mr. Douglas Pickett, NRR Senior Project Manager
Ms. Bridget Frymire, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Francis J. Murray, Jr., President and CEO NYSERDA

ATTACHMENT 1 TO NL-12-134

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE

LICENSE RENEWAL APPLICATION

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

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

RAI's Related to Reactor Vessel Internals Inspection Plan [NRC Ref. 5]

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.

Response to RAI 6

Indian Point Unit 2

1) To provide reasonable assurance that the RVI components are bounded by assumptions in the generic industry failure modes, effects and consequences analyses (FMECA), MRP-191, and functionality analyses, MRP-232, supporting the development of MRP-227-A, a response to request 2.b is provided.

2.a) Based on the response in item 1 above, no response is necessary.

2.b) The process used to verify that Indian Point Unit 2 (IP2) is reasonably represented by the generic industry program assumptions with regard to neutron fluence, temperature, materials, and stress values used in the development of MRP-227-A is as follows:

1. Identification of typical Westinghouse pressurized water reactor (PWR) internal components in MRP-191, Table 4-4.
2. Identification of IP2 PWR internal components.
3. Comparison of the typical Westinghouse PWR internal components to the IP2 PWR internal components.
    a. Confirm that no additional items were identified by this comparison
    b. Confirm that the materials identified for IP2 are consistent with those materials identified in MRP-191, Table 4-4.
    c. Confirm that the IP2 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.
4. Confirmation that the IP2 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.
5. Confirmation that the IP2 RVI materials operated at temperatures within the original design basis parameters.
6. Determination of stress values based on design basis documents.
7. Confirmation that any changes to the IP2 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

The IP2 RVI components are reasonably represented by assumptions regarding neutron fluence, temperature, materials, and stress values in the generic industry FMECA and functionality analyses based on the following:

1. IP2 operating history is consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to neutron fluence.
    a. The FMECA and functionality analyses for MRP-227-A were based on the assumption of 30 years of operation with high-leakage core loading

patterns followed by 30 years of low-leakage core fuel management strategy. IP2 started to go from high-leakage to low-leakage loading pattern in Fuel Cycle 6. In Fuel Cycle 6 (12/29/82) at 9 years of operation, IP2 switched to use of a low-low leakage loading pattern (L^3P). Then, in Fuel Cycle 12 (4/20/93) at 19 years of operation, IP2 switched to use of a low-low-low leakage loading pattern (L^4P). Therefore, IP2 meets the fluence and fuel management assumptions in MRP-191, MRP-232, and the requirements for application of the MRP-227-A aging management strategy.

b. IP2 has operated under base load conditions over the life of the plant. Therefore, IP2 satisfies the assumptions in MRP-227-A and supporting documents regarding operational parameters affecting fluence.

2. The IP2 RVI operate between T_{hot} and T_{cold}, which are not less than approximately 514°F (515.5°F prior to Stretch Power Uprate [SPU]) for T_{cold} and not higher than 605.8°F (611.7°F prior to SPU) for T_{hot}. The design temperature for the vessel is 650°F. Therefore, IP2 historical operation is within original design basis parameters and is consistent with the assumptions used to develop the aging management strategy in MPR-227-A with regard to temperature operational parameters.

3. IP2 RVI components and materials are covered by the list of generic Westinghouse-designed PWR RVI components in MRP-191, Table 4-4. Components and materials are summarized in [Entergy reference 1].

   a. As summarized in [Entergy reference 1], no additional components were identified for IP2 by this comparison.

   b. Most of the IP2 RVI component materials are consistent or nearly equivalent with those materials identified in MRP-191, Table 4-4 for Westinghouse-designed plants, except for a few components that were fabricated from CF8 cast austenitic stainless steel (CASS) material rather than the Type 304 stainless steel (SS) called out in MRP-191. These items along with the items that were fabricated from different but essentially equivalent materials are summarized in a Westinghouse proprietary document [Entergy reference 1]. Plant-specific details are proprietary and not typically released as part of RAI responses. If the NRC requires additional details, the calculation would be made available for the NRC to review the supporting technical basis. This typically occurs at either a plant location or at the Westinghouse Rockville office with Westinghouse personnel supporting the discussion.

   c. The design and fabrication of IP2 RVI components are the same as, or equivalent to, the typical Westinghouse-designed PWR RVI components.

4. Modifications to the IP2 RVI made over the lifetime of the plant are those identified in general industry guidance or specifically directed by Westinghouse. IP2 performed a SPU in 2004 where analyses were performed on RVI components and it was determined that the structural integrity of the reactor internals was maintained at the SPU conditions. The design has been maintained

over the lifetime of the plant as specified by Westinghouse, and IP2 has not made any modifications since May 2007, which meets requirements from MRP-227-A. Operational parameters with regard to fluence and temperature are compliant with requirements from MRP-227-A, and the components and materials are the same as those considered in MRP-191. Therefore, the IP2 stress values are reasonably represented by the assumptions in MRP-191, MPR-227- A, and MRP-232.

3) Per the response to item 2.b of this RAI, IP2 complied with Applicant/Licensee Action Item 1 of the NRC Safety Evaluation (SE) regarding MRP-227, Revision 0, except for the CF8 components that were called out as 304 SS in MRP-191. Those material differences were evaluated and determined, as summarized in [Entergy reference 1], to require no modifications to the generic MRP-227-A inspection strategy. Therefore, the requirement is met for application of MRP-227-A as a strategy for managing age-related material degradation in the IP2 components, and no additional aging mechanisms are applicable to components at IP2.

4) See response to item 3 of this RAI. A FMECA was done addressing the CF8 components that were called out as 304 SS in MRP-191 and determined that the components could be classified as "No Additional Measures" components based on a consideration of the likelihood of failure and the likelihood of damage. Documentation for this FMECA is summarized in [Entergy reference 1].

5) No design changes were made to IP2 that were beyond those identified in general industry guidance or as were recommended by the original vendors. There were no design changes implemented after May 2007 [Entergy reference 5].

Indian Point Unit 3

1) To provide reasonable assurance that the RVI components are bounded by assumptions in the generic industry FMECA, MRP-191, and functionality analyses, MRP-232, supporting the development of MRP-227-A, a response to request 2.b is provided.

2.a) Based on the response in item 1 above, no response is necessary.

2.b) The process used to verify that Indian Point Unit 3 (IP3) is reasonably represented by the generic industry program assumptions with regard to neutron fluence, temperature, materials, and stress values used in the development of MRP-227-A is as follows:

1. Identification of typical Westinghouse PWR internal components MRP-191, Table 4-4.

2. Identification of IP3 PWR internal components.

3. Comparison of the typical Westinghouse PWR internal components to the IP3 PWR internal components.

    a. Confirm that no additional items were identified by this comparison.

b. Confirm that the materials identified for IP3 are consistent with those materials identified in MRP-191, Table 4-4.

c. Confirm that the IP3 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Confirmation that the IP3 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.

5. Confirmation that the IP3 RVI materials operated at temperatures within the original design basis parameters.

6. Determination of stress values based on design basis documents.

7. Confirmation that any changes to the IP3 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

The IP3 RVI components are reasonably represented by assumptions regarding neutron fluence, temperature, materials, and stress values in the generic industry FMECA and functionality analyses based on the following:

1. IP3 operating history is consistent with the assumptions used to develop the MRP-227-A with regard to neutron fluence.

   a. The FMECA and functionality analyses for MRP-227-A were based on the assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. IP3 started to go from high-leakage to low-leakage loading pattern in Fuel Cycle 4. In Fuel Cycle 7 (6/24/84) at 9 years of operation, IP3 switched to use of a low-low-leakage loading pattern (L^3P). Then, in Fuel Cycle 14 (4/7/05) at 30 years of operation, IP3 switched to use of a low-low-low- leakage loading pattern (L^4P). Therefore, IP3 meets the fluence and fuel management assumptions in MRP-191, MRP-232, and the requirements for application of the MRP-227-A aging management strategy.

   b. IP3 has operated under base load conditions over the life of the plant. Therefore, IP3 satisfies the assumptions in MRP-227-A and supporting documentation regarding operational parameters affecting fluence.

2. The IP3 RVI operate between T_{hot} and T_{cold}, which are not less than approximately 517°F (541.9°F prior to SPU) for T_{cold} and not higher than 603°F (600.8°F prior to SPU) for T_{hot}. The design temperature for the vessel is 650°F. Therefore, IP3 historical operation is within original design basis parameters and is consistent with the assumptions used to develop the aging management strategy in MRP-227-A with regard to temperature operational parameters.

3. IP3 RVI components and materials are covered by the list of generic Westinghouse-designed PWR RVI components in MRP-191, Table 4-4. Components and materials are summarized in [Entergy reference 1].

a. As summarized in [Entergy reference 1], no additional components were identified for IP3 by this comparison.

b. Most of the IP3 RVI component materials are consistent or nearly equivalent with those materials identified in MRP-191, Table 4-4 for Westinghouse-designed plants, except for a few components that were fabricated from CF8 CASS material rather than the Type 304 SS called out in MRP-191. These items along with the items that were fabricated from different but essentially equivalent materials are summarized in a Westinghouse proprietary document [Entergy reference 1]. Plant-specific details are proprietary and not typically released as part of RAI responses. If the NRC requires additional details, the calculation would be made available for the NRC to review the supporting technical basis. This typically occurs at either a plant location or at the Westinghouse Rockville office with Westinghouse personnel supporting the discussion.

c. The design and fabrication of IP3 RVI components are the same as, or equivalent to, the typical Westinghouse-designed PWR RVI components.

4. Modifications to the IP3 RVI made over the lifetime of the plant are those identified in general industry guidance or specifically directed by Westinghouse. IP3 performed a SPU in 2005 where analyses were performed on RVI components and it was determined that the structural integrity of the reactor internals was maintained at the SPU conditions. The design has been maintained over the lifetime of the plant as specified by Westinghouse, and IP3 has not made any modifications since May 2007, which meets requirements from MRP-227-A. Operational parameters with regard to fluence and temperature are compliant with requirements from MRP-227-A, and the components and materials are the same as those considered in MRP-191. Therefore, the IP3 stress values are reasonably represented by the assumptions in MRP-191, MPR-227-A, and MRP-232.

3) Per the response to item 2.b of this RAI, IP3 complied with Applicant/Licensee Action Item 1 of the NRC SE regarding MRP-227, Revision 0, except for the CF8 components that were called out as 304 SS in MRP-191. Those material differences were evaluated and determined, as summarized in [Entergy reference 1], to require no modifications to the generic MRP-227-A inspection strategy. Therefore, the requirement is met for application of MRP-227-A as a strategy for managing age-related material degradation in the IP3 components, and no additional aging mechanisms are applicable to components at IP3.

4) See response to item 3 of this RAI. A FMECA was done addressing the CF8 components that were called out as 304 SS in MRP-191 and determined that the components could be classified as "No Additional Measures" components based on a consideration of the likelihood of failure and the likelihood of damage. Documentation for this FMECA is summarized in [Entergy reference 1].

5) No design changes were made to IP3 that were beyond those identified in general industry guidance or as were recommended by the original vendors. There were no design changes implemented after May 2007 [Entergy reference 5].

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.

Response to RAI 7

Indian Point Unit 2

In RAI 7, the NRC notes that all IP2 components within the scope of license renewal compare with an equivalent component designation in MRP-191 with the exception of the lower internals assembly column cap. The lower internals assembly column cap is a CASS piece welded onto the top of the core support column shaft. These two pieces together constitute the lower internals assembly – column body. The development of the MRP-227-A aging management strategy considered the lower support column as one complete unit denoted as the "lower support column assemblies – lower support column bodies." Under the lower support column bodies in MRP-191, both Type 304 SS and CF8 CASS material were considered, as summarized in [Entergy reference 1]. Since the lower internals assembly column cap is part of the lower support column body, it was addressed by the generic industry FMECA and functionality analysis; therefore, it is subject to the same inspection requirements as the lower support column bodies (cast). Since the column cap is a part of the lower support column body, it would be subject to the same inspection requirements as the lower support column body. The inspection of the lower support column body includes the column cap.

Indian Point Unit 3

In RAI 7, the NRC notes that all IP3 components within the scope of license renewal compare with an equivalent component designation in MRP-191 with the exception of the lower internals assembly column cap. The lower internals assembly column cap is a CASS piece welded onto the top of the core support column shaft. These two pieces together constitute the lower internals assembly – column body. The development of the MRP-227-A aging management strategy considered the lower support column as one complete unit denoted as the "lower support column assemblies – lower support column bodies." Under the lower support column bodies in MRP-191, both Type 304 SS and CF8 CASS material were considered, as summarized in [Entergy reference 1]. Since the lower internals assembly column cap is part of the lower support column body, it was addressed by the generic industry FMECA and functionality analysis; therefore, it is subject to the same inspection requirements as the lower support column bodies (cast). Since the column cap is a part of the lower support column body, it would be subject to the same inspection requirements as the lower support column body. The inspection of the lower support column body includes the column cap.

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?

Response to RAI 9

Indian Point Unit 2 and Indian Point Unit 3

1. The acceptable criteria are based on the measured height of the spring as a function of time relative to the required hold-down force. Applicable plant loading conditions were evaluated. Time-dependent details for the hold down spring (HDS) height measurements are summarized in [Entergy reference 1]. Confirmatory actions are included if applicable after completion of the evaluation. Plant-specific details are proprietary and not typically released as part of RAI responses. If the NRC requires additional details, the calculation would be made available for the NRC to review the supporting technical basis. This typically occurs at either a plant location or at the Westinghouse Rockville office with Westinghouse personnel supporting the discussion.

2. The decrease in HDS height is assumed to occur linearly over time. The approach used to develop the HDS height acceptance criteria is to consider the actual HDS height at plant start-up and the required HDS heights at the end of 60 years. A linear interpolation at the time of the HDS height measurement determines the required minimum HDS height.

3. The HDS height measures less than the required minimum HDS heights, as summarized in [Entergy reference 1], indicating a need for re-evaluation and successive measurements or a replacement HDS.

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.

Response to RAI 10

Indian Point Unit 2 and Indian Point Unit 3

The RVI Program is the base program that addresses the RVI components that require aging management and specifies inspection requirements. The Thermal Aging and Neutron Irradiation Embrittlement of CASS Program at Indian Point (IPEC LRA) augments the RVI program by evaluating the potential susceptibility of plant-specific CASS components based on component specific chemistry.

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 IP2/1P3 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.

Response to RAI 11

Indian Point Unit 2 and Indian Point Unit 3

1) The approach used to evaluate the susceptibility of the CASS components at IP2 and IP3 to thermal and neutron embrittlement is a screening approach. A screening approach was used as this is consistent with the MRP-227-A development methodology and a methodology and screening values have been endorsed by the NRC [Entergy reference 3] for application with CASS materials.

2) The acceptance criteria for the plant-specific CASS component susceptibility evaluations based on fabrication and chemistry, taken from [Entergy references 2 and 3], are summarized in the following table.

Molybdenum (wt %)	Casting Method	Delta-Ferrite %	NRC Susceptibility Evaluation
High 2.0-3.0	Static	>14%	Potentially susceptible to TE
		≤14%	Not susceptible to TE
	Centrifugal	>20%	Potentially susceptible to TE
		≤20%	Not susceptible to TE
Low 0.5 max	Static	>20%	Potentially susceptible to TE
		≤20%	Not susceptible to TE
	Centrifugal	all	Not susceptible to TE

3) Commitment 47 in [Entergy reference 4] states that IPEC will perform and submit analyses that demonstrate that the lower support column bodies will maintain their functionality during the period of extended operation. These analyses will consider the possible loss of fracture toughness due to thermal and irradiation embrittlement. The analyses will be consistent with the IP2/IP3 licensing basis.

NRC 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)

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