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Fred Dacimo
Vice President
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NL-12-089

June 14, 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

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment 1, a reply to the additional information requested in the referenced letter 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 questions on LRA Amendment No. 9 and the Reactor Vessel Internals (RVI) Program.

As an initial matter, with regard to the RAIs on the RVI Program, Entergy notes that Indian Point is in a unique position with respect to the timing and implementation of the generic industry guidance for reactor vessel internals aging management (MRP-227-A). The Electric Power Research Institute (EPRI) just issued the NRC-approved version of MRP-227-A in January of this year, and the industry is working, through EPRI and the Pressurized Water Reactor Owners' Group (PWROG), to develop guidance on the required plant-specific evaluations for submittal to the NRC, including evaluations referenced in the RAIs. As a result of Indian Point's unique position, however, Entergy must prepare the requested evaluations in advance of this guidance which will require additional time beyond the requested 30-day response period. Nevertheless, in this letter Entergy provides responses to RAIs 1-5, 8, and 12. Entergy will develop the required evaluations and submit responses to the remaining RAIs by 09/28/2012.

Attachment 2 provides the latest list of regulatory commitments including the commitment made in response to RAI 11 in this letter.

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

A128
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 14, 2012.

Sincerely,

Patrick W. Conroy acting for Fred Dacimo

FRD/rw

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

2. License Renewal Application IPEC List of Regulatory Commitments Revision 18.

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-12-089

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)

NRC 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.

Response to RAI 1

Table 2.3.1.4-IP2 was numbered incorrectly in Amendment 9 and should have been identified as Table 2.3.1-2-IP2. Table 2.3.1-2-IP3 was inadvertently omitted from the Amendment 9 submittal; however it would have been the same as Table 2.3.1-2-IP2. Tables 2.3.1-2-IP2 and 2.3.1-2-IP3 are presented below as they should have appeared in Amendment 9.

Table 2.3.1-2-IP2
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Lower Core Support Structure	
Core baffle/former assembly • bolts	Structural support
Core baffle/former assembly • plates	Structural support Flow distribution Shielding
Core barrel assembly • bolts and screws	Structural support
Core barrel assembly • axial flexure plates (thermal shield flexures)	Structural support
Core barrel assembly • flange	Structural support
Core barrel assembly • ring • shell • thermal shield	Structural support Flow distribution Shielding
Core barrel assembly • lower core barrel flange weld • upper core barrel flange weld	Structural support
Core barrel assembly • outlet nozzles	Flow distribution
Lower internals assembly • clevis insert bolt • clevis insert • fuel alignment pin • lower core support plate column sleeves • lower core support plate column bolt • radial key	Structural support

Table 2.3.1-2-IP2
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Lower internals assembly • intermediate diffuser plate	Flow distribution
Lower internals assembly • lower core plate • lower core support castings • column cap • lower core support • secondary core support	Structural support Flow distribution
Upper Core Support Structure—Upper Internals Assembly	
RCCA guide tube assembly • bolt	Structural support
RCCA guide tube assembly • guide tube (including lower flange welds)	Structural support
RCCA guide tube assembly • guide plates	Structural support
RCCA guide tube assembly • support pin	Structural support
Core plate alignment pin	Structural support
Head / vessel alignment pin	Structural support
Hold-down spring	Structural support
Mixing devices • support column orifice base • support column mixer	Structural support Flow distribution
Support column	Structural support
Upper core plate, fuel alignment pin	Structural support Flow distribution

Table 2.3.1-2-IP2
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Upper support plate, support assembly (including ring)	Structural support
Upper support column bolt	Structural support
Incore Instrumentation Support Structure	
Bottom mounted instrumentation column	Structural support
Flux thimble guide tube	Structural support
Thermocouple conduit	Structural support

Table 2.3.1-2-IP3
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Lower Core Support Structure	
Core baffle/former assembly • bolts	Structural support
Core baffle/former assembly • plates	Structural support Flow distribution Shielding
Core barrel assembly • bolts and screws	Structural support
~~Core barrel assembly~~ ~~• axial flexure plates~~ ~~• flange~~ ~~• ring~~ ~~• shell~~ ~~• thermal shield~~	~~Structural support~~ ~~Flow distribution~~ ~~Shielding~~
Core barrel assembly • axial flexure plates (thermal shield flexures)	Structural support
Core barrel assembly • flange	Structural support
Core barrel assembly • ring • shell • thermal shield	Structural support Flow distribution Shielding
Core barrel assembly • lower core barrel flange weld • upper core barrel flange weld	Structural support
Core barrel assembly • outlet nozzles	Flow distribution

Table 2.3.1-2-IP3
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Lower internals assembly • clevis insert bolt • clevis insert • fuel alignment pin • lower core support plate column bolt • lower core support plate column sleeves • radial key	Structural support
Lower internals assembly • intermediate diffuser plate	Flow distribution
Lower internals assembly • lower core plate • lower core support castings • column cap • lower core support • secondary core support	Structural support Flow distribution
Upper Core Support Structure—Upper Internals Assembly	
~~RCCA guide tube assembly~~ ~~• bolt~~ ~~• guide tube~~ ~~• support pin~~	~~Structural support~~
RCCA guide tube assembly • bolt	Structural support
RCCA guide tube assembly • guide tube (including lower flange welds)	Structural support
RCCA guide tube assembly • guide plates	Structural support
RCCA guide tube assembly • support pin	Structural support
Core plate alignment pin	Structural support
Head / vessel alignment pin	Structural support

Table 2.3.1-2-IP3
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Hold-down spring	Structural support
Mixing devices • support column orifice base • support column mixer	Structural support Flow distribution
Support column	Structural support
Upper core plate, fuel alignment pin	Structural support Flow distribution
Upper support plate, support assembly <u>(including ring)</u>	Structural support
Upper support column bolt	Structural support
Incore Instrumentation Support Structure	
Bottom mounted instrumentation column	Structural support
Flux thimble guide tube	Structural support
Thermocouple conduit	Structural support

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.

Response to 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 are revised as shown below to update the reference to MRP-227-A. (underline - added)

3.1.2.2.6 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling

Loss of fracture toughness due to neutron irradiation embrittlement and change in dimensions (void swelling) in stainless steel and nickel alloy reactor vessel internals components exposed to reactor coolant and neutron flux will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227<u>-A</u>. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals.

3.1.2.2.9 Loss of Preload due to Stress Relaxation

Loss of preload due to thermal stress relaxation (creep) would only be a concern in very high temperature applications (> 700°F) as stated in the ASME Code, Section II, Part D, Table 4. No IPEC internals components operate at > 700°F. Therefore, loss of preload due to thermal stress relaxation (creep) is not an applicable aging effect for the reactor vessel internals components. However, irradiation-enhanced creep (irradiation creep) or irradiation enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress; and, on void swelling if present. Therefore, loss of preload of stainless steel and nickel alloy reactor vessel internals components will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227<u>-A</u>. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals.

3.1.2.2.15 Changes in Dimensions due to Void Swelling

Changes in dimensions due to void swelling in stainless steel and nickel alloy reactor internal components exposed to reactor coolant will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227<u>-A</u>. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals.

3.1.2.2.17 Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking

Cracking due to stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), and irradiation-assisted stress corrosion cracking (IASCC) in PWR stainless steel and nickel alloy reactor vessel internals components will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227-A. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals.

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.

Response to RAI 3

1. There is no 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. In Tables 3.1.2-2-IP2 and 3.1.2-2-IP3, cracking for the "Upper support plate, support assembly (including ring)" is managed by the Water Chemistry Control – Primary and Secondary and Inservice Inspection Programs. This item is compared to NUREG-1801, Rev. 1, Volume 2 Item IV.B2-42, and aligned to Table 1 Item 3.1.1-30. Therefore, the RAI statement that "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," is incorrect.

2. LRA Section 3.1.2.2.12 was revised by Letter NL-11-101, dated August 22, 2011, to correct the omission of the section from Amendment 9. LRA Section 3.1.2.2.12, as revised by NL-11-101 is shown below. Additional revisions are shown, with strikethrough for deletion and underline for additions, to provide clarification on the use of the Inservice Inspection Program, and the updated reference to MRP-227-A.

3.1.2.2.12 Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Cracking due to SCC and IASCC in PWR stainless steel reactor internals exposed to reactor coolant will be managed by the Water Chemistry Control – Primary and Secondary <u>Program</u> and <u>the</u> Reactor Vessel Internals (RVI) <u>or Inservice Inspection (ISI)</u> Programs. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227<u>-A</u>. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals. <u>The RVI Program includes inspections of core support structures using the existing ASME Section XI, ISI Program as delineated in MRP-227-A, Table 4-9. Where credited for the management of cracking, the existing ISI Program is listed in Tables 3.1.2-2-IP2 and 3.1.2-2-IP3 in lieu of the RVI Program.</u>

3. In Tables 3.1.2-2-IP2 and 3.1.2-2-IP3, cracking for the "Upper support plate, support assembly (including ring)" is managed by the Water Chemistry Control – Primary and Secondary and Inservice Inspection Programs. This item is compared to NUREG-1801, Rev. 1, Volume 2 Item IV.B2-42, and aligned to Table 1 Item 3.1.1-30. This item corresponds to the matching entry in MRP-227-A, Table 4-9, Westinghouse Plants Existing Programs Components. Consistent with MRP-227-A, the ISI program is the "existing program" credited to manage cracking for this item. No other changes are required.

<u>RAI's Related to Reactor Vessel Internals Program</u>

<u>RAI 4</u>

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.

Response to RAI 4

The IPEC RVI Program plans to use the guidance of WCAP-17096-NP, Rev. 2 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 T J has been or will be developed for the baffle-former bolts. Therefore, the staff requests the applicant submit a T J for the IP2 and IP3 baffle-former bolts.

Response to RAI 5

MRP-227-A and its associated safety evaluation contain no requirement for the submittal of a technical justification for these inspections with the application to implement MRP-227-A. Baffle-former bolt inspections are not required to be performed until between 25 and 35 EFPY. Currently both IPEC units are at less than 28 EFPY. Therefore, inspections are required prior to 2019 at IP2 and 2021 at IP3. As a result, IPEC has not yet finalized the inspection schedule and has not yet selected the vendor to perform the inspections. Since the technical justification will be prepared by the vendor selected to perform the inspections, a technical justification has not yet been prepared for IPEC. A technical justification is planned to be developed for the baffle-former bolts when the inspection vendor is selected but no later than 6 months prior to the beginning of the outage when the inspections will be performed.

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.

<u>Response to RAI 6</u>

As noted in the cover letter the response to this RAI requires that additional evaluations be performed. The response will be submitted to the NRC by 09/28/2012.

<u>RAI 7</u>

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.

<u>Response to RAI 7</u>

As noted in the cover letter the response to this RAI requires that additional evaluations be performed. The response will be submitted to the NRC by 09/28/2012.

<u>RAI 8</u>

The staff requests the following information related to the applicant's response to Applicant/Licensee Action Item 3 from the NRC staffs 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.

Response to RAI 8

1: Cold-worked type 316 split pins have been installed at other nuclear power plants since 1997. No plants have experienced failures of cold-worked type 316 split pins to date.

2: No inspections are planned for the split pins at IP3. However, based on industry operating experience, if failures of cold-worked type 316 split pins occur, an IP3 plant specific evaluation will be performed at that time to determine if inspections are required. Since the IP3 split pins were replaced in 2009 and other plants have installed cold-worked type 316 split pins starting in 1997, failure of other plant split pins would be expected before potential failures at IP3. Any failure would be evaluated by IP3 to determine the need for an inspection and other actions. No plants have experienced any failures of cold-worked type 316 split pins to date.

3: IP2 plans to use the same replacement split pin material and design that was used for IP3. IP2 plans to use cold-worked type 316 split pins.

4: The inspection strategy for the replacement IP2 split pins during the PEO will be the same as the IP3 inspection strategy. No inspections are planned for the replacement IP2 split pins. However, based on industry operating experience, if failures of cold-worked type 316 split pins occur, an IP2 plant specific evaluation will be performed at that time to determine if inspections are warranted.

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

As noted in the cover letter the response to this RAI requires that additional evaluations be performed. The response will be submitted to the NRC by 09/28/2012.

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

As noted in the cover letter the response to this RAI requires that additional evaluations be performed. The response will be submitted to the NRC by 09/28/2012.

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

As noted in the cover letter the response to this RAI requires that additional evaluations be performed. The response will be submitted to the NRC by 09/28/2012.

Commitment 47

IPEC will perform and submit analyses that demonstrate that the lower support column bodies will maintain their functionality during the period of extended operation considering the possible loss of fracture toughness due to thermal and irradiation embrittlement. The analyses will be consistent with the IP2/IP3 licensing basis and will be submitted prior to the PEO.

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.

Response to RAI 12

IPEC will use the RVI Program to manage the effects of aging due to fatigue on the reactor vessel internals. The aging management strategy development described in MRP-227-A was based on consideration of susceptibility to eight age-related degradation mechanisms. Fatigue was one of the eight degradation mechanisms considered. As provided in Section 3.5.1 of the NRC's safety evaluation for MRP-227-A, for locations with a fatigue time-limited aging analysis, IPEC will manage the effects of aging due to fatigue through the Fatigue Monitoring Program in

accordance with 10 CFR 54.21(c)(1)(iii). For locations which do not have a current licensing basis fatigue analysis, IPEC will rely on the inspection requirements of MRP-227-A to manage the effects of aging due to fatigue.

Consistent with 10 CFR 54.21(c)(1)(iii) and the NRC's safety evaluation for MRP-227-A, the Fatigue Monitoring Program will manage the effects of aging due to fatigue on RVI components with a fatigue time-limited aging analysis. The Fatigue Monitoring Program as described in LRA Section B.1.12 provides assurance that the CUF remains below the allowable limit of 1.0.

Consistent with Section 3.5.1 of the safety evaluation for MRP-227-A, prior to entering the period of extended operation the existing RVI fatigue calculations will be reviewed to evaluate the effects of the reactor coolant system water environment on the CUF. Specifically, under Commitment 43, Entergy will review the IPEC design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the IP2 and IP3 configurations. This review includes ASME Code Class 1 fatigue evaluations for reactor vessel internals. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage.

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)

ATTACHMENT 2 TO NL-12-089

<u>LICENSE RENEWAL APPLICATION</u>
<u>IPEC LIST OF REGULATORY COMMITMENTS</u>

<u>Rev. 18</u>

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

List of Regulatory Commitments

Rev. 18

The following table identifies those actions committed to by Entergy in this document.

Changes are shown as strikethroughs for ~~deletions~~ and underlines for <u>additions</u>.

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
1	Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to perform thickness measurements of the bottom surfaces of the condensate storage tanks, city water tank, and fire water tanks once during the first ten years of the period of extended operation. Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to require trending of thickness measurements when material loss is detected.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.1 A.3.1.1 B.1.1
2	Enhance the Bolting Integrity Program for IP2 and IP3 to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and clarify the prohibition on use of lubricants containing MoS_2 for bolting. The Bolting Integrity Program manages loss of preload and loss of material for all external bolting.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.2 A.3.1.2 B.1.2 Audit Items 201, 241, 270

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
3	Implement the Buried Piping and Tanks Inspection Program for IP2 and IP3 as described in LRA Section B.1.6. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M34, Buried Piping and Tanks Inspection. Include in the Buried Piping and Tanks Inspection Program described in LRA Section B.1.6 a risk assessment of in-scope buried piping and tanks that includes consideration of the impacts of buried piping or tank leakage and of conditions affecting the risk for corrosion. Classify pipe segments and tanks as having a high, medium or low impact of leakage based on the safety class, the hazard posed by fluid contained in the piping and the impact of leakage on reliable plant operation. Determine corrosion risk through consideration of piping or tank material, soil resistivity, drainage, the presence of cathodic protection and the type of coating. Establish inspection priority and frequency for periodic inspections of the in-scope piping and tanks based on the results of the risk assessment. Perform inspections using inspection techniques with demonstrated effectiveness.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-09-106 NL-09-111 NL-11-101	A.2.1.5 A.3.1.5 B.1.6 Audit Item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
4	Enhance the Diesel Fuel Monitoring Program to include cleaning and inspection of the IP2 GT-1 gas turbine fuel oil storage tanks, IP2 and IP3 EDG fuel oil day tanks, IP2 SBO/Appendix R diesel generator fuel oil day tank, and IP3 Appendix R fuel oil storage tank and day tank once every ten years. Enhance the Diesel Fuel Monitoring Program to include quarterly sampling and analysis of the IP2 SBO/Appendix R diesel generator fuel oil day tank, IP2 security diesel fuel oil storage tank, IP2 security diesel fuel oil day tank, and IP3 Appendix R fuel oil storage tank. Particulates, water and sediment checks will be performed on the samples. Filterable solids acceptance criterion will be less than or equal to 10mg/l. Water and sediment acceptance criterion will be less than or equal to 0.05%. Enhance the Diesel Fuel Monitoring Program to include thickness measurement of the bottom of the following tanks once every ten years. IP2: EDG fuel oil storage tanks, EDG fuel oil day tanks, SBO/Appendix R diesel generator fuel oil day tank, GT-1 gas turbine fuel oil storage tanks, and diesel fire pump fuel oil storage tank; IP3: EDG fuel oil day tanks, EDG fuel oil storage tanks, Appendix R fuel oil storage tank, and diesel fire pump fuel oil storage tank. Enhance the Diesel Fuel Monitoring Program to change the analysis for water and particulates to a quarterly frequency for the following tanks. IP2: GT-1 gas turbine fuel oil storage tanks and diesel fire pump fuel oil storage tank; IP3: Appendix R fuel oil day tank and diesel fire pump fuel oil storage tank. Enhance the Diesel Fuel Monitoring Program to specify acceptance criteria for thickness measurements of the fuel oil storage tanks within the scope of the program. Enhance the Diesel Fuel Monitoring Program to direct samples be taken and include direction to remove water when detected. Revise applicable procedures to direct sampling of the onsite portable fuel oil contents prior to transferring the contents to the storage tanks. Enhance the Diesel Fuel Monitoring Program to direct the addition of chemicals including biocide when the presence of biological activity is confirmed.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-08-057	A.2.1.8 A.3.1.8 B.1.9 Audit items 128, 129, 132, 491, 492, 510

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	Enhance the External Surfaces Monitoring Program for IP2 and IP3 to include periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.10 A.3.1.10 B.1.11
6	Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures. Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.11 A.3.1.11 B.1.12, Audit Item 164
7	Enhance the Fire Protection Program to inspect external surfaces of the IP3 RCP oil collection systems for loss of material each refueling cycle. Enhance the Fire Protection Program to explicitly state that the IP2 and IP3 diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Enhance the Fire Protection Program to specify that the IP2 and IP3 diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion and cracking at least once each operating cycle. Enhance the Fire Protection Program for IP3 to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.12 A.3.1.12 B.1.13

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
8	Enhance the Fire Water Program to include inspection of IP2 and IP3 hose reels for evidence of corrosion. Acceptance criteria will be revised to verify no unacceptable signs of degradation. Enhance the Fire Water Program to replace all or test a sample of IP2 and IP3 sprinkler heads required for 10 CFR 50.48 using guidance of NFPA 25 (2002 edition), Section 5.3.1.1.1 before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner. Enhance the Fire Water Program to perform wall thickness evaluations of IP2 and IP3 fire protection piping on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function. Enhance the Fire Water Program to inspect the internal surface of foam based fire suppression tanks. Acceptance criteria will be enhanced to verify no significant corrosion.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-08-014	A.2.1.13 A.3.1.13 B.1.14 Audit Items 105, 106

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
9	Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to implement comparisons to wear rates identified in WCAP-12866. Include provisions to compare data to the previous performances and perform evaluations regarding change to test frequency and scope. Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to specify the acceptance criteria as outlined in WCAP-12866 or other plant-specific values based on evaluation of previous test results. Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to direct evaluation and performance of corrective actions based on tubes that exceed or are projected to exceed the acceptance criteria. Also stipulate that flux thimble tubes that cannot be inspected over the tube length and cannot be shown by analysis to be satisfactory for continued service, must be removed from service to ensure the integrity of the reactor coolant system pressure boundary.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.15 A.3.1.15 B.1.16

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
10	Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to include the following heat exchangers in the scope of the program. • Safety injection pump lube oil heat exchangers • RHR heat exchangers • RHR pump seal coolers • Non-regenerative heat exchangers • Charging pump seal water heat exchangers • Charging pump fluid drive coolers • Charging pump crankcase oil coolers • Spent fuel pit heat exchangers • Secondary system steam generator sample coolers • Waste gas compressor heat exchangers • SBO/Appendix R diesel jacket water heat exchanger (IP2 only) Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to perform visual inspection on heat exchangers where non-destructive examination, such as eddy current inspection, is not possible due to heat exchanger design limitations. Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to include consideration of material-environment combinations when determining sample population of heat exchangers. Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to establish minimum tube wall thickness for the new heat exchangers identified in the scope of the program. Establish acceptance criteria for heat exchangers visually inspected to include no indication of tube erosion, vibration wear, corrosion, pitting, fouling, or scaling.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-09-018	A.2.1.16 A.3.1.16 B.1.17, Audit Item 52
11	Deleted		NL-09-056 NL-11-101	

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
12	Enhance the Masonry Wall Program for IP2 and IP3 to specify that the IP1 intake structure is included in the program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.18 A.3.1.18 B.1.19
13	Enhance the Metal-Enclosed Bus Inspection Program to add IP2 480V bus associated with substation A to the scope of bus inspected. Enhance the Metal-Enclosed Bus Inspection Program for IP2 and IP3 to visually inspect the external surface of MEB enclosure assemblies for loss of material at least once every 10 years. The first inspection will occur prior to the period of extended operation and the acceptance criterion will be no significant loss of material. Enhance the Metal-Enclosed Bus Inspection Program to add acceptance criteria for MEB internal visual inspections to include the absence of indications of dust accumulation on the bus bar, on the insulators, and in the duct, in addition to the absence of indications of moisture intrusion into the duct. Enhance the Metal-Enclosed Bus Inspection Program for IP2 and IP3 to inspect bolted connections at least once every five years if performed visually or at least once every ten years using quantitative measurements such as thermography or contact resistance measurements. The first inspection will occur prior to the period of extended operation. The plant will process a change to applicable site procedure to remove the reference to "re-torquing" connections for phase bus maintenance and bolted connection maintenance.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-08-057	A.2.1.19 A.3.1.19 B.1.20 Audit Items 124, 133, 519
14	Implement the Non-EQ Bolted Cable Connections Program for IP2 and IP3 as described in LRA Section B.1.22.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.21 A.3.1.21 B.1.22

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
15	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program for IP2 and IP3 as described in LRA Section B.1.23. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E3, Inaccessible Medium-Voltage Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-11-032 NL-11-096 NL-11-101	A.2.1.22 A.3.1.22 B.1.23 Audit item 173
16	Implement the Non-EQ Instrumentation Circuits Test Review Program for IP2 and IP3 as described in LRA Section B.1.24. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E2, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.23 A.3.1.23 B.1.24 Audit item 173
17	Implement the Non-EQ Insulated Cables and Connections Program for IP2 and IP3 as described in LRA Section B.1.25. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.24 A.3.1.24 B.1.25 Audit item 173
18	Enhance the Oil Analysis Program for IP2 to sample and analyze lubricating oil used in the SBO/Appendix R diesel generator consistent with the oil analysis for other site diesel generators. Enhance the Oil Analysis Program for IP2 and IP3 to sample and analyze generator seal oil and turbine hydraulic control oil. Enhance the Oil Analysis Program for IP2 and IP3 to formalize preliminary oil screening for water and particulates and laboratory analyses including defined acceptance criteria for all components included in the scope of this program. The program will specify corrective actions in the event acceptance criteria are not met. Enhance the Oil Analysis Program for IP2 and IP3 to formalize trending of preliminary oil screening results as well as data provided from independent laboratories.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-11-101	A.2.1.25 A.3.1.25 B.1.26

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
19	Implement the One-Time Inspection Program for IP2 and IP3 as described in LRA Section B.1.27. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M32, One-Time Inspection.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.26 A.3.1.26 B.1.27 Audit item 173
20	Implement the One-Time Inspection – Small Bore Piping Program for IP2 and IP3 as described in LRA Section B.1.28. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M35, One-Time Inspection of ASME Code Class I Small-Bore Piping.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.27 A.3.1.27 B.1.28 Audit item 173
21	Enhance the Periodic Surveillance and Preventive Maintenance Program for IP2 and IP3 as necessary to assure that the effects of aging will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.28 A.3.1.28 B.1.29
22	Enhance the Reactor Vessel Surveillance Program for IP2 and IP3 revising the specimen capsule withdrawal schedules to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation. Enhance the Reactor Vessel Surveillance Program for IP2 and IP3 to require that tested and untested specimens from all capsules pulled from the reactor vessel are maintained in storage.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.31 A.3.1.31 B.1.32
23	Implement the Selective Leaching Program for IP2 and IP3 as described in LRA Section B.1.33. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M33 Selective Leaching of Materials.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.32 A.3.1.32 B.1.33 Audit item 173
24	Enhance the Steam Generator Integrity Program for IP2 and IP3 to require that the results of the condition monitoring assessment are compared to the operational assessment performed for the prior operating cycle with differences evaluated.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.34 A.3.1.34 B.1.35

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
25	Enhance the Structures Monitoring Program to explicitly specify that the following structures are included in the program. • Appendix R diesel generator foundation (IP3) • Appendix R diesel generator fuel oil tank vault (IP3) • Appendix R diesel generator switchgear and enclosure (IP3) • city water storage tank foundation • condensate storage tanks foundation (IP3) • containment access facility and annex (IP3) • discharge canal (IP2/3) • emergency lighting poles and foundations (IP2/3) • fire pumphouse (IP2) • fire protection pumphouse (IP3) • fire water storage tank foundations (IP2/3) • gas turbine 1 fuel storage tank foundation • maintenance and outage building-elevated passageway (IP2) • new station security building (IP2) • nuclear service building (IP1) • primary water storage tank foundation (IP3) • refueling water storage tank foundation (IP3) • security access and office building (IP3) • service water pipe chase (IP2/3) • service water valve pit (IP3) • superheater stack • transformer/switchyard support structures (IP2) • waste holdup tank pits (IP2/3) Enhance the Structures Monitoring Program for IP2 and IP3 to clarify that in addition to structural steel and concrete, the following commodities (including their anchorages) are inspected for each structure as applicable. • cable trays and supports • concrete portion of reactor vessel supports • conduits and supports • cranes, rails and girders • equipment pads and foundations • fire proofing (pyrocrete) • HVAC duct supports • jib cranes • manholes and duct banks • manways, hatches and hatch covers • monorails	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-08-057	A.2.1.35 A.3.1.35 B.1.36 Audit items 86, 87, 88, 417

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	• new fuel storage racks • sumps, sump screens, strainers and flow barriers Enhance the Structures Monitoring Program for IP2 and IP3 to inspect inaccessible concrete areas that are exposed by excavation for any reason. IP2 and IP3 will also inspect inaccessible concrete areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant concrete degradation is occurring. Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspections of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties and for inspection of aluminum vents and louvers to identify loss of material.			
	Enhance the Structures Monitoring Program for IP2 and IP3 to perform an engineering evaluation of groundwater samples to assess aggressiveness of groundwater to concrete on a periodic basis (at least once every five years). IPEC will obtain samples from at least 5 wells that are representative of the ground water surrounding below-grade site structures and perform an engineering evaluation of the results from those samples for sulfates, pH and chlorides. Additionally, to assess potential indications of spent fuel pool leakage, IPEC will sample for tritium in groundwater wells in close proximity to the IP2 spent fuel pool at least once every 3 months. Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspection of normally submerged concrete portions of the intake structures at least once every 5 years. Inspect the baffling/grating partition and support platform of the IP3 intake structure at least once every 5 years.		NL-08-127	Audit Item 360
	Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspection of the degraded areas of the water control structure once per 3 years rather than the normal frequency of once per 5 years during the PEO.			Audit Item 358

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	Enhance the Structures Monitoring Program to include more detailed quantitative acceptance criteria for inspections of concrete structures in accordance with ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" prior to the period of extended operation.		NL-11-032 NL-11-101	
26	Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program for IP2 and IP3 as described in LRA Section B.1.37. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.36 A.3.1.36 B.1.37 Audit item 173
27	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program for IP2 and IP3 as described in LRA Section B.1.38. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.37 A.3.1.37 B.1.38 Audit item 173
28	Enhance the Water Chemistry Control – Closed Cooling Water Program to maintain water chemistry of the IP2 SBO/Appendix R diesel generator cooling system per EPRI guidelines. Enhance the Water Chemistry Control – Closed Cooling Water Program to maintain the IP2 and IP3 security generator and fire protection diesel cooling water pH and glycol within limits specified by EPRI guidelines.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-08-057	A.2.1.39 A.3.1.39 B.1.40 Audit item 509
29	Enhance the Water Chemistry Control – Primary and Secondary Program for IP2 to test sulfates monthly in the RWST with a limit of <150 ppb.	IP2: September 28, 2013	NL-07-039	A.2.1.40 B.1.41
30	For aging management of the reactor vessel internals, IPEC will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	IP2: September 28, 2011 IP3: December 12, 2013 Complete	NL-07-039 NL-11-107	A.2.1.41 A.3.1.41

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
31	Additional P-T curves will be submitted as required per 10 CFR 50, Appendix G prior to the period of extended operation as part of the Reactor Vessel Surveillance Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.2.1.2 A.3.2.1.2 4.2.3
32	As required by 10 CFR 50.61(b)(4), IP3 will submit a plant-specific safety analysis for plate B2803-3 to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, the site may choose to implement the revised PTS rule when approved.	IP3: December 12, 2015	NL-07-039 NL-08-127	A.3.2.1.4 4.2.5
33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF. 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.	IP2: September 28, 2011 IP3: December 12, 2013 Complete	NL-07-039 NL-07-153 NL-08-021 NL-10-082	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008 Complete	NL-07-078 NL-08-074 NL-11-101	2.1.1.3.5
35	Perform a one-time inspection of representative sample area of IP2 containment liner affected by the 1973 event behind the insulation, prior to entering the period of extended operation, to assure liner degradation is not occurring in this area. Perform a one-time inspection of representative sample area of the IP3 containment steel liner at the juncture with the concrete floor slab, prior to entering the period of extended operation, to assure liner degradation is not occurring in this area. Any degradation will be evaluated for updating of the containment liner analyses as needed.	IP2: September 28, 2013 IP3: December 12, 2015	NL-08-127 NL-11-101 NL-09-018	Audit Item 27
36	Perform a one-time inspection and evaluation of a sample of potentially affected IP2 refueling cavity concrete prior to the period of extended operation. The sample will be obtained by core boring the refueling cavity wall in an area that is susceptible to exposure to borated water leakage. The inspection will include an assessment of embedded reinforcing steel. Additional core bore samples will be taken, if the leakage is not stopped, prior to the end of the first ten years of the period of extended operation. A sample of leakage fluid will be analyzed to determine the composition of the fluid. If additional core samples are taken prior to the end of the first ten years of the period of extended operation, a sample of leakage fluid will be analyzed.	IP2: September 28, 2013	NL-08-127 NL-11-101 NL-09-056 NL-09-079	Audit Item 359
37	Enhance the Containment Inservice Inspection (CII-IWL) Program to include inspections of the containment using enhanced characterization of degradation (i.e., quantifying the dimensions of noted indications through the use of optical aids) during the period of extended operation. The enhancement includes obtaining critical dimensional data of degradation where possible through direct measurement or the use of scaling technologies for photographs, and the use of consistent vantage points for visual inspections.	IP2: September 28, 2013 IP3: December 12, 2015	NL-08-127	Audit Item 361

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
38	For Reactor Vessel Fluence, should future core loading patterns invalidate the basis for the projected values of RTpts or C_VUSE, updated calculations will be provided to the NRC.	IP2: September 28, 2013 IP3: December 12, 2015	NL-08-143	4.2.1
39	Deleted		NL-09-079	
40	Evaluate plant specific and appropriate industry operating experience and incorporate lessons learned in establishing appropriate monitoring and inspection frequencies to assess aging effects for the new aging management programs. Documentation of the operating experience evaluated for each new program will be available on site for NRC review prior to the period of extended operation.	IP2: September 28, 2013 IP3: December 12, 2015	NL-09-106	B.1.6 B.1.22 B.1.23 B.1.24 B.1.25 B.1.27 B.1.28 B.1.33 B.1.37 B.1.38
41	IPEC will inspect steam generators for both units to assess the condition of the divider plate assembly. The examination technique used will be capable of detecting PWSCC in the steam generator divider plate assembly. The IP2 steam generator divider plate inspections will be completed within the first ten years of the period of extended operation (PEO). The IP3 steam generator divider plate inspections will be completed within the first refueling outage following the beginning of the PEO.	IP2: After the beginning of the PEO and prior to September 28, 2023 IP3: Prior to the end of the first refueling outage following the beginning of the PEO.	NL-11-032 NL-11-074 NL-11-090 NL-11-101	N/A

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
42	IPEC will develop a plan for each unit to address the potential for cracking of the primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds using one of the following two options. Option 1 (Analysis) IPEC will perform an analytical evaluation of the steam generator tube-to-tubesheet welds in order to establish a technical basis for either determining that the tubesheet cladding and welds are not susceptible to PWSCC, or redefining the pressure boundary in which the tube-to-tubesheet weld is no longer included and, therefore, is not required for reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary must be approved by the NRC as a license amendment request. Option 2 (Inspection) IPEC will perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. If weld cracking is identified: a. The condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and b. An ongoing monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators.	IP2: Prior to March 2024 IP3: Prior to the end of the first refueling outage following the beginning of the PEO. IP2: Between March 2020 and March 2024 IP3: Prior to the end of the first refueling outage following the beginning of the PEO.	NL-11-032 NL-11-074 NL-11-090 NL-11-096	N/A
43	IPEC will review design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the IP2 and IP3 configurations. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. IPEC will use the NUREG/CR-6909 methodology in the evaluation of the limiting locations consisting of nickel alloy, if any.	IP2: Prior to September 28, 2013 IP3: Prior to December 12, 2015	NL-11-032 NL-11-101	4.3.3

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
44	IPEC will include written explanation and justification of any user intervention in future evaluations using the WESTEMS "Design CUF" module.	IP2: Prior to September 28, 2013 IP3: Prior to December 12, 2015	NL-11-032 NL-11-101	N/A
45	IPEC will not use the NB-3600 option of the WESTEMS program in future design calculations until the issues identified during the NRC review of the program have been resolved.	IP2: Prior to September 28, 2013 IP3: Prior to December 12, 2015	NL-11-032 NL-11-101	N/A
46	Include in the IP2 ISI Program that IPEC will perform twenty-five volumetric weld metal inspections of socket welds during each 10-year ISI interval scheduled as specified by IWB-2412 of the ASME Section XI Code during the period of extended operation. In lieu of volumetric examinations, destructive examinations may be performed, where one destructive examination may be substituted for two volumetric examinations.	IP2: Prior to September 28, 2013	NL-11-032 NL-11-074	N/A
47	IPEC will perform and submit analyses that demonstrate that the lower support column bodies will maintain their functionality during the period of extended operation considering the possible loss of fracture toughness due to thermal and irradiation embrittlement. The analyses will be consistent with the IP2/IP3 licensing basis.	IP2: Prior to September 28, 2013 IP3: Prior to December 12, 2015	NL-12-089	N/A